Safety Evaluation Report

Renewal of the Facility Operating License for the University of California-Davis McClellan Nuclear Research Center TRIGA Research Reactor

License No. R-130
Docket No. 50-607

United States Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

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ABSTRACT

This safety evaluation input 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 Regents of the University of California (the licensee) for a 20-year renewal of Facility Operating License No. R-130 to continue to operate the University of California Davis McClellan Nuclear Research Center reactor (UCD/MNRC, the facility). In its safety review, the NRC staff considered information submitted by the licensee, past operating history recorded in the licensee's annual reports to the NRC, inspection reports prepared by NRC personnel, and firsthand observations. Based on 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, UCD/MNRC staff, or the environment.

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ABBREVIATIONS AND ACRONYMS

$	a unit of reactivity or money
Δk/k	absolute reactivity
%Δk/k	reactivity in percent
°C	temperature in degree(s) Celsius
°F	temperature in degree(s) Fahrenheit
°K	temperature in degree(s) Kelvin
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
AFB	Air Force Base
Al-28	aluminum-28
ALARA	as low as reasonably achievable
ALI	annual limit on intake
ANS	American Nuclear Society
ANSI	American National Standards Institute
AR	annual operating report
Ar-41	argon-41
BOL	beginning-of-life
Br	bromide
CAM	continuous air monitor
cm	centimeter
CDE	committed dose equivalent
Ci	curie
cm/s	centimeters per second
Co-60	cobalt-60
CY	calendar year
DAC	derived air concentration
DANU	Division of Advanced Reactors and Non-Power Production and Utilization Facilities
DEX	Division of Engineering and External Hazards
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DOE	U.S. Department of Energy
DSS	Division of Safety System
ECP	estimated critical position
EOL	end-of-life
EP	emergency plan
EPA	U.S. Environmental Protection Agency
ER	environmental report
FFCR	fuel-followed control rod

FR	*Federal Register*
ft	foot (feet)
FY	fiscal year
GA	General Atomics
γ	gamma
H	hydrogen
H-3	tritium
HEPA	high efficiency particulate absorption
HotSpot	DOE Lawrence Livermore National Laboratory HotSpot computer code
hr	hour
I-125	iodine-125
IFE	instrumented fuel element
in	inch(es)
IR	inspection report
ISG	interim staff guidance
kW	kilowatt
kWt	kilowatt thermal
Kr	Krypton
LCC	limiting core configuration
LEU	low-enriched uranium
l	liter
lb	pound
lbm	pound mass
LLNL	DOE Lawrence Livermore National Laboratory
LOCA	loss-of-coolant accident
LRA	license renewal application
LSSS	limiting safety system setting
m	meter
μCi/ml	microcuries per milliliter
MDNBR	minimum departure from nucleate boiling ratio
MCNP	DOE Los Alamos National Laboratory Monte Carlo N-Particle transport computer code
MHA	maximum hypothetical accident
Mg-27	magnesium-27
mm	millimeter
Mn-56	manganese-56
Mo-99	molybdenum-99
MNRC	McClellan Nuclear Research Center
mR/hr	milli-Roentgen per hour
mrem	millirem
mrem/hr	millirem per hour
MW	megawatt
MWt	megawatt thermal
Na-24	sodium-24

N-16	nitrogen-16
NMSS	Nuclear Material Safety and Safeguards
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSC	Nuclear Safety Committee
OCC	Operating Core Configuration
OSL	optically stimulated luminescent
pcm	One thousandth of a precent of reactivity, 1 pcm= $\rho(10^5)$
PDR	public document room
PSP	physical security plan
PTS	pneumatic transfer system
Radwaste	radioactive waste
RAI	request for additional information
RAM	radiation area monitor
REFS	Division of Rule Making, Environmental, and Financial Support
RF	release fraction
RO	reactor operator
RSO	radiation safety officer
RTR	research and test reactor
RWP	radiation work permit
SAR	safety analysis report
SER	safety evaluation report
SL	safety limit
SNM	special nuclear material
SOI	statement of intent
SRM	staff requirements memorandum
SRO	senior reactor operator
SS	stainless steel
S.S.	steady state
T-H	thermal-hydraulic
TCP	temperature control panel
TEDE	total effective dose equivalent
TRIGA	Training, Research, Isotope, General Atomics
TS	technical specification(s)
U	uranium
UCD	University of California – Davis
UNPL	Non-Power Production and Utilization Facility Licensing
UPS	uninterruptable power supply
USAF	United States Air Force
$U\text{-}ZrH_x$	uranium-zirconium hydride
W	watt
wt%	weight percent

Xe	xenon
Zr	zirconium

1 INTRODUCTION

1.1 Overview

By letter dated June 11, 2018, the Regents of the University of California (the licensee) submitted a license renewal application (LRA) to the U.S. Nuclear Regulatory Commission (NRC, the Commission) for a 20-year renewal of the Class 104c Facility Operating License No. R-130 (NRC Docket No. 50-607) for the University of California – Davis McClellan Nuclear Research Center (UCD/MNRC) Training, Research, Isotope, General Atomics (TRIGA) nuclear reactor (UCD/MNRC, the facility) (Ref. 1). By letter dated May 10, 2019, the licensee requested changes to the facility's LRA to better align with the anticipated facility mission for the term of the requested license renewal (Ref. 2).

By letter dated July 6, 2020, the licensee supplemented its LRA to reflect its decision to reduce the licensed thermal operating power level to 1.0 megawatt-thermal (MWt), to eliminate pulsing and square-wave operation, and to eliminate the irradiation of explosive materials in the reactor tank (Ref. 3). The updated UCD/MNRC LRA (hereafter referred to as the LRA) included an updated safety analysis report (SAR) (Ref. 4), financial qualifications (Ref. 5), environmental report (ER) (Ref. 6), proposed technical specifications (TSs) (Ref. 7), operator requalification program (Ref. 8), and emergency plan (EP) (Ref. 9). A copy of the physical security plan (PSP) (safeguards information-modified handling) was provided by separate letter.

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.51, "Continuation of license," paragraph (a) states, in part "[e]ach license will be issued for a fixed period of time to be specified in the license, but in no case to exceed 40 years from the date of issuance." The UCD/MNRC TRIGA reactor was originally licensed by the U.S. Air Force (USAF) to operate in accordance with Section 91b of the Atomic Energy Act of 1954, as noted in 10 CFR 50.11, "Exceptions and exemptions from licensing requirements," paragraph (a). The application of the USAF to construct the McClellan Air Force Base (AFB) TRIGA reactor was made in August 1987, and actual construction began the following month. The USAF issued its authorization to operate the reactor on January 19, 1990, and initial criticality followed immediately on January 20, 1990. Operation at 1.0 MWt began shortly thereafter on January 25, 1990, with power being increased to 2 MWt in April 1997. The McClellan AFB TRIGA reactor was designed and constructed to perform neutron radiography and irradiation services for the USAF and for other assigned tasks. During 1997, construction began on another facility that would provide a neutron beam for tomography and boron neutron capture therapy.

Because of the impending closure of McClellan AFB, the USAF applied for an NRC facility operating license to continue its use of the UCD/MNRC reactor, by letter dated October 23, 1996, as supplemented. The NRC staff completed its review and issued Facility Operating License No. R-130 on August 13, 1998, and documented its evaluation in NUREG-1630, "Safety Evaluation Report Related to the Issuance of a Facility Operating License for the Research Reactor at McClellan Air Force Base" (Ref. 10). The application for the NRC facility operating license was submitted, and the NRC staff review was performed, in accordance with the guidance in NRC, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," NUREG-1537, Parts 1 and 2, February 1996 (Ref. 11).

The McClellan AFB was scheduled to close in calendar year (CY) 2002, and the USAF was seeking to divest itself of the McClellan AFB MNRC reactor facility. By letter dated April 13, 1999, as supplemented, the USAF and the Regents of the University of California submitted an application to the NRC requesting approval of a proposed transfer of Facility Operating License No. R-130 from the USAF to the Regents of the University of California (Ref. 12). By letter dated February 1, 2000, the NRC issued an order approving the transfer of the facility operating license from the USAF to the Regents of the University of California and issued License Amendment No. 3 to amend the facility operating license and TSs to conform to the changes contained in the transfer order. Due to administrative errors, the NRC order was reissued in its entirety on February 17, 2000 (Ref. 13) and the USAF MNRC facility became the UCD/MNRC facility.

Because the licensee submitted the LRA 30 days before the expiration of the facility operating license, which was August 13, 2018, the timely renewal provision provided in 10 CFR 2.109(a), authorizes the licensee to continue operating the UCD/MNRC facility under the terms and conditions of the current license until the NRC staff completes action on the renewal request. A renewal would authorize continued operation of the UCD/MNRC facility for an additional 20 years from the issuance of the renewed license.

A Notice of Opportunity to Request a Hearing was published in the *Federal Register* (FR) on March 9, 2022 (87 FR 13334). No requests for a hearing were received.

The NRC staff based its review of the request to renew the UCD/NMRC facility operating license on the information contained in the LRA, as well as in supporting supplements in response to the NRC staff's regulatory audits and request for additional information (RAI). As part of its review, the NRC staff also conducted a site familiarization visit on April 9-11, 2019, to observe facility conditions and to discuss potential license renewal information needed by the NRC staff.

The NRC staff conducted regulatory audits from December 14, 2020, to August 31, 2021, and from February 1, 2022, to June 21, 2022, in order to obtain information supporting its LRA review. The regulatory audit plans were described in NRC staff letters dated December 10, 2020 (Ref. 14) and February 1, 2022 (Ref. 15). The results of the regulatory audits were provided by NRC staff letters dated September 14, 2021 (Ref. 16), and June 28, 2022 (Ref. 17). Following the regulatory audits, the licensee provided supplemental information by letters dated September 22, 2021 (Ref. 18), and June 3, 2022 (Ref. 19), and June 21, 2022 (Ref. 70).

The NRC staff also issued RAI letters dated November 30, 2021 (Ref. 20); and February 8 (Ref. 21), June 3 (Ref. 54), and June 24, 2022 (Ref. 71). The licensee provided its responses by letters dated December 17, 2021 (Ref. 22), March 30 (Ref. 23), June 21 (Ref. 72), and June 30, 2022 (Ref. 73). Hereafter, reference to the updated SAR includes the information provided by the supplemental information and the RAI responses.

The NRC staff reviewed the licensee's PSP, EP, and the selection and training plan for reactor personnel, to ensure that the plans were consistent with current NRC regulations and guidance, and the results of the NRC staff review are discussed below. The NRC staff also reviewed UCD/MNRC annual operating reports (ARs) from CY 2009 through CY 2020 (Ref. 24) and NRC inspection reports (IRs) from CY 2009 through CY 2021 (Ref. 25).

With the exception of the PSP, the material pertaining to this review may be examined or copied, for a fee, at the NRC's Public Document Room (PDR), located at One White Flint North, 11555 Rockville Pike, Rockville, Maryland. The NRC maintains the Agencywide Documents Access and Management System (ADAMS), which provides text and image files of the NRC's public documents. Documents related to this license renewal may be accessed through the NRC's Public Library on the Internet at http:/www.nrc.gov. If you do not have access to ADAMS or if you experience problems accessing the documents in ADAMS, contact the NRC PDR staff by telephone at 1-800-397-4209 or 1-301-415-4737, or send an email to the PDR at PDR.Resources@nrc.gov. The PSP is protected from public disclosure under 10 CFR 73.21, "Protection of Safeguards Information: Performance Requirements," and 10 CFR 2.390, "Public inspections, exemptions, requests for withholding," paragraph (d). Because parts of the SAR and RAI responses from the licensee contain security-related information and are protected from public disclosure, redacted versions are available to the public.

Section 7, "References," of this safety evaluation report (SER) contains the dates and associated ADAMS Accession numbers of the licensee's renewal application and related supplements, and documents used by the NRC staff to complete its review.

In conducting its review, the NRC staff evaluated the facility 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," 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," and 10 CFR Part 73, "Physical Protection of Plants and Materials"; the recommendations of applicable regulatory guides; and relevant accepted industry standards, such as the American National Standards Institute/American Nuclear Society (ANSI/ANS) 15 series. The NRC staff also referred to the recommendations contained in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," issued February 1996 (Ref. 11). Because no specific accident dose criterion in NRC regulations exist for research reactors, the NRC staff compared calculated dose values for accidents against the requirements in 10 CFR Part 20 (i.e., the standards for protecting employees and the public against radiation).

In SECY-08-0161, "Review of Research and Test Reactor License Renewal Applications," dated October 24, 2008 (Ref. 27), the NRC staff provided the Commission with information regarding plans to streamline the review of LRAs for research and test reactors (RTRs). The Commission issued its staff requirements memorandum (SRM)-SECY-08-0161, dated March 26, 2009 (Ref. 28). The SRM directed the NRC staff to streamline the renewal process for such RTRs, using some combination of the options presented in SECY-08-0161. The SRM also directed the NRC staff to implement a graded approach with a review scope commensurate with the risk posed by each facility. The graded approach incorporates elements of the alternative safety review approach discussed in Enclosure 1 to SECY-08-0161. In the alternative safety review approach, the NRC staff should consider the results of past NRC staff reviews when determining the scope of the review. A basic requirement, as contained in the SRM, is compliance with applicable regulatory requirements.

The NRC staff developed interim staff guidance (ISG) 2009-001, "Interim Staff Guidance on the Streamlined Review Process for License Renewal of 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 thermal power level of 2.0 MWt and greater, or requesting a power level increase, undergo a full review using NUREG-1537. Facilities with a licensed thermal power level less than 2.0 MWt undergo a focused review that centers on the most safety-significant aspects of the renewal application and relies on past NRC reviews for certain safety findings. The NRC staff issued a draft of the ISG available for public comment and the considered public comments received in its development of the final ISG. The NRC staff reviewed the UCD/MNRC LRA using the guidance in the final ISG, dated October 15, 2009 (Ref. 29). Since the licensed thermal power level requested for UCD is less than 2 MWt, the NRC staff performed a focused review of the licensee's LRA. Specifically, the NRC focused on reactor design and operation, accident analysis, TSs, radiation protection, waste management programs, financial requirements, environmental assessment, and changes to the facility made after submittal of the application.

The licensee is required to maintain a 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. Changes to the PSP can be made, by the licensee, in accordance with 10 CFR 50.54, "Conditions of licenses," paragraph (p), as long as those changes do not decrease the effectiveness of the plan. In its LRA (Ref. 1), the licensee provided a copy of its current PSP and indicated that no changes were needed as a result of the LRA. However, following NRC staff review conducted during the NRC audit, the license provided an updated PSP, revised in accordance with 10 CFR 50.54(p) (Ref. 30 - cover letter only). The NRC staff reviewed the UCD/MNRC PSP and found it in compliance with the applicable regulations of 10 CFR Part 73. Based on that finding, the NRC staff approved the UCD/MNRC PSP by letter dated May 3, 2022 (Ref. 31). In addition, the NRC staff performs routine inspections of the licensee's compliance with the requirements of the PSP. The NRC staff's review of UCD/MNRC's IRs for the past several years identified no violations.

As part of the LRA, the licensee submitted its current EP entitled, "University of California, Davis McClellan Nuclear Radiation Center (UCD/MNRC) Emergency Plan, MNRC-0001-DOC-09," dated June 2018 (Ref. 9). The NRC staff reviewed the UCD/MNRC EP, found that it met the applicable regulations, and based on that finding, approved the UCD/MNRC EP, documented by letter dated September 3, 2020 (Ref. 32). In response to NRC staff RAI letter (Ref. 54), the licensee provided an updated EP (Ref. 72), which incorporated changes to its response to a loss-of-coolant accident (LOCA) (discussed in SER section 4.2.2, which was evaluated by the NRC staff and found acceptable). The licensee indicated that the changes to the EP did not constitute a reduction in effectiveness and were implemented in accordance with the requirements in 10 CFR 50.54(q), "Emergency plans." In addition, the NRC routinely inspects the licensee's compliance with the EP requirements. The NRC staff's review of UCD/MNRC's IRs for the past several years identified no violations. The licensee maintains an EP in compliance with 10 CFR 50.54(q) and appendix E to 10 CFR Part 50, "Emergency Planning and Preparedness for Production and Utilization Facilities," which provides reasonable assurance that the licensee will be prepared to assess and respond to emergency events.

As part of the LRA, the licensee submitted its current "University of California - Davis/McClellan Nuclear Radiation Center Selection and Training Plan for Reactor Personnel Document No. MNRC-0009-DOC-04," dated October 1999 (Ref. 8), and subsequently provided an updated training plan, dated September 2021 (Ref. 33). The NRC staff review found the updated training plan acceptable and notified the licensee by letter dated October 5, 2021 (Ref. 34).

The NRC staff also evaluated the environmental impacts of the renewal of the license for UCD/MNRC reactor in accordance with 10 CFR Part 51. The NRC staff published an Environmental Assessment and Finding of No Significant Impact in the *Federal Register* on October 20, 2022, (87 FR 63820), which concluded that renewal of the UCD/MNRC nuclear reactor operating license will not have a significant effect on the quality of the human environment.

The purpose of this SER is to summarize the findings resulting from the UCD/MNRC nuclear reactor safety review and to delineate the technical details that the NRC staff considered in evaluating the radiological safety aspects of continued operation. This SER and the Environmental Assessment and Finding of No Significant Impact will serve as the basis for issuance of a renewed license authorizing the operation of the UCD/MNRC nuclear reactor up to a steady-state thermal power level of 1.0 MWt.

This SER was prepared by Justin Hudson, and Geoffrey Wertz, Project Managers in the NRC's Office of Nuclear Reactor Regulation (NRR), Division of Advanced Reactors and Non-Power Production and Utilization Facilities (DANU), Non-Power Production and Utilization Facility Licensing (UNPL) Branch; Robert Beaton and Adam Rau, Nuclear Engineers in the NRC's NRR, Division of Safety Systems, Nuclear Systems Performance Branch; Richard Clement, Senior Health Physicist, and Zachary Gran, Health Physicist in NRC's NRR, Division of Risk Assessment, Radiation Protection and Consequence Branch; David Heeszel, Geophysicist and Rao Tammara, Physical Scientist in NRC's NRR, Division of Engineering and External Hazards (DEX), External Hazards Branch; Jorge Cintron-Rivera, Electrical Engineer in the NRC's NRR, DEX, Electrical Engineering Operating Reactors Branch; and Emil Tabakov, Financial Analyst in the NRC's Office of Nuclear Material Safety and Safeguards (NMSS), Division of Rulemaking, Environmental, and Financial Support (REFS), Financial Assessment Branch.

Major contributors to the UCD/MNRC renewal reviews include Paulette Torres, Project Manager and Nicole Parker, Licensing Assistant in the NRC's NRR, DANU, UNPL Branch; Beth Reed and Michael Takacs, Security Specialists (RTR) from the NRC's NRR, DANU, Non-Power Production and Utilization Facility Oversight Branch; and Kara McCullough, Emergency Preparedness Specialist and Michael Norris, Senior Emergency Preparedness Specialist in the NRC's Office of Nuclear Security and Incident Response, Reactor Licensing Branch. The Environmental Assessment and Finding of No Significant Impact supporting the UCD/MNRC renewal was prepared by Phyllis Clark, Nuclear Engineer, Briana Arlene, Biologist, Kevin Folk, Environmental Scientist, William Rautzen, Health Physicist, and Jeffrey Rikhoff, Senior Environmental Scientist in the NRC's NMSS, REFS, Environmental Review License Renewal Branch.

1.2 Summary and Conclusions on Principal Safety Considerations

The NRC staff's review and evaluation considers the information submitted by the licensee, including past operating history recorded in the licensee's ARs 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 UCD/MNRC reactor, the NRC staff concludes the following:

- The design and use of the reactor structures, systems, and components important to safety during normal operation discussed in chapter 2 of the SAR 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 teaching, research, training, and radionuclide production activities, as described in SAR section 1.3.

- The licensee considered the expected consequences of a broad spectrum of postulated credible accidents and a maximum hypothetical accident (MHA), emphasizing those that could lead to a loss of integrity of fuel element cladding and a release of fission products.

- The licensee performed analyses using conservative assumptions of the most serious credible accidents and the MHA and determined that the calculated potential radiation doses for the facility staff, and members of the public would not exceed 10 CFR Part 20 doses for unrestricted areas.

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

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

- The licensee's TSs, which provide limits controlling operation of the facility, offer a high degree of assurance that the facility will be operated safely. No significant degradation of the reactor has occurred, as discussed in chapter 4 of the SAR, and the TSs will continue to help ensure that no significant degradation of safety-related equipment will occur.

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

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

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

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

On the basis of these findings, the NRC staff concludes that there is reasonable assurance that the licensee can continue to operate the UCD/MNRC reactor in accordance with the Atomic Energy Act (AEA) of 1954, as amended, NRC regulations, and the renewed facility operating license without endangering public health and safety, facility personnel, or the environment. The issuance of the renewed license will not be inimical to the common defense and security.

1.3 General Description of the Facility

SAR Chapter 1 (Ref. 4) provides a general description of the UCD/MNRC reactor. SAR section 1.1, "Introduction," describes the UCD/MNRC reactor as a steady state 1,000 kilowatt (kW) TRIGA reactor. The reactor is operated by the UCD for neutron radiography and irradiation services for both the university and for non-university tasks. The facility is known as the UCD/MNRC. In SAR section 1.1.2, "Location of the Facility," the licensee states that the reactor is located in the UCD/MNRC Building on the former McClellan AFB, an industrial park of 2,600 acres located approximately 8 miles (13 kilometers (km)) northeast of Sacramento, California. The licensee indicated that the industrial park is adequately suited for the location of the UCD/MNRC reactor.

SAR section 1.2, "General Plant Description," states that the UCD/MNRC facility is a three-level 18,000 square foot (1,672 square meters) rectangular shaped building that incorporates the TRIGA reactor. The facility provides space, shielding, and environmental control for the radiography and irradiation services work. Room space has been provided to handle the experiments and components in the facility in a safe manner. The ground-level elements of the UCD/MNRC are constructed of reinforced concrete and concrete unit masonry with minor elements of exposed steel. The exterior walls of the upper portions feature factory-colored metal panels, concrete, and concrete unit masonry walls. The exterior walls of the radiography bays are made of reinforced concrete and vary in thickness from 2 to 3 feet (ft) (0.6 to 0.9 meters (m)). The interior walls and the roofs of the radiography bays are constructed of 2-ft (0.6-m) thick reinforced concrete. The reactor room is above the radiography bays. Its walls are constructed of standard-filled reinforced concrete block, and it has a typical metal deck built-up roof. The reactor is located in a cylindrical aluminum walled tank with the core positioned approximately 4.5 ft (1.4 m) below grade. The reactor tank is surrounded by a monolithic block of reinforced concrete. The basic purpose of the massive concrete structures is to provide biological shielding for personnel working in and around the UCD/MNRC. In addition, due to the massiveness of these structures, they provide excellent protection for the reactor core against natural phenomena. The facility exhaust systems are designed to maintain the reactor room and radiography bays at a slightly negative pressure with respect to surrounding areas to prevent the spread of radioactive contamination. These systems also maintain concentrations of radioactive gases in the reactor room and the radiography bays to levels that are below the 10 CFR Part 20 limits for restricted areas. The reactor and radiography control rooms each have its own air handling systems.

SAR section 1.2.2, "Reactor," states that the UCD/MNRC reactor is a 1.0 MWt steady state, natural-convection-cooled TRIGA reactor with a graphite reflector presently designed to accept the source ends of the four neutron radiography beam tubes which terminate in four separate neutron radiography bays. The reactor is located near the bottom of a water-filled aluminum tank 7 ft (2 m) in diameter and about 24.5 ft (7.5 m) deep. Direct visual and mechanical access to the core and mechanical components are available from the top of the tank for inspection, maintenance, and fuel handling. The water provides adequate shielding for personnel standing at the top of the tank. The control rod drives are mounted above the tank on a bridge structure spanning the diameter of the tank. The reactor is monitored and controlled by a computer-based instrumentation and control system featuring color graphics display and automatic logging of vital information. Both manual and automatic control options are available to the operator. The reactor console is located in the reactor control room and manages all control rod movements, accounting for such things as interlocks and choice of particular operating modes. It processes and displays information on control rod positions, power level, fuel temperatures, pulse characteristics, and other system parameters. The reactor console performs many other functions, such as monitoring reactor usage and storage of historical operating data for replay at a later time.

The licensee also states, in SAR section 1.2.2, that fuel for the UCD/MNRC reactor is standard TRIGA reactor fuel having uranium enriched to less than 20 percent uranium enriched in the isotope uranium-235. TRIGA reactor fuel is characterized by inherent safety, high fission product retention, and the demonstrated ability to withstand water quenching with no adverse reaction from temperatures to 1,150 degrees Celsius (°C) (2,102 degrees Fahrenheit (°F)). The inherent safety of TRIGA reactors has been demonstrated by extensive experience acquired from similar TRIGA systems throughout the world. This safety arises from the large prompt negative temperature coefficient that is characteristic of uranium-zirconium hydride fuel-moderator elements used in TRIGA systems. As the fuel temperature increases, this coefficient immediately compensates for reactivity insertions. This results in a mechanism whereby reactor power excursions are limited/terminated quickly and safely. Heat produced by the reactor core is removed by the primary and secondary cooling systems. The primary system circulates tank water through a water-to-water heat exchanger. The secondary water system gains heat in the heat exchanger and rejects it by use of a cooling tower. A purification system circulates a small amount of tank water through a filter and resin tanks to maintain purity and optical clarity. All of these systems contain the necessary instruments and controls for operations and monitoring performance.

1.4 Shared Facilities and Equipment

Shared facilities and equipment are described in SAR section 1.4. As the UCD/MNRC reactor is a natural convection open pool research reactor, offsite utilities are not required to place the reactor in a safe and secure shutdown configuration. Further, only the electrical service is shared with the building directly south of the UCD/MNRC, which is owned by the McClellan Business Park. The loss of offsite utilities cannot generate any accident scenario. The two utilities of most interest to all reactors are electricity and water. In the event of loss of electrical power, the facility maintains a backup battery supply which provides power to the console, reactor magnets, and all reactor indication for 15 minutes. This gives the reactor operator ample time to shut the reactor down in a controlled manner and verify the reactor has been placed in a secure condition. No electrical service (onsite or offsite) is required to maintain this secure condition.

Offsite water service is not required to shut down the reactor and to maintain it in a shutdown (subcritical) condition, nor to prevent the potential for fuel failure following a LOCA. In chapter 13, “Accident Analyses,” of the SAR, the analysis shows that the reactor does not need additional water to prevent unacceptable heating of the core during a complete LOCA event, but the ability to reflood the tank could mitigate LOCA doses to members of the public, as discussed in SER section 4.2.2, which was evaluated by the NRC staff and found acceptable.

The NRC staff reviewed the shared facilities and equipment and finds that the licensee provided a complete listing in the SAR and in its response to RAI 1 as stated above. Further, the NRC staff performed a site familiarization visit April 9 -11, 2019, and no shared utility concerns were observed or identified.

Based on its review, the NRC staff also finds that a malfunction or a loss of function of these shared facilities would not affect the operation of the non-power reactor, nor would it damage the UCD/MNRC reactor or its capability to be safely shut down. Additionally, a loss of function of the shared facilities would not create the potential or result in an uncontrolled release of radioactive material from the licensed facility to unrestricted areas.

1.5 Comparison with Similar Facilities

SAR section 1.3, “Relation of UCD/MNRC to Other TRIGA® Reactors,” states that the design of the UCD/MNRC fuel is similar to those of approximately thirty (30) TRIGA-type reactors currently operating world-wide with sixteen (16) in the United States. Most of these reactors were constructed in the late 1950s and 1960s. Since a large number of these reactors have been in operation for many years, considerable operational information is available, and their characteristics are well documented. Four of the 10 TRIGA reactors licensed for 1.0 MWt steady-state operation in the United States have characteristics similar to the UCD/MNRC reactor. These four reactors are located at: Pennsylvania State University, College Station, PA (1966); the U.S. Geological Survey Center, Denver, CO (1969); Oregon State University, Corvallis, OR (1967); and the University of Texas – Austin, TX (1990).

The NRC staff has compared the UCD/MNRC design bases and safety considerations with the facilities listed above that are also TRIGA-type reactors, and have similar fuel type, thermal power level, and siting considerations. The NRC staff finds that the history of these facilities demonstrates consistently safe operation that is acceptable. Further, the NRC staff finds that the UCD/MNRC reactor's design does not differ in any substantive way from similar TRIGA facilities that have been found acceptable to NRC and should be expected to perform in a similar manner when operated in accordance with its facility operating license.

1.6 Summary of Operations

As discussed in SAR section 1.1.1, UCD/MNRC provides a broad range of radiographic and irradiation services to the military and non-military sector. The facility has four radiography bays, each with a neutron beam for radiography purposes. The UCD/MNRC reactor core and associated experiment facilities are completely accessible for the irradiation of material, which include silicon doping, isotope production, both medical and industrial, and neutron activation analysis (e.g., geological samples). All radiography bays contain the equipment required to position parts for inspection as well as the radiography equipment. Further, the licensee indicated that, for the foreseeable future, the UCD/MNRC reactor will primarily function to support commercial neutron radiography and education/outreach programs. These programs can be accomplished by 1.0 MWt single shift operations.

SAR section 1.5, “Operational History,” describes that while the UCD/MNRC reactor was licensed for continuous 2.0 MWt steady state operation, since 2007, it has essentially operated as a single shift 1.0 MWt reactor. The change in reactor operation at 1.0 MWt was due to historical decrease in workload and little need for higher fluxes (i.e., silicon doping). By letters dated May 10 (Ref. 2) and June 14, 2019 (Ref. 35), the licensee stated that it planned to request changes to the LRA that would reduce the nominal steady-state operating power level, and to eliminate pulsing and square-wave operation, and irradiation of explosive materials in the reactor tank, and to provide an updated LRA SAR. As stated above, the licensee supplemented the LRA by letter dated July 6, 2020 (Ref. 3).

1.7 Compliance with the Nuclear Waste Policy Act of 1982

Section 302(b)(1)(B) of the Nuclear Waste Policy Act of 1982, as amended, specifies that the NRC may require, as a precondition to issuing or renewing an operating license for a research reactor, that the applicant reaches an agreement with the U.S. Department of Energy (DOE) for the disposal of high-level wastes and spent nuclear fuel. In a letter dated May 3, 1983, R.L. Morgan of DOE informed H. Denton of the NRC that DOE has determined that universities and other Government agencies operating non-power reactors have entered into contracts with DOE providing that DOE retains title to the fuel and is obligated to store or reprocess the spent fuel and high-level waste (Ref. 36). An email from DOE to the NRC reconfirms this obligation with respect to the fuel at UCD/MNRC (DOE Contract No. 74301, valid from August 25, 2008, to December 31, 2025) (Ref. 37). By entering into this contract with DOE, the licensee has satisfied the applicable requirements of the Nuclear Waste Policy Act of 1982, as amended.

1.8 Facility History and Modifications

The UCD/MNRC reactor was originally licensed by the USAF to operate in accordance with Section 91b of the Atomic Energy Act of 1954, as noted in 10 CFR 50.11(a). The application to construct the McClellan AFB TRIGA reactor was made in August 1987, and construction began the following month. The USAF issued its authorization to operate the reactor on January 19, 1990, and initial criticality followed immediately on January 20, 1990. Operation at 1.0 MWt began a shortly thereafter on January 25, 1990, with power being increased to 2.0 MWt in April 1997. The McClellan AFB TRIGA reactor was designed and constructed to perform neutron radiography and irradiation services for the USAF.

Because of the impending closure of McClellan AFB, the USAF McClellan AFB applied for an NRC license to continue operating the MNRC reactor, by letter dated October 23, 1996, as supplemented. The NRC staff completed its review and issued Facility Operating License No. R-130 on August 13, 1998, and documented its evaluation in NUREG-1630, “Safety Evaluation Report Related to the Issuance of a Facility Operating License for the Research Reactor at McClellan Air Force Base” (Ref. 10).

The McClellan AFB was scheduled to close in CY 2002, and the USAF was seeking to divest itself of the McClellan AFB MNRC reactor facility. By letter dated April 13, 1999, as supplemented, the USAF and the Regents of the University of California submitted an application to the NRC requesting approval of a proposed transfer of Facility Operating License No. R-130 from the USAF to the Regents of the University of California. By letter dated February 1, 2000, the NRC issued an order approving the transfer of the facility operating license from the USAF to the Regents of the University of California and issued License Amendment No. 3 to amend the facility operating license and technical specifications to conform to the changes contained in the transfer order. Due to administrative errors, the NRC order was reissued in its entirety on February 17, 2000 (Ref. 13) and the USAF MNRC facility became the UCD/MNRC facility.

As described in SAR section 1.6, "Facility Modifications," the UCD/MNRC has undergone relatively few facility modifications over the past 20 years. Facility modifications of low significance and facility modification requiring NRC approval were not included in the SAR. Facility modifications of low significance include air conditioning unit replacement, reroofing of the main building, upgrading internet network, and other equipment replacement unrelated to structures, systems, and components related to the operation of the reactor.

The licensee provided the following facility modifications in its LRA SAR:

- Bay 4 Reflector Insert: In February of 1999 the bay 4 beamline insert located in the UCD/MNRC reflector assembly was changed out with an insert that contained a sapphire crystal. This new reflector insert produces a much more thermalized (and attenuated) neutron beam in bay 4. The old beamline insert remains in shielded storage in radiography bay 1. The sapphire containing bay 4 beamline insert has remained in place to this day.

- Bay 2 Fuel Storage Area: In January of 2003 a fuel storage area was made for a subcritical assembly containing low-enriched uranium (LEU). The assembly was intended to be a flux booster placed in the reactor tank between the core and bay 5. The goal of the assembly was to boost neutron flux to provide shorter irradiation times for boron neutron capture therapy. The assembly was never placed in the UCD/MNRC reactor tank. The facility modification was closed out in 2007 when the assembly was returned to DOE.

- Iodine (I)-125 Production System: An I-125 production system was approved by the NRC in early 2000 and operated several dozen times before operational issues became too serious to continue. Use of the system was discontinued in 2006. UCD/MNRC currently has no capability or planned capability to produce I-125.

<u>I-125 Production System</u>

By letter dated August 9, 2001 (ML011580157), the NRC staff issued License Amendment No. 4 to modify TS 3.8.2, "Materials Limit," by adding specifications c and d, which provided limits on the production of I-125. By letter dated December 30, 2003 (ML033421339), the NRC staff issued License Amendment No. 7, amending the facility operating license to add LC 2.B.(4), which authorized the licensee to possess 40 curies of I-125. Based on the information provided in SAR section 1.6, "Facility Modification," which states that the UCD/MNRC facility currently has no capability or planned to produce I-125, the NRC staff has removed LC 2.B.(4) and TSs 3.8.2, Specifications c and d, as part of the license renewal (discussed in SER section 1.10, which was evaluated by the NRC staff and found acceptable).

The NRC staff reviewed the information described above, including NUREG-1630, and finds the facility changes described above to be consistent with the licensee's LRA SAR and planned operation of the facility. Further, the changes described above appear to be consistent with changes to the facility performed in accordance with the requirements of 10 CFR 50.59, "Changes, tests and experiments."

Current TS 6.7.1, "Annual Operating Reports," item 3, requires the licensee to describe in its AR any facility changes performed in accordance with the requirements of 10 CFR 50.59. The NRC staff reviewed UCD/MNRC ARs from CY 2009 through CY 2020 (Ref. 24) and NRC IRs from CY 2009 through CY 2021 (Ref. 25) and finds no discrepancies with the facility as described in the LRA SAR. The NRC staff concludes that all changes performed under 10 CFR 50.59 appear to be reasonable.

1.9 Financial Considerations

1.9.1 Financial Ability to Operate the Facility

The financial requirements for non-electric utility licensees are in 10 CFR 50.33, "Contents of applications; general information," paragraph (f) states, in part:

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

The UCD/MNRC reactor does not qualify as an "electric utility," as defined in 10 CFR 50.2, "Definitions." The application to renew or extend the term of any operating license for a non-power reactor shall include the financial information that is required in an application for an initial license. UCD/MNRC must meet the financial qualifications requirements pursuant to 10 CFR 50.33(f) and is subject to a full financial qualification review by the NRC. As required by 10 CFR 50.33(f)(2), the licensee must provide information to demonstrate 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 submit estimates of the total annual operating costs for each of the first 5 years of facility operations from the expected license renewal date and indicate the source(s) of funds to cover those costs. This is consistent with the guidance described in NUREG-1537 as it pertains to the NRC staff's review of the licensee's financial qualifications.

By letter dated June 10, 2020, "McClellan Nuclear Research Center Financial Qualification Report," (Ref. 5), the licensee provided financial data in support of its LRA. The applicant provided projected operating costs for UCD/MNRC for each of the fiscal years (FY) 2020 through FY 2025. The operating costs for the UCD/MNRC are projected to be:

Table 1 - McClellan Nuclear Research Center
Projected Annual Sources and Uses (in $1,000)

FY	2020	2021	2022	2023	2024	2025
Sources of Funds						
1. Provost Funding	$805	$805	$833	$861	$889	$916
2. Commercial Income and Recharge Activities	$1,299	$1,650	$2,063	$2,269	$2,269	$2,475
3. Research Income and Recharge Activities	$188	$154	$191	$250	$250	$250
Total Sources	**$2,292**	**$2,609**	**$3,087**	**$3,380**	**$3,408**	**$3,641**

FY	2020	2021	2022	2023	2024	2025
Uses of Funds						
4. Core Operations (non-profit) and Research Support	$899	$1,041	$969	$1003	$942	$970
5. For Profit Programs Support	$689	$776	$799	$828	$855	$881
6. Decommissioning Fund Augmentation and Debt Repayment	$704	$792	$1,319	$1,549	$1,611	$1,790
Total Uses	**$2,292**	**$2,609**	**$3,087**	**$3,380**	**$3,408**	**$3,641**
Percentage of Total Funding Dedicated to For Profit Programs (Item No. 4 divided by Total Sources/Uses)	30%	30%	26%	24%	25%	24%
10 CFR 50.22 Limit	No more than 50%					
Percentage of Total Funding/Revenue from Commercial Services (Item No. 2 divided by Total Sources/Uses)	57%	63%	67%	67%	67%	68%
NEIMA Limit	75% or less					

According to the licensee, UCD/MNRC operating revenues (i.e., sources of funds) are derived from multiple sources by category as: Provost Funding; Commercial Income and Recharge Activities; and Research Income and Recharge Activities. UCD/MNRC expenses are listed by category as: Core Operations (non-profit) and Research Support; For Profit Programs Support; and Decommissioning Fund Augmentation and Debt Repayment. As part of its review, the NRC staff considered guidance in NUREG-1537, as well as the projected operating costs and associated funding for similar research reactor facilities. The NRC staff finds the licensee's estimates for expenses and sources of funds to be reasonable.

The UCD/MNRC is currently licensed under Section 104c of the AEA, as amended, 42 U.S.C. §2234(c), as a facility that is useful in research and development. Pursuant to 10 CFR 50.21, "Class 104 licenses; for medical therapy and research and development facilities," paragraph (c) and 50.22, "Class 103 licenses; for commercial and industrial facilities," if a facility is to be licensed under Section 104c, as a non-commercial, non-power reactor facility that is useful in the conduct of research and development, then the facility is to be used so that not more than 50 percent of the annual cost of owning and operating the facility is devoted to the production of materials, products, or energy for sale or commercial distribution, or to the sale of services, other than research and development or education or training. Additionally, pursuant to Section 104c of the AEA, as amended, if a facility is to be licensed under Section 104c, then the licensee shall recover not more than 75 percent of the annual costs to the licensee of owning and operating the facility through sales of nonenergy services, energy, or both, other than research and development or education and training, of which not more than 50 percent may be through sales of energy.

The UCD/MNRC facility was originally built in the late 1980s by the USAF. This reactor was built to perform neutron radiography on airplane structures. According to the licensee, the UCD/MNRC provides irradiation services for various researchers at universities worldwide, U.S. national laboratories, and U.S. private industry. In its application, the licensee has confirmed that the annual cost of conducting the commercial activities at the UCD/MNRC facility is less than 50 percent of the annual cost of owning and operating the UCD/MNRC facility. The licensee provided financial information, in SER table 1, "McClellan Nuclear Research Center Projected Annual Sources and Uses (in $1,000)," to demonstrate compliance with the requirement in 10 CFR 50.21 by the ratio of Core Operations (non-profit) and Research Support to the total cost of owning the facility (Uses).

Additionally, the UCD application shows that 75 percent or less of the annual costs of owning and operating the UCD/MNRC are recovered through sales of nonenergy services, energy, or both, other than research and development or education and training. The licensee provided financial information, as shown in table 1 above, to demonstrate compliance with the requirement in Section 104c of the AEA by the ratio of Commercial Income and Recharge Activities to the Total Sources (or costs of owning and operating the facility). Because the licensee confirmed in the LRA, as supplemented, that the UCD/MNRC is used so that it meets the statutory requirements in Section 104c of the AEA and the regulatory requirements in 10 CFR 50.21(c), the NRC staff concludes that the renewed license can be issued pursuant to Section 104c of the AEA.

Based on the above discussion, the NRC staff finds that the licensee has provided the appropriate information for operating costs and has also demonstrated reasonable assurance for obtaining the necessary funds to cover these costs for the period of the renewed facility operating license. Accordingly, the NRC staff finds that the licensee has met the acceptance criteria in NUREG-1537 and financial qualifications requirements in 10 CFR 50.33(f).

1.9.2 Financial Ability to Decommission the Facility

Pursuant to 10 CFR 50.33(k), the NRC requires that an applicant for an operating license for a utilization facility submit information to demonstrate how reasonable assurance will be provided that funds will be available to decommission the facility.

