

Safety Evaluation Report Related to the Renewal of
the Facility Operating License for the TRIGA Nuclear Reactor at the
University of Utah

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ABSTRACT

This safety evaluation report summarizes the findings of a safety review conducted by the staff of the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Reactor Regulation. The NRC staff conducted this review in response to an application filed by the University of Utah (the licensee) for a 20-year renewal of Facility Operating License No. R-126 to continue to operate the University of Utah TRIGA Nuclear Reactor. 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, and inspection reports prepared by NRC personnel, as well as that gained by firsthand observations. On the basis of its review, the NRC staff concludes that the University of Utah can continue to operate the facility for the term of the renewed facility operating license, in accordance with the license, without endangering public health and safety, facility personnel, or the environment.

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ABBREVIATIONS AND ACRONYMS

ADAMS	Agencywide Documents Access and Management System
AEA	Atomic Energy Act of 1954, as amended
ALARA	as low as reasonably achievable
ALI	annual limit on intake
Am-Be	americium-beryllium
ANL	Argonne National Laboratory
ANS	American Nuclear Society
ANSI	American National Standards Institute
ARI	all rods in
ARI-1	all rods in with the strongest rod withdrawn
ARM	area radiation monitor
CAM	continuous air monitor
CFR	<i>Code of Federal Regulations</i>
CHF	critical heat flux
DAC	derived air concentration
DCF	dose conversion factors
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DOE	U.S. Department of Energy
DPR	Division of Policy and Rulemaking
EP	emergency plan
EPA	U.S. Environmental Protection Agency
ERI	Energy Research, Inc.
FGR	Federal Guidance Report
FNIF	Fast Neutron Irradiation Facility
FTC	fuel temperature coefficient
GA	General Atomics
He	helium
HEPA	high-efficiency particulate air
HVAC	heating ventilation and air conditioning

IFE	instrumented fuel elements
ISG	interim staff guidance
LCC	limiting core configuration
LCO	limiting condition for operation
LEU	low-enriched uranium
LOCA	loss-of-coolant accident
LRA	license renewal application
LSSS	limiting safety system setting
MCNP5	Monte Carlo N-Particle Transport Code
MEB	Merrill Engineering Building
MEHL	Mechanical Engineering Heat Lab
MHA	maximum hypothetical accident
NaI	sodium iodine
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
PDR	Public Document Room
PI	pneumatic irradiator
PRLB	Research and Test Reactors Licensing Branch
PTS	pneumatic transfer system
Pu-Be	plutonium-beryllium
RAI	request for additional information
RG	regulatory guide
RHD	Radiological Health Department
RO	reactor operator
RS	reactor supervisor
RSC	Reactor Safety Committee
RSO	radiation safety officer
RTRs	research and test reactors
SAR	safety analysis report
SDM	shutdown margin
SER	safety evaluation report
SL	safety limit

SNM	special nuclear material
SOI	statement of intent
SRM	staff requirements memorandum
SRO	senior reactor operator
TEDE	total effective dose equivalent
TI	thermal irradiator
TLD	thermoluminescence dosimeter
TS	technical specifications
U	uranium
U-Zr	uranium-zirconium
UNEF	Utah Nuclear Engineering Facility
UNEP	Utah Nuclear Engineering Program
U-235	uranium isotope 235
UU	University of Utah
UUTR	University of Utah TRIGA Nuclear Reactor
ZrH	zirconium hydride

TECHNICAL PARAMETERS AND UNITS

\$	a unit of reactivity where absolute reactivity is divided by the total effective delayed neutron fraction β_{eff}
\$/%	reactivity in \$ per % void in coolant or moderator
% burnup	the change in fuel composition due to depletion expressed as a % of the original U-235 content
°C	temperature in degrees Celsius
μmhos	micromhos
ARI-1	all rods in minus the strongest rod (hypothetical stuck rod used for shutdown margin calculations)
C ₀	heat capacity of core at 0 °C for Fuchs-Nordheim analysis
C ₁	heat capacity as a function of temperature for Fuchs-Nordheim analysis
CFM	cubic feet per minute
Ci	curies
cm	centimeter
cps	counts per second
cm ³ /s	cubic centimeters per second
dpm	disintegrations per minute
dpm/100 cm ²	disintegrations per minute per 100 square centimeters
ft	feet
g	gram
h	hour
Hz	hertz
K	temperature in kelvin
kg	kilograms
kW	kilowatts
m	meter
MPa	megapascals
MWd	megawatt days
MW(t)	megawatt thermal
pH	potential of hydrogen
psi	pounds per square inch
rem	Roentgen Equivalent Man
mrem	millirem