Under 10 CFR 50.75, "Reporting and recordkeeping for decommissioning planning," paragraph (d)(1), each non-power reactor applicant for or holder of an operating license for a production or utilization facility shall submit a decommissioning report as required by 10 CFR 50.33(k). Pursuant to 10 CFR 50.75(d)(2), the report must contain a cost estimate for decommissioning the facility, an indication of the funding method(s) to be used to provide funding assurance for decommissioning, and a description of the means of adjusting the cost estimate and associated funding level periodically over the life of the facility. The acceptable method for providing financial assurance for decommissioning are described in 10 CFR 50.75(e)(1). The NRC staff used the guidance in NUREG-1537, chapter 15, "Financial Qualifications," to complete its review of the UCD/MNRC LRA as it pertains to financial assurance for decommissioning. The UCD/MNRC decommissioning cost estimate was developed using the methodology of NUREG/CR-1756, "Technology, Safety and Costs of Decommissioning Reference Research and Test Reactors," for a reference test reactor using the DECON option (Ref. 75). The licensee provided a decommissioning cost estimate of $25.7 million in 2020 dollars, by letter dated June 10, 2020 (Ref. 5). The cost estimate included itemized costs for staffing, labor, equipment service, indirect costs, radioactive waste, burial activities, and a 3 percent compounded contingency. The estimate did not include the cost associated with nuclear fuel removal and transport to the DOE facility. Based on the NRC staff's review of the UCD/MNRC LRA using guidance in NUREG-1537 and NUREG-1756, the NRC staff concludes that the decommissioning approach and decommissioning cost estimate submitted for the UCD/MNRC are reasonable.

The licensee has elected to use a statement of intent (SOI) to provide financial assurance, as allowed by 10 CFR 50.75(e)(1)(iv) for a Federal, State, or local government licensee. The SOI must contain or reference a cost estimate for decommissioning and indicate that funds for decommissioning will be obtained when necessary. The licensee provided an updated SOI in Attachment 2, "Statement of Intent (SOI) Regarding Decommissioning Funding for the UC Davis McClellan Nuclear Research Center and 1.0 MW TRIGA Reactor," in its letter dated June 10, 2020 (Ref. 5), stating that the Regents of the University of California will make funds available for decommissioning when necessary.

To support the SOI and qualifications to use an SOI, the licensee states that UCD/MNRC is a non-profit educational institution and a part of the State of California government in Attachment 2 of its Financial Qualification Report (Ref. 5). The Financial Qualification Report also provided the information needed to support UCD's representations that the decommissioning funding obligations of UCD/MNRC are backed by the full faith and credit of the State of California.

The NRC staff reviewed the licensee's information on decommissioning funding assurance as described above and finds that the licensee is a state government licensee under 10 CFR 50.75(e)(1)(iv), the SOI is acceptable, the decommissioning cost estimate appears reasonable, and the licensee's means of adjusting the cost estimate for UCD/MNRC and associated funding level periodically over the life of the facility is reasonable.

The NRC staff concludes that funds will be made available to decommission the facility and that the financial status of the licensee regarding decommissioning costs consistent with the guidance in NUREG-1537 and meets the requirements of 10 CFR 50.33(k) and 10 CFR 50.75. Therefore, the NRC staff concludes that the licensee has reasonable assurance that funds will be provided for decommissioning of the facility.

1.9.3 Foreign Ownership, Control, or Domination

Section 104d 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 regulations at 10 CFR 50.33(d) and 10 CFR 50.38, "Ineligibility of Certain Applicants," contain language to implement this prohibition. The Financial Qualification Report (Ref. 5) states that UCD/MNRC is owned and operated by the Regents of the University of California, an entity of the State of California. Further, the Financial Qualification Report states that UCD is a State of California government entity and is not owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government. Based on the above discussion, the NRC staff concludes that UCD/MNRC is not owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government.

1.9.4 Nuclear Indemnity

Pursuant to the requirements of the Price-Anderson Act (Section 170 of the AEA) and the NRC's implementing regulations at 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," the NRC staff notes that the Regents of the University of California has a current indemnity agreement for the UCD/MNRC with the Commission that will not expire until the license terminates. Therefore, UCD/MNRC will continue to be a party to the indemnity agreement following issuance of the renewed operating license. As required by Subpart D, "Provisions Applicable Only to Nonprofit Educational Institutions," of 10 CFR Part 140, the Commission will indemnify UCD/MNRC for any claims arising under its indemnity agreement pursuant to 10 CFR 140.95, "Appendix E - Form of Indemnity Agreement with Nonprofit Educational Institutions," above $250,000 and up to $500 million. In accordance with Subpart B, "Provisions Applicable Only to Applicants and Licensees Other Than Federal Agencies and Nonprofit Educational Institutions," to 10 CFR Part 140, the Regents of the University of California UCD/MNRC, as a nonprofit educational institution, is not required to provide nuclear liability insurance. Finally, as a research reactor UCD/MNRC is not required to maintain property insurance pursuant to 10 CFR 50.54(w).

1.9.5 Financial Consideration Conclusions

Based on its review as discussed above, the NRC staff concludes that there is reasonable assurance that the necessary funds will be available to support the continued safe operation of the UCD/MNRC and, when necessary, to shut down the facility and carryout the decommissioning activities in accordance with the requirements in 10 CFR 50.33(f), 10 CFR 50.33(k), and 10 CFR 50.75 and per guidance in NUREG-1537 and NUREG-1756. In addition, the NRC staff concludes that there are no problematic foreign ownership or control issues or insurance issues that would preclude the issuance of a renewed license.

1.10 Facility Operating License Possession Limits and License Changes

The NRC staff revised the UCD Facility Operating License No. R-130, by reformatting and renumbering the license conditions (LCs) for consistency with current reactor operating licenses, and incorporating changes requested by the licensee in its RAI response, by letter dated June 30, 2022 (Ref. 73). A description of these changes is provided below.

The current LC is followed by the proposed renewal LC.

Current UCD LC 2.B.(ii) states:

2.B.(ii) Pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," to receive, possess, and use up to 21.0 kilograms of contained uranium-235 enriched to less than 20 percent in the isotope uranium-235 in the form of reactor fuel; up to 4 grams of contained uranium-235 of any enrichment in the form of fission chambers; up to 16.1 kilograms of contained uranium-235 enriched to less than 20 percent in the isotope uranium-235 in the form of plates; and to possess, but not separate, such special nuclear material as may be produced by the operation of the facility.

Proposed renewal LC 2.B.2 states:

2.B.2 Pursuant to the Act and 10 CFR Part 70, to receive, possess, and use, but not separate, in connection with operation of the facility:

a. up to 16.0 kilograms of contained uranium-235 enriched to less than 20 percent in the isotope uranium-235 in the form of reactor fuel;

b. up to 4 grams of contained uranium-235 of any enrichment in the form of fission chambers;

c. up to 2 grams of special nuclear material of any enrichment in the form of foils for use in reactor-based experiments;

d. such special nuclear material as may be produced by the operation of the facility.

In its response to RAI 1 (Ref. 73), the licensee requested a reduction in the possession limit currently stated from 21.0 kilograms to 16.0 kilograms. The NRC staff revised renewal LC 2.B.2.a to 16.0 kilograms of contained uranium-235 enriched to less than 20 percent in the isotope uranium-235 in the form of reactor fuel.

The NRC staff moved the existing LC 2.B.(4).E. (see below), to proposed renewal LC 2.B.2.c, and modified it from “Special Nuclear Material, Any form, 2 grams per radionuclide not to exceed 5 grams total” to “up to 2 grams of special nuclear material of any enrichment in the form of foils for use in reactor-based experiments” to maintain consistency with the current license format for SNM authorized by 10 CFR Part 70.

The NRC staff deleted the authorization for up to 16.1 kilograms of contained uranium 235 enriched to less than 20 percent in the isotope uranium 235 in the form of plates, as stated in LC 2.B.(ii), as the licensee indicated that the material was no longer possessed at the UCD facility.

Current UCD LC 2.B.(iii) states:

2.B.(iii) Pursuant to the Act and 10 CFR Part 30, “Rules of General Applicability to Domestic Licensing of Byproduct material,” to receive, possess, and use a 4-curie sealed americium-beryllium neutron source in connection with operation of the facility; a 55-millicurie sealed cesium 137 source for instrument calibrations; small instrument calibration and check sources of less than 0.1 millicurie each; and to possess, use, but not separate, except for byproduct material produced In reactor experiments, such byproduct material as may be produced by the operation of the facility.

Proposed renewal LC 2.B.3 states:

2.B.3 Pursuant to the Act and 10 CFR Part 30, to receive, possess, and use, in connection with the operation of the facility:

a. a 4-curie sealed americium-beryllium neutron source;

b. a 55-millicurie sealed cesium-137 source for instrument calibrations;

c. up to 10.0 millicuries total of instrument calibration and check sources, with each source less than 0.1 millicurie;

d. up to 5.0 curies of byproduct material used in reactor-based experiments, in sources for calibration of radiation detectors, and references for use in reactor-based analytic techniques;

e. such byproduct material as may be produced by the operation of the facility, which cannot be separated except for byproduct material produced in reactor experiments.

In proposed renewal LC 2.B.3.c, the NRC staff added a total possession limit of 10 millicuries for instrument calibration and check sources, with each source less than 0.1 millicurie, as requested in the licensee's response to RAI 3 (Ref. 73)

In proposed renewal LC 2.B.3.d, the NRC staff added a possession limit of 5.0 curies of byproduct material used in reactor-based experiments, in sources for calibration of radiation detectors, and references for use in reactor-based analytic techniques, as requested in the licensee's response to RAI 4 (Ref. 73).

<u>Current UCD LC 2.B.(4) states:</u>

2.B.(4) In addition to those items specified in 2.B.(1), 2.B.(2) and 2.B.(3) the following radioactive materials may be received, possessed, and used at the facility.

Radioactive Material (element and mass number)	Chemical and/or Physical Form	Maximum Quantity Licensee May Possess at Any One Time
A. Any radioactive material between atomic number 1 through 83, inclusive	A. Any	A. A. 20 Curies (1 Curie each, except as provided below)
B. Any radioactive material with atomic numbers 84 and above	A. Any	A. 4 Curies (100 millicuries each, except as provided below) or up to 20 micrograms
C. Iodine-125	C. Iodide/Liquid	C. 40 Curies
D. Source material (but only trace amounts of Th-234	D. Any	D. 4 grams per radionuclide, not to exceed 10 grams total
E. Special nuclear material	E. Any	E. 2 grams per radionuclide, not to

exceed 5 grams total

Proposed renewal LC 2.B.4 states:

2.B.4. Pursuant to the Act and 10 CFR Part 40, in connection with the operation of the facility, to receive, possess, and use, up to 4 grams per radionuclide, not to exceed 10 grams total of source material for reactor-based experiments.

By License Amendment No. 4, dated August 9, 2001 (ML011580157), the NRC staff amended TS 3.8.2, “Materials Limit,” specifications c and d, providing limits on the production of I-125. By License Amendment No. 7, dated December 30, 2003 (ML033421339), the NRC staff amended the facility operating license, by adding the LC 2.B.(4).

SAR section 1.6, “Facility Modification,” states that the UCD/MNRC facility currently has no capability or planned to produce I-125. The NRC staff deleted the LCs associated with current LC 2.B.(4), A, B, C, and E, since the licensee states that the system had not been used and it currently has no capability or planned to produce I-125. Further, the licensee indicated that only current LC 2.B.(4) D was needed for current or planned experimental work activities. The NRC staff renumbered existing LC 2.B.(4) D to proposed renewal LC 2.B.4 to account for the material which is needed by the licensee to support experimental activities.

Other changes to the proposed renewal LCs:

The NRC staff revised and renumbered current LC 2.C.(i), “Maximum Power Level,” to proposed renewal LC 2.C.1 to reflect the licensee’s requested licensed power level limit of 1,000 kW (thermal) and remove the authorization to operate in pulse mode.

The NRC staff revised and renumbered current LC 2.C.(ii), “Technical Specifications,” to proposed renewal LC 2.C.2 to indicate the renewed license TSs.

The NRC staff revised and renumbered LC 2.C.(iii), “Physical Security Plan,” to proposed renewal LC 2.C.3 to indicate the updated PSP for the UCD/MNRC.

2 REACTOR DESCRIPTION

2.1 Summary Description

Safety Analysis Report (SAR) section 4.1 (Ref. 4), “Introduction,” states that the University of California – Davis/McClellan Nuclear Research Center UCD/MNRC reactor is a hexagonal grid, natural convection water cooled Training, Research, Isotopes, General Atomic (TRIGA) reactor. The reactor was originally licensed, as documented in NUREG-1630, “Safety Evaluation Report Related to the Issuance of a Facility Operating License for the Research Reactor at McClellan Air Force Base” (Ref. 10) to operate at a 2.3 megawatt thermal (MWt) steady state power and to operate in pulse and square wave modes. Where applicable, this safety evaluation report (SER) uses relevant information from NUREG-1630, given that the description provided has not changed since the issuance of NUREG-1630 in 1998. The primary function of the UDC/MNRC reactor is to support commercial and research neutron radiography and education/outreach programs, which can be accomplished at 1.0 MWt without pulsing or square wave operation. As such, the licensee has requested a license from the U.S. Nuclear Regulatory Commission (NRC) to continue steady-state operations at a licensed power level of 1.0 MWt and to eliminate pulsing and square-wave operation as specified in the proposed technical specifications (TSs).

NUREG-1630, section 4.1, “Introduction,” states that the UCD/MNRC reactor is a fixed-core, pool-type research reactor that uses light water as the moderator, coolant, and shielding. The reactor core is immersed in a reinforced concrete, water-filled, open pool. The pool is spanned by a fixed structure that supports the control rod systems, reactor instrumentation, and some experimental facilities. The core itself is located near the bottom of the pool, where it is supported on a structure that rests on the pool floor. The reactor uses standard TRIGA low enriched fuel with stainless steel cladding.

NUREG-1630, section 4.1 also states that reactor control is achieved by inserting or withdrawing up to six neutron-absorbing control rods suspended from the drive mechanisms. Heat generated by fission is transferred from the fuel to the pool water. Flow is driven by natural circulation. Pool water circulates through a heat exchanger in which the heat is transferred to the secondary system and released to the environment by the cooling tower. The instrumentation and control system for the UCD/MNRC reactor is a computer-based design incorporating a multifunction microprocessor-based neutron monitor channel developed by General Atomics (GA) and an analog-type neutron monitoring channel.

The NRC staff finds that safety of TRIGA reactors has been demonstrated by the extensive experience gained from TRIGA designs used throughout the world. TRIGA fuel is characterized by a strongly negative prompt temperature coefficient characteristic of uranium-zirconium hydride (U-ZrH) fuel moderator elements that contributes to safe operation. As the fuel temperature increases, this coefficient quickly responds with a sizable negative change in core reactivity. TRIGA fuel is also characterized by a high degree of fission product retention and the ability to withstand water quenching with no adverse reaction. NUREG-1282, “Safety Evaluation Report on High-Uranium Content, Low-Enriched Uranium-Zirconium Hydride Fuels for TRIGA Reactors” (Ref. 39), provides regulatory approval for TRIGA fuel. The GA and NRC reports documenting these features are listed below:

- The reactor kinetic behavior is discussed in GA-7882, “Kinetic Behavior of TRIGA Reactors, dated March 31, 1967 (Ref. 38);

- The fission product retention is discussed in NUREG-1282, “Safety Evaluation Report on High-Uranium Content, Low-Enriched Uranium-Zirconium Hydride Fuels for TRIGA Reactors,” issued August 1987 (Ref. 39), and “The U-ZrHx Alloy: Its Properties and Use in TRIGA Fuel,” M.T. Simnad, 1980 (Ref. 40); and

- The fuel performance during the Accident Analysis is discussed in NUREG/CR-2387, “Credible Accident Analyses for TRIGA and TRIGA-Fueled Reactors,” issued April 1982 (Ref. 41).

The NRC staff review focused on elements of the design analyses, which have not changed since the issuance of the NRC facility operating license for the UCD/MNRC reactor in 1998 as described in NUREG-1630. Although much of the design is unchanged, the licensee has requested that the licensed power level be reduced from 2.3 MWt to 1.0 MWt. The core configurations, as defined in the current TS 5.3.1, “Reactor Core,” have changed, and the licensee has described new operating core configuration (OCC) and limiting core configuration (LCC) in the updated LRA SAR to support these changes. The T-H analysis model has also been updated since the issuance of the NRC license in 1998 (Ref. 10).

2.2 Reactor Core

SAR section 4.2, “Reactor Core, Associated Structures, and Reactor Experiment Facilities,” describes the UCD/MNRC reactor core assembly, the reflector assembly, the grid plates, the safety plate, the fuel-moderator elements (also referred to as fuel elements or fuel element assemblies), including instrumented elements, the neutron source, the graphite dummy elements, the control rods and drives, the experiment facilities, and the beam tubes. The UCD/MNRC reactor core consists of the fuel-moderator assemblies (including the instrumented fuel element), reflector assemblies, grid plates, safety plate, neutron source, graphite elements, control rods, experimental facilities, and beam tubes.

2.2.1 Reactor Fuel

Fuel Types

SAR section 4.2.1, “Reactor Fuel,” provides details of the five TRIGA fuel types used at UCD/MNRC, which include TRIGA fuel types 8.5, 12, 20, 30 and 45 weight percent uranium, each having an enrichment slightly less than 20 percent uranium (U)-235, which were developed by GA, the fuel vendor. These fuel types are referred to as 8.5/20, 12/20, 20/20, 30/20, and 45/20 fuel, with the first number indicating the weight percent of uranium and the second number indicating the nominal percent enrichment of U-235. The licensee has proposed core configurations that use only the 20/20 and 30/20 TRIGA fuels. SAR section 4.1 states that the 8.5/20 fuel was previously used in the UCD/MNRC reactor core, and proposed TS 3.1.3, Specification 1, will require the license to limit the fuel types to 20/20 and 30/20. SAR section 4.6.4.3, “Operating Core Configuration (OCC),” states that the licensee plans to continue current operation using a mixed core of 20/20 and 30/20 fuel loading. SAR section 4.6.4.4, “Future Cores and the Limiting Core Configuration (LCC),” states that the licensee intends to acquire additional fresh 30/20 fuel elements, which will be loaded into the core as needed for continued operation.

Because the licensee has no plans to use and did not analyze cores containing fuel types other than 20/20 and 30/20 fuels, the use of other fuels in the reactor was not evaluated by the NRC staff and will not be authorized for use at UCD/MNRC, as limited by proposed TS 3.1.3, “Core Configuration Limitations,” (discussed in SER section 2.5.1, which was evaluated by the NRC staff and found acceptable).

Fuel Characteristics

SAR section 4.6.4.2, “Prompt Negative Temperature Coefficient,” states that the safety of the fuel arises from the strongly negative prompt temperature coefficient of reactivity characteristic of $U\text{-}ZrH_x$ fuel-moderator elements and discusses mechanisms of fuel temperature reactivity feedback. This temperature coefficient primarily arises from a reduction in the fuel utilization factor (that is, the fraction of neutrons absorbed in the fuel isotope instead of other materials) resulting from the heating of the $U\text{-}ZrH_x$ fuel-moderator elements. The coefficient is prompt because the fuel is intimately mixed with a large portion of the moderator; thus, fuel and solid moderator temperatures rise simultaneously. The heating of the fuel-moderator mixture causes the spectrum to harden in the fuel, which increases the leakage of neutrons from the fuel into the water, where some are absorbed. This yields a loss of reactivity.

As described in SAR section 4.6.4.2, spectrum hardening in the fuel also leads to preferential absorption in the burnable poison erbium, which reduces reactivity further. This feedback mechanism is inherently dependent on the concentration of erbium and has a decreasing impact as the concentration of erbium is reduced with burnup as shown by the burnup-dependent reactivity coefficients provided by the licensee in SAR table 13-4, “Maximum Reactivity Insertion and Related Quantities for Various Fuels and Burnups,” and SAR figure 13.1, “Prompt Negative Temperature Coefficient For TRIGA® Fuels.” An additional contribution to the prompt, negative temperature coefficient is the Doppler broadening of the U-238 resonances at high temperatures, which increases nonproductive neutron capture in these resonances.

SAR section 4.2.1.1, “Fuel-Moderator Element,” provides a description of the fuel element and states that the active part of each fuel-moderator element has a diameter of approximately 1.43 inches (3.63 centimeters (cm)) and is 15 inches (38.1 cm) long. The reactor fuel is a solid, homogeneous mixture of a $U\text{-}ZrH_x$ alloy. The hydrogen-to-zirconium atom ratio within the UCD/MNRC fuel varies from 1.6 to 1.7. The hydrogen in the alloy is a neutron moderator. Mixture of the moderator with the fuel in a solid form result in the moderator having the same operating conditions as the fuel. This design feature of the fuel contributes to the ability to safely pulse some TRIGA reactors. Erbium is also homogeneously mixed throughout the fuel (Ref. 51). Erbium acts as a burnable poison, allowing for longer core lifetimes by reducing reactivity early in the cycle. Erbium also contributes to the negative prompt temperature feedback coefficient.

SAR section 4.2.1.1 also states that each fuel element is clad with a 0.020 inches (0.0508 cm) thick stainless steel can, and all closures are made by heliarc (also known as Tungsten Inert Gas) welding. Two sections of graphite are inserted in the can, one above and one below the fuel, to serve as neutron reflectors for the core. Stainless steel end fixtures are attached to both ends of the can, making the fuel-moderator element approximately 29.0 inches (73.66 cm) long. Standard reactor fuel element physical dimension limits such as transverse bend and elongation are specified in proposed TS 3.1.4, "Fuel Parameters," (discussed in SER section 5.3.1.1, which was evaluated by the NRC staff and found acceptable).

In NUREG-1282 (Ref. 39), the NRC staff approved the generic use of TRIGA fuels with uranium loadings of up to 30 weight percent in licensed TRIGA reactors with the provision that case-by-case analyses discuss individual operating conditions in applications for authorization to use 20/20 and 30/20 fuels. The use of 8.5/20, 12/20, 20/20, 30/20, and 45/20 fuels was approved for the UCD/MNRC reactor with the issuance of the NUREG-1630 (Ref. 10). As stated above, UCD/MNRC has proposed to only use 20/20 and 30/20 fuels.

Based on the NRC staff's review of NUREG-1282 and NUREG-1630, the UCD/MNRC's record of safe operation with 20/20 and 30/20 fuels, and the NRC staff's review of the neutronic and T-H evaluations presented in the SER section 2.6, the NRC staff finds continued use of 20/20 and 30/20 fuels acceptable in the UCD/MNRC reactor. The NRC staff finds proposed TS 3.1.3, Specification 1, limiting the fuel type used in the UCD/MNRC reactor to 20/20 and 30/20, is also acceptable.

Instrumented Fuel-Moderator Element

SAR section 4.2.1.2, "Instrumented Fuel-Moderator Element," states that an instrumented fuel-moderator element has three thermocouples embedded in the fuel. Proposed TS 2.2, "Limiting Safety Systems Setting," requires that one instrumented element shall be placed in the peak power location in the core to monitor fuel temperature, which is the variable upon which the safety limit is placed (discussed in SER section 5.2.2, which was evaluated by the NRC staff and found acceptable). SAR section 4.2.1.2 states that the sensing tips of the fuel element thermocouples are located about 0.3 inches (0.762 cm) radially from the vertical centerline and SAR Figure 4.6, "Typical TRIGA® Instrumented Fuel Element," shows that the elements are located one inch above, below, and at the axial center of the fuel. Proposed TS 2.2 and proposed TS 3.2.3, "Reactor Scrams and Interlocks," table 3.2, "Minimum Number of Scrams," Item f. defines operability requirements of instrumented fuel element (discussed in SER section 5.3.2.3, which was evaluated by the NRC staff and found acceptable). SAR section 4.2.1.2 also states that the thermocouple readout wires pass through a seal in the upper end fixture. A lead-out tube provides a watertight conduit that carries the lead-out wires above the surface of the water in the reactor tank. In other respects, the instrumented fuel-moderator element is identical to the standard element.

As stated in NUREG-1630 (Ref. 10), the NRC staff found the instrumented fuel elements acceptable during the previous licensing period for the UCD/MNRC reactor. Given that the licensee has proposed no changes to the instrumented fuel element, the NRC staff continues to find the TRIGA instrumented fuel element acceptable.

Graphite Dummy Elements

SAR section 4.2.2, "Graphite Dummy Elements," states that graphite dummy elements may be used to fill grid positions not filled by the fuel-moderator elements or other core components. Filled entirely with graphite and clad with aluminum, these components are of the same general dimensions and construction as the fuel-moderator elements.

As stated in NUREG-1630 (Ref. 10), the NRC staff's previous evaluation found the graphite dummy elements acceptable for use during the previous licensing period for the UCD/MNRC reactor. As the licensee has proposed no changes to the graphite dummy elements, the NRC staff continues to find the graphite dummy elements acceptable.

2.2.2 Control Rods

SAR section 4.2.3, "Control Rods," states that the reactivity of the UCD/MNRC reactor is controlled by up to five standard control rods and a transient rod. The control and transient rod drives are mounted on a bridge at the top of the reactor tank. The drives are connected to the control and transient rods through a connecting rod assembly. Every core loading includes up to five fuel-followed control rods (FFCRs) (i.e., control rods that have a fuel section below the absorber section). The top section is a solid boron carbide neutron absorber. Immediately below the absorber is a fuel section. The bottom section of the rod has an air-filled void. The fuel and absorber sections are sealed in Type 304 stainless steel tubes that are approximately 43 inches (109.22 cm) long with a diameter of about 1.35 inches (3.429 cm). In its response to the request for additional information (RAI) 3, by letter dated December 17, 2021 (Ref. 22), the licensee states that only 20/20 FFCRs would be used in the UCD/MNRC core.

SAR section 4.2.3 states that the rods are attached to drive assemblies mounted on a raised bridge. The drive assembly consists of a stepping motor and reduction gear driving a rack and pinion. The control rods and rod extensions are connected to the rack through an electromagnet and armature. The transient rod was formerly used for operating in pulse and square wave mode, but the licensee indicated that pulse and square wave modes of operation would no longer not be permitted. In SAR section 4.1, the licensee indicated that the transient function (rapid rod movement by the use of compressed air) would be disabled, and the transient rod would be used as a non-FFRC. In its response to RAI 2, by letter dated December 17, 2021 (Ref. 22), the licensee indicated that the compressed gas cylinder needed to actuate the transient rod for pulse and square wave modes of operation would be physically disconnected and removed and disabled the pulse and square wave console buttons so that they cannot be depressed, ensuring that pulse and square wave operation is not possible. Further, the licensee removed the pulse and square wave modes of operation from the proposed TSs, which will also help ensure that the pulse and square wave modes of operation are not permitted. The control rods are designed and will be tested to ensure operability (proposed TS 3.2.1, "Control Rods," Specification 2.b, the maximum permissible drop time is one (1) second or less) (discussed in SER section 5.3.2.1, which was evaluated by the NRC staff and found acceptable).

As stated in NUREG-1630, the NRC staff found the FFCRs acceptable (during the previous licensing period for the UCD/MNRC reactor). In addition, the NRC staff evaluated the licensee's proposed elimination of the pulse and square wave operation of the transient rod, and finds that the licensee's planned use of the transient rod as a non-FFCR is also acceptable. Further, the NRC staff evaluated the licensee's planned physical changes to eliminate pulsing and square wave operation, including disconnecting the compressed gas cylinder from the transient rod and disabling the associated pulse and square wave buttons acceptable in preventing the actuation of the transient function (i.e., pulse and square wave operation), and finds these changes acceptable.

2.2.3 Neutron Moderator and Reflector

Reflector

SAR section 4.2.4, “Reflector Assembly,” states that the neutron reflector is a ring-shaped block of graphite that surrounds the core radially. The graphite has a radial thickness of 12.625 inches (32.0675 cm), with an inside diameter of 21.5 inches (54.61 cm) and a height of about 22.125 inches (56.1975 cm). The graphite is protected from water penetration by a leak-tight welded aluminum can. Vertical tubes attached to the outer diameter of the reflector assembly permit accurate and reproducible positioning of fission and ion chambers used to monitor reactor operation. The reflector currently accommodates four tangential neutron radiography beam tubes. Additionally, each fuel element contains graphite reflector sections above and below the fuel section, within the cladding. The coolant surrounding the fuel elements also acts as a reflector.

As stated in NUREG-1630, the NRC staff found the reflector acceptable during the previous licensing period for the UCD/MNRC reactor. As the licensee has proposed no changes to the reflector, the NRC staff continues to find the reflector acceptable.

Moderator

SAR section 4.5, “Primary Coolant,” states that the neutron moderation in the UCD/MNRC reactor is primarily achieved through interaction with hydrogen in the coolant and the uranium-zirconium-hydride fuel mixture. Moderation and reflection by the reactor coolant leads to the reactor’s negative void coefficient of reactivity. The combination of the fuel and the solid moderator in the fuel element contributes to the prompt reactivity feedback coefficient of the fuel.

As stated in NUREG-1630, the NRC staff found the moderator acceptable during the previous licensing period for the UCD/MNRC reactor. As the licensee has proposed no changes to the moderator, the NRC staff continues to find the moderator acceptable.

2.2.4 Neutron Startup Source

SAR section 4.2.5, “Neutron Source and Holder,” states that an americium-beryllium neutron source is used for reactor startup. The source material is triple encapsulated in welded stainless steel. The capsule has a diameter of approximately 1 inch (2.54 cm) and is approximately 3 inches (7.62 cm) long. The neutron source holder is an aluminum cylinder that can be installed at any fuel location in the top grid plate.

As stated in NUREG-1630, the NRC staff found the neutron startup source acceptable during the previous licensing period for the UCD/MNRC reactor. As the licensee has proposed no changes to the neutron startup source, the NRC staff continues to find the neutron startup source acceptable.

2.2.5 Core Support Structure

SAR section 4.2.6, “Grid Plates,” provides a description of the top, bottom and safety plates which constitute the UCD/MNRC reactor core support structure. The UCD/MNRC reactor fuel is fixed in place in a uniformly spaced hexagonal pattern (1.714 inches [4.354 cm] pitch) by two grid plates. The top grid plate is aluminum with a diameter of 21 inches (53.34 cm) and a thickness of 1.25 inches (3.175 cm) (0.75 inches [1.905 cm] thick in the central region). The plate provides accurate lateral positioning for the core components and is supported by six 0.5-inch (1.27-cm) stainless steel rods attached to the bottom grid plate. Both plates are anodized to resist wear and corrosion. The bottom grid plate is an aluminum plate with a thickness of 1.25 inches (3.175 cm) that supports the entire weight of the core and provides accurate spacing between the fuel-moderator elements. The bottom grid plate is bolted to an assembly that is welded to the core barrel. The core barrel supports the bottom grid plate and is supported by the reflector assembly base support. This arrangement is pictured in Figure 3.3, "TRIGA® Reactor," of the SAR (Ref. 4). A safety plate with a thickness of 1 inch (2.54 cm) is provided to preclude the possibility of control rods falling out of the core. The machined aluminum plate is suspended from the lower grid plate by stainless steel rods that are 18.25 inches (46.355 cm) long.

As stated in NUREG-1630, the NRC staff found the core support structure acceptable during the previous licensing period for the UCD/MNRC reactor. As the licensee has proposed no changes to the core support structure, the NRC staff continues to find the core support structure acceptable.

2.3 Reactor Tank or Pool

SAR sections 4.3, "Reactor Tank," and 5.1, "Reactor Tank," describe the UCD/MNRC reactor tank. The UCD/MNRC reactor core is located in a cylindrical aluminum tank surrounded by a reinforced concrete structure. The reactor tank is a welded aluminum vessel with 0.25 inches (0.635 cm) walls, a diameter of approximately 7 ft (2.218 m), and a depth of approximately 24.5 ft (7.448 m). The tank is welded for water tightness. The integrity of the weld joints has been verified by radiographic testing, dye penetrant checking, and leak testing. The outside wall of the tank is coated with a tar material for corrosion protection. Four beam tubes are attached to the reactor tank at 90 degrees intervals tangential to the reflector assembly and core. The tank wall section of the beam tubes consists of a pipe with a diameter of 12.5 inches (31.75 cm) welded to the tank wall with a flange at its end. Flanges are welded to the in-tank end to ensure water tightness inside the beam tubes without penetrating the tank wall. The beam tubes clamp onto the tank wall flanges and extend through the bulk shielding concrete that surrounds the reactor tank. The outside of the tank is coated with epoxy and a double layer of roofing felt to prevent corrosion. Any leakage or tank overflow would be collected in a drain around the base of the tank. This section is designed so it can be routinely monitored for evidence of leakage. Additionally, the entrance to pump suction lines are located less than 3 ft (0.9 m) below the normal tank water level, which limits loss of tank inventory in the event that a leak develops in other portions of the primary coolant system.

As stated in NUREG-1630, the NRC staff found the reactor tank acceptable during the previous licensing period for the UCD/MNRC reactor. As the licensee has proposed no changes to the reactor tank, the NRC staff continues to find the reactor tank acceptable.

2.4 Biological Shield

SAR section 4.4, “Biological Shield,” describes the UCD/MNRC reactor tank that is surrounded by a concrete shield of various thickness and sits on a concrete pad. This structure provides radiation shielding for personnel working in and around the reactor room, as well as protection for the reactor core against natural phenomena which could potentially damage the reactor core.

SAR section 11.1.6.1, “Shielding,” states that the UCD/MNRC reactor bulk shield is very similar, in material type and thickness, to other proven TRIGA shields. The one significant difference is the beam tube penetrations, where supplemental shielding has been added, which was designed to provide the same attenuation to both neutrons and gammas as the basic unpenetrated shield. The UCD/MNRC has eight areas with specially designed shielding: the reactor bulk shield, the four radiography bays, the demineralizer resin cubicle, the continuous air monitor (CAM) room, and the second-floor hand and foot monitor. The reactor bulk shield is similar to other above ground TRIGA reactors, which consists of approximately 20 ft (6.1 m) of water above the core to protect personnel in the reactor room. The shield is 11 ft (3.4 m) thick below ground level. Above ground level, it varies in thickness from 10 ft (3 m) to 3.25 ft (1 m), with the smaller dimension at the top of the reactor tank. Additional shielding is provided by the water in the reactor tank. The radial shielding, which protects personnel in the adjoining radiography bays, is provided by the graphite reflector and pool water to a radius of 3.5 ft (1.1 m) and by standard reinforced concrete extending to a radius of 10.5 ft (3.2 m) (7 ft (2.1 m) thick in Bay 1). This basic shield has been augmented in the areas of beam tube penetration with shadow shields of steel. Actual measured radiation levels at the surface of this shield at 1.0 MWt show 1 millirem per hour (mrem/hr). The reinforced concrete pad below the tank is approximately 10 ft (3 m) thick and prevents soil and ground water activation. The 20 ft (6.1 m) of water above the core provides the bulk shielding for personnel in the reactor room. Included in the radiography bays’ shielding are the shutter biological shields, the beam stops, and the walls and roof of the individual bays. Shielding has been designed so that radiation levels in areas occupied by personnel are as low as reasonably achievable (ALARA). The radiography bays shielding was designed to reduce the radiation levels at 1.0 MWt operation to less than 5 mrem/hr, which the licensee states have been confirmed by actual radiation measurement.

The licensee states that, in addition to the primary biological shielding for the reactor and radiography bays, certain auxiliary systems required shielding. An additional 1 ft (0.3 m) of concrete was installed around the demineralizer system in order to keep the radiation levels on the second floor of the reactor building ALARA, and to maintain an acceptable radiation background for health physics instrumentation in the general area. In order to achieve a background reduction sufficient to maintain adequate counting sensitivity, in addition to the demineralizer resins, the east wall of the CAM room (containing the reactor room and the stack CAMs) was shielded with 1 ft (0.3 m) of concrete. The lower level in Bay 4 has a large cavity cut up to the reactor tank wall for planned neutron cancer therapy research. The cavity is approximately 10 ft x 10 ft x 10 ft (0.3 m x 0.3 m x 0.3 m) and is currently filled with concrete blocks stacked in overlapping layers to prevent radiation streaming. The radiation levels at 1.0 MWt are less than 0.5 mR/hr [milli-Roentgen per hour] gamma and less than 0.1 mrem/hr neutron on the outside of the concrete blocks.

The NRC staff reviewed the description for the UCD/MNRC biological shield and finds that the description provided in the SAR is typical of other TRIGA reactors. The results of facility operations, as reported in the licensee's annual operating reports (ARs) (Ref. 24), or observed by the NRC staff during inspections, as documented in the inspection reports (IRs) (Ref. 25) indicate the biological shield acceptably limits radiation exposure from the reactor. As stated in NUREG-1630, the NRC staff found the biological shield acceptable during the previous licensing period for the UCD/MNRC reactor. As the licensee has proposed no changes to the biological shield, the NRC staff continues to find the biological shield acceptable.

2.5 Nuclear Design

The information discussed in this section establishes the design bases for the content of other chapters in this SER. The UCD/MNRC reactor will operate at a steady-state thermal power level of less than or equal to 1.0 MWt.

2.5.1 Normal Operating Conditions

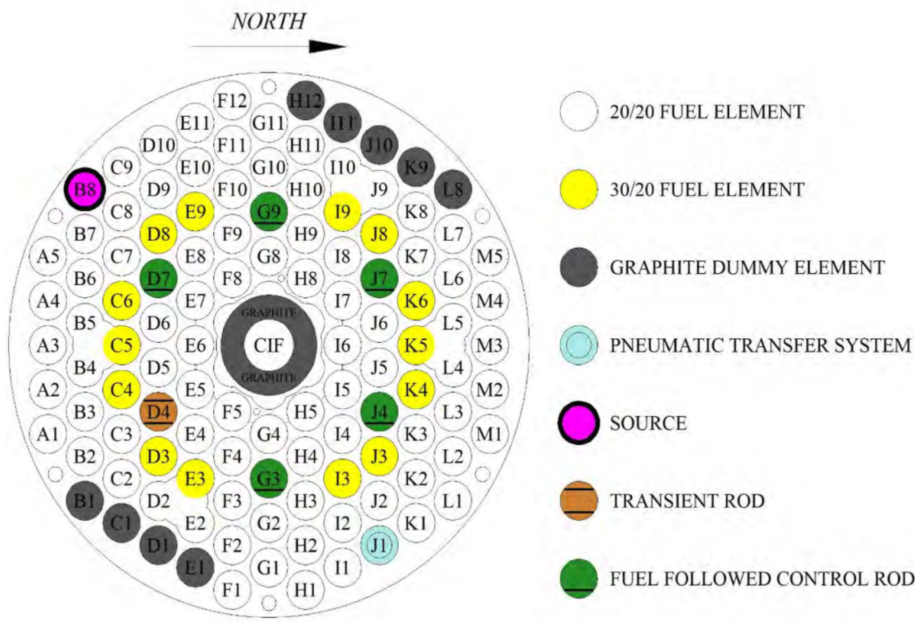
SAR section 4.6.4.3 defines an operating core configuration (OCC). The OCC is an as-built core that provides benchmarking information for reactor neutronic and T-H calculations. The results of the OCC analyses are compared to measurements which help to validate that the codes and methods used are accurate. Using the same codes and methods for the OCC to analyze the LCC helps to provide confidence in the predicted results of the LCC analysis.

Operating Core Configuration

SAR section 4.6.3, "Design Criteria - Operating Core Configuration (OCC), Limiting Core Configuration (LCC), Planned Future Operating Core Configuration, and End of Life Planned Future Operating Core Configuration," states that the three separate core loadings are defined for use in the safety calculations to arrive at the most limiting thermal hydraulic conditions to be analyzed. The OCC is the current UCD/MNRC reactor core configuration. For this analysis, the OCC and LCC use a core-power level of 1.1 MWt, which is higher than the proposed 1.0 MWt.

The NRC staff finds that the licensee's use of 1.1 MWt is this acceptable because the analytical results for the thermal limits of the fuel will be more conservative when compared to the licensed power level of 1.0 MWt. Use of the higher power level also provides additional margin of safety to the TS safety limits. The OCC and all subsequent cores are assembled with both the 20/20 and 30/20 type TRIGA fuel elements, together with five control rods having 20/20 type fuel as FFCRs, one transient rod, and in some cases dummy graphite elements. The transient rod was used for pulsing runs in the past, but now serves as one additional control rod.

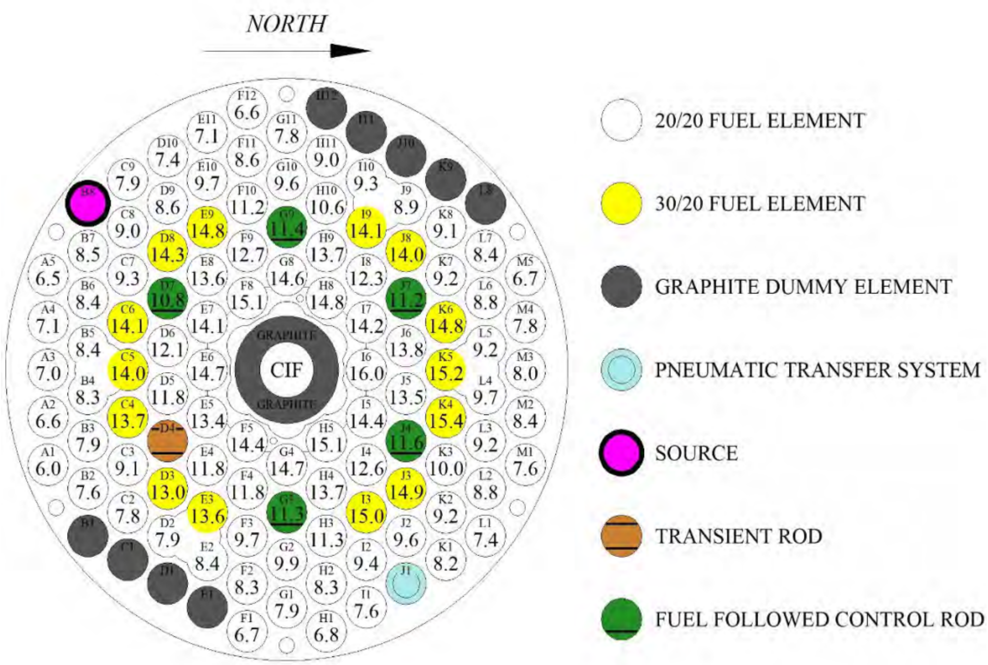
Figure 2-1 UCD MNRC Operating Core Configuration



UCD LRA SAR Figure 4.33. OCC's fuel map. "There are 121 positions in total in the fuel grid plate; 83 20/20 type fuel elements, 14 30/20 type fuel elements, 1 transient rod and 5 fuel-followed control rods, 9 graphite dummy rods, 1 neutron source, 1 pneumatic transfer system (PTS), and aluminum thimble and cylindrical graphite sleeve in the central irradiation facility, occupying 7 positions."

SER Figure 2-1, "UCD MNRC Operating Core Configuration," above was reproduced from the UCD LRA SAR Figure 4.33, "OCC's fuel map," to provide the licensee's OCC, illustrating the locations of the 20/20 and 30/20 fuel elements, graphite dummy element, transient rod (no longer used as a transient rod as result of the elimination of pulsing and square wave operation, but used as a shim rod), fuel follower control rods, the pneumatic transfer system, and the source.

Figure 2-2 UCD MNRC OCC Power Distribution at 1.1 MWt



UCD LRA SAR Figure 4.35. OCC's power distribution at 1.1 MWt. For all 20/20 type fuel elements, the maximum power per fuel element is 16.0 kW, which is at I6 location in C-hex. For all 30/20 type fuel elements, the maximum power per fuel element is 15.4 kW, which is at K4 location in E-hex. Currently, two instrumented fuel elements (IFEs) are used; one 20/20 type IFE are located at I6 position in C-hex and one 30/20 type IFE are located at E9 position in E-hex. Both IFEs are located in the opposite positions of the reactor core with close to the highest values of power per fuel element. Their normal readings are about 320 to 330 degrees C [608 to 626 degrees F] at 1.0 MWt steady state operation.

SER Figure 2-2, "UCD MNRC OCC Power Distribution at 1.1 MWt," above was reproduced from the UCD LRA SAR Figure 4.35, "OCC's power distribution at 1.1 MWt," to provide the power distribution for the OCC at 1.1 MWt operation.

<u>Calculational Methodology</u>

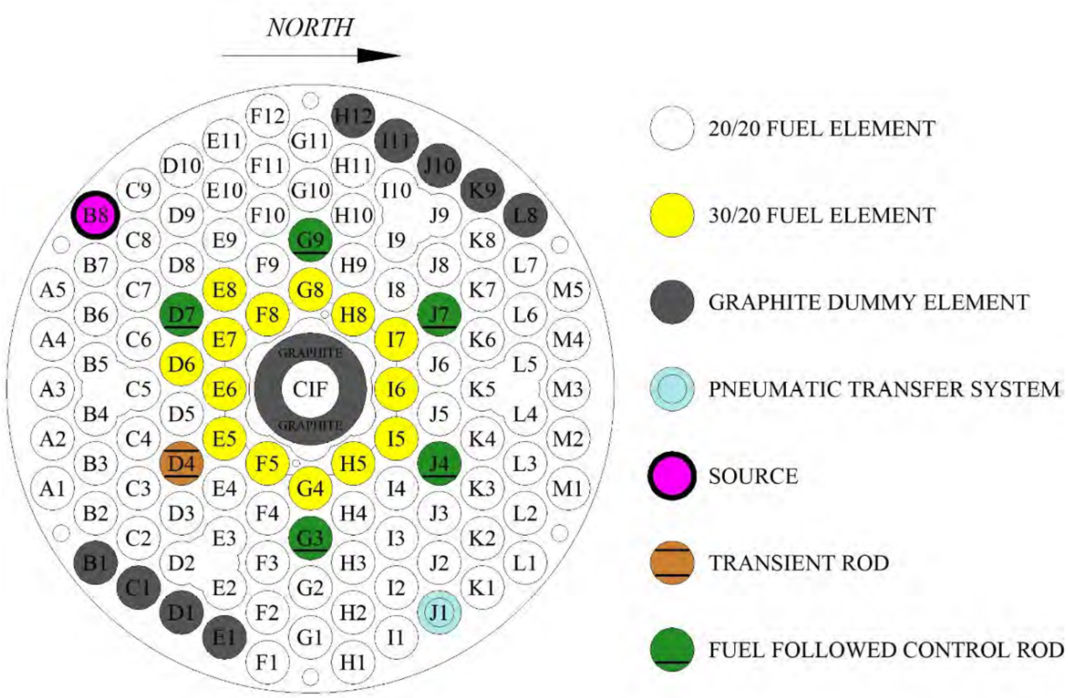
SAR section 4.6.4.2.1, "Validation of MNRC MCNP Core Model," indicates that the licensee performed analyses of core configurations using the Monte Carlo N-Particle Transport code (MCNP), version 5 (Ref. 64), which solves the Boltzmann transport equation through stochastic simulation of individual particle histories. SAR section 4.6.4.3 states that the configuration of fuel that was loaded in the UCD/MNRC reactor at the time of submission of the 2020 SAR was used as the OCC. The licensee used measured critical control rod positions and the reactivity worth of control rods from the OCC to validate the UCD/MNRC core model, as documented in SAR section 4.6.4.2.1. The UCD/MNRC model predicted a K-effective (K_{eff}) multiplication factor of 1.00087 ± 0.00011 when control rods were modeled at the measured critical positions, meaning that core reactivity was predicted within 0.1 percent, or \$0.13. The reactivity worth of each control rod was predicted within \$0.16.

The NRC staff finds, that based on the consistency of analysis methods with those used at similar facilities and the favorable comparisons between these model predictions and experimental data, the analysis methods used for nuclear design of the UCD/MNRC reactor have been justified and validated. The NRC staff finds that this code has been used extensively for nuclear reactor applications and has been used in the licensing of other TRIGA reactors, such as TRIGA reactors at Texas A&M University (Ref. 42) and U.S. Geological Survey (Ref. 43), and is acceptable for this application. The NRC staff reviewed the configurations of the OCC and finds that the licensee followed the guidance in NUREG-1537, Part 1, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors” (Ref. 11) section 4.5.1, Normal Operating Conditions,” which states that the licensee should propose an initial core configuration, analyze all reactivity conditions and possible configurations during the lifetime of the reactor, and using assumptions and methods that are justified and validated.

<u>Limiting Core Configuration</u>

SAR section 4.6.4.4, “Future Cores and the Limiting Core Configuration (LCC),” defines the LCC. The licensee designed the LCC by moving the most reactive fuel to the regions with the highest neutron flux (i.e., the C and D rings of the core). Additionally, the licensee determined that acquisition of additional fuel will be required to support operation of the core over the license renewal period. The potential limiting configurations based on the planned fuel stock have also been considered, although the loading pattern developed using the current fuel stock is limiting.

Figure 2-3 UCD MNRC LCC Power Distribution at 1.1 MWt



UCD LRA SAR Figure 4.36. LCC’s fuel map: (14) 30/20 type fuel elements are relocated to 12 positions in C-hex[agonal ring] and 2 positions in D-hex[agonal ring].

SER Figure 2-3, “UCD MNRC LCC Power Distribution at 1.1 MWt,” above was reproduced from the UCD LRA SAR Figure 4.36, “CC’s fuel map: (14) 30/20 type fuel elements are relocated to 12 positions in C-hex[agonal ring] and 2 positions in D-hex[agonal ring],” to show the LCC at 1.1 MWt operation.

The LCC is defined in NUREG-1537 as the core configuration that would yield the highest power density using the fuel authorized for use by the TSs. The LCC is used in the T-H analyses to establish the limiting operating conditions for the reactor. The NRC staff reviewed the configurations of the LCC and finds that the licensee followed the guidance in NUREG-1537, section 4.5.1 to describe a LCC with the highest power density for use in the T-H calculations (discussed in SER section 2.6, which was evaluated by the NRC staff and found acceptable).

The NRC staff finds that the licensee modeled the LCC core with isotopic compositions reflecting the initial loading (Beginning-of-life [BOL] LCC) and compositions that will occur later in life (End-of-life [EOL] LCC) due to the effects of burnup. The NRC staff also finds that the requirements in proposed TS 5.3.1 (discussed in SER Chapter 5, which was evaluated by the NRC staff and found acceptable) placed on design of UCD/MNRC cores help ensure that any future core configurations are bounded by the LCC.

The NRC staff finds that the licensee has provided reasonable assurance that the LCC peak fuel element power, as limited by proposed TS 4.1.4, “Fuel Parameter,” Specification 3, to 17.69 kilowatt (kW), (evaluated by the NRC staff in SER section 5.4.1.4 and found acceptable) will bound all planned reactor operating conditions. This finding is based on justification and validation of the licensee’s analysis methods, the licensee’s method of developing the LCC core discussed above, including consideration of fuel burnup and planned fuel acquisitions, and the restrictions put in place by proposed TS 5.3.1. For the NRC staff’s evaluation of the peak fuel element power of 17.69 kW, see SER section 2.6, which was evaluated by the NRC staff and found acceptable.

2.5.2 Core Physics Parameters

The core physics parameters of temperature coefficient of reactivity, effective delayed neutron fraction, and void coefficient, are discussed below.

<u>Temperature Coefficient of Reactivity</u>

UCD supplemental information, dated September 22, 2021, “Maximum Reactivity Insertion,” figure 13.1, “Prompt Negative Temperature Coefficient for TRIGA® Fuels,” (Ref. 44), provides the prompt negative temperature coefficient of reactivity, calculated over a range of fuel temperatures and burnups for each licensed fuel type (20/20 and 30/20). The licensee states that the fuel temperature coefficient of reactivity will vary with burnup since depletion of the burnable neutron poison erbium diminishes the strength of the associated reactivity feedback mechanism. Further, the licensee provides the variation of the UCD/MNRC fuel temperature coefficient of reactivity (Δk/k/°C) with burnup in UCD supplemental information in figure 13.1.

The NRC staff’s review finds that the licensee’s accounting for the burnup effect is necessary to ensure that analysis using this coefficient will remain bounding throughout the life of the UCD/MNRC reactor. Further, the NRC staff reviewed the values of the reactivity coefficient reported by Texas A&M University (Ref. 42), the U.S. Geological Survey (Ref. 43), and the Armed Forces Radiobiology Research Institute (Ref. 45) and finds the UCD values are consistent with other TRIGA reactors. The NRC staff finds that the fuel temperature coefficients of reactivity are also comparable to those used in the safety analysis for the previous licensing period for the UCD/MNRC reactor. SER table 2.1, “Fuel Temperature Coefficients of Reactivity Reported by Other TRIGA Reactors,” provides the fuel temperature coefficients of reactivity reported by other TRIGA reactors and for the previous licensing period of the UCD/MNRC reactor. Based on the information above, the NRC staff finds the UCD/MNRC fuel temperature coefficients of reactivity acceptable.

Table 2.1 Fuel Temperature Coefficients of Reactivity Reported by Other TRIGA Reactors

Reactor	Fuel Type	Fuel Temperature Coefficient of Reactivity [Δk/k/°C]	Fuel Temperature [°C]
Texas A&M University (TRIGA) (Ref. 42)	30/20	-5.3×10^{-5}	23
		-13.1×10^{-5}	700
U.S. Geological Survey (Ref. 43)*	8.5/20, 12/20	-8.6×10^{-5} to -9.5×10^{-5}	73.65
		-12×10^{-5} to -13×10^{-5}	226.85
		-10×10^{-5} to -11×10^{-5}	726.85
Armed Forces Radiobiology Research Institute (Ref. 45)	8.5/20, 12/20	-11×10^{-5}	700
UCD/MNRC, SER 1998 (Ref. 10)	20/20	-4.7×10^{-5}	20
		-7.0×10^{-5}	200
UCD/MNRC, current (Ref. 44)**	20/20, 30/20	-5.2×10^{-5} to -5.4×10^{-5}	20
		-5.6×10^{-5} to -8.4×10^{-5}	200

*Values were read from figures and are approximate. Range of FTC values reflects FTCs calculated for different core configurations and FTC error bars
**Range of FTC values reflects variation in fuel element burnup

Effective Delayed Neutron Fraction

In SAR section 4.6.4.2.1 the licensee provides the calculated value of 0.0075 for the effective delayed neutron fraction using the MCNP model of the UCD/MNRC core. The licensee uses this value of effective delayed neutron fraction in modeling reactivity insertion accidents, as provided in supplemental information, “Maximum Reactivity Insertion” (Ref. 44) and “Uncontrolled Withdrawal of a Control Rod Nonlinear Worth” (Ref. 46), dated September 22, 2021.

The NRC staff finds that this value of effective delayed neutron fraction is within the range generally recommended by GA (i.e., 0.0071 to 0.0075), and is comparable to values used in other 1.0 MWt TRIGA research reactors at Texas A&M University (Ref. 42), U.S. Geological Survey (Ref. 43), and the Armed Forces Radiobiology Research Institute (Ref. 45). The NRC staff provides the effective delayed neutron fractions reported by other TRIGA reactors in SER table 2.2, “Effective Delayed Neutron Fraction Reported by Other TRIGA Reactors.” The

NRC staff finds the effective delayed neutron fraction to be consistent with the range of values generally expected for a TRIGA core, and therefore, acceptable.

Table 2.2 Effective Delayed Neutron Fraction Reported by Other TRIGA Reactors

Reactor	Effective Delayed Neutron Fraction
Texas A&M University (TRIGA) (Ref. 42)	0.0070
U.S. Geological Survey (Ref. 43)	0.00728
Armed Forces Radiobiology Research Institute (Ref. 45)	0.0068-0.0075
UCD/MNRC (Refs. 47 and 49)	0.0075

Void Coefficient of Reactivity

UCD supplemental information, dated September 22, 2021, "Calculation of Negative Void Coefficient of MNRC Core" (Ref. 52), provides the licensee's calculated void coefficient of -$0.25/% void, which corresponds to a void coefficient of reactivity coefficient of -0.1875 %Δk/k/% void, when using the licensee's value for effective delayed neutron fraction.

The NRC staff finds that the UCD/MNRC value of the void coefficient reactivity is comparable to values predicted for other TRIGA reactors (Refs. 45, 46, and 48). The void coefficients of reactivity reported by other TRIGA reactors are shown in SER table 2.3, "Void Coefficient of Reactivity Reported by Other TRIGA Reactors."

Table 2.3 Void Coefficient of Reactivity Reported by Other TRIGA Reactors

Reactor	Void Coefficient of Reactivity [%Δk/k/% void]
Texas A&M University (TRIGA) (Ref. 42)	-0.130
Armed Forces Radiobiology Research Institute (Ref. 45)	-0.080
UCD/MNRC, current (Ref. 52)	-0.1875

The NRC staff reviewed the "Calculation of Negative Void Coefficient of MNRC Core," and finds that this value of void coefficient of reactivity adequately demonstrates that the reactor will become substantially subcritical in the event of a loss-of-coolant accident, even without the insertion of control rods. Further, the guidance in NUREG-1537 indicates that the void coefficient of reactivity should be negative of the significant operating ranges of the reactor. Based on the value of the UCD/MNRC void coefficient of reactivity of -0.1875 %Δk/k/% void, the NRC staff finds that UCD/MNRC void coefficient of reactivity is consistent with guidance in NUREG-1537. Based on the information above, the NRC staff finds that the UCD/MNRC void coefficient of reactivity is acceptable.

2.5.3 Operating Limits

Excess Reactivity

In SAR section 4.6.4.3, the licensee states that the MCNP code was used to evaluate the clean reactor core excess reactivity compared to the reactor operational data, and to benchmark the reactivity of all six control rods, including one transient rod and five FFCRs, when completely withdrawn out of the reactor core (i.e., in the full up position). The licensee states that the clean excess reactivity was calculated to be $5.16, which is consistent with the measured value of $5.13 in the startup of Monday morning on October 1, 2018. The licensee proposes TS 3.1.1, “Excess Reactivity,” which will limit the total excess reactivity that the UCD/MNRC is authorized to have loaded into its reactor during operation to not to exceed $7.50.

The NRC staff reviewed the licensee’s excess reactivity and finds that the proposed limit as stated in proposed TS 3.1.1 provides sufficient reactivity to compensate for various negative reactivity effects associated with operation and use of the reactor, as well as allowing some operational flexibility. The NRC staff finds that the essential parameter needed to ensure the capability to shut down the reactor is the minimum shutdown margin of $0.50, as required by proposed TS 3.1.2, “Shutdown Margin.”

The NRC staff performed calculations using the licensee’s excess reactivity of $7.50 to demonstrate that the licensee could ensure that the shutdown margin of $0.50 was maintained. The result of the NRC staff calculations is provided in SER table 2.5, “UCD/MNRC Shutdown Margin Calculations.” Based on the information described above, the NRC staff finds the proposed TS 3.1.1, excess reactivity of $7.50, is acceptable.

Shutdown Margin

SAR section 4.6.4.2.1 describes the licensee’s analytical methods and measurements used to demonstrate the ability to maintain the minimum shutdown margin to ensure that the reactor can be safely shutdown from any operational configuration. The licensee has a minimum shutdown margin of $0.50, as limited by proposed TS 3.1.2, which helps to ensure that the reactor can be shut down and remain shut down, with the reactor in any core condition, with the most reactive control rod assumed to be fully withdrawn, and with the absolute value of all movable experiments in their most reactive condition or $1.00, whichever is greater.

The NRC staff reviewed the licensee’s shutdown margin of $0.50 and finds that it is consistent with the guidance in NUREG-1537, Part 1, chapter 14, appendix 14.1, “Technical Specifications,” section 3.1, “Reactor Core Parameters,” item (2) “Shutdown Margin,” which states that the value of the shutdown margin should be readily determined, e.g., ≥$0.50. The NRC staff finds the proposed TS 3.1.2 value of $0.50 for the shutdown margin, is acceptable.

Validation of the UCD/MNRC MCNP Core Model

SAR section 4.6.4.2.1 describes the benchmarking performed to validate the MCNP model. The licensee used data from reactor operation in October 2018 and determined that the estimated critical position was accomplished when the transient rod at the D4 location and 4 other control rods at D7, G3, G9, and J4 locations were banked at 60 percent withdrawal, and regulating rod at J7 location was withdrawn at 48 percent. The central irradiation facility is occupied by the aluminum thimble and cylindrical graphite sleeve. The MCNP simulated core configuration represented the same OCC with the same control rods withdrawn to the same heights. The K_{eff} estimated critical position was calculated to be 1.00087 +/- 0.00011. The difference is within 0.001.

The licensee also states that the MCNP simulation of the OCC, using the 2018 data, was benchmarked by evaluating each individual control rod worth and compare to the measured values during the annual shutdown for reactor maintenance in August 2018. The control rod worth measurements were accomplished by banking 5 control rods at 60 percent withdrawal and slowly raising the “evaluated” control rod from 100 percent insertion to 100 percent withdrawal. Sequentially, the reactor core begins in subcritical condition with 5 control rods at 60 percent withdrawal, becomes critical at low power, and continues its power increase up to about 900 watts, but less than 1 kW without adding detectable heat to the reactor core, when the “evaluated” control rod is 100 percent withdrawal. The effective delayed neutron fraction is 0.0075, chosen from the range of 0.0071 to 0.0075, originally recommended by GA. As indicated in SER table 2.4, “UCD/MNRC Control Rod Worths – Measured vs. Calculated,” below, the values for each calculated control rod worth compared to the measured value, is within 10 percent.

Table 2.4 UCD/MNRC Control Rod Worths – Measured vs. Calculated

Control Rod Worth	D4	D7	G3	G9	J4	J7
Measured ($)	1.83	2.49	2.61	2.56	2.91	2.78
Calculated ($)	1.78±0.03	2.65±0.03	2.49±0.03	2.54±0.03	2.78±0.03	2.62±0.03
Difference (percent)	2.7	6.4	4.6	0.8	4.5	5.7

The NRC staff finds that SAR section 4.6.4.2.1 provides the results of the licensee’s calculations to determine the control rod worth, excess reactivity, and shutdown margin of the OCC. Further, the NRC staff finds that these calculations generally conservatively overestimate excess reactivity and conservatively underestimate the shutdown margin, although the accuracy is consistent with other neutronics models of TRIGA reactors. The calculated excess reactivity and shutdown margin are within the proposed TS limits. The excess reactivity of the LCC is also within the proposed TS limit.

The NRC staff performed confirmatory calculations using licensee supplied reactivity values, reproduced in SER table 2.5, “UCD/MNRC Shutdown Margin Calculations,” that demonstrated that the licensee’s reactor can be operated with wide margins to the excess reactivity and shutdown margin limits.

Table 2.5 UCD/MNRC Shutdown Margin Calculations

	OCC	Proposed TS 3.1.2 Limit $7.50
Total Control Rod Worth	$15.18	$15.18
Core Excess Reactivity calculated (subtract)	- $5.16	-$7.50
Non-Secured Experiment (subtract)	- $1.00	-$1.00
Most Reactive Control Rod Worth (subtract)	- $2.91	-$2.91
Shutdown Margin (October 2018)	= $6.11	= $3.77
Shutdown Margin Limit TS 3.1.2	$0.50	$0.50

Based on the information provided above, the NRC staff finds that the licensee's MCNP model provides results that are consistent with the measured reactor core parameters for control rod worth. The NRC staff finds that the excess reactivity and shutdown margin were evaluated using methods consistent with industry standards for research reactors (e.g., MCNP). The NRC staff finds the proposed values of excess reactivity and shutdown margin are consistent with other TRIGA research reactors and the guidance in NUREG-1537. The NRC staff finds that the validation data presented for the UCD/MNRC neutronics model provides reasonable assurance that core reactivity and control rod worth can be assessed with acceptable accuracy such that a shutdown margin of $0.50 is sufficient to ensure that the reactor can be shut down under all credible conditions. Based on the information described above, the NRC staff concludes that the core excess reactivity and shutdown margin are acceptable.

Reactivity Limits on Experiments

The licensee proposed TS 3.8.1, "Reactivity Limits," Specifications 1 and 2, limit the absolute reactivity worth of movable experiments to less than $1.00 per experiment and the total absolute reactivity worth of all experiments to less than $1.75, respectively. The proposed TS definition, "Movable Experiment," defines a movable experiment as one that is not secure and intended to be moved in or near the reactor core or into and out of the reactor experiment facilities while the reactor is operating. The proposed TS definition, "Secured Experiment," defines secured experiments as those mechanically held in a stationary position relative to the reactor, where the restraining forces must be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces that are normal to the operating environment of the experiment or forces which can arise as a result of credible conditions. Individual secured experiments are limited by the proposed TS 3.8.1, to a worth of $1.75. This worth is less than the positive reactivity insertion limit of the pulse analyzed in the SAR chapter 13.2.2, "Insertion of Excess Reactivity," (i.e., pulse that would be needed to reach the fuel temperature safety limit).

The NRC staff reviewed proposed limitations on the worth of experiments. On the basis of its review, the NRC staff finds these limitations are conservative and provide reasonable assurance that failure of a single experiment resulting in a positive reactivity insertion would not result in damage to the fuel or reactor components. Also, in the extremely unlikely event of simultaneous multiple failures of all in-tank experiments, the positive reactivity insertion would not result in a reactivity addition of more than $1.90, the pulse analyzed in SAR chapter 13, as supplemented

(Ref. 46). Based on the information discussed above, the NRC staff also finds that proposed TS 3.8.1, Specifications 1 and 2, are acceptable. Further, the NRC staff finds that reasonable assurance exists that these experiments will not lead to a reactivity insertion that will pose a threat to the health and safety of the public.

2.6 Thermal-Hydraulic Design

The license provides in UCD supplemental information, dated September 22, 2021, “TH RAI,” the T-H analysis for operation of the UCD/MNRC reactor at a 1.0 MWt using RELAP5-3D computer program (Ref. 47). The licensee indicates that this power level reflects the nominal operating condition for the UCD/MNRC reactor. RELAP5-3D is developed from the RELAP5/MOD3 code (which was used during the previous licensing period for the UCD/MNRC reactor), and developers have made effort to avoid compromising the validation undertaken in development of RELAP5/MOD3. RELAP5-3D has been used in the licensing of other TRIGA reactors (Ref. 45).

A RELAP5-3D model consists of a system of control volumes connected by flow junctions. The fluid mass, momentum, and energy equations and the appropriate equation of state are solved for the user-defined geometry. The RELAP5-3D code uses a full nonhomogeneous, non-equilibrium, six-equation, two-fluid model for simulation of two-phase system transient behavior. User-defined heat structures are used to simulate the reactor fuel rods. Heat transfer coefficients are computed, as appropriate, for the channel flow and fluid state.

Some of the RELAP5-3D features important for simulating a natural circulation reactor such as the TRIGA at the UCD/MNRC are as follows:

- an ability to compute the system density distribution and the gravity force terms in the coolant momentum equation
- an ability to compute implicitly the local pool or convective subcooled boiling, which might occur in TRIGA reactors
- temperature-dependent material properties

When power in the UCD/MNRC reactor core is increased, nucleation will occur on the fuel rod surfaces and fully developed nucleate boiling may eventually occur. As long as the surface heat flux remains below the critical heat flux (CHF), it is possible to increase the heat flux without an appreciable increase in fuel rod (cladding) surface temperature. If the CHF is exceeded, film boiling occurs, the surface temperature almost immediately increases to a much higher value, and fuel rod damage may occur. The transition from nucleate boiling to film boiling is referred to as departure from nucleate boiling (DNB). The safe operation of the reactor depends on maintaining the operating heat flux safely below the CHF. The ratio of the CHF to the peak core heat flux is thus a measure of the safety margin. This quantity is referred to as the DNB ratio (DNBR).

The RELAP5-3D code directly computes DNBR according to the Groeneveld 1986 CHF correlation. The licensee also used RELAP5-3D output to compute DNBR according to the Groeneveld 1986, Groeneveld 1995, Groeneveld 2006, and Bernath correlations. The NRC staff understands that the use of multiple correlations helps to account for the fact that CHF is a complex phenomenon that is dependent on many variables and is generally predicted using models developed from empirical data. Of these correlations, Bernath is the oldest and generally the most conservative. The licensee confirmed that Groeneveld 1986 DNBR computed by RELAP5-3D match those manually computed using RELAP5-3D output.

The net driving force for flow within the tank of the UCD/MNRC reactor is the difference between the net buoyancy of the water heated in the core and the friction within the flow paths. Both are implicitly computed by the RELAP5-3D code (Ref. 10). Friction losses consist mainly of the wall friction within the fuel pin flow channels and form losses in the upper and lower grid plate regions. Friction losses in other flow paths are computed but are small because of the low coolant velocities. The wall friction is computed directly by RELAP5-3D.

The form loss coefficients for the upper and lower grid regions are supplied as inputs to the code. Form loss coefficients were recalculated in response to the NRC staff audit question 4-42 (Ref. 14), and provided by licensee in its supplemental information (Ref. 47). The inlet form loss coefficients are appreciably lower than that included in SAR table 4.13, “Single channel model geometric thermal hydraulic properties summary,” (0.58 vs. 2.26), but are comparable to that calculated in Table B-1, “Summary of CHF Results and Hydraulic Parameter,” of ANL/RERTR/TM-07-01, “Fundamental Approach to TRIGA Steady-State Thermal-Hydraulic CHF Analysis,” dated December 2007, done by GA for the UCD/MNRC reactor in coordination with Argonne National Laboratory (Ref. 51). This report compares CHF predictions between four TRIGA reactors, noting specifically that UCD/MNRC loss coefficients are substantially smaller than the other three TRIGA reactors considered. The report also studied the sensitivity of the CHF to these loss coefficients using a RELAP5/MOD3.2 model, noting that an increase in form loss coefficients from 0.58 or 0.59 to 1.50 results in 3.1 percent and 2.1 percent reductions in CHF predicted with Groeneveld 2006 and Bernath correlations, respectively. Because the value is comparable to an independently calculated form loss coefficients and because of the appreciable margin to CHF discussed below, the NRC staff finds this acceptable.

The NRC staff finds that the licensee’s model simulates flow in a single hot channel of the reactor (that is, the channel with the most conservative boundary conditions), as opposed to the entire reactor core. The NRC staff notes that this approach is conservative as it neglects cross-flow from adjacent cooler channels that will act to lower coolant temperature, and this approach has been used in licensing of similarly designed research reactors (see section 2.6, “Thermal-Hydraulic Design,” of the SER for Renewal of the Facility Operating License for the United States Geological Survey TRIGA Research Reactor, October 2016, (Ref. 43)). The NRC staff review also finds that the RELAP5-3D code was properly used for the analysis of the T-H performance of the UCD/MNRC TRIGA, with natural convection cooling.

The licensee also performed RELAP5-3D calculations for the OCC, LCC at BOL and LCC at EOL. For each operating condition, a hot channel power was used that corresponds to a core operating at a total power level of 1.0 MWt. The results from these analyses are summarized and provided by licensee in its supplemental information (Ref. 47). As summarized in SER table 2.6, “UCD/MNRC RELAP5-3D Result Summary,” below, the maximum fuel centerline temperature is 410.0 degrees Celsius (°C) (770 degrees Fahrenheit (°F)) from the LCC at BOL.

The NRC staff finds that these results demonstrate significant margin to the 750 °C (1,382 °F) fuel temperature limiting safety system setting (LSSS) from proposed TS 2.2. The NRC staff used the UCD/MNRC input model with an NRC version of RELAP5 (Mod 3.3) and obtained results consistent with the licensee's calculated results.

Table 2.6 UCD/MNRC RELAP5-3D Result Summary

Parameter	OCC	LCC at BOL	LCC at EOL
Hot channel power (kilowatts thermal [kWt])	14.81	17.69[(1)]	17.59[(2)]
Maximum fuel centerline temperature (°C)	387.6	410.0	408.4
Maximum clad temperature (°C)	129.9	131.5	131.4
Minimum DNBR – Bernath correlation	3.16	2.26	2.28

(1) Maximum allowed by TS 5.3.1.
(2) Slightly lower than power at BOL due to the effects of burnup.

The NRC staff notes that the guidance in NUREG-1537, Part 2, Section 4.6, "Thermal-Hydraulic Design," states that a DNBR above 2.0 is conservative and acceptable to the NRC staff for non-power reactors. As presented in SER table 2-6 above, the minimum calculated DNBR is 2.26, given by the Bernath correlation, at 1.0 MWt nominal power and the limiting inlet temperature for the LCC at BOL core. For the OCC, a minimum DNBR (MDNBR) of 3.16 is calculated, also by the Bernath correlation. The NRC staff also notes that the DNBR given by the Groeneveld models are substantially higher. All DNBR given by the Groeneveld correlations for all conditions considered (OCC BOL, LCC BOL, and LCC EOL) are greater than 5.0 (Ref. 47). The NRC staff notes that appreciable uncertainty exists in the correlations used to calculate CHF for a TRIGA reactor, as evidenced by the wide variation in DNBR values.

For each of the three operating conditions, the licensee also analyzed hot channel powers ranging from 15 kWt to 30 kWt in order to determine the power level where the MDNBR equals 2.0. As presented in SER table 2.7, "UCD/MNRC Results of Hot Chanel Power for MDNBR Equals 2.0," the hot channel power can be increased to 108 percent of the nominal power at the point where the MDNBR equals 2.0.

Table 2.7 UCD/MNRC Results of Hot Chanel Power for MDNBR Equals 2.0

Parameter	OCC	LCC at BOL	LCC at EOL
Hot channel power resulting in an MDNBR of 2.0 (kWt)	21.0 (141.8 percent of nominal)	19.1 (108.0 percent of nominal)	19.1 (108.6 percent of nominal)

Further, the renewed license application requested operation of the UCD/MNRC at 1.0 MWt; however, the power level scram in proposed TS 3.2.3, "Reactor Scrams and Interlocks," table 3.2, "Minimum Number of Scrams," provides a reactor power level safety scram set point of 1.02 MWt. Although this power level is slightly higher than the base cases simulated, the NRC staff finds that the DNBR value calculated at 1.0 MWt, as well as additional T-H cases run at higher power levels indicate that significant margin exists between the power level SCRAM set point of 1.02 percent and the power level that could result in exceeding the CHF. Based on the sensitivity calculations performed by the licensee, for the hot assembly at 30 kWt, the peak fuel temperature would increase approximately 8 °C (14.4 °F) for a power increase to 1.03 MWt. This would result in a peak fuel temperature of 418 °C (784 °F), for the LCC at BOL, which remains significantly less than the 750 °C (1,382 °F) limiting safety limit setting operating limit. The NRC staff notes that the magnitude of the CHF depends upon local fluid conditions, as well as on channel inlet conditions and local fission power density. Limits placed on the pool water temperature and level (proposed TS 3.3, "Reactor Coolant Systems,") operating power (proposed TS 3.1.1), and hot element power (proposed TS 5.3.1) help ensure that actual operating conditions are equally or more favorable to DNBR than those assumed in the T-H analysis.

The NRC staff finds that, for steady-state operation at 1.0 MWt, the maximum predicted fuel temperature is 410 °C (770 °F). The NRC staff also finds that the calculated coolant temperature distribution in the hot channel for both the nominal and limiting cases are acceptable. As specified in proposed TS 2.2, the fuel temperature LSSS is 750 °C (1,382 °F). Given the significant margin in the T-H design calculations at the operating limit, the NRC staff finds that operation of the UCD/MNRC reactor at a power level of 1.0 MWt will maintain fuel temperatures below the operating limit. Operating at power levels up to 1.03 MWt would see a steady-state peak fuel temperature of 418 °C (784 °F), for the LCC at BOL, which is significantly below the LSSS of 750 °C (1,382 °F). The NRC staff also finds that the licensee's conservative T-H analysis provides reasonable assurance that operating up to 1.03 MWt, will not result in the CHF being exceeded.

2.7 Safety Limit

NUREG-1537, section 4.2.1, "Reactor Fuel," states that limits on operating conditions should ensure that the fuel cladding will remain intact and not allow the escape of any fission products. The proposed safety limit for the temperature of the fuel should be chosen such that a failure of the fuel cladding will not occur as a result of internal pressure or clad melting. As the temperature of a U-$ZrH_{1.7}$ fuel element increases, the internal pressure inside the fuel cladding also increases because of the presence of air, fission product gases, and hydrogen from the disassociation of hydrogen and zirconium in the fuel-moderator with hydrogen being the most important contributor to the internal pressure. If the temperature becomes high enough, the stress on the cladding as a result of the internal pressure can exceed the ultimate strength of the stainless-steel cladding, and the cladding will fail, releasing fission products from the fuel. The ultimate strength of the cladding material is also temperature-dependent and decreases with increasing temperature. The licensee has proposed a safety limit of 930 °C (1,706 °F) in proposed TS 2.1, "Safety Limits," on fuel temperature for steady-state operation. The NRC staff finds the licensee's proposed safety limit is less than the limit provided in NUREG-1537, appendix 14.1, section 2.1 (Ref. 11), of 950 °C (1,742 °F) for TRIGA fuel with a cladding temperature of less than 500 °C (932 °F).

The NRC staff notes that, although it is not a safety limit in proposed TS, the licensee uses a fuel temperature limit of 1,100 °C (2,012 °F) when evaluating reactivity insertion accidents, as discussed in SAR section 4.6.4.1.3, "ZrH Fuel Temperature Limits." This temperature limit is relevant during pulse-like transients initiated by a rapid accidental insertion of more than one dollar in reactivity. In such a transient, the fuel temperature increases abruptly during the power pulse, and the cladding temperature peaks later as heat is transferred out of the fuel pin. The cooler clad temperature results in a higher ultimate stress for the stainless-steel cladding, which allows a higher internal pressure to be present to the fuel cladding, and allows the fuel to reach a higher temperature without challenging the integrity of the cladding.

The temperature limits proposed by the licensee are more limiting than those provided in the guidance in NUREG-1537, appendix 14.1, section 2.1 (Ref. 11). On the basis of theoretical and experimental evidence (Refs. 40 and 49), the NRC staff finds that the proposed safety limit represents a conservative value to provide confidence that the fuel elements will maintain their integrity and that no cladding damage should occur. Therefore, the NRC staff finds that the safety limit in proposed TS 2.1 conforms to the definition of safety limits in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36, "Technical specifications," paragraph (c)(1)(i)(A) and is acceptable.

2.8 Limiting Safety System Settings

In accordance with 10 CFR 50.36 the licensee proposes a LSSS designed to ensure that automatic protective action (reactor shutdown) would occur in sufficient time to prevent safety limits from being exceeded.

The value used by the licensee to set the reactor instrumentation is a fuel temperature of 750 °C (1,382 °F), as indicated in proposed TS 2.2 which also requires the instrumented element to be located in the peak power location of the core. This value is consistent with the guidance in NUREG-1537, appendix 14.1, section 2.2, and American National Standards Institute/American Nuclear Society (ANSI/ANS-15.1-2007) (Ref. 55), section 2.2, "Limiting safety system settings," which states, in part, "LSSSs are those limiting values for settings of the safety channels by which point protective action must be initiated. The LSSS are chosen so that automatic protective action will terminate the abnormal situation before a safety limit is reached."

The NRC staff finds that the LSSS of 750 °C (1,382 °F), as indicated in proposed TS 2.2, located in the peak power (i.e., highest temperature) location of the UCD/MNRC core will help ensure that the safety limit of 930 °C (1,706 °F) in proposed TS 2.1, will not be exceeded. The NRC staff also finds that the temperature provides a significant safety margin to allow for any difference between true and measured values and allow automatic protective action to be initiated before safety limits are exceeded. Therefore, the NRC staff finds proposed TS 2.2 conforms to the definition of 10 CFR 50.36(c)(1)(ii)(A) and is acceptable.

2.9 Reactor Description Conclusions

The NRC staff reviewed the information pertaining to the design, construction, function, and operation of the reactor fuel, neutron reflectors, grid and safety plates, moderator/graphite elements, neutron source, control rods and reactor core support structure, focusing on elements that have changed since the issuance of UCD/MNRC license. On the basis of its review, the NRC staff concludes that the design of these core-related components for the UCD/MNRC facility are acceptable and continue to permit safe operation and shutdown of the reactor. The NRC staff also concludes, as limited by proposed TS 5.3.2, "Reactor Fuel," that the use of 20/20 and 30/20 fuels in the UCD/MNRC TRIGA reactor remains acceptable.

The licensee discussed and proposed minimum shutdown margin and excess reactivity limits, in proposed TS 3.1.2, "Shutdown Margin," and TS 3.1.1, "Excess Reactivity," respectively, that are acceptable to the NRC staff. The NRC staff concludes that the minimum shutdown margin ensures that the reactor can be shut down from any operating condition with the highest worth control rod stuck out of the core. The limit on excess reactivity allows operational flexibility while limiting the reactivity available for reactivity addition accidents.

Reactivity limits on experiments, in proposed TS 3.8.1, “Reactivity Limits,” limit the absolute value of all experiments, movable experiments, and secured experiments. The licensee has proposed values that are bounded by pulse reactivity insertion accident analysis. Therefore, the NRC staff concludes that failure of experiments will not add unacceptable amounts of reactivity to the reactor.

The information provided in the UCD/MNRC SAR, as supplemented, includes T-H analysis for the UCD/MNRC reactor. The NRC staff concludes that the licensee has justified the assumptions and methods used. The T-H analysis provides reasonable assurance that the reactor can be operated at its licensed power level without undue risk to the health and safety of the public.

The fuel and core design, when considered with the restrictions and requirements on the operation of the reference cores and variations of the reference cores, in proposed TS 3.1.3, “Core Configuration Limits,” and TS 3.1.4, “Fuel Parameters,” and the LSSS, in proposed TS 2.2, “Limiting Safety System Setting,” will ensure that the maximum fuel temperature will not exceed the safety limit for steady-state operation of 930 °C (1,706 °F). The LSSS is set at 750 °C (1,382 °F) which is 180 °C (324 °F) less than the safety limit (930 °C (1,706 °F)) to provide a safety margin; therefore, the reactor will be shut down before reaching the safety limit in proposed TS 2.1,”Safety Limits.” Given the operational conditions specified in proposed TS 2.1, TS 2.2, TS 3.1.1, TS 3.1.2, TS 3.1.3, TS 3.1.4, TS 3.8.1, and TS 5.3.2, the NRC staff concludes that there is reasonable assurance that the UCD/MNRC TRIGA research reactor can be operated safely at power levels up to 1.03 MW(t).

3 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

3.1 Radiation Protection

Title 10 of the *Code of Federal Regulations* (10 CFR) 20.1101, “Radiation protection programs,” specifies, 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 upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA). The basic aspects of the radiation protection program include occupational and general public exposure limits, surveys and monitoring, and personnel dosimetry.

Safety analysis report (SAR) chapter 11.0, “Radiation Protection and Waste Management,” describes the radiation protection and waste management program at the University of California-Davis/McClellan Nuclear Research Center (UCD/MNRC).

The U.S. Nuclear Regulatory Commission (NRC) inspection program routinely reviews radiation protection and radioactive waste management at the UCD/MNRC. Initially, the NRC staff reviewed the licensee’s annual operating reports (ARs) (Ref. 24) summarizing UCD/MNRC operations, and the NRC inspection reports (IRs) (Ref. 25) from calendar years (CYs) 2012 to 2020. However, as described below, the NRC staff identified discrepancies with the Argon (Ar)-41 dose calculations to a member of the public reported in its ARs. As a result, the NRC staff expanded its review of the ARs and IRs to include CY 2009 to CY 2011 for the radiation protection area to ensure accurate Ar-41 dose methodology and calculational results. The licensee provided updated Ar-41 dose calculations in its response to the NRC staff’s requests for additional information (RAIs), by letter dated March 31, 2022 (Ref. 23). Following its review of the updated Ar-41 doses (discussed in SER section 3.1.1, which was evaluated by the NRC staff and found acceptable), the NRC staff finds that the licensee’s radiation protection program demonstrates that adequate measures are in place to minimize radiation exposure to UCD/MNRC 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 UCD/MNRC are described in SAR section 11.1.2, “Radiation Sources.” The NRC staff reviewed the description of radiation sources, the inventories of each physical form and the sources’ locations, and potential radiation hazards as presented in Chapter 11 of the SAR, and verified that the hazards were accurately depicted and comprehensively identified. Primary radiation sources are directly related to reactor operation that include activation of air in the radiography bays and activation of air dissolved in the primary coolant. Liquid sources are limited at the UCD/MNRC facility and are typically limited to the primary coolant since no routine liquid effluent releases are planned at the facility.

Airborne Radiation Sources

SAR section 11.1.2.1, “Airborne Radiation Sources,” states that during normal operations, the primary airborne sources of radiation are Ar-41 and nitrogen-16 (N-16). Ar-41 results principally from irradiation of the air in experimental facilities and activation of dissolved air in the reactor pool water. Ar-41 is generated in several ways at the UCD/MNRC facility, which include the activation of air in the radiography bays, the reactor room, the Pneumatic Transfer System, and the primary coolant water around the reactor core.

N-16 is produced when oxygen in the pool water is irradiated in the reactor core that then must diffuse to the pool surface before it is released to the atmosphere. When N-16 is produced in the core it rises to the top of the reactor tank. SAR section 5.6, "Nitrogen-16 Control System," states that the facility is equipped with an in-tank diffuser to increase the time it takes for N-16 to rise to the top. This diffuser is operating during all operations of the reactor. Given that N-16 has a very short half-life (7.14 seconds) the nitrogen produced from the core essentially decays away before rising to the surface.