s	second
W	watts
w%	weight percent
yr	year
α_F	fuel temperature coefficient
α_M	moderator temperature coefficient
α_V	moderator void coefficient
β_{eff}	effective delayed neutron fraction
$\Delta k/k$	absolute reactivity
ρ_{EXP}	reactivity of the experiments
ρ_F	excess reactivity of the fuel
ρ_F	worth of the safety control rod
ρ_{FUEL}	reactivity of the fuel
ρ_R	worth of the regulating control rod
ρ_{SDM}	reactivity of the reactor system for comparison to the shutdown margin requirement
ρ_{SH}	worth of the shim control rod
ρ_S	reactivity of the safety rod
ρ_X	total excess reactivity
X/Q	atmospheric relative concentration, s-cm ³

1. INTRODUCTION

1.1 Overview

By a letter dated March 25, 2005, as supplemented on June 1, 2009; February 9, March 10, May 13, May 27, October 4, 2010 (two documents); and June 8, July 15, August 23, and August 31, 2011, the University of Utah (UU, or the licensee) submitted to the U.S. Nuclear Regulatory Commission (NRC or the Commission) an application for a 20-year renewal of the Class 104c Facility Operating License No. R-126, Docket No. 50-407, for the UU TRIGA Nuclear Reactor (UUTR).

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.51(a) states that each license will be issued for a period of time to be specified in the license but in no case to exceed 40 years from the date of issuance. UU is the holder of Facility Operating License No. R-126 (the license), which was originally issued on September 30, 1975, for a period of 10 years from the issuance of the construction permit on April 24, 1973. The license was renewed on April 17, 1985, for a period of 20 years, with an expiration of April 17, 2005. UU submitted its renewal application, "License Renewal and Power Up Rate of the University of Utah Nuclear Reactor Facility," dated March 25, 2005 (Ref. 2), 24 days before the license expiration, which exceeded the 30 days required to maintain the timely renewal provision contained in 10 CFR 2.109, "Effect of Timely Renewal Application," item (a). UUTR requested an exemption to the 30-day requirement in a letter dated April 13, 2005. The Commission approved the exemption request in a letter dated April 15, 2005. Because the exemption request restored the timely renewal provision contained in 10 CFR 2.109(a) to UUTR, the licensee was permitted to continue operation of UUTR under the terms and conditions of the current license until the NRC staff completes action on the renewal request. A renewal would authorize the licensee to continue operation of UUTR for an additional 20 years.

UUTR was licensed in 1975 at a maximum steady-state power level of 100 kilowatts (kW) as a teaching and research facility. UUTR is not licensed to have reactor power pulsing capability. Some of the reactor components, such as the control rods, control console, and most of the fuel elements, were operated in an NRC licensed reactor at the University of Arizona from 1958 to 1971. In 1971, these components were transferred to UU and installed in their current location within the Merrill Engineering Building (MEB) on the UU campus. NUREG-1096, "Safety Evaluation Report Related to the Renewal of the Operating License for the TRIGA Training and Research Reactor at the University of Utah," issued March 1985 (Ref. 3), contains more details on the earlier operation of the UUTR.

The initial license renewal application (LRA) was submitted on March 25, 2005. The LRA also requested an increase in the licensed power level from 100 kW thermal (kW(t)) to 250 kW(t). During the NRC staff review, UUTR withdrew the power uprate portion of the LRA in its letter, "Request for Postponing Power Uprate of the UUTR Reactor," dated May 27, 2010 (Ref. 4). The NRC staff acknowledged this request in a letter dated July 6, 2010.