Occupational Dose

Nitrogen-16

In SAR section 11.1.2.1.5, "Production and Evolution of N-16 in the Reactor Room," the licensee describes that the reactor room dose rates are primarily from N-16 in the air. The calculated N-16 concentrations, while being diluted by air, are calculated by the licensee to be 30 mrem/hr 1 feet (ft) (0.30 meters (m)) above the reactor pool, 4-8 mrem/hr at the reactor room radiation area monitors (RAM) 5 ft (1.5 m) above the side of the reactor tank, and 1-2 mrem/hr in the rest of the reactor room. Furthermore, in SAR section 11.1.2.1.5, the licensee states that the dose rates are mitigated by minimizing the amount of time workers and visitors are allowed to stay in the reactor room and by closely monitoring recorded worker and visitor dose.

The NRC staff finds that the licensee's N-16 concentration calculations are acceptable, and the licensee's program to mitigate worker doses by limiting their stay time in the reactor bay is also acceptable. Additionally, the NRC staff finds that the licensee's calculated N-16 doses to an occupational worker is below the 10 CFR 20.1201, "Occupational dose limits for adults," occupational dose limit of 5 rem TEDE in a year. Therefore, the NRC staff concludes that the licensee's occupational airborne dose estimates of N-16 for the operation of the UCD/MNRC are acceptable.

Argon-41

The licensee provides the following assessment describing the main source of Ar-41 at the UCD/MNRC facility. SAR section 11.2.1.1, "Argon-41 in the Radiography Bays," provides the results of the Ar-41 analyses, which indicates that the Ar-41 concentrations in the radiography bays are very low (orders of magnitude less than the derived air concentrations (DAC).

SAR section 11.1.2.1.2, "Production and Evolution of Ar-41 in the Reactor Room," indicates that actual measurements of Ar-41 in the reactor room after reactor operations indicates that the Ar-41 contributes very little to the reactor room dose of 3-4 mrem/hr. SAR section 11.1.2.1.2, also states that the production of Ar-41 in the reactor room was verified by measurements after 9 hours of operation at 1.0 megawatt-thermal (MWt). The air concentration measured at this time was $1.5x10^{-6}$ microcuries per milliliter (µCi/ml) for areas which are occupied during normal work in the reactor room. SAR section 11.1.2.1.2, states that in measurements after reactor operation at 1.0 MWt with the reactor room exhaust operating resulted in Ar-41 concentrations averaging $1.5x10^{-6}$ µCi/ml, which corresponds to a dose rate of 1.25 millirem/hour (mrem/hr).

SAR section 11.1.2.1.3, “Ar-41 from the Pneumatic Transfer System,” states that, during operation, very small amounts of Ar-41 is exhausted from the Pneumatic Transfer System, and not a measurable contributor to the Ar-41 doses. Also, the licensee indicates in SAR appendix A, section A.3, “Production Rate of Ar-41 From Coolant Water,” that the vast majority of Ar-41 effluents to the environment is from activation of Ar-41 in the primary coolant, and not from the activation of air in the neutron beams.

The maximum air concentration limit for occupational workers is 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, “Standards for Protection Against Radiation.” For Ar-41, the DAC value is $3x10^{-6}$ µCi/ml. The DAC concentration limit corresponds to a 5-rem occupational dose for an individual worker assumed to be exposed to the limiting DAC continuously for 2,000 hours.

The NRC staff finds that the Ar-41 doses from the radiography bays are of low significance to the dose received at the facility based its review of the calculations used for the neutron beamline assessments for activation of the Ar-40 in air. The licensee’s assessment indicates that the maximum calculated Ar-41 concentration is $2.24x10^{-8}$ µCi/ml, which is a value that is over 100 times less than the DAC value for Ar-41 of $3x10^{-6}$ µCi/ml.

The NRC staff review of the licensee’s ARs from CY 2009 through CY 2020 confirmed that the highest reported individual occupational total effective dose equivalent (TEDE) was 169 mrem for this time period. SER table 3.1, “Summary of Data from Annual Reports for Personnel Radiation Exposures from 2009 to 2020,” provides the average TEDE and the greatest individual TEDE reported in the ARs for each year.

Table 3.1 - Summary of Data from Annual Reports for Personnel Radiation Exposures from 2009 to 2020

Year of Operation	Average TEDE (mrem)	Greatest Individual TEDE (mrem)
2009	28	63
2010	59	169
2011	13	50
2012	29	56
2013	34.8	162
2014	23	79
2015	19	55
2016	55	120
2017	87.4	115
2018	65	149
2019	46	107
2020	23	105

Based on its review of the information described above, including the licensee's Ar-41 measurements, the NRC staff finds that the majority of the Ar-41 dose to the workers comes from the reactor pool water, and not the neutron radiography bays, reactor bay, or the pneumatic transfer system. The NRC staff also finds that the licensee's dose rate estimates from Ar-41 are conservative and satisfy the requirements of 10 CFR Part 20. The NRC staff finds that the licensee provided estimates and measured doses to an occupational worker is below the 10 CFR 20.1201, "Occupational dose limits for adults," occupational dose limits of 5 rems TEDE in a year. Therefore, the NRC staff concludes that the licensee's occupational airborne dose estimates for the operation of the UCD/MNRC are acceptable.

Dose to Members of the Public

The regulations, 10 CFR Part 20, "Subpart D—Radiation Dose Limits for Individual Members of the Public," 10 CFR 20.1301, "Dose limits for individual members of the public," limit the total effective dose equivalent to individual members of the public from the licensed operation to 0.1 rem (100 mrem) in a year. The effluent concentration limit for Ar-41, provided in appendix B to 10 CFR Part 20, is 1×10^{-8} µCi/ml, which if inhaled or ingested continuously over the course of a year, would produce a TEDE of 0.05 rem (50 mrem or 0.5 millisieverts). Paragraph (d) of 10 CFR 20.1101 limits radiation dose from gaseous effluents to 10 mrem/yr.

In SAR section 11.1.2.1.4, "Ar-41 Release to the Unrestricted Area," the licensee indicates that radioactive air concentrations produced by the UCD/MNRC facility are discharged through the facility's exhaust stack, which is 60 ft (18.3 m) above ground level. Dilution with other building ventilation air and atmospheric dilution reduces the Ar-41 concentration considerably before the exhaust plume returns to ground level locations that could be occupied by UCD/MNRC personnel or the general public. SAR figure 9.11, "UCD/MNRC Air Handling System," provides a high-level depiction of the air handling system and describes the facility stack as being the combined release point for all air systems. The licensee uses the 2019 effluent data from the UCD/MNRC facility to report that the facility released 27.6 curies (Ci) of Ar-41 over 1,430 hours of operation at 1.0 MWt. SAR section 11.1.2.1.4 indicates that the facility's typical stack flow rate is 5,678 CFM [cubic feet per minute]. Based on this flow rate, the licensee uses a concentration of 2×10^{-6} μCi/ml for Ar-41 discharged from the facility's exhaust stack, for its calculated dose to a member of the public in the unrestricted area. In SAR appendix A, section A.4, "Maximum Impact of Ar-41 Outside the Operations Boundary," the licensee provides the methodology and results of its calculation of the Ar-41 doses to members of the public, using the HotSpot computer code.

To verify the licensee's calculated values, the NRC staff reviewed information from the ARs for CY 2009 through CY 2020. In SER table 3.2, "Summary of AR Data for Gaseous Effluents from CY 2009 to CY 2020," the NRC staff summarized the Ar-41 gaseous effluent data and provided a comparison for annual average concentrations of Ar-41 released, as well as the total doses reported for each year.

Table 3.2 Summary of AR Data for Gaseous Effluents from CY 2009 to CY 2020

Year of Operation (CY)	**Total Ci Released per Year (Ci)**	**Annual Average Concentration of Ar-41 Released at Unrestricted Area (μCi/ml)**	**Total Dose from Ar-41 at Unrestricted Area (mrem)**
2009	27.38	3.22×10^{-7}	10.36
2010	17.20	7.85×10^{-11}	5.73
2011	14.49	6.50×10^{-11}	4.74
2012	16.95	8.81×10^{-11}	6.43
2013	10.79	5.02×10^{-11}	3.67
2014	14.20	6.67×10^{-11}	4.87
2015	18.50	8.75×10^{-11}	6.38
2016	19.27	9.09×10^{-11}	6.64
2017	19.12	9.44×10^{-11}	6.89
2018	24.87	1.20×10^{-10}	8.84
2019	27.58	1.34×10^{-10}	9.79
2020	35.91	3.98×10^{-7}	12.80

During its review, the NRC staff noted that the Ar-41 doses to a member of the public for CY 2009 and CY 2020 (highlighted SER table 3.2) were above the 10 CFR 20.1101, paragraph (d) limit of 10 mrem, and, following discussions with the licensee, requested clarification by RAI letter, dated February 8, 2022 (Ref. 53). The licensee indicates that some of the Ar-41 calculations had been done incorrectly and provided corrected Ar-41 concentration and dose data in its RAI response dated March 30, 2022 (Ref. 23) for CY 2009 through CY 2020. The corrected information is provided in SER table 3.3, "Corrected AR Data for Gaseous Effluents from CY 2009 to CY 2020."

Table 3.3 Corrected AR Data for Gaseous Effluents from CY 2009 to CY 2020

Year of Operation (CY)	Total Ci Released per Year (Ci)	Annual Average Concentration of Ar-41 Released at Unrestricted Area (μCi/ml)	Total Dose from Ar-41 at Unrestricted Area (mrem)
2009	27.38	2.29×10^{-10}	1.14
2010	17.2	1.44×10^{-10}	0.72
2011	14.5	1.21×10^{-10}	0.61
2012	16.97	1.42×10^{-10}	0.71
2013	10.78	9.03×10^{-11}	0.45
2014	14.21	$1.19\text{E} \times 10^{-10}$	0.60
2015	18.51	$1.55\text{E} \times 10^{-10}$	0.78
2016	19.27	1.61×10^{-10}	0.81
2017	19.11	1.6×10^{-10}	0.80
2018	24.88	2.08×10^{-10}	1.04
2019	27.59	2.31×10^{-10}	1.16
2020	35.9	3.01×10^{-10}	1.50

The NRC staff's review of the licensee's Ar-41 dose calculations finds that the licensee's use of the HotSpot computer code is an industry accepted method for calculating radiation doses from the facility, and that the licensee used conservative assumptions in the HotSpot computer code calculations. The NRC staff reviewed the corrected doses (listed in SER table 3.3) and performed confirmatory calculations using the methodology described in the licensee's response to the NRC's RAIs (Ref. 23). The NRC staff's finds that the updated Ar-41 dose calculations values were consistent with the NRC staff's confirmatory calculations, and therefore, is acceptable. The NRC staff finds that the results of the licensee's annual average Ar-41 concentrations released to the public remain below the limit in 10 CFR Part 20, appendix B, table 2 of 1×10^{-8} μCi/ml, and the annual doses are also below the 10 mrem limit specified in 10 CFR 20.1101.

Proposed technical specification (TS) 3.7.2, "Effluents-Argon-41 Discharge Limit," states that the total Ar-41 released by UCD/MNRC shall not exceed 118 Ci per calendar year. In its response to the NRC's RAIs (Ref. 23), the licensee calculated a radiation dose, to the maximum exposed member of the public, of 1 mrem for every 23.7 Ci of Ar-41 released, which equates to 5 mrem at the proposed TS 3.7.2 limit of 118 Ci. The NRC staff finds that the licensee's limit of 118 Ci in proposed TS 3.7.2 is well below the limit of 100 mrem in 10 CFR 20.1301, and below to ALARA constraint 10 mrem in 10 CFR 20.1101(d). Based on its review, the NRC staff concludes that proposed TS 3.7.2 will limit the release of routine effluent releases of Ar-41 in accordance with the requirements of 10 CFR Part 20 and is acceptable.

Liquid Radiation Sources

SAR section 11.1.2.2.1, "Radioactivity in the Primary Coolant," states that the only significant liquid radioactive source at the UCD/MNRC is the reactor primary coolant. Radioactivity in this liquid source occurs due to neutron activation of Ar-40 in entrained air (creating Ar-41); neutron interactions with oxygen in the water molecule (creating N-16); and neutron interactions with tank and structural components with subsequent transfer of the radioactivity into the primary coolant. Radionuclides such as Manganese (Mn)- 56 and Sodium (Na)-24 are common examples of waterborne radioactivity created in this manner. Tritium (H-3) is also present in the primary coolant due to activation of natural deuterium in water. Because the pool and primary system piping contains water that has been circulated through the reactor core, radioactive corrosion products produced during normal operation may be capable of producing radiation exposure to personnel. The activated corrosion products are deposited in a mechanical filter and demineralizer resins therefore, this waste is treated as solid wastes.

The primary sources of radioactivity include Aluminum (Al)-28, Ar-41, H-3, Magnesium (Mg)-27, Molybdenum (Mo)-99, Cobalt (Co)-60, Mn-56 and Na-24. "Airborne Radiation Sources," in SER section 3.1.1, states activation of the oxygen in water also creates N-16 in the coolant. SAR table 11-4, "Predominant Radionuclides and their Equilibrium Concentration in MNRC Reactor Primary Coolant at 1 MW," contains the typical concentrations of the radionuclides in the primary coolant.

The licensee states that its policy is not to release liquid radioactivity as an effluent or as liquid waste. As such, the liquid sources of radioactivity are not a source of public exposure. In scenarios where occupational workers are required to perform activities such as maintenance, the licensee states that the primary coolant could be allowed to decay for several days to reduce the worker exposures.

The NRC staff's review of the licensee's ARs reports from CY 2009 through CY 2020 finds that no liquid releases were performed, and that the liquid radioactive sources generated from reactor operations are consistent with other research reactors. Based on the NRC staff's review as described above, the NRC staff finds that liquid radioactive sources from continued normal operation of the UCD/MNRC are acceptably controlled, and do not pose a hazard to the UCD/MNRC staff or member of the public.

Solid Radiation Sources

In the SAR section 11.1.2.3, "Solid Radioactive Sources," the licensee provides a representation of the solid waste generation rates expected at the UCD/MNRC facility in SAR table 11-5, "Representative Radioactive Sources for the UCD/MNRC." This table includes the anticipated curies expected as well as the physical characteristics expected for the radioactive material which includes sealed sources and other wastes encountered during operations. Routinely produced solid waste mainly consists of water purification system demineralizer resin bottles, mechanical filters, rags, paper towels, plastic bags, rubber gloves, along with other materials that may be used for contamination control or decontamination. The licensee stated that it anticipates generating one or two 55-gallon drums of solid waste and two resin bottles. The licensee states it disposes of one B-25 waste container box of radioactive waste from the facility every 5 years.

As part of its license renewal application (LRA) review, the NRC staff reviewed UCD/MNRC ARs from CY 2009 through CY 2020 (Ref. 24) and NRC IRs from CY 2009 through CY 2021 (Ref. 25) and finds that solid radioactive waste handling has not resulted in any significant personnel exposure at the UCD/MNRC facility. The NRC IRs concluded that the licensee's program was acceptable, directed toward the protection of public health and safety, and no violations involving radioactive waste were identified. The IRs documented that these operations were conducted in accordance with the license and regulatory requirements and were within the specified regulatory and TS requirements. In addition, the NRC staff noted that the only shipment noted in these IRs took place in 2014 which is consistent with the licensee's SAR that shipments of solid waste occur about every 5 years.

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

3.1.2 Radiation Protection Program

The regulation, 10 CFR 20.1101(a) requires each licensee to develop, document, and implement a radiation protection program. The NRC inspection program routinely reviews the radiation protection program at the MNRC facility for compliance. The licensee stated that the Health Physics branch within the UCD/MNRC is the organization that administers the radiation safety program. The radiation protection program is discussed in SAR section 11.1.3, "Radiation Protection Program." The radiation protection program at the UCD/MNRC facility is implemented by the Health Physics Supervisor who is also the Radiation Safety Officer (RSO). The UCD/MNRC facility has a structured radiation protection program, which is implemented by qualified health physics staff that is equipped with radiation detection capabilities to determine, control, and document occupational radiation exposures at the facility. Their responsibilities include directing and developing the radiation protection program; implementing radiation protection policies and procedures; and providing radiological control during reactor operations and maintenance.

Program Controls, Organization, and Responsibilities

SAR section 11.1.3.1, "Organization of the Health Physics Branch," provides the positions of authority and responsibility within the Health Physics Branch are as follows:

- Health Physics Supervisor (Radiation Safety Officer) - The Health Physics Supervisor reports directly to the MNRC Facility Director. The Health Physics Supervisor is responsible for directing the activities of the Health Physics Branch including the development and implementation of the MNRC Radiation Protection Program.

- Health Physicist - Health Physicists report to the Health Physics Supervisor. Health Physicists are responsible for implementing the MNRC Radiation Protection Program policies and procedures, and directing the activities of the Health Physics Technicians.

- Health Physics Technicians - Health Physics Technicians report directly to the Health Physicist on-duty. Health Physics Technicians are responsible for providing radiological control during reactor operations and maintenance. This includes radiological monitoring, surveillance checks on radiological monitoring equipment and radiological control oversight of operations involving radiation and/or contamination. The position description for the health physics technician specifies the authority to interdict perceived unsafe practices. Typically, ROs and SROs are trained to perform some of the tasks described here. ROs and SROs trained in this manner are to report health physics issue to the RSO and other operational issue to the reactor supervisor.

Procedures

SAR section 11.1.3.3, “Health Physics Procedures and Radiation Work Permits (RWP),” states that the operation of the radiation protection program is the responsibility of the Health Physics Supervisor and is implemented by the facility’s health physics personnel using health physics (HP) procedures and the radiation work permits (RWP). These HP procedures and RWP are reviewed by the health physics and operations supervisors and are approved by the UCD/MNRC Director. These procedures and RWP are audited on an annual basis by a Nuclear Safety Committee (NSC), as well as reviewed annually by health physics staff. The radiation work permit program covers a wide range of tasks which can be observed in SAR section 11.1.3.3.

SAR section 11.1.3.1 states that the RSO is responsible for the implementation of the program and will report to the UCD/MNRC Director. The program is implemented using written HP procedures as required by proposed TS 6.4, “Procedure,” item 5. The procedure topic areas stated in proposed TS for health physics includes the following:

- Testing and calibration of facility radiation monitoring
- Work performed in laboratories and other areas where radioactive material is used
- Facility radiation monitoring programs including surveys, monitoring and handling of radioactive waste, and the sampling and analysis of solid, liquid, and gaseous effluents
- Monitoring of radioactivity in the environment around the facility
- Administrative guidelines for the facility protection program, which includes personnel orientation and training
- Receipt of radioactive materials at the facility, and the unrestricted release of materials and items from the facility
- Leak testing of sealed sources
- Special nuclear material accountability
- Transportation of radioactive materials

The NSC of the UCD/MNRC facility periodically reviews the program. The NRC inspection program routinely reviews the radiation protection program. The NRC staff reviewed the information provided in the SAR, together with the licensee’s ARs and NRC IRs from CY 2009 through CY 2020 and finds that the licensee demonstrated that adequate measures are in place to minimize radiation exposure to personnel and provide adequate protection against operational releases of radioactivity to the environment.

Training

SAR section 11.1.3.4, “Radiation Protection Training,” states that radiation protection training is conducted by the Health Physics Branch and is structured at different levels in order to meet the needs of different categories of facility staff and researchers using the reactor. All personnel and visitors entering the UCD/MNRC facility shall receive training in radiation protection sufficient for the work/visit or shall be escorted by an individual who has received such training. Further, annual refresher training is required for all personnel permitted unescorted access and covers review of proper radiation safety practices, occurrences at the MNRC facility over the past year, ALARA summary, and notable changes in procedures, equipment, and the facility.

The general levels of training are as follows:

Initial Training for all personnel permitted unescorted access to the UCD/MNRC facility:

- Storage, transfer, and use of radiation and/or radioactive material in portions of the restricted area, including radioactive waste management and disposal;
- Health protection problems and health risks (including prenatal risks) associated with exposure to radiation and/or radioactive materials;
- Precautions and procedures to minimize radiation exposure (ALARA);
- Purposes and functions of protective devices;
- Applicable regulations and license requirements for the protection of personnel from exposure to radiation and/or radioactive materials;
- Responsibility to report potential regulatory and license violations or unnecessary exposure to radiation or radioactive materials;
- Appropriate response to warnings in the event of an unusual occurrence or malfunction that involves radiation or radioactive materials; and
- Radiation exposure reports which workers will receive or may request.

Specialized training is for individuals who require more in-depth training than described in initial training. These topics include:

- Principles of Atomic Structure;
- Radiation Characteristics;
- Sources of Radiation;
- Interaction of Radiation with Matter;
- Radiation Measurements;
- Biological Effects of Radiation;
- Radiation Detection;
- Radiation Protection Practices;
- ALARA; and
- Radioactive Waste Management and Disposal.

Audit Function

SAR section 11.1.3.5, “Audits of the Health Physics Program,” states that the NSC provides timely, objective, and independent reviews, audits, recommendations, and approvals on matters affecting nuclear safety at the UCD/MNRC. The NSC charter requires that membership shall consist of individuals who have the extensive experience necessary to evaluate the safety of the UCD/MNRC. The chairman of the NSC is appointed by the UCD/MNRC license holder. Voting membership on the NSC is specified in the NSC Charter. The independent members are voting members and are selected based on their technical qualifications. NSC meetings are held at least semi-annually (the period between meetings cannot exceed 7.5 months). The NSC is chartered to conduct an annual on-site audit/inspection of the UCD/MNRC health physics and reactor operations programs and associated records. The annual health physics inspection is performed by an independent member of the NSC and normally covers all aspects of the radiation protection program. The audit typically covers areas such as actions on NSC recommendations from previous audits, health physics staffing, the interface between health physics and reactor operations, health physics training for UCD/MNRC staff and UCD/MNRC users, health physics procedures, personnel monitoring, environmental monitoring, effluent monitoring, operational radiological surveys, instrument calibration, radioactive waste management and disposal, radioactive material transportation, special nuclear material accountability, and a review of unusual occurrences. The audit reports are sent to the chairman of the NSC, who in turn presents a report of the audit findings to the full NSC at the next NSC meeting. Copies of the audit findings are provided to the UCD/MNRC Facility Director who is responsible for ensuring that corrective actions are taken.

The audit and inspection function is described in proposed TS 6.2.4, "Audit Function," discussed in SER section 5.6.2.4, which was evaluated by the NRC staff and found acceptable.

The NRC staff reviewed the UCD/MNRC radiation protection program, as described in the SAR, and finds that the program controls, organization, responsibilities, training, procedures, and audit requirements complies with the requirements in 10 CFR 20.1101(a). The NRC staff finds that the UCD/MNRC radiation protection program is implemented in an acceptable manner, and provides reasonable assurance that, for all facility activities, the program will protect the UCD/MNRC staff, the environment, and the public from unacceptable radiation exposures. On this basis, the NRC staff concludes that the UCD/MNRC radiation protection program is acceptable.

3.1.3 ALARA Program

The regulation, 10 CFR 20.1101(b), requires that licensees use, to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are ALARA.
SAR section 11.1.4, "ALARA Program," states the ALARA program for the UCD/MNRC has been established in accordance with 10 CFR 20.1101. The bases for this program are the guidelines found in American National Standards Institute/American Nuclear Society (ANSI/ANS)-15.11-2009, "Radiation Protection at Research Reactor Facilities" (Ref. 50). The ALARA program incorporates a review of all UCD/MNRC operations with emphasis on operational procedures and practices that might reduce UCD/MNRC staff exposures to radiation and lower potential radioactive effluent releases to unrestricted areas. Personnel radiation doses at the UCD/MNRC are minimized by considering use of the following ALARA actions when performing work with radiation or radioactive materials:

- Reviewing records of similar work previously performed;
- Eliminating unnecessary work;

- Preparing written procedures;
- Using special tools;
- Installing temporary shielding;
- Performing as much work as possible outside of radiation areas;
- Performing mockup training;
- Conducting prework briefings and post-work critiques; and
- Keeping unnecessary personnel out of areas where radiation exposure may occur.

In addition to the above actions, the UCD/MNRC ALARA program also contains the following elements which are designed to enhance the effectiveness of the overall program:

- Exposure investigations are conducted when an individual receives greater than 100 mrem in one month or 300 mrem in one quarter. The investigation is focused on determining the cause of the exposure so that appropriate ALARA actions, if any, can be applied;
- ALARA dose trend analysis charts are prepared quarterly and posted for review by all UCD/MNRC personnel; and
- An annual inspection of the UCD/MNRC ALARA program wherein a health physicist is required to be involved during planning, design approval, and construction of new UCD/MNRC facilities; during planning and implementation of new UCD/MNRC reactor use; during maintenance activities; and during the management and disposal of radioactive waste. In addition, written procedures pertaining to the preceding operational facilities are required to be reviewed by the Health Physics Supervisor for ALARA considerations prior to implementation.

The NRC staff reviewed the ARs and NRC IRs from CY 2009 to CY 2020 and finds that the program provided guidance for keeping doses ALARA and was consistent with the requirements in 10 CFR Part 20. The NRC staff also finds that radiation doses to UCD/MNRC 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 UCD/MNRC ALARA program complies with 10 CFR 20.1101 (b) and provides reasonable assurance that radiation exposure will be maintained ALARA for all facility activities. Therefore, the NRC concludes that the UCD/MNRC ALARA program is acceptable.

3.1.4 Radiation Monitoring and Surveying

The regulation, 10 CFR 20.1501 states:

(a) Each licensee shall make, or cause to be made, surveys of areas, including the subsurface, that -

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

(2) Are reasonable under the circumstances to evaluate—

(i) The magnitude and extent of radiation levels; and

(ii) Concentrations or quantities of radioactive material; and
(iii) The potential radiological hazards of the radiation levels and residual radioactivity detected

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

SAR section 11.1.5, “Radiation Monitoring and Surveying,” states that the radiation monitoring program for the UCD/MNRC reactor is structured to ensure that all three categories of radiation sources (air, liquid, and solid) are detected and assessed in a timely manner.

Radiation Surveys

SAR section 11.1.5.1, “Monitoring for Radiation Levels and Contamination,” states that the routine monitoring program at UCD/MNRC is structured to make sure that adequate radiation measurements of both radiation fields and contamination are made on a regular basis. This program includes, but is not limited to, the following:

Typical surveys for radiation fields as follows:

1. Surveys whenever operations are performed that might significantly change radiation; levels in occupied areas;
2. Daily surveys at temporary boundaries (e.g., rope barriers);
3. Monthly surveys in accessible radiation areas and high radiation areas, and in all other occupied areas of the UCD/MNRC facility;
4. Annual surveys outside of the UCD/MNRC facility, but within the facility fence;
5. Annual surveys in radioactive material storage areas;
6. Annual surveys of potentially contaminated ventilation ducting outside of the UCD/MNRC facility;
7. Surveys upon initial entry into a radiography bay after the shutter is closed or upon entry into the demineralizer cubicle;
8. Surveys in surrounding areas where personnel could potentially be exposed when radioactive material is moved;
9. Surveys when performing operations that could result in personnel being exposed to small intense beams of radiation (e.g., when transferring irradiated fuel, when removing shielding, or when opening shipping/storage containers);
10. Surveys of packages received from another organization;
11. Surveys when irradiated parts or equipment are removed from a radiography bay, or from the reactor core, from a fuel storage pit, from the pneumatic transfer system terminal, or from the reactor room;
12. Surveys as necessary to control personnel exposure. Such surveys may include the following:
    a) Gamma surveys of potentially contaminated exhaust ventilation filters when work is performed on these filters;
    b) Gamma and neutron surveys on loaded irradiated fuel containers;

c) Gamma and neutron surveys when handling an unshielded neutron source.

Typical surveys for contamination as follow:

1. Surveys at the exits to the UCD/MNRC facility once per shift;
2. Daily surveys in accessible contaminated areas and occupied areas surrounding contaminated areas;
3. Monthly surveys in occupied non-contaminated areas of the UCD/MNRC;
4. Annual surveys in areas outside of the UCD/MNRC facility, but within the facility fence;
5. Annual surveys in radioactive material storage areas;
6. Surveys as necessary to control the spread of contamination whenever operations are performed that are known to result in, or expected to result in, the spread of contamination;
7. Surveys prior to removal of paint from areas where contaminated paint is possible;
8. Surveys as part of the following operations:
    a) Decontamination of equipment;
    b) Removal of irradiated parts or equipment from a radiography bay, from the reactor core, from a fuel storage pit, from the pneumatic transfer system terminal, from the reactor room, or from the UCD/MNRC facility;
    c) inspection, maintenance, or repair of the primary cooling system;
    d) initial opening of the secondary cooling system (heat exchanger) for inspection, maintenance, or repair;
    e) When working in or entering areas where radioactive leaks or airborne radioactivity has occurred previously;
    f) Upon initial entry into potentially contaminated exhaust ventilation ducting;
    g) Prior to replacing filters or ducting in potentially contaminated exhaust ventilation systems.

The NRC staff reviewed IRs for the CYs 2009 through 2020 which documented radiation surveys performed at the UCD/MNRC facility, and no findings nor violations were noted. Further, the NRC staff finds that UCD/MNRC's radiation survey program is consistent with the guidance in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors" (Ref. 11) section 11.1.4, "Radiation Monitoring and Surveying," which states that the licensee should document the types of surveys performed and the associated areas of the facility.

Radiation Monitoring Equipment

SAR section 11.1.5.2, “Radiation Monitoring Equipment,” provides information related to the radiation monitoring equipment used in the UCD/MNRC reactor program. The licensee provides a summary table of the radiation monitoring in SAR table 11-6, “Radiation Monitoring and Related Equipment used in the MNRC Radiation Protection Program,” which describes the types of radiation monitors used as well as the location of the monitors with their intended functions.

SAR section 11.1.5.3, “Instrument Calibration,” states that radiation monitoring instrumentation is calibrated according to written procedures. It is the policy of the MNRC to use National Institute of Standards and Technology traceable sources for instrument calibrations whenever possible. The licensee uses a combination of calibration at its own facility and calibration at a contractor calibration facility to perform equipment calibrations. Instrument calibrations are tracked by a computer-based tracking system. Instrument calibration records are maintained by the Health Physics Branch and calibration stickers showing pertinent calibration information (e.g., counting efficiency, the most recent calibration date, and the date the next calibration is due) is attached to all instruments.

The NRC staff finds that UCD/MNRC the instrument calibrations are tracked by the licensee’s computer-based tracking system and routinely checked by NRC inspectors, as described in their IRs. Based on its review of the NRC IRs, the NRC staff finds that the licensee’s radiation monitoring equipment is routinely tracked, documented, and maintained. Further, the NRC staff finds that the placement, use, and control of the radiation monitoring and surveying equipment are in accordance with applicable standards, guidance, and regulations. The equipment selected is appropriate for detecting the types and intensities of radiation likely to be encountered within the facility at appropriate frequencies to help ensure compliance with 10 CFR Part 20 requirements and the facility ALARA program under all operating conditions.

The requirements for the UCD/MNRC radiation monitoring systems are provided in proposed TS 3.7.1, “Monitoring Systems,” (discussed in SER section 5.3.7.1, which was evaluated by the NRC staff and found 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 at appropriate frequencies to help ensure compliance with 10 CFR 20.1501(a) and (b). Therefore, the NRC staff concludes that the UDC/MNRC radiation monitoring and surveying programs are acceptable.

Personnel Dosimetry

SAR section 11.1.6.5.2, “Personnel Dosimetry Devices,” provides information on the personnel dosimetry devices in use at the UCD/MNRC facility. Personnel exposure is monitored by beta-gamma and neutron optically stimulated luminescent (OSL) badges, OSL finger rings, and CR-39 Track Etch dosimeters. Dose rates in radiation areas are measured using survey meters and the measured dose rates are posted where required. In regard to tour groups, the licensee states that these individuals typically spend 30 minutes in the reactor room tour and receive a 1 mrem dose.

The NRC staff review of the annual exposure results recorded in the UCD/MNRC ARs for the period 2009 through 2020, confirms that worker doses were kept below the limits and were on average less than 100 mrem with the greatest individual TEDE for these years being less than 200 mrem. When compared to the occupational dose limit for adults in 10 CFR 20.1201(a)(1)(i) of 5 rem/yr TEDE, the NRC staff finds that the values observed in the licensee's ARs are much less than the regulatory dose limit. In addition, the NRC staff confirms that the visitor doses were on average less than 1 mrem with the highest TEDE being 8 mrem in this time period which is much less than the 100 mrem/yr TEDE dose limit for members of the public in 10 CFR 20.1301(a)(1).

The NRC staff finds that radiation and contamination surveys are performed on a regular basis by the health physics staff at the UCD/MNRC facility, which provides adequate oversight of areas where work with radioactive materials is performed. The NRC staff review showed that the placement, use, and control of the radiation monitoring and surveying equipment are in accordance with applicable standards, guidance, and regulations. The NRC staff review also verified that the selection of equipment used is appropriate for detecting the types and intensities of radiation likely to be encountered within the facility at appropriate frequencies to help ensure compliance with 10 CFR Part 20 requirements and the facility ALARA program under all operating conditions. The NRC staff also reviewed NRC IRs for CY 2009 through 2021 and finds that the licensee performed radiation and containment surveys adequately.

Based on its review of the information provided above, the NRC staff concludes that the licensee's equipment for detecting the types and intensities of radiation likely to be encountered within the facility and the surveillance frequencies are appropriate to help ensure compliance with 10 CFR 20.1501, paragraphs (a) and (b) and the facility ALARA program.

3.1.5 Radiation Exposure Control and Dosimetry

SAR section 11.1.6, "Radiation Exposure Control and Dosimetry," states that radiation exposure control depends on many different factors including facility design features, operating procedures, training, proper equipment, etc. The UCD/MNRC facility includes several design features used to limit radiation exposures to workers and the public. In SAR section 11.1.6, the licensee also provides details regarding its design features of shielding, ventilation, containment and entry controls for high radiation areas, protective equipment, personnel dosimetry, and estimates for annual radiation exposures at various locations within the plant.

<u>Shielding</u>

SAR section 11.1.6.1, "Shielding," states that shielding available at the plant includes the concrete and water surrounding the reactor core (reactor bulk shield), four radiography bays, the demineralizer resin cubicle area, the CAM room, and the second-floor area near the hand and foot monitors. The shielding available at the radiological bays consists of shutter biological shields, the beam stops, and the walls and roof of each bay. The licensee describes the reactor bulk shield as being like other TRIGA reactor designs. The shield consists of 20 ft (6.1 m) of water above the core. Radially, personnel is protected by 3.5 ft (1.1 m) of water and the graphite reflector shield as well as 7 ft (2.1 m) of concrete. The stated dose rates during operations is 1 mrem/hr. The neutron beam shutter, biological shield, and radiography bay interior walls were designed to reduce radiation levels to less than 5 mrem/hr in the areas that are routinely used when the reactor is in operation. The less than 5 mrem/hr radiation levels were measured and confirmed during actual operations.

Ventilation System

SAR section 11.1.6.2, “Ventilation System,” describes the design features used to maintain Ar-41 and N-16 concentrations in the reactor room below the limits of 10 CFR Part 20. One design feature for the ventilation includes maintaining differential air pressure in the reactor room, the equipment room, and the sample preparation rooms with respect to their surrounding areas. This is consistent with ALARA principles in ensuring the spread of contamination is not spread from contaminated areas to less contaminated areas. The reactor room contains a high efficiency filter to remove radioactive particles. In addition, the reactor room exhaust can also recirculate through the high efficiency particulate absorption (HEPA) and charcoal filters if the reactor room CAM exceeds the preset limits. When the exhaust is recirculated, no reactor room air is exhausted through the facility stack.

Entry Control

SAR section 11.1.6.4, “Entry Control – Radiography Bays and Demineralizer Cubicle,” describes the five main areas within the UCD/MNRC facility, which require entry control in order to ensure that the 10 CFR Part 20 requirements for limiting access into high radiation areas are maintained. Specifically, these areas are the four radiography bays and the small cubicle containing the demineralizer resins. Access into the radiography bays is controlled by a system of interlocks and warning devices incorporated into the facility design and described in SAR section 9.6, “Interlocks/Controls - Bay Shutters/Doors.” These interlocks prevent the radiography bay doors from opening when the beam shutter is open, and the reactor is operating. The reactor also SCRAMs if both the beam shutter and the bay door are open. An audible alarm is activated, and a red flashing light will indicate when the shutter starts to open. The licensee also has an interlock system requiring keys being used for power to be delivered to the bay door drive mechanism to open.

Regarding the demineralizer cubical, the licensee states that this is a locked area during operations. Given that this area will contain the buildup of radioactive materials cleaned from the primary coolant, this is an area of focus for controlling doses. The licensee provides shielding around this cubical and provides access control by using a locked barrier at the point of entry into this area. Entry procedures incorporate 10 CFR Part 20 access requirements.

Protective Equipment

SAR section 11.1.6.5, “Protective Equipment,” describes the personnel dosimetry devices used at the MNRC facility. Respiratory protection and personnel dosimetry are handled by the UCD/MNRC radiation protection program. The radiation protection program does not anticipate the use of respiratory protection equipment given that the only airborne concerns at the facility are Ar-41 and N-16. Given the ventilation systems ability to maintain low concentrations of these radionuclides, the NRC staff agrees that respiratory equipment is not a concern.

SAR table 11-10, “Summary of Typical Protective Equipment Used in the MNRC Radiation Protection Program,” describes the protective equipment used in the MNRC Radiation Protection Program. Personnel dosimetry used at the UCD/MNRC facility includes OSL monitoring badges, OSL finger rings, and personnel fast neutron monitoring. The licensee describes the annual occupational whole-body exposure for 2 MWt operations as being 217 mrem and the annual extremity and eye dose being 181 mrem and 195 mrem, respectively. Given that this renewal will have operations at 1.0 MWt, the licensee's estimates for the whole-body, extremity and eye doses are 25 mrem, 50 mrem, and 50 mrem, respectively. The licensee provides the results of the doses in its ARs. The NRC staff review finds the doses are maintained well below the limits in 10 CFR 20.1201.

The regulations, 10 CFR 20.1502, "Conditions requiring individual monitoring of external and internal occupational dose," require monitoring of workers likely to receive, in one year from sources external to the body, a dose in excess of 10 percent of the specified limits. Licensee procedures require monitoring of individuals entering the facility. In addition, electronic dosimetry is given to all individuals who enter radiation areas. The UCD/MNRC facility uses portable equipment to perform radiation surveys. Personnel protective equipment is used as needed. Facilities and equipment to decontaminate persons are available if needed. Procedures exist that govern the use of this equipment.

The licensee states that it uses survey meters to measure dose rates from radiation fields, and these measured rates are posted where required. 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. Limited access to radiography bays while neutron shutters and the gamma shields are open also help to keep doses lower. The licensee states that it also maintains personnel exposure records and effluent and environmental monitoring readings for the life of the UCD/MNRC facility.

The NRC staff performed its review of the licensee's Radiation Exposure Control and Dosimetry using the guidance in NUREG-1537, section 11.1.5, "Radiation Exposure Control and Dosimetry," and finds that the facility design includes shielding and ventilation systems that help ensure the uncontrolled release of radioactivity to the environment or work areas is mitigated. Further, the entry control system interlocks should prevent unauthorized entry into high radiation areas by the UCD/MNRC workers. The NRC staff finds that the shielding help reduce unnecessary radiation exposure in accordance with ALARA principles. The NRC staff finds that the personnel dosimetry is typical of a research reactor and provides effective radiation dose monitoring for the UCD/MNRC staff and workers.

The NRC staff reviewed the licensee's ARs and NRC IRs from CY 2009 through CY 2020, and finds that the UCD/MNRC staff annual doses consistent with the facility's ALARA program. The NRC staff finds that all UCD/MNRC staff received significantly less radiation dose than the 10 CFR 20.1201 limits. The NRC staff finds that the radiation doses to the UCD/MNRC staff, and the application of the equipment and procedures used to be acceptable. The personnel exposures at UCD/MNRC facility are controlled through satisfactory radiation protection and ALARA programs. Based on its review of the information provided above, the NRC staff concludes that the licensee's exposure control and dosimetry programs are acceptable.

3.1.6 Contamination Control

SAR section 11.1.7, "Contamination Control," states that radioactive contamination is controlled using written procedures for radioactive material handling, by using trained personnel, and by using a monitoring program designed to detect contamination in a timely manner. The licensee identifies the two known contaminated area as the reactor tank and the pneumatic transfer system hood. For work in areas where contamination is likely, the licensee has procedures established to maintain control over the contamination. The licensee states this document to be the UCD/MNRC Health Physics Procedures, UCD/MNRC-0029-DOC. Handling of any radioactive material within the UCD/MNRC facility is controlled by this written procedure. Workers are trained in working with radioactive materials, including how to limit its spread when entering and exiting an area containing radioactive material. The facility surveys have routinely shown no detectable contamination in non-radiological areas of the facility.

The NRC staff reviewed the licensee's contamination control using the guidance in NUREG-1537, section 11.1.6, "Contamination Control," and finds that the licensee identified and understands the potential problems caused by radioactive contamination and has procedures to control contamination. The NRC staff reviewed the licensee's ARs and NRC IRs from CY 2009 through CY 2020 and finds that the UCD/MNRC staff handled contamination in an acceptable manner, and no violations were identified. The NRC staff finds that adequate controls exist to prevent the spread of radiological contamination within the facility given the results recorded show the average and highest contamination levels are all below the lower limit of detection for surveys. Based on its review of the UCD/MNRC radiation protection program and on a history of satisfactory contamination control, the NRC staff concludes that adequate controls exist to prevent the spread of contamination within the UCD/MNRC facility.

3.1.7 Environmental Monitoring

SAR section 11.1.8, "Environmental Monitoring," describes the environmental radiation monitoring program at the UCD/MNRC. The UCD/MNRC staff established an environmental radiation program that is conducted to measure the integrated radiation exposure in and around the environment surrounding the facility since 1988. The licensee states that there will be no liquid releases from the plant into the environment or the sewer system. The only airborne radionuclide with potential to be released to the environment is Ar-41. As stated in SAR section 11.1.6, the licensee uses CAMs to detect radioactivity in excess of release limits and recirculates the reactor bay air through HEPA and charcoal filters to prevent excess releases to the environment.

SAR table 11-12, "Environmental Monitoring and Sampling Program," describe the locations of the OSL dosimetry set up at 37 onsite locations and 7 offsite locations. The dosimeters are exchanged quarterly. Water samples are also obtained by the licensee to ensure no releases into water pathways. The water samples are analyzed for gross alpha, beta, and tritium. Soil and vegetation samples are analyzed for gross beta and undergo gamma spectroscopy. These samples and OSL dosimetry are analyzed by offsite contractors. SAR table 11-12 also provides a list of sampling locations.

On an annual basis, the NRC staff has performed inspections of the UCD/MNRC environmental monitoring program. The NRC staff reviewed UCD/MNRC ARs from CY 2009 through CY 2020 (Ref. 24) and NRC IRs from CY 2009 through CY 2021 (Ref. 25). Based on its review, the NRC staff find that releases of radioactive material satisfied regulatory requirements and that the releases were within the limits specified in the NRC regulations.

The environmental monitoring program is reviewed and audited by the licensee's NSC on an annual basis as part of its review and audit of the radiation protection program as required by proposed TS 6.2.4. The NSC review helps ensure that the environmental monitoring program contains an adequate number of locations and sufficient frequency of collection such that the analysis of the data has sufficient sensitivity to ensure that the overall program complies with 10 CFR 20.1301, and will provide an early indication of any environmental impact caused by the reactor facility operation.

The administrative requirements for monitoring and reporting radioactive releases to the environment are provided in proposed TS 6.7.1, “Annual Operating Reports,” items 5 through 7 (discussed in SER section 5.6.7.1, which was evaluated by the NRC staff and found acceptable).

The NRC staff finds that the environmental reporting criteria are effective to understand the radioactivity released from the facility and are consistent with the guidance in NUREG-1537. Based on its review of the information provided above, the NRC staff concludes that the environmental monitoring program is sufficient to assess the radiological impact of the operation of the UCD/MNRC on the environment and is acceptable.

3.2 Radioactive Waste Management

SAR section 11.2, “Radioactive Waste Management,” describes the licensee’s radioactive waste management program. The licensee states that the objective of the radioactive waste (radwaste) management program is to minimize radioactive waste and ensure its proper handling, storage, and disposal. The UCD/MNRC facility primarily generates radwaste in solid and gaseous form. The licensee states that operation of the facility results in the generation of very little liquid waste, which it then converts into solid waste for disposal.

3.2.1 Radioactive Waste Management Program

SAR section 11.2.1, “Radioactive Waste Management Program,” states that all radioactive waste handling operations are controlled by procedure and overseen by the Health Physics branch at the UCD/MNRC facility. All radwaste handling operations are controlled by procedures to help ensure compliance with the requirements of 10 CFR Part 20 and other appropriate NRC regulations. The radioactive waste management program is audited by the Nuclear Safety Committee (NSC). Waste management training is part of both the initial radiation protection training and the specialized training. Radioactive waste management records are maintained by the Health Physics Branch. Radioactive waste packages in storage are tracked by a computer based radioactive material accountability system until shipment for disposal or transfer to an authorized broker. Radioactive material shipment and transfer records are also maintained by the Health Physics Branch. All records are retained for the life of the facility.

The NRC staff reviewed the information provided in SAR section 11.2.1 that describes the radioactive waste management program. The NRC staff finds the radioactive waste management program is controlled by the use of facility procedures, as stated in SAR section 11.2.1. Radioactive waste management program is audited by the NSC, and workers are trained on radiative waste management during initial training and specialized training. The NRC staff finds that the licensee maintains records of the radioactive waste. Based on its review of the information above, the NRC staff finds that the licensee’s radioactive waste management program demonstrates reasonable assurance that radiological releases from the facility will not exceed applicable regulatory limits nor pose unacceptable radiation risk to the environment and the public. The NRC staff also finds that the licensee has adequate controls in place to prevent uncontrolled personnel exposures from radioactive waste operations and to provide the necessary accountability to prevent any potential unauthorized release of radioactive waste. On this basis, NRC staff concludes that the licensee's radioactive waste management program is acceptable.

3.2.2 Radioactive Waste Control

SAR section 11.2.2, "Radioactive Waste Controls," states that radioactive waste is considered to be any material that is no longer used, and which contains radioactivity above the background radioactivity. When possible, radioactive waste is initially segregated at the point of origin from items that will not be considered waste. Screening is based on the presence of detectable radioactivity using appropriate monitoring and detection techniques and on the projected future need for the items and materials involved. All items and materials initially categorized as radioactive waste are monitored a second time before packaging for disposal to confirm data needed for waste records, and to provide a final opportunity for decontamination/reclamation of an item, which helps reduce the volume of radioactive waste.

SAR section 11.2.2.1," Gaseous Waste," states that gaseous waste generated at the facility is considered as a normal effluent release during reactor operation. The primary radionuclide released is Ar-41, which is monitored by the plant stack for effluent releases (evaluated in SER section 3.1.1 and found acceptable).

SAR section 11.2.2.2, "Liquid Waste," states that it is the licensee's policy to minimize the release of radioactive liquid waste. Since MNRC operations create only small volumes of liquid which contain radioactivity, the licensee converts the liquids to a solid waste form for disposal. In special cases, the MNRC may generate a large volume of radioactive liquid waste which cannot be converted to a solid waste. In these cases, disposal by the sanitary sewer in accordance with 10 CFR 20 may be required.

SAR section 11.2.2.3, "Solid Waste," states that solid waste is generated from reactor maintenance operations and irradiations of various experiments. No solid radioactive waste is intended to be retained or permanently stored on site. Most solid waste generated at the facility is in the form of demineralizer waste from maintenance activities and various irradiation experiments. The licensee indicates that solid waste shipments occur about every 5 years.

Based on its review of the information provided above, the NRC staff finds that the UCD/MNRC facility has adequate radioactive waste controls in place to minimize the amount of radioactive waste generated, limit any liquid radioactive effluent released from the facility, and to properly prepare solid radioactive waste for transfer to an offsite disposal facility.

3.2.3 Release of Radioactive Waste

SAR section 11.2.3, "Release of Radioactive Waste," provides information on the licensee's release of radioactive waste. SAR section 11.2.2.2 indicates that normal operation of the UCD/MNRC facility does not produce significant liquid radioactive waste. Small quantities of liquid waste are periodically generated by maintenance operations but converted to solid waste for disposal. Occasionally, a large volume of liquid waste may be generated, (e.g., heat exchanger cleaning) and disposal by sanitary sewer in accordance with the limitations in 10 CFR Part 20 is the only viable disposal option.

SAR section 11.2.3 describes that the primary release observed at the UCD/MNRC facility is the release of gaseous Ar-41 from the ventilation exhaust. As previously mentioned, the licensee's ventilation system will recirculate the air through the ventilation system if the reactor room CAMs alarm. This will divert air through the HEPA and charcoal filters once again and avoid a release above 10 CFR Part 20 release limits. As stated in SAR section 11.2.2.3, solid waste generated by this facility is stored and shipped every 5 years.

The NRC staff finds that the annual reporting requirement related to radioactive releases is consistent with NUREG-1537 (Ref. 11) and ANSI/ANS-15.1-2007 (Ref. 55). The reporting requirement is found within proposed TS 6.7, "Reports," (discussed in SER section 5.6.7, which was evaluated by the NRC staff and found acceptable), for reporting of wastes which includes the amount, activity, and the dates of shipment and disposition. Review of the licensee's ARs from CY 2009 through CY 2020 indicates that radioactive materials released are recorded and tracked.

Based on its review of the information provided above, the NRC staff finds that the licensee's controls and techniques for release of radioactive waste are acceptable. Furthermore, the NRC staff finds that the UCD/MNRC facility has adequate controls in place to minimize releases of radioactive material into the environment.

3.3 <u>Conclusions</u>

Based on its review of the information presented in the SAR, documented NRC inspector observations of the licensee's operations, licensee's ARs, and the results of the NRC inspection program, the NRC staff concludes the following:

- The UCD/MNRC radiation protection program complies with the requirements in 10 CFR 20.1101(a), is acceptably implemented, and provides reasonable assurance that the UCD/MNRC staff, the public, and the environment are protected from unacceptable radiation exposures. The radiation protection staff has adequate lines of authority and communication to implement the program.

- The licensee's ALARA program complies with the requirements of 10 CFR 20.1101(b). Review of controls for radioactive material at the UCD/MNRC provides reasonable assurance that radiation doses to the UCD/MNRC staff, the public, and the environment will be ALARA.

- The results of radiation surveys carried out at the UCD/MNRC facility, doses to the persons issued dosimetry, and the results of the environmental monitoring program confirm that the radiation protection and ALARA programs are effective, and in compliance with the requirements of 10 CFR 20.1501(a).

- Potential radiation sources have been adequately identified and described by the licensee. The licensee controls radiation sources under a radiation protection program that meets the requirements of 10 CFR Part 20 and the guidance in ANSI/ANS-15.11-2016, “Radiation Protection at Research Reactor Facilities.”

- Facility design and procedures provide adequate control of the potential exposures from N-16 and Ar-41 to the UCD/MNRC staff, the public and the environment. Review of licensee ARs as well as the NRC staff’s evaluation results confirm that the quantities of these gases released into restricted and unrestricted areas provide reasonable assurance that doses to UCD/MNRC staff and the 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 neither exceed applicable regulations nor pose unacceptable radiation risk to the public and the environment.

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

4 Accident Analyses

4.1 Accident Analyses

The accident analysis presented in safety analysis report (SAR) Chapter 13, "Accident Analysis," as supplemented, helps to establish safety limits (SLs) and limiting safety system settings (LSSS) that are imposed on the University of California-Davis/McClellan Nuclear Research Center (UCD/MNRC) reactor through implementation of the technical specifications (TSs). The SAR provides the licensee's analyzed potential reactor transients and other hypothetical accidents. The licensee's SAR also analyzed the consequences of a maximum hypothetical accident (MHA) that could lead to the maximum potential radiation hazard to facility staff and members of the public. The radiological consequences of the licensee's MHA doses to members of the public bounds all credible accident scenarios, as well as the potential effects of natural hazards involving the operation of the UCD/MNRC reactor. The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the licensee's analytical assumptions, methods, and results. In addition, the NRC staff performed confirmatory calculations and reviewed accident analyses for other Training, Research, Isotope, General Atomics (TRIGA) reactors and compared those results with accidents analyzed by the licensee.

As discussed below, none of the potential accidents considered in the SAR would lead to significant occupational or public radiation exposure.

4.2 Accident-Initiating Events and Scenarios

The NRC staff has previously prepared an independent analysis of credible accidents for TRIGA reactors. This study was documented in NUREG/CR-2387, "Credible Accident Analyses for TRIGA and TRIGA-Fueled Reactors" (Ref. 41). The NRC staff used applicable information from NUREG/CR-2387 as a basis for evaluating information presented in SAR Chapter 13. In addition, NUREG-1537, Part 2, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Standard Review Plan and Acceptance Criteria" (Ref. 11), identifies nine potential credible accidents for research reactors as follows:

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

4.2.1 Maximum Hypothetical Accident

SAR chapter 13, as supplemented, evaluates the category of accidents listed in NUREG-1537, and describes the accident initiating events and scenarios for the applicable and credible accidents for the UCD/MNRC TRIGA reactor. SAR section 13.2.1, "Maximum Hypothetical Accident," and supplemental information provided by letter dated September 22, 2021, “Appendix B, Radiological Impact of Accidents” (Ref. 56), and “New MNRC MHA Source Term” (Ref. 57), provided the analysis of the MHA. The MHA for TRIGA reactors, including the UCD/MNRC TRIGA reactor, is a cladding failure of a single irradiated element in air in the reactor room, assuming no radioactive decay of contained fission products as described in NUREG/CR-2387 (Ref. 41).

Maximum Hypothetical Accident Scenario

The MHA scenario, described in the SAR, as supplemented, assumes a cladding rupture of one highly irradiated fuel element with no decay followed by instantaneous release of fission products into the air for a bounding analysis of the radiological dose consequence to the UCD/MNRC reactor staff and members of the public. The MHA source term is the fission product inventory for the single irradiated fuel element, operated with the highest core power density for a continuous period of 1 year at 1.0 megawatt thermal (MWt). In SAR appendix, “Radiological Impact of Accidents,” B.1, “Maximum Hypothetical Accident,” the licensee states that this operating history is very conservative in nature as it is unlikely the facility is ever operated more than 2,000 MWt-hrs in a year for the remainder of the facility’s life.

Table 4.1 Source Term for Single Fuel Element MHA in Air

Nuclide	Activity (Ci)	Nuclide	Activity (Ci)
Bromide (Br)-83	113.4	Krypton (Kr)-83m	113.4
Br-84	208.4	Kr-85m	272.9
Br-84m	35.5	Kr-85	3.7
Br-85	272.9	Kr-87	541.6
Iodine (I)-131	611.4	Kr-88	751.0
I-132	911.8	Kr-89	954.1
I-133	1,417.4	Xenon (Xe)-131m	8.6
I-134	1,656.5	Xe-133m	40.0
I-134m	77.0	Xe-133	1,417.4
I-135	1,328.6	Xe-135m	232.7
		Xe-135	1,383.6
		Xe-137	1,296.8
		Xe-138	1,332.8
Total (Ci)	6,632.7		8,348.6

In appendix B of the SAR, as supplemented, the licensee provided an updated fission product source term, which is calculated based on the highest fission rate fuel element in the limiting core configuration (LCC) of 17.69 kilowatt (kW) (Ref. 57). This element would theoretically operate at the highest temperature and have the highest fission inventory. Fission product yield and half-life data were taken from published International Atomic Energy Agency data. Very short half-lived isotopes (i.e., less than 1 minute) were excluded from the source term calculation due to the unavailability of some dose factors and their low expected contribution to radiological consequences. To help compensate for this underestimation a 25 percent overestimation is made in all other fission product inventories. The source term for the single fuel element MHA in air is shown in SER table 4.1, “Source Term for Single Fuel Element MHA in Air” (reproduced from table B-1 Source Terms for One-Element Accident for MHA, Ref. 56).

Source Term 24 Hours After Shutdown

In SAR section 13.2.1.1, the licensee indicates that a realistic scenario for the MHA is difficult to establish since fuel handling, the activity frequently associated with this accident, would be unlikely to occur immediately after reactor shutdown, and fuel elements would not be moved out of the reactor tank into air with no time to decay. The licensee considers the single fuel element failure in the reactor pool water 24 hours after shutdown a more realistic accident than the MHA in air, which occurs immediately after reactor shutdown. Therefore, the licensee also evaluated a less severe accident involving a single fuel element cladding failure in the reactor pool water following operation at 1.0 MWt, but not initiating until 24 hours after reactor shut down in SAR appendix B (Ref. 56). The licensee indicates that this evaluation was provided to gain insights into the radiological consequences of a damaged a fuel element during handling operations in the reactor tank (i.e., during fuel handling). The source term for the single fuel element cladding failure accident in the reactor pool water 24 hours after reactor shutdown is shown in SER table 4.2, “Source Term for Single Fuel Element Cladding Failure Accident in the Reactor Pool Water 24 Hours After Reactor Shutdown,” below (reproduced from table B-2 Source Terms for One-Element Accident for 24-hour decay, Ref. 56).

Table 4.2 Source Term for Single Fuel Element Cladding Failure Accident in the Reactor Pool Water 24 Hours After Reactor Shutdown

Nuclide	Activity (Ci)	Nuclide	Activity (Ci)
Br-83	0.11	Kr-83m	0.015
Br-84	0	Kr-85m	6.6
Br-84m	0	Kr-85	3.7
Br-85	0	Kr-87	0.001
I-131	561	Kr-88	2.15
I-132	0.62	Kr-89	0
I-133	637	Xe-131m	8.1
I-134	0	Xe-133m	29.1
I-134m	0	Xe-133	1,242
I-135	105.6	Xe-135m	0
		Xe-135	222
		Xe-137	0
		Xe-138	0
Total (Ci)	1,304.3		1,513.7

The NRC staff reviewed the UCD/MNRC reactor MHA source term and performed confirmatory calculations on the source terms for the single fuel element MHA in air and the single fuel element cladding failure accident in the reactor pool water 24 hours after reactor shutdown, as shown in SER tables 4.1 and 4.2, respectively. The NRC staff confirmatory calculations were consistent with the licensee’s accident source terms. Based on the NRC staff’s review and confirmatory calculations described above, the NRC staff finds both accident source terms acceptable.

Release Fraction

The licensee calculated the release of noble gases and halogens from the fuel element matrix uranium-zirconium hydride (U-ZrH) to fuel element gap using the General Atomic (GA) developed correlation for fission product release fraction (RF) (Ref. 58). This correlation estimates the fission product RF for TRIGA low-enriched uranium (LEU) fuels based on the fuel temperature using Equation 16, in SAR chapter 4, section 4.6.4.5, “Fission Product Release Fraction.” For both accidents, a RF of 2.36×10^{-5} is used for the release of noble gases and halogens from the fuel to the cladding gap. This release fraction is based on the maximum measured fuel temperature 400 degrees Celsius (°C) (752 degrees Fahrenheit (°F)) that corresponds to the average fuel temperature of the highest thermal output fuel element of the LCC core. It is assumed that 100 percent of the noble gases reach inside the reactor building and the unrestricted environment outside the reactor building for calculating doses to workers and members of the public, respectively. The NRC staff’s confirmatory calculation resulted in a similar RF of 2.31×10^{-5} using the GA correlation and maximum measured fuel temperature of 400 °C (752 °F). Therefore, the NRC staff finds the RF of 2.36×10^{-5} for both accidents acceptable.

Atmospheric Dispersion

As described in SAR section 13.2.1.2, “Accident Analysis and Determination of Consequences,” three downwind receptor distances from the reactor were selected by the licensee to calculate the dose consequence for the MHA: nearest distance to a member of the public is 92 feet (ft) (28 meters (m)) to the UCD/NMRC fence; the closest building is an industrial X-ray facility utilized by the Air Force at 184 ft (56 m) to the south of the UCD/MNRC that has been uninhabited for nearly 20 years; and the closest habited building is a regularly used large conference center at 308 ft (94 m) to the north east of the UCD/MNRC. The Gaussian plume atmospheric dispersion model contained in the Department of Energy (DOE) Lawrence Livermore National Laboratory HotSpot Version 3.1.1 computer code (Ref. 59) is used by the licensee to calculate doses at the downwind receptor distances on the plume centerline.

Air Handling System

As described in SAR section 9.5, “Air Handling System,” the design basis of the UCD/MNRC air handling system is to isolate, recirculate, and filter reactor room air should there be a release of fission products or other abnormal airborne radionuclides. Ventilation throughout the reactor facility is designed and balanced so that the reactor room and radiography bays are at a slight negative pressure with respect to their surrounding areas. Figure 9.11, “UCD/MNRC Air Handling System,” of the SAR illustrates the two operation modes (normal and recirculation) for the reactor room ventilation system.

As described in SAR section 9.5.2, “Description,” during normal reactor operation, reactor room supply air is provided by a combination heating and cooling air conditioning unit (AC-1), and reactor room air is filtered with a pre-filter and high efficiency particulate (HEPA) filter before being discharged to the exhaust stack by an exhaust fan (EF-1) that maintains the negative pressure. Reactor room air is monitored for radioactive airborne particulate, noble gas, and iodine by the reactor room continuous air monitor (CAM) that samples the exhaust effluent prior to filtration. Upon detection of airborne radioactivity above a preset level, the reactor room ventilation system automatically isolates the reactor room and enters recirculation mode. In the recirculation mode, reactor room air is drawn through a separate filtration system using the EF-1 before being returned to the reactor room until the reactor room CAM alarm is reset. This separate filtration system includes a filter bank that contains a dehumidifier, pre-filter, HEPA filter, and two carbon filters. The normally opened damper in the AC-1 makeup duct is closed and AC-1 is shut down, and AC-2 is prevented from being shut down to maintain slightly positive adjacent reactor equipment room air pressure with respect to the isolated reactor room achieved by seven dampers and two air conditioning unit interlocks that operate automatically. To return the reactor room ventilation system to normal operation, the reactor room CAM alarm must be cleared and must be reset via the CAM RESET button on the temperature control panel (TCP) in the reactor control room. The CAM RESET button restores the interlocked dampers to their normal configuration, allowing filtered reactor room exhaust air to discharge to the atmosphere through the exhaust stack. The CAM RESET also restarts AC-1 and enables AC-2 to be controlled at the TCP.

As described in SAR section 9.5.2, during a LOCA the radiation levels in the reactor room could cause the reactor CAM to alarm and force the reactor room ventilation system into recirculation mode. In this situation, reactor room ventilation remains in the normal operating mode during a LOCA. A ventilation damper control switch located on the TCP in the reactor control room enables the reactor operator to override the damper controls for recirculation and continue exhausting air from the reactor room through the normal exhaust path.

SAR section 13.2.1.2, states that the licensee calculated the MHA occupational dose assuming an instantaneous release of the radioactive source term into the reactor room and not being removed by the ventilation system. The licensee indicated that this was done to keep the radiological consequences conservative (i.e., provide an overestimation of the doses).

The NRC staff finds, as part of its evaluation of the dose consequences from the single fuel element MHA in air and the single fuel element cladding failure accident in the reactor pool water 24 hours after reactor shutdown, that the licensee did not credit the mitigation capabilities available by reactor room ventilation system, which would have reduced the calculated doses to workers by recirculating the reactor room air and would have reduced the calculated doses to members of the public from filtered exhaust air to the environment in the event of a fission product release. Because the mitigation capabilities of the reactor room ventilation system were not credited in the MHA dose calculation by the licensee, the NRC staff finds that the dose calculations that follow represent conservative estimates and are acceptable for bounding all other potential accident doses.

Doses to Members of the Public

The licensee assumed that all of the MHA fission products were released to the unrestricted area instantaneously, which would maximize the dose rate to persons exposed to the plume during the accident and minimize the exposure time to receive the highest estimated dose from this accident. The licensee also assumed that the release height of the radioactive material to be the height of the floor of the reactor room (19 feet above ground level) to keep the radiological results as conservative as possible. Additionally, a very calm wind speed of 1 m/s was selected as lower wind speeds in these types of releases produce greater radiological consequences.

The licensee states that offsite doses to the public due to the postulated MHA are evaluated using the HotSpot code. In the HotSpot code model, the licensee applied the MHA source term with gap release activity reproduced in table 4.1 of this SER, using all atmospheric stability classes (A, B, C, D, E, and F) along the plume centerline, a wind speed of 3.3 ft/second (s) (1 m/s), a receptor height of 3.3 ft (1.0 m), an effective release height of 18.7 ft (5.70 m) above ground level (conservative compared to the stack height of 60 ft [18.3 m]), a receptor height of 3.3 ft (1.0 m), a sample time of 10 minutes (default value in HotSpot code), the dose factors in U.S. Environmental Protection Agency (EPA) Federal Guidance Report No. 11 (Ref. 60) and No. 12 (Ref. 61), and a breathing rate of 1.059×10^{-2} ft^3/s (3.33×10^{-4} m^3/s). Atmospheric modeling values used in the HotSpot code calculation are shown in table B-4, “Radiation Doses to Members of the General Public Under Different Atmospheric Conditions and at Different Distances from the MNRC Following a Fuel Element Cladding Failure in Air (MHA),” in SAR appendix B.

The NRC staff performed confirmatory calculations using the source term in SER table 4-1, with gap release activities for a single fuel element MHA, and the licensee’s HotSpot code modeling parameter values and assumptions described above, except for an effective release height of 19 ft (5.79 m) and receptor height of 4.9 ft (1.5 m) (default value in HotSpot code). The licensee’s and NRC staff’s results for calculated offsite doses to members of the public at downwind receptor distances from the reactor are shown in SER table 4.3, “MHA Doses to Members of the Public.” Included in table 4.3 is the licensee’s and NRC staff’s calculated maximum doses and distances to a member of the public at 19 m (62 ft). The NRC staff notes that the 19 m (62 ft) location is within the licensee’s protected area fence, and any members of the public would be evacuated in accordance with the licensee’s emergency plan, and thus, unlikely to receive the postulated dose at that distance.

Table 4.3 MHA Doses to Members of the Public

Location	**33 meters (UCD/MNRC Fence Line)**		**93 meters (Closest Inhabited Building)**		**480 meters (Closest Residence)**		**Maximum Dose 19 meters**	
Dose (mrem)	Licensee	NRC	Licensee	NRC	Licensee	NRC	Licensee	NRC
TEDE	0.56	0.75	0.46	0.61	0.18	0.24	0.69	0.92
10 CFR 20.1301 Limit - 100 mrem								

The NRC staff finds that the licensee’s methodology for calculating the MHA doses to members of the public to be consistent with industry practices, including the release fraction, the HotSpot computer code, and the modeling parameter values and assumptions used by the licensee. The NRC staff’s confirmatory calculations for the doses for the MHA provided in SER table 4.3 are consistent when compared to the licensee’s calculations and results. The NRC staff finds that the doses to any member of the public from the MHA are below the limits provided in Title 10 of the *Code of Federal Regulations* (10 CFR) 20.1301, “Dose limits for individual members of the public,” of 100 millirem (mrem), and therefore, are acceptable.

Occupational Doses

The licensee evaluated occupational doses to workers inside the UCD/MNRC building using the MHA source term with gap release activities assuming 50 percent suppression of halogens (i.e., iodine and bromide isotopes) in the reactor pool water and 50 percent plate-out of halogens in the reactor building and reactor room volume of 7.50×10^3 ft^3 (212 m^3). The reactor room is conservatively assumed to be large enough to approximate submersion in a semi-infinite cloud from external radiation exposure.

The licensee compared reactor building radionuclide concentrations with respective derived air concentrations (DACs) in table 1, column 3 in appendix B of 10 CFR Part 20, “Standards for Protection Against Radiation,” to calculate the worker doses equivalent to 5 rem (5,000 mrem) total effective dose equivalent (TEDE) and 50 rem (50,000 mrem) committed dose equivalent (CDE) thyroid for times of 2 minutes and 5 minutes to evacuate the reactor room following the MHA. SER table 4.4, “Worker Doses for MHA” (with information taken from SAR table B-6 Worker dose for MHA, (Ref. 56)) provides the results of the licensee’s calculations for evacuation of the workers at 2 and 5 minutes. Evacuation occurs within 5 minutes which yields worker doses of 300 mrem TEDE and 2,495 mrem CDE thyroid. If evacuation occurs within 2 minutes, these doses reduce to 120 mrem TEDE and 998 mrem CDE thyroid considering a small reactor room and easy egress.

Table 4.4 Worker Doses for MHA

Occupancy Time (minutes)	**CDE Thyroid (mrem)**		**10 CFR 20.1201 Limit (mrem)**	**TEDE (mrem)**		**10 CFR 20.1201 Limit (mrem)**
	Licensee	NRC		Licensee	NRC	
2	998	968	50,000	120	116	5,000
5	2,495	2,420	50,000	300	289	5,000

SER table 4.5, “Worker Doses Following a Single Fuel Cladding Failure in the Reactor Pool Water 24 Hours After Reactor Shutdown” (with information taken from SAR table B-7 Radiation Dose Worker Following an Element Failure in Water with 24 hours of Decay (Ref. 56)), provides the workers’ doses for the more realistic accident scenario involving a single fuel element cladding failure at 1.0 MWt operation with 24 hours decay after reactor shut down, and water remaining in the reactor pool. The licensee calculated occupational doses to workers inside the reactor building given evacuation times of 2 and 5 minutes, respectively.

Table 4.5 Worker Doses Following a Single Fuel Cladding Failure in the Reactor Pool Water 24 Hours After Reactor Shutdown

Occupancy Time (minutes)	**CDE Thyroid (mrem)**		**10 CFR 20.1201 Limit (mrem)**	**TEDE (mrem)**		**10 CFR 20.1201 Limit (mrem)**
	Licensee	NRC		Licensee	NRC	
2	75.3	73.0	50,000	8.4	8.2	5,000
5	188	183	50,000	21.1	20.5	5,000

The NRC staff’s confirmatory calculations for both accident scenarios are also shown in tables 4.4 and 4.5. The NRC staff calculated the radionuclide concentrations from the respective accident source terms with gap release activities, reactor room volume, assumed percentages of suppression of halogens in reactor pool water and plate-out in the reactor building, and evacuation times.

The NRC staff finds that the licensee's methodology for calculating the MHA doses to the workers to be consistent with industry practices, including the release fraction, the HotSpot computer code, and the modeling parameter values and assumptions used by the licensee. The NRC staff's confirmatory calculations for the doses for the MHA provided in SER tables 4.4 and 4.5 are consistent compared to the licensee's calculations and results. The NRC staff finds that the doses to a worker, given evacuation times of 2 and 5 minutes, are below the limits provided in 10 CFR 20.1201, "Occupational dose limits for adults," of 50,000 mrem CDE Thyroid and 5,000 mrem TEDE. Based on the information above, the NRC staff finds the MHA doses to the workers to be acceptable.

The NRC staff finds, based on its review and confirmatory calculations described above, the dose consequence analyses from the single fuel element MHA in air and the single fuel element cladding failure accident 24 hours after reactor shutdown demonstrate that the maximum doses are conservative and well below the occupational dose limit in 10 CFR 20.1201 and well below the public dose limit in 10 CFR 20.1301. Therefore, the NRC staff concludes that the dose consequences from both accidents are acceptable.

4.2.2 Loss-of-Coolant Accident

The NRC staff reviewed the LOCA using the guidance in NUREG-1537, Part 2, chapter 13, "Accidents Analyses," "Loss of Coolant," which includes evaluating consequences of a loss of primary coolant, the resulting maximum peak fuel cladding temperature, and the potential for cladding damage and releasing fission products to the environment. The guidance in NUREG-1537, also directs the NRC staff to review the licensee's LOCA analysis for potential doses to reactor staff and members of the public, the ability of the operators to provide make-up water to recover the reactor core, the maximum potential dose rates in the unrestricted area, and if sufficient time exists for protective actions to ensure that no doses to any members of the public exceed the limits in 10 CFR Part 20.

SAR section 13.2.3, "Loss of Coolant Accident (LOCA)," states that a LOCA could be initiated through either pumping water from the reactor tank, or a failure of the reactor tank. These accident scenarios are summarized below.

<u>Pumping of Water from the Reactor Tank</u>

SAR section 13.2.3.2.1, "Pumping of Water from the Reactor Tank," states that the intake for the primary-cooling-system pump is located about 3 ft (1 m) below the normal tank water level. The line is perforated from about 8 inches (0.2 m) below the normal tank water level to the intake line entrance. The intake for the purification-system pump is through a short, flexible line attached to a skimmer that floats on the surface of the tank water. However, the length of the flexible line will cause a loss of pump suction if the tank water level is lowered about 4 ft (1.3 m). SAR section 13.2.3.2.1 also indicates that given the design features of the primary coolant pump and the purification pump, the reactor tank cannot be accidentally pumped dry by either the primary pump or the purification-system pump. In addition, it is not possible for other cooling system or water cleanup system components to fail and syphon water from the tank since all of the primary-water-system and purification-system piping and components are located above the normal tank water level.

The NRC staff reviewed the SAR description and finds that the likelihood of a LOCA caused by pumping water out of the reactor tank is not likely to occur nor is credible.

Reactor Tank Failure

SAR section 13.2.3.2.2, “Reactor Tank Failure,” states that there are no nozzles or penetrations in the reactor tank below the normal water level, so the only mechanisms that could cause tank failure are corrosion or a mechanical failure of the tank. The licensee has provisions to monitor and collect tank leakage in the facility design, described in SAR chapter 5, “Reactor Coolant Systems.” Small leaks are expected to be discovered prior to a significant drop in water level with makeup water supplied by the auxiliary makeup water system until the leak is repaired.

The licensee indicates that the facility is designed to withstand the earthquake intensity of the Uniform Building Code Zone 3. The tank is supported by the biological shield structure also designed to withstand this magnitude earthquake. The licensee also indicates that an earthquake of greater magnitude appears to be the only credible mechanism to cause a large tank rupture tank.

A LOCA with a complete draining of the reactor coolant water would expose the TRIGA fuel elements to air that would provide cooling. Previous calculations and experiments on similar TRIGA reactors have shown that air circulation would be adequate to prevent fuel damage by removing the decay heat of the fuel. The results of one of these studies (Ref. 67) found that operation with a TRIGA fuel element power of 19.7 kW, the corresponding peak LOCA fuel surface temperature would reach 585 °C (1,085 °F), which is well below the SL of 930 °C (1,706 °F) and would ensure that the cladding integrity is maintained. The maximum peak fuel element power in UCD/MNRC is 17.69 kW, which, being bounded by the previous study value of 19.7 kW, would result in a lower temperature. Early studies by GA, as discussed in NUREG/CR-2387 (Ref. 41) demonstrated that an instantaneous loss of all the water for a reactor operating continuously at 1.5 MWt produced a maximum temperature of 460 °C (860 °F) and that radiative loss of the core heat would be sufficient to ensure cladding integrity. While the peak fuel element power was not determined, the total reactor power of 1.5 MWt is significantly larger than the 1.0 MWt of the UCD/MNRC reactor and there was significant margin to the 930 °C (1,706 °F) temperature limit.

The licensee provides in UCD supplemental information, dated September 22, 2021, “UCD Analysis of Fuel Temperature after LOCA” (Ref. 62), a description of a LOCA that involves reactor coolant water draining from the reactor coolant tank through a drain valve (1.5 inches [3.81 cm] diameter) and emptying into Bay 1. For this smaller break LOCA (not the instantaneous loss of all reactor tank water), the licensee’s LOCA analysis assumes that the reactor had been operating indefinitely at 1.0 MWt, the hottest TRIGA fuel element operated at 17.69 kW, the reactor scram occurs when the reactor tank water level alarms, and the reactor core begins to be uncovered from water 60 minutes following the start of the draining. The licensee’s LOCA analysis calculated the TRIGA fuel element decay heat power to be 210.5 watts at the time the core begins to uncover.

The NRC staff’s confirmatory calculations confirmed that the core begins to uncover after approximately 60 minutes from the initiation of the leak. The NRC staff review also notes that previous studies by GA, as documented in GA-6596, “Simulated Loss-of-Coolant Accident for TRIGA Reactors,” dated August 18, 1965 (Ref. 68), demonstrate that air cooling is sufficient with a TRIGA fuel element power of 267 W, which bounds the UCD/MNRC power of 210.5 W. The peak fuel temperature measured in this study was 600 °C (1,112 °F), below the SL of 930 °C (1,706 °F). In addition, the NRC staff finds that the GA studies considered a partial loss-of-coolant simulation with the water level 1.5 inches (3.81 centimeters (cm)) above the bottom of the fuel elements at 270 W. The results showed that steam cooling was more effective than air cooling and resulted in fuel temperatures 60 °C (108 °F) cooler compared to the full loss-of-coolant case.

SAR section 13.2.3.2.2.1, “Air Cooling,” describes a unique aspect of the UCD/MNRC reactor facility is the relatively small reactor room and should a LOCA occur, without operation of the reactor room ventilation system, the reactor room air temperature will rise noticeably.

The NRC staff reviewed the licensee’s analysis (Ref. 62) and finds that it conservatively assumed all decay heat from the core was transferred to the reactor room air, the reactor room volume and surface area were included, and the analysis modeled the heat transfer through the concrete walls using a conservative thermal resistance value in order to maximize the air temperature. The results from the licensee’s analysis showed the air temperature available for cooling the fuel increased by 34 °C (61 °F) at its peak. The NRC staff performed confirmatory calculations of the air temperature increase and finds that the values agreed with the licensee’s analysis. The NRC staff finds that the licensee’s analysis of the air temperature increase to be accurate and, given the margin in the LOCA analysis to the SL of 930 °C (1,706 °F), the air temperature increase does affect the safety of the facility and is acceptable.

On the basis of its review of the licensee’s analysis, previous experimental studies as discussed above, and NRC staff confirmatory calculations, the NRC staff concludes that the licensee’s LOCA analyses would not result in damage to the reactor fuel or result in a release of radioactive effluents.

<u>Ground Water Contamination</u>

SAR section 13.2.3.2.2.2, “Ground Water Contamination,” and UCD supplemental information, dated September 22, 2021, “MNRC Soil Permeability Information” (Ref. 63), evaluates the postulated LOCA dose consequence from a break in the UCD/MNRC reactor tank. Primary coolant is assumed to leak from the reactor tank and migrate into the ground water. The reactor tank has no breaks in its structural integrity (i.e., there are no beam tube protrusions or other discontinuities in the reactor tank surface), and the reactor core is below ground level, so the potential for most types of leaks is minimized.

The licensee calculated the equilibrium concentration of activation products in primary coolant water at 1.0 MWt operation shown in SER table 4.6, “Predominate Radionuclides in Primary Coolant at Equilibrium and Upon Reaching Ground Water.”

Table 4.6 Radionuclides in Primary Coolant at Equilibrium and Upon Reaching Ground Water

Radionuclide	Half-Life	Typical Concentration at 1.0 MWt (µCi/ml)	Concentration After 36 Hours of Decay (µCi/ml)	10 CFR Part 20 Liquid Effluent Concentration Value (µCi/ml)
Aluminum-28	2.3 min	$3.0x10^{-3}$	0	-
Argon-41	1.8 hr	$1.0x10^{-3}$	$<1.0x\ 0^{-10}$	-
Hydrogen-3	12 yr	$2.0x10^{-2}$	$2.0x10^{-2}$	$1x10^{-3}$
Magnesium-27	9.46 min	$2.0x10^{-4}$	0	-
Molybdenum-99	2.75 d	$1.0x10^{-5}$	$6.9x10^{-6}$	$2x10^{-5}$
Cobalt-60	5.27 y	$1.0x10^{-6}$	$1.0x10^{-6}$	$3x10^{-6}$
Manganese-56	2.58 hr	$5x10^{-5}$	$3.1x10^{-9}$	$7x10^{-5}$
Sodium-24	14.96 hr	$6.0x10^{-4}$	$1.1x10^{-4}$	$5x10^{-5}$

SAR chapter 13, section 13.2.3.2.2.2, Equation 7, assumes a migration time greater than 36 hours for the lost primary coolant to move from a point under the reactor tank to ground water, hydraulic gradient of 1.0, hydraulic conductivity of $4.57x10^{-4}$ ft/s ($1.39x10^{-4}$ m/s), and ground water depth of 80 ft (24 m) below the UCD/MNRC reactor site. Primary coolant radionuclide concentrations are calculated assuming 7,000 gallons (3,182 liters) of water released from the reactor tank that will likely escape from the reactor radiography bays and migration time of 36 hours and compared to the liquid effluent concentration values in table 2, column 2 in appendix B of 10 CFR Part 20, that are equivalent to the radionuclide concentrations that, if ingested continuously over the course of a year, would produce a TEDE of 0.05 rem (50 mrem).

In addition, in the UCD supplemental information, dated September 22, 2021, "MNRC Soil Permeability Information (Ref. 63), the licensee states that the permeability of the soil on which the UCD/MNRC site is constructed is given as 2.0 inches/hr (5 cm/hr) or 1 ft/6 hrs (0.3 m/6 hrs) which results in a water migration time of 480 hours or 20 days compared to the determined migration time of 36 hours at the same water table depth of 80 ft (24 m).

Based on its review of information above, the NRC staff finds the migration time of 36 hours for use as the decay time for evaluating predominate radionuclides in primary coolant reaching the ground water from the postulated LOCA is acceptable.

During the first 36 hours, short-lived radionuclides such as aluminum-28, magnesium-27, and nitrogen-16 (gaseous) in primary coolant are completely decayed away by the time the reactor tank water reaches the ground water. After 36 hours, other radionuclides have undergone some degree of decay based on their half-lives with manganese-56, molybdenum-99, sodium-24 (after an additional 24 hours of decay), and cobalt-60 in primary coolant water determined to be less than their respective liquid effluent concentration values in table 2, column 2 in appendix B of 10 CFR Part 20. Argon-41 (gaseous) insoluble in water is also determined to be less than its air effluent concentration value in table 2, column 1 in appendix B of 10 CFR Part 20.

The NRC staff notes that only H-3 (tritium), with a half-life of 12.3 years remains in the primary coolant, is readily soluble in water and could present a potential exposure concern to members of the public. In SAR section 13.2.3.2.2.2, the licensee states that the equilibrium radionuclide concentration of tritium is 2.0×10^{-2} microcuries/milliliters (µCi/ml) for tritium and 1.0×10^{-6} µCi/ml for Co-60 operating approximately 1,200 MWt hours per year for nearly a decade. The NRC staff finds that a dilution factor of 20 reduces the tritium concentration of 2.0×10^{-2} µCi/ml, which meets the liquid effluent concentration limit in table 2, column 2 in appendix B of 10 CFR Part 20. Further, the NRC staff finds, that in the event of an accidental release of liquid radioactivity to the unrestricted environment, the tritium concentration would be diluted by a factor much greater than 20 when mixed with ground water at the closest drinking water well over one mile from the facility.

The NRC staff also performed confirmatory calculations of the equilibrium radionuclide concentrations in primary coolant upon reaching the ground water due to the postulated LOCA using the migration time, hydraulic gradient, hydraulic conductivity, and ground water depth, and compared them to the liquid effluent concentration values in table 2, column 2 in appendix B of 10 CFR Part 20. The NRC staff's confirmatory calculations, in general, agreed with licensee's values. The NRC staff finds, based on its review and confirmatory calculations described above, the equilibrium radionuclide concentrations in primary coolant upon reaching the ground water due to the postulated LOCA are below the limits in 10 CFR Part 20, and thus, are acceptable.

Radiation Levels from an Uncovered Core

SAR section 13.2.3.2.2.3, "Radiation Levels from the Uncovered Core," and UCD supplement, dated June 21, 2022, "Responses to NRC Request for Additional Information" (Ref. 72), provide estimated direct and scattered dose rates due to the postulated LOCA resulting in a loss of primary coolant water and uncovered core following one year of operation at 1.0 MWt. As part of the UCD supplement, the licensee provided an update to the UCD SAR "Appendix B, Radiological Impact of Accidents." The licensee evaluated dose rates for an individual directly standing over the core in the reactor room, inside the reactor room but not in direct line-of-sight of the core, just outside the reactor room, inside the control room, at the UCD/MNRC fence line, in the closest building, and the closest in habited building.

In the updated appendix B of the SAR (Ref. 56), the licensee states that it constructed a simple conservative model using the DOE Los Alamos National Laboratory Monte Carlo N-Particle (MCNP) transport computer code (Ref. 64) to estimate direct and scattered dose rates to members of the public at three receptor distances from the reactor and to workers at four locations inside the reactor building.

In the MCNP model, each radioactive decay is assumed to produce a single 1 MeV gamma-ray. The MCNP model utilizes a homogenized reactor (10 inches height [26 cm] by 15 inches diameter [38 cm]) to simplify the geometry and approximate some self-shielding in the reactor. The 9-ft-thick (2.7 m) concrete shielding for the base of the reactor tank and the minimum 9-ft-thick (2.7 m) concrete cylinder surrounding the reactor tank was modeled. The 6-inch-thick (15 cm) concrete walls of the reactor room were as modeled; however, the roof of the facility was not modeled because it is a thin steel corrugated structure that would provide little shielding or scatter. A 1-inch-thick (2.54 cm) iron disk, 3 ft (1 m) in diameter was placed 10 ft (3 m) above the top of the reactor tank to simulate the reactor "bridge" where the reactor control drives are located. This disk was used to simulate the scatter that would take place during this scenario and contribute to the dose rates of workers not directly located in direct line-of-sight with the reactor. No other structural materials were included in the MCNP model such as walls and ceilings in the reactor building to minimize the attenuation of scattered radiation to yield conservative dose estimates. Air attenuation and scatter was included in the MCNP model.

In the updated appendix B, the licensee calculates and provides the total fission product activity in the core as a function of time after shutdown in table B-8, “Total Fission Product Activity After Shutdown,” by using the SAR chapter 13, Equation 8. SAR table B-8 is reproduced below in SER table 4.7, “Total Fission Product Activity After Shutdown.”

Table 4.7 Total Fission Product Activity After Shutdown

Time After Shut Down	Total Activity (Ci)	Source Strength (γ/s)
1 hour	2.74×10^6	1.01×10^{17}
1 day	9.70×10^5	3.59×10^{16}
1 week	5.20×10^5	1.92×10^{16}
1 month	2.86×10^5	1.06×10^{16}

Occupational Dose

In updated appendix B of the SAR, the licensee provides its calculated dose rates for workers at the facility. SER table 4.8, “Dose Rates in the UCD/MNRC Reactor Building After a Loss of Pool Water Following 1.0 MWt Operation,” provides a summary of the occupational dose rates.

To estimate scattered dose rates for “Inside Reactor Room,” the licensee used the MCNP model at a position located just inside the reactor room 7 ft (2 m) from the edge of the reactor tank. This location was selected to be far enough away from the reactor tank so that there is no line-of-sight to the exposed core. To estimate dose rates for “Outside Reactor Room,” the licensee used MCNP model at a position just outside 7 ft (2 m) from the entrance to the reactor room to provide a worst-case dose rate in the equipment room. Should this accident scenario occur, workers may need to enter these areas to replenish water to the reactor tank. To estimate dose rates for “Control Room,” the licensee used the MCNP model at the reactor control room located at ground level approximately 46 ft (14 m) east of the reactor core.

The NRC staff review finds, using the results of the 2.0 MWt Torrey Pines TRIGA reactor (Ref. 65) LOCA dose rates when scaled to UCD/MNRC reactor operations at 1.0 MWt for the “Reactor Top,” the UCD dose rates summarized in SER table 4.8, are in general agreement. The NRC staff also finds the other dose rates (inside reactor room, outside reactor room, and control room) are reasonable in comparison to the dose rates for the “Reactor Top.”

Table 4.8 Dose Rates in the UCD/MNRC Reactor Building After a Loss of Pool Water Following 1.0 MWt Operation

Time After Shut Down	Effective Dose Equivalent Rate			
	Reactor Top (rem/hr)	Inside Reactor Room (mrem/hr)	Outside Reactor Room (mrem/hr)	Control Room (mrem/hr)
1 hour	$1.68x10^3$	235	89.8	17.5
1 day	$5.94x10^2$	83.9	32.1	6.2
1 week	$3.18x10^2$	46.1	17.7	3.4
1 month	$1.74x10^2$	24.5	9.4	1.8

Dose to Members of the Public

In updated SAR appendix B, the licensee provides its calculated dose rates for members of the public at various locations around the facility.

SER table 4.9, "Dose Rates at Receptor Distances from the UCD/MNRC Reactor After a Loss of Pool Water Following 1.0 MWt Operation," summarizes these dose rates.

The licensee uses the MCNP model with the same three receptor distances from the reactor as the single fuel element MHA in air with gap release activities for the single fuel element cladding failure accident in the reactor pool water 24 hours after reactor shutdown (discussed in SER section 4.2.1, which was evaluated by the NRC staff and found acceptable). The shortest distance to the UCD/MNRC fence line is 92 ft (28 m) and is considered the highest possible dose rates for members of the public in this scenario. The closest building to the reactor is an industrial X-ray facility utilized by the Air Force located 184 ft (56 m) to the south of the reactor that has been uninhabited for nearly 20 years. The closest inhabited building to the reactor is a large conference center located 308 ft (94 m) to the northeast of the reactor that is regularly used. The licensee also states that estimated dose rates calculated using the MCNP code are conservative since the roof and walls of the reactor facility are not modeled to consider shielding and attenuation that would result in reduced dose rates.

Table 4.9 Dose Rates at Receptor Distances from the UCD/MNRC Reactor After a Loss of Pool Water Following 1.0 MWt Operation

Time After Shut Down	Effective Dose Equivalent (mrem/h) at Distances (m) from UCD/MNRC		
	UCD/MNRC Fence Line (28 m)	Old Industrial X-ray Facility (56 m)	Large Conference Center (94 m)
1 hour	16.1	8.4	3.5
1 day	5.7	3.0	1.2
1 week	3.0	1.6	0.69
1 month	1.6	0.85	0.36

The NRC staff review finds that the MCNP model used by the licensee to calculate the dose to workers within the UCD/MNRC reactor was the same MNCP model used to evaluate direct and scattered dose rates (which the licensee terms skyshine) to members of the public following a postulated LOCA (see SER tables 4.8 and 4.9). The NRC staff review also finds that the MCNP model methodology is acceptable.

<u>LOCA Event Response by UCD Staff</u>

As described in its updated SAR appendix B, section B.2.5, “Mitigation of LOCA Skyshine by Reflooding MNRC Reactor Core,” the licensee indicates that the probability of a complete LOCA with the resulting uncovering of the core is very low and poses no threat of fuel overheating and the potential doses to the public can exceed 100 mrem (approximately 10 hours after the core becomes uncovered). Doses to the public may also eventually exceed 500 mrem, though this would require a member of the public to stand at the MNRC fence line, beginning at the start of the LOCA, and remaining there for nearly 100 hours. However, due to the potential to exceed the limits in 10 CFR 20.1301(d) of 500 mrem, the licensee developed a method to reflood the core to reduce doses to any member of the public.

Updated SAR appendix B, section B.2.5, describes the LOCA event which drains the 7,000-gallon (26,498-liter) reactor tank in approximately one hour. The previous LOCA event described in the SAR required an emergency core cooling system (ECCs) to prevent fuel overheating as a result of operation in excess of 1.5 MWt. Since the licensee requested a power level reduction in the licensed limit to 1.0 MWt, the ECCs is no longer needed to ensure that the peak fuel temperature remains below the SL at the proposed licensed power limit of 1.0 MWt (versus 2.0 MWt). As proposed in its updated SAR appendix B, section B.2.5, the licensee has chosen to implement a modification of the previous ECCS that it now describes as the “Core Reflooding System” and will provide the capability to reflood the core cavity and limit the skyshine doses to members of the public.

The core reflooding system consists of a 1.35-inch (3.43-cm) diameter pipe, that after attaching to the municipal city water supply, will flood directly over the core from above through a two-foot-high chimney structure. The core reflooding system requires operating staff to connect a fire hose from the city water supply to a core inlet pipe located just outside of the reactor room, which the licensee estimates can be done in less than 15 minutes. The licensee indicates that a reactor operator should have ample time to begin the reflood process as it calculated that it would take approximately 1 hour after receiving the low tank level alarm before the core would become uncovered.

Further, the licensee describes in its updated SAR appendix B, section B.2.5, that in the event of the postulated LOCA, the primary coolant would fill the void between the reactor tank and the surrounding concrete monolith before flooding Bay 5 located directly below. Since the core reflooding system cannot provide a flow rate greater than the postulated leak rate of the LOCA, the licensee indicates that reactor core will be uncovered for about 22 hours before the core can be recovered following the filling of Bay 5 with water. Once Bay 5 has been completely filled with water, the reactor core will be covered with approximately 4 ft (1.2 m) of water. The licensee’s dose calculations indicate that the reflooding will reduce the doses by a factor of 100 to 1,000 and essentially end the dose hazard from the reactor.

As described in the licensees’ updated SAR appendix B, section B.2.5, Bay 5 is an excavated concrete lined room 24 ft (7.3 m) wide by 27 ft (8.2 m) long by 14 ft (4.3 m) in depth. This corresponds to a volume of 8,064 cubic feet (228 cubic meters) or 60,320 gallons (228,336 liters). Based on a water pressure of 20 pounds per square inch (psi) (138 kilopascal (kPa)) and utilizing Bernoulli’s equation, the licensee calculated a flow rate of 43 gallons (163 liters) per minute for the core reflooding system, which would reflood Bay 5 (and the core) under 24 hours.

Updated appendix B, table B-16, “Integrated Scattered Radiation Doses for Various Location After 24 hours and Reactor Core has been Reflooded,” shows the integrated doses for the postulated LOCA and reflooding the core event at various locations, as shown in SER table 4.10, “Total Doses Around the UCD/MNRC Facility with the Reflood System.” Public doses are limited to less than 500 mrem.

Table 4.10 Total Doses Around the UCD/MNRC Facility with the Reflood System

Location	Licensee’s Effective Dose Equivalent (mrem)	NRC Staff Confirmatory Calculation of Effective Dose Equivalent (mrem)
Control Room	208	239
Fence Line	192	220
Closest Building (not inhabited)	100	115
Closest Building (inhabited)	42	47

The NRC staff review of the licensee’s LOCA event finds that the reactor tank water level is monitored (as required by proposed TS 3.3, “Reactor Coolant Systems,” Specification 3) and would provide an alarm to the operators. The NRC staff notes that the height of the primary coolant water (TS 3.3, Specification 3) is maintained 19 ft (5.8 m) above the core, and thus finds the licensee’s assumption that the operators would have an hour to diagnose and possibly terminate the reactor tank water leak is acceptable.

The NRC staff reviewed the licensee’s plan to use the core reflood system to mitigate the LOCA event where the leak cannot be stopped (as described in the updated SAR appendix B). The NRC staff finds that the activities needed to use the core reflooding system to be commensurate with activities normally conducted by operations staff. The NRC staff also finds that the licensee’s assumption that the LOCA event would result in a complete reactor tank drain down in approximately 1 hour to be consistent with the licensee’s calculations described above.

The licensee states that the reactor coolant water will initially collect into and fill Bay 5. The licensee indicates that the core reflooding system will take 15 minutes to initiate, by operator action, at which time it will provide a makeup flowrate of 43 gallons per minute (163 liters per minute) into the reactor tank. Based the makeup flow rate, the licensee indicates that the core will be covered with water from the core reflood system in less than 24 hours.

The NRC staff reviewed the licensee’s calculations for the reflooding of the core, and finds the licensee’s statements and estimates to be accurate. The NRC staff reviewed the postulated leak path into Bay 5 and finds that the licensee’s assumption that the core reflood system will refill the reactor tank, in approximately 24 hours, to a water level of approximately 4 ft (1.2 m) above the core, consistent with the design of the facility. The NRC staff performed confirmatory dose calculations and finds that 4 ft (1.2 m) of water above the core will provide sufficient shielding to reduce the radiation dose by factor of 1/512, or 0.002 percent, as stated in SER table 4.10.

<u>Emergency Plan Response</u>

The regulation, 10 CFR 20.1301(d), states, in part, that a licensee or license applicant may apply for prior NRC authorization to operate up to an annual dose limit for an individual member of the public of 0.5 rem (5 millisieverts (mSv)) if the licensee: (1) demonstrates the need for and the expected duration of operation in excess of the 100 mrem limit in paragraph (a); (2) has a program to assess and control dose within the 0.5 rem (5 mSv) annual limit; and (3) has procedures to be followed to maintain the dose as low as reasonably achievable (ALARA).

SER table 4.10 shows that the dose to a member of the public located at the fence (92 ft [28 m]) will exceed 100 mrem. In its response to the NRC staff request for additional information (RAI) (Ref. 72), the licensee provides an updated emergency plan (EP) that provides an assessment and corrective actions for an activation of the reactor room radiation area monitor (RAM) that would alarm on high radiation following a LOCA. The response actions in EP section 7, “Emergency Response,” c., “Class II Emergency – Alert,” ii, “Assessment Actions for Alert,” (1) “Loss of Reactor Tank Water,” and (2) “Elevated Radiation Levels at the Site Boundary,” directs UCD emergency staff to assess the reactor tank low water level alarm actuation, evaluate the loss of water, and to measure and characterize the radiation fields at the site boundary using portable dose rate instruments. The licensee indicates that these emergency response actions help to ensure the protection of the workers and any members of the public in the vicinity. If the leak cannot be isolated, EP section 7.c.iii, “Corrective Actions for Alert,” (1) Loss Reactor Tank Water directs the emergency staff to initiate the core reflooding system using procedure MNRC-0071-OMM-01, “Core Reflooding System Operation and Maintenance Manual.” If the radiation doses are expected to exceed the emergency action thresholds, the UCD emergency staff will coordinate with the local sheriff’s office to evacuate members of the public to a safe distance.

The NRC staff reviewed the EP as described above and the licensee’s procedure MNRC-0071-OMM-01 and finds the information specified adequate to respond to the highly unlikely occurrence of a LOCA resulting in complete uncovering of the reactor core. The NRC staff finds that the licensee has the capability and EP programs and procedures in place to minimize any potential radiation doses to ALARA with its proposed emergency response actions. Further, the NRC licensee’s EP response would ensure the safety and protection of the public, and limit any possible doses to a member of the public to less than 500 mrem in accordance with the limits provided in 10 CFR 20.1301(d). Based on the information above, the NRC staff finds that the licensee has satisfied the criteria in 10 CFR 20.1301(d)(1) - (3).

Conclusions

The NRC staff reviewed the UCD LOCA analysis using the guidance in NUREG-1537 and finds that the peak fuel cladding temperature that a TRIGA fuel element would reach is 585 °C (1,085 °F), which is well below the SL temperature limit of 930 °C (1,706 °F) in TS 2.2. Thus, the LOCA accident scenario will not challenge the integrity of the fuel cladding, and will not result in the release of any fission products to the reactor room or environment. Further, the NRC staff has evaluated the potential LOCA dose rates to the unrestricted area (fence boundary) and finds that the licensee has included the necessary protective actions specified in its EP to ensure that any potential radiation doses to a member of the public are maintained below the limits in 10 CFR 20.1301(d). On the basis of its review, the NRC staff concludes that the LOCA does not pose significant risk to the health and safety of the public or personnel.

4.2.3 Reactivity Insertion Event

SAR section 13.2.2, “Insertion of Excess Reactivity,” states that the most credible generic accident is the inadvertent rapid insertion (pulse insertion) of positive reactivity that, if large enough, could produce a transient resulting in fuel overheating and a possible breach of cladding integrity. Operator error or failure of the automatic power level control system could cause a slower event to occur because of the uncontrolled withdrawal of multiple control rods. Flooding or removal of beam tube inserts could also have a positive effect on reactivity but not as severe as the rapid removal of a control rod. The inherent prompt negative temperature response characteristics of TRIGA fuels clearly is a safety factor for this type of postulated accident.

SAR section 4.6.4.1.3, “ZrH Fuel Temperature Limits,” states that the TRIGA fuel cladding will not rupture if the fuel temperatures are never greater than 1,200 °C (2,192 °F) to 1,250 °C (2,282 °F), providing that the cladding temperature is less than about 500 °C (932 °F). However, for TRIGA fuel with a Zr-to-H ratio of 1.7, the license has chosen a conservative limit of 1,100 °C (2,012 °F). At this temperature, the internal fuel gas pressure is about a factor of 4 lower than would be necessary for cladding failure. This factor of 4 is more than adequate to account for uncertainties in cladding strength and manufacturing tolerances.

The NRC staff review of the reactivity insertion accident finds that the licensee’s limit of 1,100 °C (2,012 °F) will preserve the fuel cladding integrity and is acceptable.

SAR section 13.2.2.2.1, “Maximum Reactivity Insertion,” as supplemented by an updated analysis of the insertion of excess reactivity accident (Ref. 44), uses the Nordheim-Fuchs model, as described in GA-7882 (Ref. 38), to compute the maximum reactivity pulse that can occur without exceeding the pulse temperature limit of 1,100 °C (2,012 °F), which has been established to ensure cladding integrity. Multiple calculations were performed by the licensee for both 20/20 and 30/20 fuel at various burnups and initial temperatures (Ref. 44).

The NRC staff reviewed the licensee’s analysis, performed confirmatory calculations, and compared the results of the analysis with the findings of NUREG/CR-2387 (Ref. 41), and NUREG-1282, “Safety Evaluation Report on High-Uranium Content, Low-Enriched Uranium-Zirconium Hydride Fuels for TRIGA Reactors” (Ref. 39). In summary, the licensee’s analysis concludes that the maximum rapid reactivity insertion under the worst conditions (30/20 fuel, end-of-life conditions) that can be allowed is $1.90 at which point the fuel temperature could reach the limit of 1,100 °C (2,012 °F). Since the absolute maximum reactivity of all experiments which could potentially cause an inadvertent reactivity pulse insertion event, is limited per proposed TS 3.8.1, “Reactivity Limit,” Specification 2, to $1.75, the NRC staff finds that the UCD/MNRC reactor fuel will not approach the temperature limit. Further, the NRC staff performed confirmatory calculations that indicated that an inadvertent reactivity pulse of $1.80 would result in a maximum fuel temperature of 1,000 °C (1,832 °F), which is lower than the 1,100 °C (2,012 °F) limit. Therefore, given the proposed TS 3.8.1, Specification 2 limit of $1.75, and the margin to the fuel temperature limit of 1,100 °C (2,012 °F), the NRC staff finds that there is reasonable assurance that no radiation releases will occur as a result of this event.

The licensee provided an updated analysis of the uncontrolled withdrawal of both a single and all control rods (see Ref. 46, sections 13.1.1.1.1 and 13.1.1.1.2). The updated analysis conservatively assumes a single rod withdrawn at the maximum speed of 0.70 inches/s (1.78 cm/s), as opposed to the administratively limited withdraw speed of 0.40 inches/s (1.02 cm/s). The maximum single rod worth as described in SAR chapter 4 is $2.90, but a conservative rod worth of $3.00 was used to allow for variations in core loadings.

The licensee’s analysis indicated that the differential reactivity worth was assumed to be equal to the measured differential worth of the most reactive control rod between 50 percent and 55 percent withdrawn from the core and normalized to correspond to a $3.00 total rod worth, as this is the largest differential worth expected to be encountered while the reactor is critical. This maximum differential rod worth was determined to be $0.274/in or $0.192/s for a rod speed equal to 0.70 inches/s (1.78 cm/s). Initial power conditions were analyzed at both 100 W and 1.0 MWt using the single delayed neutron group model with prompt jump approximation and linear reactivity increase. With initial power at 100 watts thermal and a trip setpoint of 1.1 MWt, power level reaches the trip setpoint at 5.08 seconds. (Note that the 1.1 MWt trip setpoint used in this analysis is conservative relative to the actual trip setpoint of ≤1.02 MWt as specified in proposed TS 3.2.3, table 3.2, item d. Adding an additional 0.5 seconds for the actual release of the rods, the peak reactivity inserted was found to be $1.07, which is less than the limiting rapid reactivity insertion for the pulse accident of $1.90. With initial power at 1.0 MWt and a trip setpoint of 1.1 MWt, power level reaches the trip setpoint in 0.44 seconds. Adding an additional 0.5 seconds for the actual release of the rods, the peak reactivity inserted was found to be $0.18, which is less than the limiting rapid reactivity insertion for the pulse accident of $1.90.

The NRC staff reviewed proposed TS 3.2.1, Specification 2.c, which limits the maximum reactivity insertion rate $0.19 per second of any shim or regulating rod is consistent with the assumption used in the Insertion of Excess Reactivity accident scenario discussed above, and acceptable.

The NRC staff finds that the control rods are interlocked so the operator can only withdraw one at a time and, when in automatic power demand, a maximum of 3 rods can be withdrawn at one time. In order to envelope the accidents associated with uncontrolled withdrawal of control rods, the licensee conservatively analyzed the withdrawal of five control rods. The NRC staff finds that the same approach was used as for withdrawal of a single control rod and the accident assumes the maximum rod withdraw speed of 0.70 inches/s (1.78 cm/s). For the calculation case beginning from 100 W, the NRC staff finds that the trip setpoint was reached in 0.87 seconds with an additional 0.5 second delay for the actual release of the rods and the peak reactivity inserted was found to be $1.58, which is less than the limiting rapid reactivity insertion for the pulse accident of $1.90. For the calculation case beginning from 1.0 MWt, the NRC staff finds that the trip setpoint was reached in 0.08 seconds with an additional 0.5 second delay for the actual release of the rods. The NRC staff finds that the peak reactivity inserted was found to be $0.67, which is less than the limiting rapid reactivity insertion for the pulse accident of $1.90. Therefore, the NRC staff finds that there is no safety concern associated with the uncontrolled withdrawal of control rods.

SAR section 13.2.2.2.4, “Beam Tube Flooding or Removal,” states, that in the event of flooding of one or more beam tubes, air or inert gas would be replaced with water, which constitutes a positive reactivity addition. The licensee has previously estimated that the worth of one flooded beam tube is about $0.25. During the removal of the in-tank section of a beam tube, air and graphite will be replaced by water because a portion of the graphite reflector is removed with this section of the beam tube. As with the flooding of a beam tube, replacement of the air/gas with water results in a positive effect on reactivity. Upon flooding of one or more beam tubes (total of four), the amount of excess reactivity is well below the limit of $1.90 as discussed in the rapid reactivity insertion accident.

The NRC staff review finds the beam tube flooding or removal accident scenario does not represent a safety-significant event. In the case of removal of the in-tank section of the beam tube, the net result will be a smaller reactivity addition than for beam tube flooding, as the replacement of the graphite with water results in a negative effect on reactivity. Therefore, NRC staff finds there is no safety concern associated with beam tube flooding or removal. Based on the information provided above, the NRC concludes that the postulated reactivity insertion event would not result in damage to the fuel cladding or release of fission products, and is acceptable.

4.2.4 Loss-of-Coolant Flow

SAR section 13.2.4.1, "Accident Initiating Events and Scenarios," indicates that a loss of coolant flow could occur because of failure of a key component in the reactor primary or secondary cooling system (e.g., a pump), loss of electrical power, or blockage of a coolant flow channel. Operator error could also cause a loss of coolant flow. SAR section 13.2.4.1, provides an analysis that indicates that the bulk water temperature adiabatically increases at a rate of 0.55 °C/min (at a power level of 1.0 MWt). Under these conditions, the operator has ample time to reduce the power and place the heat-removal system into operation or shut down the reactor before any abnormal temperature is reached in the reactor water. A core inlet temperature alarm at 45 °C (113 °F) and primary and secondary low flow alarms will alert the operator to an abnormal condition and should allow for corrective action before reaching the bulk water temperature limit.

SAR section 13.2.4.2, "Accident Analysis and Determination of Consequences," describes a situation where there is a loss of coolant flow (i.e., no heat removal by reactor coolant systems) without immediate operator action. In this case, the reactor would continue to operate at 1.0 MWt, assuming automatic control of the control rods, and would heat the coolant at a rate of 0.55 °C/min. After approximately 2 hours, the coolant would reach saturated conditions and begin to boil off at a rate of approximately 3,180 kilograms/hour (kg/hr) (7,011 pounds mass per hour (lbm/hr)). At this rate, it would take about 18 hours to drain the tank. As the water level drops below the top of the core, the reactor would shut down due to the voiding in the core. The licensee considers this a non-credible event as this operating condition would not go undetected by the operators. There are many alarms that would alert the operators to the situation including low water level, low water flow, high water temperature and radiation monitors.

The NRC staff finds that a loss of coolant flow would be a slow developing event and provide sufficient time for the operators to take corrective actions or shutdown the reactor, and therefore, would not result in a loss of fuel integrity. A localized loss of coolant flow could occur due to a foreign object or debris being dropped into the reactor tank and either landing on top of the core or being drawn up from below. While the probability of this is low, it is not zero and is considered a credible event. As stated by the licensee in SAR section 13.2.4.3, "Localized Loss of Coolant Flow," if a foreign object or debris is observed, the standard response is to SCRAM the reactor so the object can be removed.

The NRC staff reviewed the licensee's description associated with a potential reactor core loss of coolant flow event due to a foreign object, as described in the SAR, as supplemented by "Analysis for Blockage of Fuel Channel Potential" (Ref. 66) and finds that the open lattice reactor design provides multiple flow paths for coolant, which would minimize the potential for a cooling flow blockage to occur that could affect fuel cooling. Based on the information described above, the NRC staff concludes that, should a localized loss-of-coolant flow condition occur, the reactor core would retain adequate cooling and a localized loss-of-flow condition poses no adverse risk to the health and safety of the public or UCD/MNRC staff.

4.2.5 Mishandling or Malfunction of Fuel

SAR section 13.2.5, "Mishandling or Malfunction of Fuel," describes the accident initiating events and scenarios and dose consequence from mishandling or malfunction of fuel at the

UCD/MNRC reactor, which include: 1) fuel handling accidents where an element is dropped underwater and damaged severely enough to breach the cladding, 2) simple failure of the fuel cladding due to a manufacturing defect or corrosion, and 3) overheating of fuel with subsequent cladding failure during steady state operations or pulsing; overheating might occur due to incorrect loading of fuel elements with different uranium (U)-235 enrichments in a mixed core.

SAR appendix B provides a description of the mishandling or malfunction of fuel and the licensee's calculated doses to UCD/MNRC reactor staff from a single fuel element cladding failure accident in the reactor pool water at 1.0 MWt operation 24 hours after reactor shutdown (Ref. 56). In SAR appendix B, the licensee states that this accident is considered a less severe, but more credible, accident evaluated to provide a better understanding of the radiological consequences of damaging a fuel element while being handled remotely in the reactor tank.

The NRC staff finds, as shown in SER table 4.1 and SER table 4.2, that the calculated worker doses from a single fuel element cladding failure accident in the reactor pool water 24 hours after reactor shutdown are less (by more than a factor of 10) compared to worker doses from the single fuel element MHA calculated for the 2-minute and 5-minute evacuation times. Therefore, the NRC staff concludes that mishandling or malfunction of fuel will not result in consequences more severe than those calculated for the single fuel element MHA in air. The NRC staff's evaluation of the single fuel element MHA in air and the single fuel element cladding failure accident in the reactor pool water 24 hours after reactor shutdown is discussed in section 4.2.1, which was evaluated by the NRC staff and found acceptable.

Further, in SAR chapter 4, the licensee states that under the current 1.0 MWt core where only 20/20 and 30/20 fuel elements are permitted, there is no credible fuel loading errors as the LCC was established to intentionally produce the most peaking possible in a single element. The licensee described two fuel loading errors of significance, but are not considered credible. One involves the physical removal of the aluminum slug located in the middle of the central irradiation facility and replacing it with a fuel element. The other involved the placement of older low-burnup 8.5 percent fuel inside the core. However, the licensee states that the older 8.5 percent fuel looks significantly different than the current 20 weight percent (wt%) and 30 wt% fuel elements, and the two different fuel types are kept in spent fuel storage areas.

Based on the information described above, the NRC staff finds that a fuel loading error is highly unlikely, and, if it occurred, the radiological consequences would be bounded by the MHA in SER section 4.2.1, which was evaluated by the NRC staff and found acceptable.

4.2.6 Experiment Malfunction

SAR section 13.2.6, "Experiment Malfunction," describes the accident initiating events and scenarios and dose consequence from improperly controlled experiments involving the UCD/MNRC reactor that could potentially result in damage to the reactor such as corrosion and large reactivity changes to the reactor, unnecessary radiation exposure to facility staff and members of the public, and unnecessary releases of radioactivity into the unrestricted area.

SAR chapter 10, "Experimental Facilities and Utilization, states that because of the potential for accidents that could damage the reactor if experiments are not properly controlled, there are strict procedural and regulatory requirements addressing experiment review and approval. These requirements focus on ensuring that experiments will not fail their containment and that the licensee will also incorporate requirements to assure that there is no reactor damage and no radioactivity releases or radiation doses that exceed the occupational and public dose limits in 10 CFR Part 20, should failure occur.

SAR section 13.2.6 provides the safety analyses which indicates that three (3) pounds (1.4 kg) of trinitrotoluene (TNT) equivalent explosives may be safely irradiated in radiography Bays 1, 2, 3 and 4, provided the beam tube cover plates are at least 0.5 inches (1.3 cm) thick. The licensee also obtained an independent safety review performed by the Southwest Research Institute (SRI). SRI completed a safety analysis to determine the maximum amount of TNT equivalent explosive allowable in radiography Bay 3, (i.e., the amount that will not cause failure of the beam tube cover plate and will cause only repairable structural damage to the bay). Bay 3 is the smallest in volume of all the radiography bays at the UCD/MNRC. The study concluded that Bay 3 can withstand a detonation of six (6) pounds (2.7 kg) of TNT equivalent explosive with certain modifications. The study performed by SRI concluded that the Bay 3 door track must be strengthened. The recommended strengthening consists of welding three additional anchor bolt plates to the door track and bolting these plates into the wall with additional drilled anchor bolts. This strengthening assures that the door will respond in a ductile manner to an unexpected high blast load, absorbing the additional load with larger deflections rather than responding in a brittle failure mode.

SAR section 13.2.6 also states that the licensee completed a similar study to determine the maximum amount of TNT equivalent explosives allowable in all radiography bays. This study concluded that Bays 1, 2 and 4 can withstand a detonation of 6 pounds (2.7 kg) of TNT equivalent explosives without any damage provided that the criteria in SAR table 13-15, "Changes to Beam Tube Cover Plates," are implemented in each bay. However, to meet category 1 protection requirements for 6 pounds (2.7 kg) of explosives, the west door of Bay 2 also requires modification by means of an additional wheel and post assembly. The analysis performed by the licensee demonstrates that for 3 pounds (1.4 kg) of TNT equivalent explosives, no modifications are necessary to the radiography bay doors for Bays 1, 2 or 4. These doors will also respond in a ductile manner. As a result of the above studies, the licensee concluded that installation of beam tube cover plates with the thicknesses shown in SAR table 13-15 and implementing an explosives limitation of 3 pounds (1.4 kg) of TNT equivalent for each of the four radiography bays will satisfy the safety limitations established by the two (2) previous safety analyses.

The NRC staff reviewed the licensee's limit of 6 pounds (2.7 kg) of TNT equivalent explosive material in radiography Bays 1-4 and finds that the licensee's analysis was thorough, as validated by an independent external organization specializing in explosive analyses (SRI). The result of the analysis confirmed that 6 pounds (2.7 kg) of TNT equivalent explosives could be allowed in all four radiography bays. Further, the licensee has incorporated a lower limit of 3 pounds (1.4 kg) of TNT equivalent explosives into proposed TS 3.8.2, "Materials," Specification 3 hence providing a safety factor of 2. The NRC staff finds that proposed TS 3.8.2, Specification 3, which limits the explosive material to 3 pounds (1.4 kg) of TNT equivalent explosive material in radiography Bays 1-4 is acceptable.

The NRC staff review finds that proposed TS 3.8.2, Specification 2, limits each fueled experiment such that the total inventory of iodine (I) isotopes I-131 through I-135 is no greater than 1.5 Ci and the maximum strontium (Sr) inventory is no greater than 5 mCi. In comparison, these activity limits are much less (by several orders of magnitude) than the respective accident source term activities shown in SER tables 4.2 and 4.3 and as described in NUREG/CR-2387, (Ref. 41). In addition, proposed TS 3.8.3, "Failure and Malfunction," limits the quantity and type of material in the experiment such that the airborne radioactivity in the reactor room will not result in exceeding the applicable dose limits in 10 CFR Part 20 in the unrestricted area, assuming 100 percent of the gases or aerosols escapes. The NRC staff finds that the experiment malfunction will not result in consequences more severe than those calculated for the single fuel element MHA in air. Based on the information above, the NRC staff finds proposed TS 3.8.2, Specification 2, acceptable.

The NRC staff finds that the inadvertent insertion of a negative reactivity worth experiment into the reactor core was a credible mechanism for an accidental positive reactivity insertion. However, reactivity insertion accidents are discussed in SER section 4.2.3, which was evaluated by the NRC staff and found acceptable, and the licensee determined limitations on the reactivity worth of experiments so that such an accident would have less severe consequences than that analyzed.

The NRC staff review finds that the limitations, as specified in proposed TS 3.8.1 of $1.75 for a single secured experiment, $1.00 for individual movable experiments, and $1.90 for the absolute total worth of all experiments in the core help to assure that the fuel temperature SL is not exceeded by an experimental malfunction. Based on the information described above, the NRC staff concludes that potential consequences from an experiment malfunction is limited by proposed TS 3.8.2, and the potential radiation dose consequences are bounded by the MHA.

4.2.7 Loss of Normal Electrical Power

SAR chapter 3, “Design of Structures, Components, Equipment and Systems,” criterion 5, “Sharing of Structures, Systems, and Components,” and SAR chapter 8, “Electrical Power,” section 8.1, “Introduction,” state, in part, that the loss of power results in the shutdown of the reactor since all control circuits are fail-safe, and no power is required for safe shutdown or to maintain safe shutdown conditions. As such, an electric power failure at any point in the UCD/MNRC network will not detrimentally affect the reactor. The design of the UCD/MNRC reactor does not require electrical power to safely shut down the reactor, nor does it require electrical power to maintain acceptable shutdown conditions.

SAR section 13.2.7, “Loss of Normal Electrical Power,” section 13.2.7.2, “Accident Analysis and Determination of Consequences,” states, in part, that since the UCD/MNRC does not require emergency backup power systems (see chapter 6, “Engineered Safety Features”) to safely maintain core cooling, there are no credible reactor accidents associated with the loss of electrical power. A backup power system is present at the UCD/MNRC, which mainly provides conditioned power to the reactor console and control instrumentation. Therefore, the reactor will not automatically scram when there is a loss of normal electrical power. In fact, the backup power system is capable of providing electrical power for the reactor control and various operational measurements for a period of time after loss of normal electrical power and until its battery power supply is exhausted. Loss of normal electrical power during operations is addressed in the reactor operating procedures, which require that upon loss of normal power an orderly shutdown is to be initiated by the operator on duty. The battery backup power will allow monitoring of the orderly shutdown of the reactor and verification of the reactor’s shutdown condition.

The NRC staff reviewed the loss of normal electrical power and finds that a loss of normal electric power poses no undue risk to the operation of the UCD/MNRC. Backup power is available from a battery powered uninterruptible power supply and the UCD/MNRC can safety shutdown and remain in a safe shutdown condition without emergency power. Based on the information above, the NRC staff concludes that the results of the licensee's analysis of loss of normal electrical power analysis are acceptable.

4.2.8 External Events

SAR section 13.2.8, "External Events," states that the UCD/MNRC reactor is located near the edge of the runway at the former McClellan Air Force Base, and therefore, airplane crashes involving the reactor may potentially impact the reactor. The probability of potential impact due to aircraft crashes was evaluated by GA Technologies as a part of the original Stationary Neutron Radiography System Proposal, and the probability of aircraft crashes was calculated to be about $5x10^{-8}$ per year (see SAR appendix C, "Probabilistic Assessment of the Airplane Crash Risk for McClellan Air Force Base TRIGA® Reactor"). Further, the licensee indicated in SAR section 2.1.1, "Site Location and Description," in tables titled "Annual Aircraft Operations-1970-1995," and "Annual Aircraft Operations 2000-2016," that the overall flight operations have been reduced by a factor of 10 times since the closure of the Airforce base in 2000. As such, the licensee indicates that the probability of an aircraft crash would be approximately $5x10^{-9}$ per year and is thus considered a non-credible event. Based on the aforementioned information on aircraft crash probability calculations by GA Technologies, and reduced flight information presented since year 2000, the licensee concluded that the probability of aircraft crash is very small and is lower than the acceptable criterion of $1x10^{-7}$ per year.

Based on the review of the information presented by the licensee, the NRC staff considers that the conclusion of the applicant is reasonable and acceptable. The NRC staff concludes that no potential adverse aircraft crash impacts are expected on the safe operation of the reactor.

SAR section 2.5, "Geology, Seismology, and Geotechnical Engineering," states that the UCD/MNRC reactor is located in a region with a history of elevated seismic hazard. The most intense historical earthquake was an Intensity VII (on the Modified Mercalli scale) and corresponds to a Magnitude 6.0 to 6.5 earthquake located approximately 20 mi (32 km) west of Sacramento. The UCD/MNRC reactor is designed to meet the requirements of the Uniform Building Code for a facility located in a Zone 3 region of seismic hazard.

The NRC staff reviewed the information in the SAR, guidance in NUREG-1537, and seismic hazard estimates available from the US Geological Survey (https:/earthquake.usgs.gov). The NRC staff finds that the licensee followed guidance for research reactors in using the applicable civil engineering codes for earthquake design. The UCD/MNRC facility was designed to withstand external events and the potential associated accidents as discussed in SAR chapter 2. The reactor facility is designed to accommodate a seismic event by shutting down, which would not pose undue risk to the health and safety of the public. For a seismic event that could cause significant facility damage, the damage would not result in a release of radioactive effluents greater than the MHA scenario (discuss in SER 4.2.1, which was evaluated by the NRC staff and found acceptable). Therefore, the NRC staff finds that any radiation exposure to the staff and the public is within acceptable limits. The NRC staff concludes that external events do not pose undue risk to the health and safety of the public.

4.2.9 Mishandling or Malfunction of Equipment

SAR section 13.2.9, "Mishandling or Malfunction of Equipment," states that no credible initiating events were identified for this accident class and that situations involving an operator error at the reactor controls, a malfunction or loss of safety related instruments or controls and an electrical fault in the control rod system were anticipated at the reactor design stage. Many safety features, such as control system interlocks and automatic reactor shutdown circuits, were designed into the overall control system as described in SAR chapter 7, "Instrumentation and Control." Malfunction of confinement or containment systems would have the greatest impact during the MHA. However, as shown in SAR section 13.2.1, no credit is taken for confinement or containment systems in the analysis of the MHA for the UCD/MNRC reactor. Furthermore, there are no accident scenarios at the UCD/MNRC that depend on confinement or containment systems, although simple confinement devices like a fume hood might be used as part of normal operations.

The review guidance in NUREG-1537, Part 2 states that initiating events under this heading would require a case-by-case, reactor specific discussion and may contain additional events that fall outside the other eight categories presented SER sections 4.2.1 through 4.2.8. The NRC staff did not identify any additional initiating events that would be specific to the UCD/MNRC reactor. Therefore, the NRC staff concludes that the consequences of mishandling or malfunction of equipment have been addressed and do not pose a risk to the health and safety of the public or to the UCD/MNRC staff.

4.3 Accident Analysis and Determination of Consequences

The NRC staff reviewed the licensee's postulated and analyzed accident scenarios at the UCD/MNRC facility. The NRC staff concludes that the licensee has postulated and analyzed sufficient accident-initiating events and scenarios. On the basis of its evaluation of the information in the licensee's SAR, as supplemented, the NRC staff concludes the following:

- The licensee considered the expected consequences of a sufficiently broad spectrum of postulated credible accidents and an MHA, emphasizing those that could lead to a loss of integrity of fuel element clad and a release of fission products.
- The licensee analyzed the most significant credible accidents and the MHA and determined that, under conservative assumptions, the most significant credible accidents and the MHA will not result in occupational radiation exposure of the UCD/MNRC staff or radiation exposure to a member of the public in excess of the applicable 10 CFR Part 20 limits.
- The licensee employed appropriate methods for accident analysis and consequence analysis.
- Licensee calculations, including assumptions, demonstrated that a LOCA would not result in unacceptable fuel element temperatures. The reactor can be safely cooled with all fuel elements in an air environment. Doses to individuals evacuating the reactor room and at the site boundary are calculated to be below the 10 CFR Part 20 limits.
- External events that would lead to fuel failure are unlikely.
- The licensee accident analysis confirms the acceptability of the licensed power of 1.0 MWt including the response to anticipated transients and accidents.
- The licensee accident analysis confirms the acceptability of the assumptions stated in the individual analyses provided in the SAR, as supplemented.

4.4 Conclusions

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

5 TECHNICAL SPECIFICATIONS

In this section of the safety evaluation report (SER), the U.S. Nuclear Regulatory Commission (NRC) staff provides its evaluation of the licensee's proposed technical specifications (TSs). The University of California-Davis/McClellan Nuclear Research Center (UCD/MNRC) TSs define specific features, characteristics, and conditions governing the safe operation of the UCD/MNRC facility. TSs are explicitly included in the renewal license as Appendix A.

The NRC staff reviewed the format and content of the proposed TSs for consistency with the guidance in NUREG-1537, Part 1, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content," chapter 14, "Technical Specifications," appendix 14.1, "Format and Content of Technical Specifications," and American Nuclear Standards Institute/American Nuclear Society (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 it meets the requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36, "Technical specifications."

5.1 Introduction

5.1.1 Scope

Proposed TS 1.1, "Scope," states:

> This document constitutes the Technical Specifications for the Facility License No. R-130 as required by 10 CFR 50.36 and supersedes all prior Technical Specifications. This document includes the "Basis" to support the selection and significance of the specifications. Each basis is included for information purposes only. They are not part of the Technical Specifications, and they do not constitute limitations or requirements to which the licensee must adhere. These specifications are formatted in a manner consistent with ANSI/ANS 15.1-2007.

The NRC staff reviewed proposed TS 1.1 and finds that the information in proposed TS 1.1 related to the scope of the TSs is consistent with the with the guidance in NUREG-1537, appendix 14.1, and ANSI/ANS-15.1-2007, and with 10 CFR 50.36. The NRC staff finds that the proposed UCD/MNRC TSs include a safety limit (SL), limiting safety system settings (LSSSs), limiting conditions for operation (LCOs), surveillance requirements (SRs), design features, and administrative controls, consistent with the requirements of 10 CFR 50.36(c). The NRC staff finds that the SL, LSSSs, LCOs, SRs, and design features in the TSs include applicability and objective statements consistent with the guidance in ANSI/ANS-15.1-2007 and NUREG-1537, appendix 14.1. The NRC staff also finds that the bases for the SL, LSSSs, LCOs, SRs, and design features in the TSs summarize the rationale for those TSs, but that the bases are not part of the TSs, as required by 10 CFR 50.36(a)(1). Therefore, the NRC staff concludes that TS 1.1 is acceptable.

5.1.2 Definitions

Proposed TS 1.2, "Definitions," states:

> Channel. A channel is the combination of sensor, line, amplifier, processor, and output devices which are connected for the purpose of measuring the value of a parameter.
>
> Channel Calibration. A channel calibration is an adjustment of the channel such that its output corresponds with acceptable accuracy to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip and shall include a Channel Test.
>
> Channel Check. A channel check is a qualitative verification of acceptable performance by observation of channel behavior. This verification, where possible, shall include comparison of the channel with other independent channels or systems measuring the same variable.
>
> Channel Test. A channel test is the introduction of a signal into the channel for verification that it is operable.
>
> Confinement. Confinement means an enclosure of the reactor room which is designed to limit the release of effluents from the enclosure to the external environment through controlled or defined pathways.
>
> Control Rod. A control rod is a device fabricated from neutron absorbing material, with or without a fuel or air follower, which is used to establish neutron flux changes and to compensate for routine reactivity losses. The follower may be a stainless steel section. A control rod shall be coupled to its drive unit to allow it to perform its control function, and its safety function when the coupling is disengaged. This safety function is commonly termed a scram.
>
>> Regulating Rod. A regulating rod is a control rod used to maintain an intended power level and may be varied manually or by a servo-controller. It may have a fueled-follower section. A regulating rod shall have scram capability.
>>
>> Shim Rod. A shim rod is a control rod having an electric motor drive and scram capabilities. It may have a fueled-follower section. Its position is varied manually. A shim rod shall have scram capability.
>
> Excess Reactivity. Excess reactivity is that amount of reactivity that would exist if all control devices were moved to the maximum reactive position from the point where the reactor is exactly critical (K_{eff} = 1) at reference core conditions.
>
> Experiment. Any operation, hardware, or target (excluding devices such as detectors) 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 a part of their design to carry out experiments is not normally considered an experiment. Specific experiments shall include:

Moveable Experiment. A movable experiment is one that is not secured and intended to be moved while near or inside the core during reactor operation.

Secured Experiment. A secured experiment is any experiment, experiment facility, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining force 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.

Experiment Facilities. Experiment facilities shall mean the pneumatic transfer tube, beam tubes, irradiation facilities in the reactor core or in the reactor tank, and radiography bays.

External Scram. External scrams may arise from the radiography bay doors, radiography bay ripcords, and bay shutter interlocks.

Instrumented Fuel Element. An instrumented fuel element is a standard fuel element fabricated with thermocouples for temperature measurements. An instrumented fuel element shall have at least one operable thermocouple embedded in the fuel near the axial and radial midpoints.

Licensed Area. The licensed area is that area inside of the fence immediately surrounding the reactor building. This fence also demarcates the property that is owned by the University of California from the surrounding area and is approximately 2.3 acres in size. Inside of the licensed area is also a restricted area.

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

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

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

Protective Action. Protective action is the initiation of a signal or the operation of equipment within the UCD/MNRC reactor safety system in response to a parameter or condition of the UCD/MNRC reactor facility having reached a specified limit.

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

Reactor Operating. The UCD/MNRC reactor is operating whenever it is not shutdown or secured.

Reactor Operator. An individual who is licensed to manipulate the controls of the facility.

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

Reactor Secured. The UCD/MNRC reactor is secured when:

1) Either 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 moderations and reflection;

2) Or the following conditions exist:

a) The minimum number of control rods are fully inserted to ensure the reactor is shutdown, as required by technical specifications; and

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

c) No experiments in any reactor experiment facility, or in any other way near the reactor, are being moved or serviced if the experiments have, on movement, a reactivity worth exceeding the maximum value allowed for a single experiment or $1.00, whichever is smaller; and

d) The console key switch is in the off position, and the key is removed from the lock.

Reactor Shutdown. The UCD/MNRC reactor is shutdown if it is subcritical by at least one dollar ($1.00) both in the Reference Core Condition and for all allowed ambient conditions with the reactivity worth of all installed experiments included.

Reference Core Condition. The condition of the core when it is at ambient temperature (cold $T<28^{o}$ C), the reactivity worth of xenon is negligible (< $0.10) (i.e., cold and clean), and the central irradiation facility contains the graphite thimble plug and the aluminum thimble plug (CIF-1).

Safety Channel. A safety channel is a measuring channel in the reactor safety system.

Scram Time. Scram time is the elapsed time between the initiation of a scram and the instant that the control rod reaches its fully-inserted position.

Senior Reactor Operator. An individual who is licensed to direct the activities of reactor operators and to manipulate the controls of the facility.

Shall, Should, and May. The word “shall” is used to denote a requirement; the word “should” to denote a recommendation; the word “may” to denote permission, neither a requirement nor a recommendation.

Shutdown Margin. Shutdown margin shall mean the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety system starting from any permissible operating condition with the most reactive rod assumed to be in the most reactive position, and once this action has been initiated, the reactor will remain subcritical without further operator action.

Surveillance Intervals. Maximum intervals are established to provide operational flexibility and not to reduce frequency. Established frequencies shall be maintained over the long term. The allowable surveillance interval is the interval between a check, test, or calibration, whichever is appropriate to the item being subjected to the surveillance, and is measured from the date of the last surveillance. Allowable surveillance intervals shall not exceed the following:

Quinquennial - interval not to exceed seventy-two (72) months

Annual - interval not to exceed fifteen (15) months.

Semiannual - interval not to exceed seven and a half (7.5) months.

Quarterly - interval not to exceed four (4) months.

Monthly - interval not to exceed six (6) weeks.

Weekly - interval not to exceed ten (10) days.

Unscheduled Shutdown. An unscheduled shutdown is any unplanned shutdown of the UCD/MNRC reactor caused by actuation of the reactor safety system, operator error, equipment malfunction, or a manual shutdown in response to conditions which could adversely affect safe operation, not including shutdowns which occur during testing or check-out operations.

The NRC staff review finds that the definitions proposed above are either standard definitions used in research reactor TSs or are facility-specific definitions that are consistent with the guidance in NUREG-1537, Part 1, and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that the licensee's proposed TS definitions are acceptable.

5.2 Safety Limits and Limiting Safety System Settings

5.2.1 Safety Limits

Proposed TS 2.1, "Safety Limits," states:

Specification - The maximum fuel temperature in a standard TRIGA fuel element shall not exceed 930°C during steady-state operation.

The licensee's proposed safety limit (SL) of 930 degrees Celsius (°C) (1,706 degrees Fahrenheit (°F)) is discussed in SER section 2.7, which was evaluated by the NRC staff and found acceptable. The NRC staff finds that this SL will reasonably protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity, as required by 10 CFR 50.36(c)(1). The NRC also finds that TS 2.1 is consistent with the guidance in NUREG-1537, Part 1, and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 2.1 is acceptable.

5.2.2 Limiting Safety System Setting

Proposed TS 2.2, "Limiting Safety System Setting," states:

> Specification - The limiting safety system setting shall be less than or equal to 750°C (operationally this may be set more conservatively) as measured in an instrumented fuel element. One instrumented element shall be located in the analyzed peak power location of the reactor operational core.

The licensee's proposed limiting safety system setting (LSSS) of 750 °C (1,382 °F) is discussed in SER section 2.8, which was evaluated by the NRC staff and found acceptable. The NRC staff finds that it will reasonably protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity, as required by 10 CFR 50.36(c)(1). The NRC also finds that TS 2.2 is consistent with the guidance in NUREG-1537, Part 1, and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that proposed TS 2.2 is acceptable.

5.3 Limiting Conditions for Operations

5.3.1 Reactor Core Parameters

5.3.1.1 Excess Reactivity

Proposed TS 3.1.1, Excess Reactivity," states:

> Specification - The maximum available excess reactivity (reference core condition) shall not exceed 5.625% Δk/k ($7.50).

The licensee's proposed excess reactivity limit of $7.50 is discussed in SER section 2.5.3, which was evaluated by the NRC staff and found acceptable. The NRC staff finds that available excess reactivity of $7.50 at reference core provides sufficient reactivity to compensate for various negative reactivity effects associated with operation and use of the reactor, as well as allowing some operational flexibility. The NRC also finds that proposed TS 3.1.1 is consistent with the guidance in NUREG-1537, Part 1, and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that proposed TS 3.1.1. is acceptable.

5.3.1.2 Shutdown Margin

Proposed TS 3.1.2, Shutdown Margin," states:

> Specification - The reactor shall not be operated unless the shutdown margin provided by the control rods is greater than 0.375% Δk/k ($0.50) with:
>
> 1. The reactor in the reference core condition where there is no ^{135}Xe poison present and the core is at ambient temperature,
>
> 2. The most reactive control rod assumed fully withdrawn, and
>
> 3. Absolute value of all movable experiments analyzed in their most reactive condition or $1.00 whichever is greater.

The licensee's proposed shutdown margin limit of $0.50 is discussed in SER section 2.5.3, which was evaluated by the NRC staff and found acceptable. The NRC staff reviewed proposed TS 3.1.2 and finds that it helps ensure that the reactor can be safely shutdown from any operational configuration and remain shutdown, even if the maximum worth control rod should stick in the fully withdrawn position. The NRC also finds that proposed TS 3.1.2 is consistent with the guidance in NUREG-1537, Part 1, and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that proposed TS 3.1.2. is acceptable.

5.3.1.3 Core Configuration Limitations

Proposed TS 3.1.3, "Core Configuration Limitations," states:

Specification-

1. The only fuel types allowed are 20/20 and 30/20 with stainless steel cladding. These elements may only be placed in any position in Hex Rings C through G.

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

3. A control rod shall not be manually removed from the core unless the core has been shown to be subcritical by at least $0.50 with the highest worth control rod in the full-out position.

The licensee's proposed fuel types and allowed locations stated in proposed TS 3.1.3, Specification 1 is discussed in SER section 2.2.1, which was evaluated by the NRC staff and found acceptable. The NRC staff reviewed proposed TS 3.1.3, Specifications 2 and 3, and finds that they are consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 3.1, "Reactor Core Parameters," item (4), "Core Configurations." The NRC also finds that proposed TS 3.1.3 is consistent with the guidance in NUREG-1537, Part 1, and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that proposed TS 3.1.3. is acceptable.

5.3.1.4 Fuel Parameters

Proposed TS 3.1.4, "Fuel Parameters," states:

Specification - The reactor shall not be used for operation with damaged fuel. All fuel elements shall be inspected visually for damage or deterioration as per Technical Specifications Section 4.1.4. A fuel element shall be considered damaged and must be removed from the core if:

1. In measuring the transverse bend, the bend exceeds 0.125 inch (3.175 mm) over the full length 23 inches (584 mm) of the cladding, or,

2. In measuring the elongation, its length exceeds its initial length by 0.125 inch (3.175 mm), or,

3. A cladding failure exists as indicated by measurable release of fission products, or,

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

5. ^{235}U burnup is calculated to be greater than 50% of initial content.

In the case of specification 3, the reactor may be operated only for the purpose of identifying the specific element that is releasing measurable fission products.

The NRC staff reviewed proposed TS 3.1.4 and finds the limits provided help ensure that operation of the UCD/MNRC remain consistent with the assumptions and analyses described in the safety analysis report (SAR), as supplemented. The NRC staff also finds that proposed TS 3.1.4 is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 3.1, “Reactor Core Parameters,” item (6), “Fuel Parameters,” and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that proposed TS 3.1.4. is acceptable.

5.3.2 Reactor Control and Safety Systems

5.3.2.1 Control Rods

Proposed TS 3.2.1, “Control Rods,” states:

Specification –

1. The reactor shall not be operated unless all six control rods are operable.

2. Control rods shall not be considered operable unless conditions a, b, and c below are met:

   a. No physical damage is apparent to the rod or drive assemblies and the control rod responds normally to control rod motion signals.

   b. The scram time measured from the instant a signal reaches the value of a limiting safety system setting to the instant that the slowest control rod reaches its fully inserted position shall not exceed one (1) second.

   c. The maximum reactivity insertion rate of any shim or regulating rod shall not exceed $0.19 per second.

The NRC staff reviewed proposed TS 3.2.1, Specification 1 and Specifications 2.a and 2.b and finds that they are consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 3.2, “Reactor Control and Safety Systems,” item (1), “Operable Control Rods,” which states that the number of operable control rods and the maximum scram time should be specified in the TSs, and item (2) “Reactivity Insertion Rates,” which states that the maximum rates of reactivity insertion should be specified.

The NRC staff reviewed proposed TS 3.2.1, Specification 2.c and finds that it helps establish the maximum reactivity insertion rate $0.19 per second of any shim or regulating rod, which is consistent with the assumption used in the Insertion of Excess Reactivity accident scenario discussed in SER section 4.2.3, which was evaluated by the NRC staff and found acceptable.

The NRC staff finds that proposed TS 3.2.1 establishes the conditions for control rod operability, which helps ensure that UCD/MNRC control rod operability is maintained consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 3.2, and assumptions used in the SAR, as supplemented. Based on the information above, the NRC staff concludes that proposed TS 3.2.1, is acceptable.

5.3.2.2 Reactor Instrumentation for Operation

Proposed TS 3.2.2, “Reactor Instrumentation for Operation,” states:

> Specification - The reactor shall not be operated unless the minimum number of channels described in Table 3.1 are operable and the information is displayed on the reactor console.

Table 3.1 Required Reactor Instrumentation

Measuring Channel	Minimum Number Required
a. Reactor Power Level Safety Channel	2
b. Linear Power Channel	1
c. Log Power Channel	1
d. Fuel Temperature Channel	1

The NRC staff reviewed proposed TS 3.2.2 and finds that it establishes the requirement to ensure the operability of the measuring channel listed in proposed TS 3.2.2, table 3.1, “Required Reactor Instrumentation,” when the reactor is operating. The NRC staff also finds that the measuring channels provide console instrumentation indicating reactor power level, coolant temperature, and coolant level.

The NRC staff also finds that proposed TS 3.2.2 helps ensure that UCD/MNRC instrumentation required for safe operation are properly controlled, and the measuring channels are consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 3.2, item (4), “Scram Channels.” Based on the information above, the NRC staff concludes that proposed TS 3.2.2 is acceptable.

5.3.2.3 Reactor Scrams and Interlocks

Proposed TS 3.2.3, “Reactor Scrams and Interlocks,” states:

> Specification - The reactor shall not be operated unless the minimum number of scrams and interlocks described in Table 3.2 and 3.3 are operable:

Table 3.2 Minimum Number of Scrams

Scrams	Function	Number Required
a. Console Manual Scram	SCRAM	1
b. Reactor Room Manual Scram	SCRAM	1
c. Radiography Bay Manual Scram	SCRAM to prevent worker radiation exposure.	1/Bay*
d. Reactor Power Level Safety Scrams	SCRAM at 1.02 MW (t) or less.	2
e. High Voltage Power Supplies Scrams	SCRAM on loss of nominal operating voltage to the NM1000 and NPP1000 power channels.	2
f. Fuel Temperature Scrams	SCRAM at 750 C or less	1
g. Watchdog Circuit Scram	Scram within 8 seconds upon lack of response in DAC or CSC computer	2
h. Magnet Power Key Switch Scram	SCRAM	1

Table 3.3 Minimum Number of Interlocks

Interlocks	Function	Number Required
i. Low Source Level Rod Withdrawal Prohibit Interlock	Prevents control rod withdrawal at $<2*10^{-7}$ % power.	1
j. Control Rod Withdrawal Interlock	Prevents simultaneous manual withdrawal of two rods.	1
k. Radiography Bay Shutter Interlock	Prevents simultaneous opening of shield door and massive shutter in the same radiography while the reactor is operating.	1/Bay*

*The reactor may be operated if an individual radiography bay manual Scram or reactor bay shutter interlock is inoperable. In this event, the affected radiography bay shall be placed out of service until the manual Scram or reactor bay shutter interlock becomes operable.

SAR section 7.2, “Reactor Protection System,” table 7-1, “Monitor Scram Window Display and Associated Reactor Protection Actions,” provides a detailed description of each of the scrams. The console and reactor room manual scram are provided for manual-initiation by the reactor operator. SAR section 9.6, “Interlocks/Controls - Bay Shutters/Doors,” provides a detailed description of the radiography bay manual scram, which is actuated by a pull-cord, provided in each of the radiography bays, if needed to terminate an irradiation.

In SER section 2.6, the NRC staff evaluated the reactor power level scram and finds that at for a power increase to 1.03 megawatt thermal (MWt), the peak fuel temperature would be 418 °C (784 °F) and well below the proposed TS 2.2 LSSS of 750 °C. SAR table 7-1 states that the high voltage power supplies scrams on a loss of voltage to the power monitoring channels, NM-1000 and NPP-1000. The fuel temperature scram provides protection for the fuel cladding by actuation of a scram should the instrumented fuel element reach its setpoint of 750 °C (1,382 °F), which is consistent with proposed TS 2.2.

SAR figure 7.1, “Block Diagram of Instrumentation and Control System,” illustrates the function of the Data Acquisition Computer (DAC), which receives and converts from analog-to-digital form or digital-to-analog form, information from the NM-1000 and NPP-1000 power level monitors. The processed information is then transmitted to the control system computer (CSC). Information transfer between the DAC and CSC is by high-speed data transmitter, and is needed for proper operation of the reactor. The watchdog circuit scram provides protection should a communication problem develop between the DAC and CSC. The magnet power key switch provides a scram if the key is not in the “on” position, necessary to provide power to the reactor console.

SAR section 7.1.2.5, “Reactor Operating Controls,” states that interlocks prevent the movement of the rods in the up direction (out of the core, or reactivity addition) under the condition that the neutron source level reading is below minimum count necessary to ensure proper operation of the neutron power monitoring channels. SAR section 13.2.2.2.2, “Uncontrolled Withdrawal of a Control Rod,” states that the control rods are interlocked so the operator can only withdraw one at a time, and SAR section 9.6, describes the radiography bay shutter door interlocks, which prevent reactor operation if the shutter door is not in the proper position.

The NRC staff reviewed the minimum number of scrams and interlocks provided in proposed TS 3.2.3, tables 3.2 and 3.3, as described above, and finds the minimum number of scrams and interlocks support the design bases provided in the SAR, as supplemented, and are consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 3.2, item (4), and table 14.1, “Typical required scrams and power reverses,” and item (5), “Interlocks,” and table 14.2, “Typical required interlocks.” The NRC staff finds that the licensee provided the type of scram and interlock, the minimum number of scram and interlock channels, and, where applicable, the appropriate setpoints. The NRC staff also finds that the type of scram and interlocks listed in proposed TS 3.2.3, is consistent with other TRIGA-type reactors. Based on the information above, the NRC staff concludes that proposed TS 3.2.3, is acceptable.

5.3.3 Reactor Coolant Systems

Proposed TS 3.3, “Reactor Coolant Systems,” states:

> Specification - The reactor shall not be operated unless the systems and instrumentation channels described below are operable, and the information is displayed locally or in the control room.
>
> 1. The bulk tank water temperature shall not exceed 45 °C;
>
> 2. The conductivity of the tank water shall be less than 5 μmhos/cm when averaged over a one month period;
>
> 3. The reactor shall not be operated if water level drops below a depth of 19 feet from the top of the fueled region of the core; and
>
> 4. The reactor shall not be operated if the radioactivity of the pool water exceeds the limits of 10 CFR 20 Appendix B Table 3 for radioisotopes with half-lives >24 hours.

5. The reactor core reflooding system is considered operable if the local pressure gauge on the system reads 20 psi or above.

The NRC staff reviewed proposed TS 3.3, Specification 1 and finds that it is consistent with the description in SAR section 13.2.4, “Loss of Coolant Flow,” which states that the core inlet temperature alarms at or above 45 °C (113 °F) and alerts the reactor operator to an abnormal reactor coolant flow condition in order to provide time for corrective action. Based on the information above, the NRC staff concludes that proposed TS 3.3, Specification 1 is acceptable.

The NRC staff reviewed proposed TS 3.3, Specification 2 and finds that the reactor coolant tank water conductivity limit of 5 micromhos per centimeter (µmhos/cm), is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 3.3, “Coolant Systems,” item (9), “Water Chemistry Requirements,” which states that acceptable values for conductivity are less than or equal to 5 µmhos/cm. Based on the information above, the NRC staff concludes that proposed TS 3.3, Specification 2 is acceptable.

The NRC staff reviewed proposed TS 3.3, Specification 3 and finds that the reactor tank low water level alarm setting of 19 feet (ft) (5.8 meter (m)) above the top of the reactor fuel is consistent with the assumption used in the updated thermal-hydraulic analysis (Ref. 47). Based on the information above, the NRC staff concludes that proposed TS 3.3, Specification 3 is acceptable.

The NRC staff reviewed proposed TS 3.3, Specification 4 and finds that limiting the radioactivity to the sewerage discharge limits in 10 CFR Part 20, “Standards for Protection Against Radiation,” appendix B, table 3, helps ensure reactor primary water radioactivity levels remain low and known in the event of any pool or primary coolant leakage. Based on the information above, the NRC staff concludes that proposed TS 3.3, Specification 4 is acceptable.

The NRC staff reviewed proposed TS 3.3, Specification 5 and finds that the reactor core reflooding system pressure gage reading of 20 pounds per square inch (psi) (138 kilopascal (kPa)) is consistent with the assumption used in the licensee’s loss-of-coolant accident (LOCA) analysis described in SER section 4.2.2, which was evaluated by the NRC staff and found acceptable. The NRC staff also finds that this minimum pressure results in sufficient water flow capability to reflood the reactor tank in just under 24 hours and reduce the potential radiation exposures to the public. Based on the information above, the NRC staff concludes that proposed TS 3.3, Specification 5 is acceptable.

5.3.4 TS 3.4 This section intentionally left blank

5.3.5 Ventilation and Confinement System

Proposed TS 3.5, “Ventilation and Confinement System,” states:

Specification -

1. This specification applies to the ventilation system under normal reactor operations. Under this normal mode of operation, air from the reactor room is exhausted from the facility stack via the reactor room exhaust fan (EF-1). The reactor shall not be operated unless the normal mode ventilation system is operable. The normal mode ventilation system shall be considered operable if

the reactor room exhaust fan (EF-1) is in operation and the pressure in the reactor room is negative relative to surrounding room (equipment room).

2. This specification applies to the ventilation system when high levels of airborne radioactivity are detected. In this mode no air from the reactor room exits via the facility stack. The transition from normal ventilation mode to recirculation mode is accomplished by a number of automated dampers triggered by the CAM alarms. The reactor room ventilation shall operate in the recirculation mode, with all exhaust air passing through a HEPA filter, whenever a high level continuous air monitor (CAM) alarm is present due to airborne radionuclides emitted from the reactor or samples in the reactor room. The recirculation ventilation system shall be considered operable when all dampers in the system are operable.

3. Movement of irradiated fuel or fueled experiments with significant fission product inventory outside of containers, systems, or storage areas within the reactor room shall not be performed unless the ventilation system is operating as described in TS 3.5, Specifications 1 and the ventilation system is operable as described in TS 3.5, Specification 2.

4. Core or control rod work that could cause a change in reactivity of more than one dollar shall not be performed unless the ventilation system is operating as described in TS 3.5, Specifications 1 and the ventilation system is operable as described in TS 3.5, Specification 2.

5. Movement of experiments within the core that could reasonably cause a change of total worth of more than one dollar shall not be performed unless the ventilation system is operating as described in TS 3.5, Specifications 1 and the ventilation system is operable as described in TS 3.5, Specification 2.

The NRC staff reviewed proposed TS 3.5, Specifications 1 and 2, and finds that these requirements help ensure that the ventilation system will function as described in SAR section 9.5, “Air Handling System,” to mitigate the consequences of any potential airborne radioactivity released within the facility (as discussed in SER section 4.2.1, which was evaluated by the NRC staff and found acceptable).

The NRC staff also finds proposed TS 3.5, Specifications 1 and 2 help ensure the ventilation system operates at a negative pressure relative to the outside environment, which ensures that any radioactive effluents produced from normal reactor operations are monitored prior to release from the facility stack as described in the guidance in NUREG-1537, Part 1, appendix 14.1, section 3.5, “Ventilation Systems.” Further, the NRC staff finds that proposed TS 3.5, Specification 2 helps ensure the ventilation system can provide confinement capability should a significant release of radioactive effluents occur in the facility, as described in the guidance in NUREG-1537, Part 1, appendix 14.1, section 3.4, “Containment or Confinement.” Based on the information above, the NRC staff concludes that proposed TS 3.5, Specifications 1 and 2 are acceptable.

The NRC staff reviewed proposed TS 3.5, Specifications 3, 4, and 5 and finds that these specifications require the ventilation system be operable in accordance with proposed TS 3.5, Specifications 1 and 2, for the movement of irradiated fuel or fueled experiments, core or control rod work, and experiments worth more than $1.00 of reactivity, consistent with the guidance in

NUREG-1537, Part 1, and ANSI/ANS-15.1-2007, section 3.4.1, “Operations that require containment or confinement.” Based on the information above, the NRC staff concludes that proposed TS 3.5, Specifications 3, 4, and 5 are acceptable.

5.3.6 TS 3.6 <u>This section intentionally left blank</u>

5.3.7 Reactor Radiation Monitoring Systems

5.3.7.1 Monitoring Systems

Proposed TS 3.7.1, “Monitoring Systems,” states:

> Specification - The reactor shall not be operated unless the channels described in Table 3.4 are operable, the readings are below the alarm setpoints, and the information is displayed in the control room. The Stack and Reactor Room Continuous Air Monitors (CAMS) shall not be placed out of service at the same time during reactor operation.

Table 3.4 REQUIRED RADIATION MONITORING INSTRUMENTATION

Measuring Equipment	Minimum Number Required	Channel Function
a. Facility Stack Continuous Air Monitor (CAM)	1*	Monitors Argon-41 and radioactive particulates, and alarms.
b. Reactor Room Radiation Area Monitor (RAM)	1	Monitors the radiation level in the reactor room and alarms
c. Demineralizer System Radiation Area Monitor (RAM)	1	Monitors radiation level at the demineralizer station and alarms
d. Reactor Room Continuous Air Monitor (CAM)	1**	Monitors air from the reactor room for particulate and xenon radioactivity and alarms
e. Environmental Dosimeters	8	Monitor radiation at facility boundary

> RAMs and CAMs may be placed out-of-service for up to 2 hours for calibration and maintenance. During this out-of-service time, no experiment or maintenance activities shall be conducted which could result in alarm conditions (e.g., airborne releases or high radiation levels).
>
> * If the Facility Stack Continuous Air Monitor CAM is out of service for more than 2 hours, the amount of Ar-41 released may be calculated based on the number of MW hours the reactor is operated. This alternative measurement method for an out of service Facility Stack Continuous Air Monitor shall not exceed 60 days.
>
> ** If the Reactor Room CAM is out of service for more than 2 hours, a manual measurement may be performed every 4 hours of operation to verify airborne levels of radiation in the reactor room are below the CAM set points. This

alternative measurement method for an out of service Reactor Room CAM shall not exceed 60 days.

In SER section 3.1.4, the NRC staff evaluated the radiation monitoring equipment, and found the licensee's equipment is appropriate for detecting the types and intensities of radiation likely to be encountered within the facility at appropriate frequencies to help ensure compliance with 10 CFR 20.1501, "General," paragraph (a) and (b).

Further, the NRC staff reviewed proposed TS 3.7.1 and finds that it helps to establish the minimum radiation monitoring requirements for operation of the facility, and proposed TS 3.7.1, table 3.4, "Required Radiation Monitoring Instrumentation," provides the required monitoring channels and the minimum number of channels to support operation at the facility. The NRC staff finds that the channels and minimum numbers in proposed TS 3.7.1, table 3.4 are typical of other TRIGA facilities and consistent with the guidance in NUREG-1537, Part 1, appendix 14.1.

The single asterisk (*) applicable to the facility stack continuous air monitor (CAM) in proposed TS 3.7.1, table 3.4, provides an exception to monitoring the Argon (Ar)-41 released by calculating the Ar-41 produced by the megawatt-hours of operation of the reactor. The NRC staff finds this exception acceptable because Ar-41 is directly produced by the operation of the reactor and can be accurately calculated using the megawatt-hours of operation (which is directly related to the Ar-41 activation in air). Further, this exception is limited to a 60-day period, such that the licensee has the flexibility to efficiently and effectively repair the CAM. The double asterisk (**) applicable to the reactor room CAM would allow a manual measurement of the airborne radiation in the reactor room every 4 hours to ensure that the airborne radiation levels are acceptable for the operators and staff. The NRC staff finds that this exception is acceptable because it provides an equivalent means to measure airborne radioactivity and is limited to a 60-day period, which provides flexibility for the licensee to efficiently and effectively repair the CAM. Based on the information above, the NRC staff concludes that proposed TS 3.7.1, is acceptable.

5.3.7.2 Effluents - Argon-41 Discharge Limit

Proposed TS 3.7.2, "Effluents - Argon-41 Discharge Limit," states:

Specification - The total Ar-41 released by MNRC shall not exceed 118 Ci per calendar year.

The NRC staff reviewed proposed TS 3.7.2 (discussed in SER section 3.11, which was evaluated by the NRC staff and found acceptable). The NRC staff also finds that the potential dose to a member of the public from the release of 118 curies (Ci) of Ar-41 as proposed TS 3.7.2 would not exceed 5 millirem (mrem). Based on the information above, the NRC staff concludes that proposed TS 3.7.2 is acceptable.

5.3.8 Experiments

5.3.8.1 Reactivity Limits

Proposed TS 3.8.1, "Reactivity Limits," states:

> Specification - The reactor shall not be operated unless the following conditions governing experiments exist:
>
> 1. The absolute reactivity worth of any single moveable experiment in the pneumatic transfer tube, the central irradiation facility, the central irradiation fixture 1 (CIF-1), or any other in-core or in-tank irradiation facility, shall be less than $1.00 (0.75% Δk/k).
>
> 2. The absolute total reactivity of all experiments positioned in the pneumatic transfer tube, and in any other reactor in-core and in-tank irradiation facilities at any given time shall be less than ($1.75) (1.31% Δk/k), including the potential reactivity which might result from malfunction, flooding, voiding, or removal and insertion of the experiments.

The NRC staff reviewed proposed TS 3.8.1 (discussed in SER section 2.5.3, which was evaluated by the NRC staff and found acceptable). Based on the information above, the NRC staff concludes that proposed TS 3.8.1 is acceptable.

5.3.8.2 Materials

Proposed TS 3.8.2, "Materials," states:

> Specification - The reactor shall not be operated unless the following conditions governing experiment materials exist:
>
> 1. 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.
>
> 2. Each fueled experiment shall be controlled such that the total inventory of iodine isotopes 131 through 135 in the experiment is no greater than 1.5 curies and the maximum strontium inventory is no greater than 5 millicuries.
>
> 3. Explosive materials in quantities of three (3) pounds of TNT equivalent or less may be irradiated in each radiography bay. All four radiography bays may contain 3 pounds TNT equivalent simultaneously.

The NRC staff reviewed proposed TS 3.8.2, Specification 1 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 3.8.2, "Materials," which states that potentially corrosive materials should be double encapsulated and the failure of an encapsulation of material that could damage the reactor should require removal and inspection of potentially damaged components. Based on the information above, the NRC staff concludes that proposed TS 3.8.2, Specification 1 is acceptable.

The NRC staff reviewed proposed TS 3.8.2, Specifications 2 and 3 (discussed in SER section 4.2.6, which were evaluated by the NRC staff and found acceptable). Based on the information above, the NRC staff concludes that proposed TS 3.8.2, Specifications 2 and 3 are acceptable.

5.3.8.3 Failure and Malfunctions

Proposed TS 3.8.3, "Failure and Malfunctions," states:

> Specification -Where the possibility exists that the failure of an experiment (except fueled experiments) under normal operating conditions of the experiment or reactor, credible accident conditions in the reactor, or possible accident conditions in the experiment could release radioactive gases or aerosols to the reactor room or the unrestricted area, the quantity and type of material in the experiment shall be limited such that 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, assuming that:
>
> 1. 100% of the gases or aerosols escape from the experiment;
> 2. If the effluent from an irradiation facility exhausts through a holdup tank which closes automatically on high radiation level, at least 10% of the gaseous activity or aerosols produced will escape;
> 3. If the effluent from an irradiation facility exhausts through a filter installation designed for greater than 99% efficiency for 0.3 micron particles, at least 10% of these aerosols can escape; and
> 4. For materials whose boiling point is above 130 °F and where vapors formed by boiling this material can escape only through an undisturbed column of water above the core, 10% of these vapors can escape.

The NRC staff reviewed proposed TS 3.8.3 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 3.8.3, "Failure and Malfunction," which helps to ensure that the radiological consequences of experiment failure are adequately considered, and the quantity of material introduced in the experiment is limited and properly controlled by the licensee. Based on the information above, the NRC staff concludes that proposed TS 3.8.3 is acceptable.

5.4 Surveillance Requirements

Proposed TS 4.0, "Surveillance Requirements," states:

> General. The surveillance frequencies denoted herein are based on continuing operation of the reactor. Surveillance activities scheduled to occur during an operating cycle which cannot be performed with the reactor operating may be deferred to the end of the operating cycle. If the reactor is not operated for a reasonable time, a reactor system or measuring channel surveillance requirement may be waived during the associated time period. Prior to reactor system or measuring channel operation, the surveillance shall be performed for each reactor system or measuring channel for which surveillance was waived. A reactor system or measuring channel shall not be considered operable until it is

> successfully tested. Discovery of non-compliance shall limit operation of the reactor to completing that specific non-compliance surveillance.
>
> Surveillance requirements may be deferred during prolonged periods in which the reactor is shutdown (except TS 4.3 and TS 4.7). However, they shall be completed prior to reactor startup unless reactor operation is required for performance of the surveillance. Such surveillance shall be performed as soon as practical after reactor startup. Scheduled surveillance which cannot be performed with the reactor operating may be deferred until a planned reactor shutdown.
>
> Any additions or modifications to the ventilation system, the core and its associated support structure, the pool or its penetrations, the primary coolant system, the rod drive mechanism or the reactor safety system shall be made and tested to assure that the systems will meet their functional requirements in accordance with manufacturer specifications or specifications reviewed by the NSC. A system shall not be considered operable until after it is successfully tested.

The NRC staff reviewed proposed TS 4.0 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 4, “Surveillance Requirements,” and ANSI/ANS-15.1-2007, section 4., “Surveillance requirements,” which states that surveillances should be performed to ensure the operability of the limiting conditions for operations, surveillance periods should be specified, surveillances may be deferred, if necessary, based on the operation of the facility, but must be performed prior to the reactor being considered operable and surveillances should be performed following modification or repairs as part of the operability for that component or system.

The NRC staff also finds that the surveillances in proposed TS 4.3, “Reactor Coolant Systems,” and proposed TS 4.7, “Reactor Radiation Monitoring Systems,” should not be deferred as a result of an extended reactor shutdown and are necessary to ensure that the reactor coolant system and radiation monitoring systems remain operable to support continue monitoring of the reactor and radioactivity. Based on the information above, the NRC staff concludes that proposed TS 4.0 is acceptable.

5.4.1 Reactor Core Parameters

5.4.1.1 Excess Reactivity

Proposed TS 4.1.1, “Excess Reactivity,” states:

> Specification - The core excess reactivity shall be verified annually or following a change in core loading, control rod configuration, or core experiment that is expected to change the reactivity by more than $0.25 (not including transient fission product poison effects).

The NRC staff reviewed proposed TS 4.1.1 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 4.1, “Reactor Core Parameters,” item (1), “Excess Reactivity,” which states the excess reactivity should be determined annually or after changes in the core, in-core experiments, or control rods when a predicted reactivity exceeds the absolute value of the shutdown margin. The NRC staff also finds that the licensee has proposed a lower reactivity ($0.25) than the shutdown margin ($0.50) which results in a more conservative surveillance frequency. Based on the information above, the NRC staff concludes that proposed TS 4.1.1 is acceptable.

5.4.1.2 Shutdown Margin

Proposed TS 4.1.2, “Shutdown Margin,” states:

> Specification- The core shutdown margin shall be determined at least annually and following a change in core or control rod configuration that is expected to change the shutdown margin by more than $0.25 (not including transient fission product poison effects).

The NRC staff reviewed proposed TS 4.1.2 and finds that it is consistent the guidance in NUREG-1537, Part 1, appendix 14.1, section 4.1, item (2), “Shutdown Margin,” which states the shutdown margin should be determined at least annually and after changes in either the core, in-core experiments, or control rods. The NRC staff also finds that the licensee’s use of $0.25 to account for core configuration changes within the annual period provides relief from the surveillance requirement for small changes but also limits core configuration changes that would involve fuel or control rods. A core configuration change that does involve fuel or control rods would require a shutdown margin verification. Based on the information above, the NRC staff concludes that proposed TS 4.1.2 is acceptable.

5.4.1.3 Core Configuration Limitations

Proposed TS 4.1.3, “Core Configuration Limitation,” states:

Specification-

1. A daily check of the core shall be made to verify only stainless steel clad 20/20 and 30/20 elements are only located Hex Rings C through G.

2. Prior to removal of any fuel element it shall be verified that the core is subcritical by more than the calculated worth of the most reactive fuel element being moved.

3. Prior to manual removal of any control rod it shall be verified that the core is subcritical by at least $0.50 with the highest worth control rod in the full-out position.

The NRC staff reviewed proposed TS 4.1.3, Specification 1, and finds that it is consistent with the guidance in ANSI/ANS-15.1-2007, section 4.1, “Reactor core parameters,” item (3), which recommends the licensee to perform the surveillance following significant core or control rod changes. The NRC staff also finds that the daily core verification of the stainless steel clad 20/20 and 30/20 fuel is more frequent than the guidance in NUREG-1537, Part 1, appendix 14.1, section 4.1, item (4), “Core Configuration,” which states only verify when any changes occur in the reactor core configuration. The NRC staff also finds that the verifications performed in proposed TS 4.1.3, Specifications 2 and 3, prior to removal of fuel or control rods, are consistent with, and help to ensure that the respective TS 3.1.3, Specifications 2 and 3 are satisfactory. Based on the information above, the NRC staff concludes that proposed TS 4.1.3 is acceptable.

5.4.1.4 Fuel Parameters

Proposed TS 4.1.4, “Fuel Parameters,” states:

Specification -

1. All fuel elements shall be inspected for damage or deterioration and measured for length and transverse bend at least at quinquennial intervals.

2. An analysis of any irradiation facility installed in the central cavity of this core shall be done before it is used with this core.

3. No single element may be operated at a power level above 17.69 kW (as analyzed) at a steady state power level of 1.0 MW.

The NRC staff reviewed proposed TS 4.1.4 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 4.1, item (6), “Fuel Parameters,” to inspect the fuel on a 5-year cycle. The licensee included an analysis of any irradiation facility installed in the central cavity of the core to ensure acceptability prior to use. The licensee also imposed a limit of 17.69 kilowatt thermal for any fuel element to ensure that the design assumptions used in the SAR analyses, as supplemented, for the limiting core configuration are maintained (discussed in SER section 2.6, which was evaluated by the NRC staff and found acceptable). Based on the information above, the NRC finds proposed TS 4.1.4 acceptable.

5.4.2 Reactor Control and Safety Systems

5.4.2.1 Control Rods

Proposed TS 4.2.1, “Control Rods,” states:

Specification -

1. Control rod worths shall be determined annually or after physical removal or any significant (>$0.25 expected reactivity change) change in core or control rod configuration.

2. Each control rod shall be inspected at annual intervals by visual observation of the fueled sections and absorber sections plus examination of the linkages and drives.

3. The scram time of each control rod shall be measured semiannually.

4. The maximum reactivity insertion rate of the highest worth control rod shall be measured annually.

The NRC staff reviewed proposed TS 4.2.1, Specification 1 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 4.2, “Reactivity Control and Safety Systems,” item (1), “Reactivity Worth of Control Rods,” which states control rod worths should be determined annually, and after changes in the core or control rod configuration. The NRC staff finds proposed TS 4.2.1, Specification 2 follows the guidance in NUREG-1537, Part 1, appendix 14.1, section 4.2, item (9), “Rod Inspection,” to perform an annual inspection. The NRC staff finds proposed TS 4.2.1, Specification 3 consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 4.2, item (4), “Scram Times of Control and Safety Rods,” and ANSI/ANS-15.1-2007, section 4.2, “Reactor control and safety systems,” item (4) to measure scram times semi-annually. The licensee added proposed TS 4.2.1, Specification 4 to help ensure that the maximum reactivity insertion rate was known by the reactor staff.

Based on the information above, the NRC staff concludes that proposed TS 4.2.1 is acceptable.

5.4.2.2 Reactor Instrumentation for Operation

Proposed TS 4.2.2, “Reactor Instrumentation for Operation,” states:

Specification –

1. Reactor power level safety channels (linear and log) shall undergo a daily test prior to reactor startup and an annual calibration. If a channel is removed, replaced, or unscheduled maintenance is performed, or a significant (>$0.25 expected change in reactivity) change in core configuration occurs, a channel calibration shall be required.

2. Fuel temperature channel shall undergo a daily test prior to reactor startup and an annual calibration.

The NRC staff reviewed proposed TS 4.2.2 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 4.2, item (5), “Scram and Power Measuring Channels,” which states that scram channels should be tested prior to reactor startup and ANSI/ANS-15.1-2007, section 4.2, item (5), “Scram channels,” which states that the scram channels should be calibrated annually. Based on the information above, the NRC staff concludes that proposed TS 4.2.2 is acceptable.

5.4.2.3 Reactor Scrams and Interlocks

Proposed TS 4.2.3, “Reactor Scrams and Interlocks,” states:

Specification –

1. Scram circuits a, b, d, e, f, g, and h required in section 3.2.3 shall undergo a daily channel test prior to operation. Scram circuits d and f shall undergo an annual calibration.

2. Scram circuit c required in section 3.2.3 shall undergo a monthly channel test.

3. All Interlock circuits required in section 3.2.3 shall undergo an annual channel test.

The NRC staff reviewed proposed TS 4.2.3, Specifications 1 and 2 and finds that these are, consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 4.2, item (5), which states that scram channels should be tested prior to reactor startup and ANSI/ANS-15.1-2007, section 4.2, item (5), which states that the scram channels should be calibrated annually.

The NRC staff reviewed proposed TS 4.2.3, Specification 3 and finds that it is consistent with the guidance in ANSI/ANS-15.1-2007, section 4.2, item (9), to annually test interlocks. Based on the information above, the NRC staff concludes that proposed TS 4.2.3 is acceptable.

5.4.3 Reactor Coolant Systems

Proposed TS 4.3, “Reactor Coolant Systems,” states:

Specification –

1. A channel check of the reactor tank bulk water temperature alarm setpoint shall be performed quarterly. A channel calibration of the reactor tank bulk water temperature system shall be performed at least annually.

2. A channel test of the reactor tank water level alarm setpoint shall be performed at least semi-annually.

3. The reactor tank water conductivity shall be measured monthly. Multiple measurements taken in one month shall be averaged to determine the monthly value.

4. The pool water radioactivity shall be measured at least semi-annually.

5. The local pressure gauge to the reactor reflooding system shall be checked to verify it is reading 20 psi or above prior to operation. Valve cycling and flow verification checks shall be made quarterly.

The NRC staff reviewed proposed TS 4.3, Specifications 1 and 2 and finds that these are consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 4.3, “Coolant Systems,” item (8), “Primary Coolant Sensors and Channels,” to perform channel checks of sensors and channels quarterly, and calibrations annually.

The NRC staff reviewed proposed TS 4.3, Specification 3 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 4.3, item (6), “Conductivity and pH,” to measure the conductivity monthly. Further, the NRC staff notes that since the UCD/MNRC reactor coolant system is open to the atmosphere, and the reactor coolant conductivity is limited by proposed TS 3.3.2 to less than 5 μmhos/cm, the information provided in NRC staff memorandum, dated May 11, 2015 (Ref. 74), states that the pH values will remain between 5.6 and 5.8. Based on this information, the licensee is not required to maintain a pH measurement in the TSs. The NRC staff finds proposed TS 4.3, Specification 3 acceptable.

The NRC staff reviewed proposed TS 4.3, Specification 4 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1 section 4.3, item (4), “Analysis of Coolants for Radioactivity,” and ANSI/ANS-15.1-2007, section 4.3, item (4), “Analysis of Coolants for Radioactivity,” to measure the pool water coolant for radioactivity at least annually. Based on this information, the NRC staff finds proposed TS 4.3, Specification 4, acceptable.

The NRC staff review proposed TS 4.3, Specification 5 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 4.3, item (2), “Test of Emergency Coolant Sources and System,” and ANSI/ANS-15.1-2007, section 4.3, item (2), “Test of emergency coolant source(s)," which checks the local pressure gauge prior to every reactor operation and cycles the valves annually. The licensee has chosen to perform the pressure check prior to reactor operation, and to cycle the valves quarterly. Further, the NRC staff finds that the core reflooding system is not intended necessary to provide emergency core cooling but to provide shielding in response to a complete primary coolant loss due to a LOCA, reflood the pool, and cover the core with water to reduce the potential doses to the members of the public. Thus, the NRC staff finds the requirements proposed in TS 4.3, Specification 5 consistent with the planned use of the core reflooding system as described in SER section 4.2.2, which was evaluated by the NRC staff and found acceptable, and the guidance in NUREG-1537, Part 1, and ANSI/ANS-15.1-2007. Based on the information described above, the NRC staff concludes proposed TS 4.3 is acceptable.

5.4.4 Ventilation and Confinement System

Proposed TS 4.4, "Ventilation and Confinement System," states:

Specification –

1. The reactor room exhaust system shall have a channel check during each day's operation.

2. A channel test of the reactor room ventilation system's ability to automatically switch to the recirculation mode (HEPA filtered confinement) mode upon actuation of the CAM high alarm shall be performed quarterly.

The NRC staff reviewed proposed TS 4.4, Specification 1 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 4.5, "Ventilation," and ANSI/ANS-15.1-2007, section 4.5, "Ventilation systems," item (1) to perform an operability check of the exhaust system quarterly by performing a channel check during each day's operation. The NRC staff also reviewed proposed TS 4.4, Specification 2, and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 4.4.2, "Confinement," to perform a functional test quarterly. Based on the information above, the NRC staff concludes that propose TS 4.4 is acceptable.

5.4.5 TS 4.5 This section intentionally left blank.

5.4.6 TS 4.6 This section intentionally left blank.

5.4.7 Reactor Radiation Monitoring Systems

Proposed TS 4.7, "Reactor Radiation Monitoring Systems," states:

Specifications-

1. A channel test of the Facility Stack Continuous Air Monitor, Reactor Room Continuous Air Monitor, Reactor Room Radiation Area Monitor, and Demineralizer System Radiation Area Monitor (RAM) shall be performed monthly.

2. A channel calibration of the Facility Stack Continuous Air Monitor, Reactor Room Continuous Air Monitor, Reactor Room Radiation Area Monitor, and

Demineralizer System Radiation Area Monitor (RAM) shall be performed annually.

3. The environmental dosimeters shall be changed and evaluated at least quarterly.

The NRC staff reviewed proposed TS 4.7 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 4.7.1, "Monitoring Systems," and ANSI/ANS-15.1-2007, section 4.7.1, "Monitoring systems," to perform an operability check, or channel test, monthly, and perform a channel calibration annually, and sample the environmental dosimeters quarterly. Based on the information above, the NRC staff concludes that proposed TS 4.7 is acceptable.

5.4.8 Experiments

Proposed TS 4.8, "Experiments," states:

Specification -

1. The reactivity worth of an experiment shall be estimated or measured, as appropriate, before routine reactor operation with that experiment to ensure that the limits of TS 3.8.1 are not exceeded.

2. An experiment shall not be installed in the reactor or its irradiation facilities unless a safety analysis has been performed and reviewed for compliance with TS 3.8.2 and TS 3.8.3 by the Facility Director or NSC in full accord with TS 6.2.3, and the procedures which are established for this purpose.

The NRC staff reviewed proposed TS 4.8 and finds that it helps ensure that the requirements of TS 3.8 are satisfied prior to the performance of an experiment. The NRC staff notes that these surveillance requirements are not specifically described in the guidance in NUREG-1537, Part 1, or ANSI/ANS-15.1-2007, but the licensee found them useful and included them in proposed TS 4.8. Based on the information above, the NRC staff concludes that proposed TS 4.8 is acceptable.

5.5 Design Features

5.5.1 Site and Facility Description

Proposed TS 5.1, "Site and Facility Description," states:

Specification -

1. The site location is situated approximately 8 miles (13 km) north-by-northeast of downtown Sacramento, California on the former McClellan AFB.

2. The licensed area is that area inside of the fence surrounding the reactor building. This fence also demarcates the property that is owned by the University of California from the surrounding area and is approximately 2.3 acres in size. Inside of the licensed area is also a restricted area. The

unrestricted area is that area outside the fence surrounding the reactor building.

3. The reactor facility shall be equipped with a ventilation system designed to exhaust air and other gases from the reactor room/radiography bays and release them from a vertical level at least 60 feet above ground level

4. Emergency controls for the exhaust system shall be located in the reactor control room.

The NRC staff reviewed proposed TS 5.1 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 5, “Design Features,” and ANSI/ANS-15.1-2007, section 5.1, “Site and facility description,” which states that a general description of the site and facility, including the licensed and restricted areas, and features such as the ventilation system release point and emergency controls, are described. The NRC staff also finds that the descriptions provided in TS 5.1 are consistent with the descriptions provided in the licensee’s SAR for the site description. Based on the information above, the NRC staff concludes that proposed TS 5.1 is acceptable.

5.5.2 Reactor Coolant System

Proposed TS 5.2, “Reactor Coolant System,” states:

Specification -

1. During reactor operation the reactor core shall be cooled by a natural convection flow of water.

2. The tank water inlet pipe to the heat exchanger and to the demineralizer shall be equipped with a siphon break 16 feet above the top of the core or higher.

The NRC staff reviewed proposed TS 5.2 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 5 and ANSI/ANS-15.1-2007, section 5.2, “Reactor coolant system,” which states that the reactor coolant system should be described. The NRC staff also finds that the descriptions provided in TS 5.2 are consistent with the descriptions provided in the licensee’s SAR, as supplemented, for the reactor coolant system. Based on the information above, the NRC staff concludes that proposed TS 5.2 is acceptable.

5.5.3 Reactor Core and Fuel

5.5.3.1 Reactor Core

Proposed TS 5.3.1, “Reactor Core,” states:

Specifications-

1. The core shall be an arrangement of TRIGA uranium-zirconium hydride fuel-moderator elements positioned in the reactor grid plate.

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

3. The reflector, excluding experiments and irradiation facilities, shall be graphite. A reflector is not required if the core has been defueled.

The NRC staff reviewed proposed TS 5.3.1 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 5, and ANSI/ANS-15.1-2007, section 5.3, "Reactor core and fuel," which states that a description of the core configuration, type of fuel, fuel arrangement in the reactor core, and other significant core components should be provided. The NRC staff also finds that the descriptions provided in TS 5.3.1 are consistent with the descriptions provided in the licensee's SAR, as supplemented, for the reactor core and fuel. Based on the information above, the NRC staff concludes that proposed TS 5.3.1 is acceptable.

5.5.3.2 Reactor Fuel

Proposed TS 5.3.2, "Reactor Fuel," states:

Specification - The individual unirradiated TRIGA fuel elements shall have the following characteristics:

1. Uranium content: 20 or 30 wt % uranium enriched nominally to less than 20% U-235.

2. Hydrogen to zirconium atom ratio (in the ZrH_x): 1.60 to 1.70 (1.65+/- 0.05).

3. Cladding: stainless steel, nominal 0.5mm (0.020 inch) thick.

The NRC staff reviewed proposed TS 5.3.2 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 5, and ANSI/ANS-15.1-2007, section 5.3, which states that a description of the type of fuel authorized for use in the reactor be specified. The NRC staff review also finds that the descriptions provided in TS 5.3.2 are consistent with the descriptions provided in the licensee's SAR, as supplemented, for the fuel. Based on the information above, the NRC staff concludes that proposed TS 5.3.2 is acceptable.

5.5.3.3 Control Rods and Control Rod Drives

Proposed TS 5.3.3, "Control Rods and Control Rod Drives," states:

Specification -

1. All control rods shall have scram capability and contain a neutron poison such as stainless steel, borated graphite, B_4C powder, or boron and its compounds in solid form. The shim and regulating rods shall have fuel followers sealed in stainless steel. The transient rod shall have an air filled follower and be sealed in an aluminum tube.

2. The control rod drives shall be the standard GA rack and pinion type with an electromagnet and armature attached.

The NRC staff reviewed proposed TS 5.3.3 and finds that the description of the control rods and control rod drives are consistent with the descriptions provided in the licensee's SAR, as supplemented. Based on the information above, the NRC staff conclude that proposed TS 5.3.2 is acceptable.

5.5.4 Fissionable Material Storage

Proposed TS 5.4, "Fissionable Material Storage," states:

> Specification -
>
> 1. All fuel elements not in the reactor core shall be stored (wet or dry) in a geometrical array where the k_{eff} is less than 0.90 for all conditions of moderation.
>
> 2. Irradiated fuel elements shall be stored in an array which shall permit sufficient natural convection cooling by water or air such that the fuel element temperature shall not exceed the safety limit.
>
> 3. If stored in water, the water quality shall be maintained according to TS 3.3, Specification 2.

By letter dated September 22, 2021, the licensee provided supplemental information titled, "UCD Criticality safety analysis for MNRC spent fuel pits" (Ref. 69), the licensee uses the MCNP computer code to calculate the k-effective (k_{eff}) for the most limiting configuration of fuel (i.e., no burn-up, fully moderated water-flooded storage location), which resulted in a k_{eff} of 0.824. Further, the licensee states that the fuel is stored in dry (little moderation) conditions, which the NRC staff notes will significantly reduce the k_{eff}.

The NRC staff reviewed the licensee's analysis and finds that the k_{eff} are similar to other TRIGA research reactor facilities. The NRC staff also reviewed proposed TS 5.4 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 5 and ANSI/ANS-15.1-2007, section 5.4, "Fissionable material storage," which states that fuel, fueled experiments, and fuel devices are stored in a geometric array where k_{eff} is less than 0.90 for all conditions of moderation and reflection. Based on the information above, the NRC staff concludes that proposed TS 5.4 is acceptable.

5.6 Administrative Control

5.6.1 Organization

Proposed TS 6.1, "Organization," states:

> The Regents of the University of California shall be the licensee (license holder) for the UCD/MNRC. The Regents delegate the license holder duties to the UC Davis Chancellor who delegates the license holder duties to the Vice Chancellor of Research. The UCD/MNRC facility shall be under the direct control of the UCD/MNRC Director or a licensed Senior Reactor Operator (SRO) designated by the UCD/MNRC Director to be in direct control.

The NRC staff reviewed proposed TS 6.1 it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.1, “Organization,” and ANSI/ANS-15.1-2007, section 6.1, “Organization,” which states that the responsibilities should be specified. Based on the information above, the NRC staff concludes that proposed TS 6.1. is acceptable.

5.6.1.1 Structure

Proposed TS 6.1.1, Structure,” states:

> The UCD/MNRC management organization is shown in Figure 6.1.

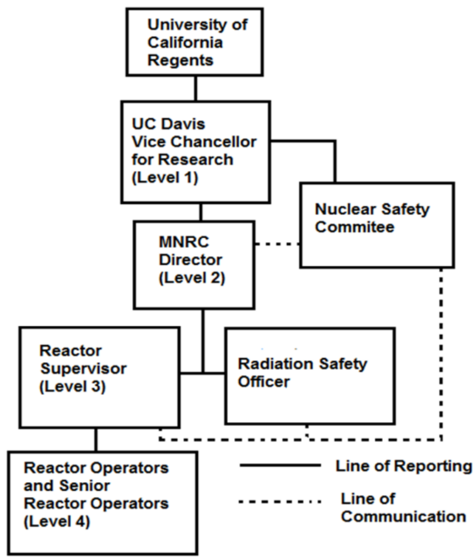


Figure 6.1 UC Davis McClellan Nuclear Research Center Organization for Reactor Operation, Licensing, and Safety

The NRC staff reviewed proposed TS 6.1.1 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.1.1, “Structure,” and ANSI/ANS-15.1-2007, section 6.1.1, “Structure,” which states, that the structure should follow the guidance in Figure 1 - Organization chart. Based on the information above, the NRC staff concludes that proposed TS 6.1.1 is acceptable.

5.6.1.2 Responsibilities

Proposed TS 6.1.2, “Responsibilities,” states:

> The UCD/MNRC Director shall be accountable to the Vice Chancellor for Research for the safe operation and maintenance of the MNRC facility. The UCD/MNRC Director, or his designated alternate, shall review and approve all experiments and experiment procedures prior to their use in the reactor. Individuals in the management organization (e.g., Reactor Supervisor and Radiation Safety Officer) shall be responsible for implementing UCD/MNRC policies and for operation of the facility, and shall be responsible for safeguarding the public and facility personnel from undue radiation exposures and for adhering to the operating license and technical specifications.

The following specific organizational levels and responsibilities shall exist:

1. Vice Chancellor for Research (Level 1): The Vice Chancellor for Research has the ultimate responsibility for the safe operation and maintenance of the MNRC. The Vice Chancellor for Research is also responsible for the facility license.

2. MNRC Director (Level 2): The UCD/MNRC facility shall be under the direct control of the UCD/MNRC Director. The UCD/MNRC Director, is responsible for the day-to-day operation of the facility. The UCD/MNRC Director is a direct report to the Vice Chancellor for Research.

3. Reactor Supervisor (Level 3): The Reactor Supervisor is a direct report to the MNRC Director. The reactor supervisor is responsible for directing the activities of the Reactor Operators and Senior Reactor Operators and for the day-to-day operation and maintenance of the reactor.

4. Reactor Operators and Senior Reactor Operators (Level 4): Senior Reactor Operators and Reactor Operators report to the Reactor Supervisor (or the MNRC Director) and are primarily involved in the direct manipulation of reactor controls, monitoring of instrumentation, and direct operation and maintenance of reactor-related equipment.

The NRC staff reviewed proposed TS 6.1.2 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.1.2, “Responsibility,” and ANSI/ANS-15.1-2007, section 6.1.2, “Responsibility,” which states that the review and audit group (Nuclear Safety Committee) should report to the Level 1, radiation safety personnel (Radiation Safety Officer) should report to Level 2 or higher, and the Levels 1-4 have the following responsibilities:

- Level 1, Individual responsible for the license, which is the UCD Vice Chancellor for Research;
- Level 2, Individual responsible for the reactor facility operation, which is the MNRC Director;
- Level 3, Individual responsible for day-to-day operation, which is the MNRC Reactor Supervisor; and
- Level 4, Operating staff, which are the UCD MNRC Reactor Operators and Senior Reactor Operators.

Based on the information above, the NRC staff concludes that proposed TS 6.1.2 is acceptable.

5.6.1.3 <u>Staffing</u>

Proposed TS 6.1.3, “Staffing,” states:

A. The minimum staffing when the reactor is not secured shall be:

1. A Licensed Operator in the control room;

2. A second person present within the MNRC facility who is able to carry out prescribed instructions;

3. If neither of these two individuals is a Senior Reactor Operator, a Senior Reactor Operator shall be readily available on call. Readily available on call means an individual who:

   i. Has been specifically designated and the designation is known to the operator on duty;

   ii. Can be contacted by phone, within 5 minutes, by the operator on duty; and

   iii. Is capable of getting to the reactor facility within a reasonable time under normal conditions (e.g., 30 minutes or within a 15-mile radius).

4. A list of management personnel, radiation personnel, and reactor staff along with their contact information shall be available to the operator on duty.

B. Events requiring the direction of a Senior Reactor Operator:

1. Initial approach to critical after each completed shutdown checklist;

2. Initial approach to power after each completed shutdown checklist;

3. All fuel or control rod relocations within the reactor core region;

4. Relocation of any in-core components (other than normal control rod movements) or experiment with a reactivity worth greater than one dollar; or

5. Recovery from an unscheduled shutdown or an unscheduled significant (>50%) power reduction.

The NRC staff reviewed proposed TS 6.1.3 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.1.3, “Staffing,” and ANSI/ANS-15.1-2007, section 6.1.3, “Staffing,” which provides the minimum staffing levels for the reactor when it is not secured, the requirements of the second person at the facility and the on-call person who is not at the facility, the on-call list of management, radiation and reactor personnel, and the list of events which require the presence of the senior reactor operator. Based on the information above, the NRC staff concludes that proposed TS 6.1.3 is acceptable.

5.6.1.4 <u>Selection and Training of Personnel</u>

Proposed TS 6.1.4, “Selection and Training of Personnel,” states:

> The selection, training and requalification of operations personnel shall meet or exceed the requirements of the American National Standard for Selection and Training of Personnel for Research Reactors ANSI/ANS 15.4-2016.

The NRC staff reviewed proposed TS 6.1.4 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.1.4, “Selection and Training of Personnel,” and ANSI/ANS-15.1-2007, section 6.1.4, “Selection and training of personnel,” which states that the selection, training, and requalification of operations personnel should meet or exceed the requirements of ANSI/ANS-15.4, “Selection and Training of Personnel for Research Reactors.” Based on the information above, the NRC staff concludes that proposed TS 6.1.4 is acceptable.

5.6.2 Review and Audit

Proposed TS 6.2, ‘Review and Audit,” states:

> The Nuclear Safety Committee (NSC) has been chartered to assist in meeting this responsibility by providing timely, objective, and independent reviews, audits and recommendations on matters affecting nuclear safety. The following describes the composition and conduct of the NSC.

The NRC staff reviewed proposed TS 6.2 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.2, “Review and Audit,” and ANSI/ANS-15.1-2007, section 6.2, “Review and audit,” which states that a method for independent review and audit of the safety aspects of the reactor facility operation should be established, and the responsible committee’s authority should be specified. Based on the information above, the NRC staff concludes that proposed TS 6.2 is acceptable.

5.6.2.1 Composition and Qualifications

Proposed TS 6.2.1, “Composition and Qualification,” states:

> The NSC shall be composed of at least four voting members, including the Chairperson. All members of the Committee shall be knowledgeable in subject matter related to reactor operations. To expedite Committee business, a Committee Chairperson shall be appointed. The Committee shall be appointed by the Vice Chancellor for Research. No definite term of service shall be specified; but should a vacancy occur in the Committee, the Vice Chancellor for Research shall appoint a replacement. The remaining members of the Committee shall be available to assist the Vice Chancellor for Research in the selection of new members. The Reactor Supervisor and the radiation safety officer shall be ex-officio members of the Committee. The NSC advises the MNRC Director and shall report any concerns to the Vice Chancellor for Research.

The NRC staff reviewed proposed TS 6.2.1 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.2.1, “Composition and Qualifications,” and ANSI/ANS-15.1-2007, section 6.2.1, “Composition and qualifications,” which states that the review and audit group should be composed of a minimum of three members if a single group is used; the members should collectively represent a broad spectrum of expertise in the appropriate reactor technology; members and alternates should be appointed by and report to Level I management; and individuals may be either from within or outside the operating organization. Based on the information above, the NRC staff concludes that proposed TS 6.2.1 is acceptable.

5.6.2.2 NSC Charter and Rules

Proposed TS 6.2.2, “NSC Charter and Rules,” states:

> The NSC consists of MNRC members and non-MNRC members, and the Committee shall meet at least annually.
>
> The review and audit functions shall be conducted in accordance with an established charter for the Committee. Dissemination and review of Committee minutes shall be done within 60 days of each respective Committee meeting.
>
> A quorum for review, audit, and approval purposes shall consist of not less than one-half of the voting membership where the operating staff does not constitute a majority. The Chairperson or an alternate must be present at all meetings in which the official business of the committee is being conducted. Approvals by the committee shall require an affirmative vote by a majority of the non-MNRC members present and an affirmative vote by a majority of the MNRC members present.

The NRC staff reviewed proposed TS 6.2.2 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.2.2, “Charter and Rules,” and ANSI/ANS-15.1-2007, section 6.2.2, “Charter and rules,” which states that the meeting frequency will not be less than once per year; a quorum will consist of not less than one half of the voting membership, where the operating staff does not constitute a majority; and the meeting minutes should be reviewed and approved in a timely manner. Based on the information above, the NRC staff concluded that proposed TS 6.2.2 is acceptable.

5.6.2.3 Review Function

Proposed TS 6.2.3, “Review Function,” states:

> The following items shall be reviewed:
>
> 1. Determinations that proposed changes in equipment, systems, test, experiments, or procedures are allowed without prior authorization by the NRC as detailed in 10 CFR 50.59;
>
> 2. All new procedures and major revisions thereto having safety significance, proposed changes in reactor facility equipment, or systems having safety significance;
>
> 3. All new experiments or classes of experiments that could affect reactivity or result in the release of radioactivity;
>
> 4. Proposed changes in technical specifications, license, or charter;
>
> 5. Violations of technical specifications, license, or charter. Violations of internal procedures or instructions having safety significance;
>
> 6. Operating abnormalities having safety significance;
>
> 7. Reportable occurrences listed in Sec. 6.7.2;

8. Audit reports.

A written report or minutes of the findings and recommendations of the review group shall be submitted to the Vice Chancellor for Research and the NSC members within 3 months after the review has been completed.

The NRC staff reviewed proposed TS 6.2.3 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.2.3, “Review Function,” and ANSI/ANS-15.1-2007, section 6.2.3, “Review function,” which states that the review function required by 10 CFR 50.59, “Changes, tests and experiments,” is explicitly stated (proposed TS 6.2.3.1); all new procedures (proposed TS 6.2.3.2); all new experiments (proposed TS 6.2.3.3); proposed changes to the TSs (proposed TS 6.2.3.4; violations of TSs or the license (proposed TS 6.2.3.5); operating abnormalities (proposed TS 6.2.3.6); reportable occurrences (proposed TS 6.2.3.7); audit reports (proposed TS 6.2.3.8); and a written report submitted to the Level 1 in a timely manner. Based on the information above, the NRC staff concludes that proposed TS 6.2.3 is acceptable.

5.6.2.4 Audit Function

Proposed TS 6.2.4, “Audit Function,” states:

The audit function shall include selective (but comprehensive) examination of operating records, logs, and other documents. Discussions with cognizant personnel and observation of operations should be used also as appropriate. In no case shall the individual immediately responsible for the area perform an audit in that area. The following items shall be audited:

1. Facility operations for conformance to the technical specifications and applicable license or charter conditions: at least once per calendar year (interval between audits not to exceed 15 months);

2. The retraining and requalification program for the operating staff: at least once every other calendar year (interval between audits not to exceed 30 months);

3. The results of action taken to correct those deficiencies that may occur in the reactor facility equipment, systems, structures, or methods of operations that affect reactor safety: at least once per calendar year (interval between audits not to exceed 15 months);

4. The reactor facility emergency plan and implementing procedures: at least once every other calendar year (interval between audits not to exceed 30 months);

5. The reactor security plan and implementing procedures: at least once every calendar year.

Deficiencies uncovered that affect reactor safety shall immediately be reported to the Vice Chancellor for Research. A written report of the findings of the audit shall be submitted to the Vice Chancellor for Research and the NSC within 3 months after the audit has been completed.

The NRC staff reviewed proposed TS 6.2.4 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.2.4, “Audit Function,” and ANSI/ANS-15.1-2007, section 6.2.4, “Audit function,” which states that the audit function will include an examination of operating records, with the individual responsible for the audit area not involved with the audit; facility operations for conformance to the TSs; retraining and requalification of operating staff; results of past corrective actions; the emergency and security plans; and any deficiencies should be immediately report to the Level 1. Based on the information above, the NRC staff concludes that proposed TS 6.2.4 is acceptable.

5.6.3 Radiation Safety

Proposed TS 6.3, “Radiation Safety,” states:

> The radiation safety officer shall be responsible for implementation of the radiation safety program. The requirements of the radiation safety program are established in 10 CFR 20. The program should use the guidelines of the ANSI/ANS 15.11, “Radiation Protection at Research Reactor Facilities." The radiation safety officer reports directly to the Director.

The NRC staff reviewed proposed TS 6.3 finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.3, “Radiation Safety,” and ANSI/ANS-15.1-2007, section 6.3, “Radiation safety,” which states that the individual responsible for implementation of the radiation safety program should use the guidance in ANSI/ANS-15.11, and should report to the Level 1 or 2. Based on the information above, the NRC staff concludes that proposed TS 6.3 is acceptable.

5.6.4 Procedures

Proposed TS 6.4, “Procedures,” states:

> Written procedures shall be prepared, reviewed, and approved prior to initiating any of the activities listed in this section. The procedures shall be reviewed by the NSC (as applicable) and approved by Director or designated alternates, and such reviews and approvals shall be documented in a timely manner. Minor modifications to the original procedures that do not change their original intent may be made by the Reactor Supervisor. Temporary deviations from the procedures may be made by the Reactor Supervisor in order to deal with special or unusual circumstances or conditions. Such deviations shall be documented and reported within 24 hours or the next working day to the director or designated alternates. Procedures shall be in effect and in use for the following items:
>
> 1. startup, operation, and shutdown of the reactor;
>
> 2. fuel loading, unloading, and movement within the reactor;
>
> 3. maintenance of major components of systems that could have an effect on reactor safety;
>
> 4. surveillance checks, calibrations, and inspections required by the technical specifications or those that may have an effect on reactor safety;

5. personnel radiation protection, consistent with applicable regulations or guidelines. The procedures shall include management commitment and programs to maintain exposures and releases as low as reasonably achievable in accordance with the guidelines of ANSI/ANS-15.11;

6. administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect reactor safety or core reactivity;

7. implementation of emergency or security plans;

8. use, receipt, and transfer of by-product material, if appropriate.

The NRC staff reviewed proposed TS 6.4 and finds that it follows the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.4, “Procedures,” and ANSI/ANS-15.1-2007, section 6.4, “Procedures,” which states that written procedures should be prepared and approved prior to use by the Level 2, for: operation of the reactor; fuel movement; maintenance of major components; surveillances required by the TSs; personnel radiation protection; administrative controls for operations and experiments; implementation of the security and emergency plans; and use of by-product material. Based on the information above, the NRC staff concludes that proposed TS 6.4 is acceptable.

5.6.5 Experiment Review and Approval

Proposed TS 6.5, “Experiment Review and Approval,” states:

> Approved experiments shall be carried out in accordance with established and approved procedures.
>
> 1. All new experiments or class of experiments shall be reviewed by the NSC and approved in writing by Director or designated alternates prior to initiation;
>
> 2. Substantive changes to previously approved experiments shall be made only after review by the NSC and approved in writing by Director or designated alternates. Minor changes that do not significantly alter the experiment may be approved by reactor supervisor.

The NRC staff reviewed proposed TS 6.5 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.5, “Experiments Review and Approval,” and ANSI/ANS-15.1-2007, section 6.5, “Experiments review and approval,” which states that experiments should be conducted in accordance with approved procedures; all new experiments should be reviewed by the Level 2; substantive changes to previously approved procedures should be reviewed by the review group and approved by the Level 2; and minor changes may be made by the Level 3. Based on the information above, the NRC staff concludes that proposed TS 6.5 is acceptable.

5.6.6 Required Actions

5.6.6.1 Action to be taken in case of a safety limit violation

Proposed TS 6.6.1, "Action to be taken in case of a safety limit violation," states:

> In the event of a safety limit violation (fuel temperature), the following action shall be taken:
>
> 1. The reactor shall be shut down and reactor operation shall not be resumed until authorized by the NRC.
>
> 2. The safety limit violation shall be promptly reported to the UCD/MNRC Director.
>
> 3. The safety limit violation shall be reported to the Chairperson of the NSC and to the NRC by the UCD/MNRC Director.
>
> 4. A safety limit violation report shall be prepared. The report shall describe the following:
>
>   a. Applicable circumstances leading to the violation, including when known, the cause and contributing factors.
>
>   b. Effect of the violation upon reactor facility components, systems, or structures, and on the health and safety of personnel and the public.
>
>   c. Corrective action to be taken to prevent recurrence.
>
> The safety limit violation report shall be reviewed by the NSC and then be submitted to the NRC when authorization is sought to resume operation of the reactor.

The NRC staff reviewed proposed TS 6.6.1 that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.6.1, "Action To Be Taken in Case of Safety Limit Violation," and ANSI/ANS-15.1-2007, section 6.6.1, "Action to be taken in case of a safety limit violation," which states that in the event of a SL violation, the reactor shall be shut down and not operated until approved by the NRC; the SL violation shall be promptly reported to the Level 2 and NRC; a report shall be prepared describing the violation and corrective actions; and SL violation will be reviewed by the review group and submitted to the NRC when authorization for operation is requested. Based on the information above, the NRC staff concludes that proposed TS 6.6.1 is acceptable.

5.6.6.2 Action to be taken in the event of an occurrence of the type identified in Secs. 6.7.2 other than a safety limit violation

Proposed TS 6.6.2, "Action to be take in the event of an occurrence of the type identified in Secs. 6.7.2 other than a safety limit violation," states:

> 1. Reactor conditions shall be returned to normal or the reactor shall be shut down. If it is necessary to shut down the reactor to correct the occurrence, operations

shall not be resumed unless authorized by the UCD/MNRC Director or his designated alternate.

2. The occurrence shall be reported to the UCD/MNRC Director or the designated alternate. The UCD/MNRC Director shall report the occurrence to the NRC.

3. Occurrence shall be reviewed by the NSC at its next scheduled meeting.

The NRC staff reviewed proposed TS 6.6.2 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.6.2, “Action To Be Taken in the Event of an Occurrence of the Type Identified in Sections 6.7.2(1)(b) and 6.7.2(1)(c),” and ANSI/ANS-15.1-2007, section 6.6.2, “Action to be taken in the event of an occurrence of the type identified in Secs. 6.7.2(1)(b) and 6.7.2(1)(c),” and therefore, is acceptable.

5.6.7 Reports

5.6.7.1 Annual Operating Reports

Proposed TS 6.7.1, “Annual Operating Reports,” states:

> An annual report covering the previous calendar year shall be created and submitted, no later than May 31 of the year following the report period to the NRC consisting of:
>
> 1. A brief summary of operating experience including the energy produced by the reactor and the hours the reactor was critical;
>
> 2. The number of unplanned shutdowns, including corrective actions taken (when applicable);
>
> 3. A tabulation of major preventative and corrective maintenance operations having safety significance;
>
> 4. A brief description, including a summary of the safety evaluations, of changes in the facility or in procedures and of tests and experiments carried out pursuant to 10 CFR 50.59;
>
> 5. A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as measured at or prior to the point of such release or discharge. The summary shall include to the extent practicable an estimate of individual radionuclides present in the effluent. If the estimated average release after dilution or diffusion is less than 25% of the concentration allowed or recommended, a statement to this effect is sufficient;
>
> 6. A summarized result of environmental surveys performed outside the facility; and;
>
> 7. A summary of exposures received by facility personnel and visitors where such exposures are greater than 25% of that allowed.

The NRC staff reviewed proposed TS 6.7.1 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.7.1, "Operating Reports," and ANSI/ANS-15.1-2007, section 6.7.1, "Operating reports," and therefore, is acceptable.

5.6.7.2 Special Reports

Proposed TS 6.7.2, "Special Report," states:

> In addition to the requirements of applicable regulations, and in no way substituting therefore, reports shall be made to the NRC as follows:
>
> 1. A report within 24 hours by telephone, confirmed by digital submission or fax to the NRC Operations Center and followed by a report in writing to the NRC, Document Control Desk, Washington, D.C. within 14 days that describes the circumstances associated with any of the following:
>
>   a. Any violation of a safety limit;
>
>   b. Any release of radioactivity above applicable limits into unrestricted areas, whether or not the release resulted in property damage, personal injury, or exposure;
>
>   c. Operation with the actual safety system setting less conservative than the LSSS;
>
>   d. Operation in violation of a Limiting Condition for Operation;
>
>   e. Malfunction of a required reactor safety system component which renders or could render the system incapable of performing its intended safety function unless the malfunction or condition is caused by maintenance, then no report is required;
>
>   f. Any unanticipated or uncontrolled change in reactivity greater than $1.00. Reactor trips resulting from a known cause are excluded;
>
>   g. An observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy causes or could have caused the existence or development of a condition which results or could result in operation of the reactor outside the specified safety limits; or
>
>   h. Abnormal and significant degradation in reactor fuel, cladding, or coolant boundary.
>
> 2. A report within 30 days in writing to the NRC, Document Control Desk, Washington, D.C. of:
>
>   a. Permanent changes in the facility organization involving Level 1-2 personnel; or
>
>   b. Significant changes in the transient or accident analyses as described in the SAR.

The NRC staff reviewed proposed TS 6.7.2 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.7.2, “Special Reports,” and ANSI/ANS-15.1-2007, section 6.7.2, “Special reports,” and therefore, is acceptable.

5.6.8 Records

5.6.8.1 Records to be Retained for a Period of at Least Five Years or for the Life of the Component Involved if Less than Five Years

Proposed TS 6.8.1, “Records to be Retained for a Period of at Least Five Years or for the Life of the Component Involved if Less than Five Years,” states:

1. Normal reactor facility operation (but not including supporting documents such as checklists, data sheets, etc., which shall be maintained for a period of at least two years);

2. Principal maintenance activities;

3. Reportable occurrences;

4. Surveillance activities required by the Technical Specifications;

5. Reactor facility radiation and contamination surveys;

6. Experiments performed with the reactor;

7. Fuel inventories, receipts, and shipments;

8. Approved changes to the operating procedures; and

9. NSC meetings and audit reports.

The NRC staff reviewed proposed TS 6.8.1 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.8, “Records,” and ANSI/ANS-15.1-2007, section 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,” and therefore, is acceptable.

5.6.8.2 Records to be Retained for at Least one Operator License Term

Proposed TS 6.8.2, “Records to be Retained for at Least One Operator License Term,” states:

1. Records of retraining and requalification of Reactor Operators and Senior Reactor Operators shall be retained for at least one license term; and

2. Records of retraining and requalification of licensed operators shall be maintained while the individual is employed by the licensee, or until that operator’s license is renewed, whichever is shorter.

The NRC staff reviewed proposed TS 6.8.2 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.8 and ANSI/ANS-15.1-2007, section 6.8.2, "Records to be retained for at least one certification cycle," and therefore, is acceptable.

5.6.8.3 Records to be Retained for the Lifetime of the Reactor Facility

Proposed TS 6.8.3," Records to be Retained for the Lifetime of the Reactor Facility," states:

1. Gaseous and liquid radioactive effluents released to the environs;

2. Offsite environmental monitoring surveys as required by Technical Specifications;

3. Radiation exposures for all personnel monitored; and

4. Drawings of the reactor facility.

The NRC staff reviewed proposed TS 6.8.3 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.8 and ANSI/ANS-15.1-2007, section 6.8.3, "Records to be retained for the lifetime of the reactor facility," and therefore, is acceptable.

5.7 Conclusions

The NRC staff reviewed and evaluated the proposed TSs as part of its review of the license renewal application for Facility Operating License No. R-130, NRC Docket No. 50-607. The proposed TSs define certain features, characteristics, organizational, reporting requirements, and conditions governing the operation of the UCD MNRC facility. The proposed TSs are included in the renewed license as appendix A. The NRC staff reviewed and evaluated the content of the proposed TSs to determine whether they met the requirements in 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, Part 1 and ANSI/ANS-15.1-2007. The NRC staff finds that the proposed TSs are consistent with the guidance. The NRC staff concludes that the UCD/MNRC proposed TSs are acceptable for following reasons:

- To satisfy the requirements of 10 CFR 50.36(a)(1), the licensee provided proposed TSs with the license renewal application. As required by the regulation, a summary statement of the bases or reasons for the TSs were submitted. The summary bases are included in the TSs, but are not part of the TSs as required by 10 CFR 50.36(a)(1).

- The UCD/MNRC is a facility of the type described in 10 CFR 50.21 "Class 104 licenses; for medical therapy and research and development facilities," paragraph (c); therefore, 10 CFR 50.36(b), requires that the facility operating license include TSs. To satisfy the requirements of 10 CFR 50.36(b), the licensee provided proposed TSs derived from analyses in the SAR, as supplemented.

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

- The proposed TS specify SLs on the fuel temperature and an LSSS for the reactor protection system to preclude reaching the SLs and satisfy 10 CFR 50.36(c)(1) requirements.

- The proposed TSs contain limiting conditions for operation on each item that meets one or more of the criteria specified in 10 CFR 50.36(c)(2)(ii).

- The 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 for 10 CFR 50.36(c)(5). The proposed administrative controls contain requirements for initial notification, written reports, and records that satisfy 10 CFR 50.36(c)(1), (2), and (7); and the NRC staff deemed necessary in accordance with 10 CFR 50.36(c)(8).

The NRC staff reviewed the proposed TSs and finds that they are acceptable. The NRC staff concludes that normal operation of the UCD/MNRC 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 public or for the UCD MNRC staff. The NRC staff concludes that the proposed TSs provide reasonable assurance that the UCD MNRC will be operated as analyzed in the SAR, as supplemented; that adherence to the proposed TSs during the license renewal period will limit the likelihood of malfunctions and the potential accident scenarios discussed in SER chapter 4; and that the conduct of activities by the licensee will not endanger the facility staff or members of the public.

6 CONCLUSIONS

On the basis of its evaluation of the application for license renewal as discussed in the previous chapters of this safety evaluation report, the U.S. Nuclear Regulatory Commission (NRC) staff concludes the following:

- The application for license renewal dated June 11, 2018, as supplemented on July 6, 2020; January 25, September 22, and December 17, 2021; and January 17, March 30, June 3, June 21, and June 30, 2022, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (AEA) and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations* (10 CFR).

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

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

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

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

- The applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," have been satisfied.

- The issuance of this license is in accordance with 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," of the NRC's regulations and all applicable requirements have been satisfied.

- The receipt, possession and use of byproduct and special nuclear materials as authorized by this facility operating license will be in accordance with the NRC's regulations in 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material."

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

7 REFERENCES

1. University of California – Davis, Renewal of Facility Operating License No. R-130, Regents of the University of California, McClellan Nuclear Research Center Training, Research, Isotope, General Atomics, June 11, 2018, Agencywide Documents Access and Management System (ADAMS) Accession No. ML18179A500 (package).

2. University of California – Davis, McClellan Nuclear Research Center (UCD/MNRC) Requested Changes to Facility License Renewal Application, May 10, 2019, ML19132A147.

3. University of California – Davis, Updated UCD/MNRC [McClellan Nuclear Research Center] License (R-130) Renewal Application Package Docket Number 50-607, July 6, 2020, ML20188A367 (package); ML20188A368 (cover letter).

4. University of California – Davis, University of California at Davis License Renewal Application Updated Safety Analysis Report (redacted), July 6, 2020, ML20238B984.

5. University of California – Davis, University of California Davis McClellan Nuclear Research Center Financial Qualification Report, June 10, 2020, ML20188A370.

6. University of California – Davis, University of California at Davis License Renewal Application Environmental Report (redacted), July 6, 2020, ML20238B993.

7. University of California – Davis, UCD/MNRC-0004-DOC-14, "Technical Specifications for the University of California - Davis McClellan Nuclear Radiation Center (UCD/MNRC)," July 6, 2020, ML20188A371.

8. University of California – Davis, UCD/MNRC-0009-DOC-04, "University of California - Davis/Mcclellan Nuclear Radiation Center Selection and Training Plan for Reactor Personnel," January 12, 2000, ML20188A374.

9. University of California – Davis, University of California at Davis License Renewal Application Emergency Plan (redacted), dated June 21, 2018, ML20238B990.

10. U.S. Nuclear Regulatory Commission, NUREG-1630, "Safety Evaluation Report Related to the Issuance of a Facility Operating License for the Research Reactor at McClellan Air Force Base," August 13, 1998, ML053210295.

11. U.S. Nuclear Regulatory Commission, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," NUREG-1537, Parts 1 and 2, February 1996, ML042430055 and ML042430048.

12. Department of the Air Force, Memorandum for U.S. Nuclear Regulatory Commission, RE: USAF request to convey the McClellan Nuclear Research Center to the Regents of the University of California at Davis, April 13, 1999, ML20205Q191.

13. U.S. Nuclear Regulatory Commission, "Re-issuance of Notice of Consideration of Approval of Transfer of Facility Operating License No. R-130 for the McClelland Nuclear

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