The NRC staff conducted its review, with respect to renewing the UUTR facility operating license, on the basis of information contained in the LRA, as well as in supporting supplements in response to the NRC staff's request for additional information (RAI). Specifically, the initial LRA dated March 25, 2005, included a SAR with technical specifications (TSs), an environmental report (Ref. 5), an emergency plan (EP), and an operator requalification plan (Ref. 6). Through the review process, the licensee submitted their RAI responses along with an updated SAR. UUTR TSs were subsequently updated after the June 8, 2011 SAR, and provided separately via letter dated July 15, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML11207A429). Clarifying information was provided by a letter dated August 23, 2011 (ADAMS Accession No. ML11249A053), and an email response dated August 31, 2011 (ADAMS Accession No. ML112490384).

The NRC review also included information from UUTR annual reports covering the years 2004 to 2010, and NRC inspection reports (IRs) covering the years 2003 through 2010. Site visits were conducted December 9, 2009, March 10, 2010, August 24 and 25, 2010, December 7, 2010, May 4, 2011, and, July 13, 2011, to observe facility conditions.

With the exception of the security plan and the EP, material pertaining to this review may be examined or copied, for a fee, at the NRC's Public Document Room (PDR), located at One White Flint North, 11555 Rockville Pike, Rockville, MD. The NRC maintains 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 there are problems in accessing the documents located in ADAMS, contact the NRC PDR staff by telephone at 1-800-397-4209 or 301-415-4737, or send an e-mail to the PDR at PDR_Resources@nrc.gov. The physical security plan is protected from public disclosure under 10 CFR 73.21, "Requirements for the Protection of Safeguards Information," and the EP is withheld from public disclosure, as it is considered security-related information. Because parts of the SAR and RAI responses from the licensee contain security-related information and are protected from public disclosure, redacted versions are available.

The Reference Section contains the dates and associated ADAMS accession numbers of the licensee's renewal application and associated supplements.

In conducting its safety 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;" and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material." The NRC staff also considered recommendations of applicable regulatory guides (RGs) and relevant accepted industry standards, such as the American National Standards Institute/American Nuclear Society (ANSI/ANS) 15 series. The NRC staff specifically referred to the recommendations provided in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," issued February 1996 (Ref. 7). Because there are no specific accident-related regulations for research reactors, the NRC staff compared calculated dose values for accidents against the requirements in 10 CFR Part 20 (i.e., the standards for protecting employees and the public against radiation).

In SECY-08-0161, "Review of Research and Test Reactor License Renewal Applications," dated October 24, 2008 (Ref. 8), the NRC staff provided the Commission with information regarding plans to revise the review of LRAs for research and test reactors (RTRs). The Commission issued its staff requirements memorandum (SRM) for SECY-08-0161 on March 26, 2009 (Ref. 9). The SRM directed the NRC staff to streamline the renewal process for such reactors, using some combination of the options presented in SECY-08-0161. The SRM also directs the NRC staff to implement a graded approach with a scope commensurate with the risk posed by each facility. The graded approach incorporates elements of the alternative safety review approach discussed in Enclosure 1 of SECY-08-0161. In the alternative safety review approach, the NRC staff should consider the results of past NRC staff reviews when determining the scope of the review. A basic requirement, as contained in the SRM, is that licensees must be in compliance with applicable regulatory requirements.

The NRC staff developed the interim staff guidance (ISG), "Interim Staff Guidance on the Streamlined Review Process for License Renewal for Research Reactors," to assist in the review of LRAs. The streamlined review process is a graded approach based on the licensed power level. Under the streamlined review process, the facilities are divided into two tiers. Facilities with a licensed power level of 2 megawatts thermal (MW(t)) and greater, or those requesting a power level increase, undergo a full review using NUREG-1537. Facilities with a licensed power level less than 2 MW(t) undergo a focused review that centers on the most safety-significant aspects of the renewal application and will rely on past NRC reviews for certain safety findings. The NRC staff made a draft of the ISG available for public comment, and the NRC staff considered public comments in its development of the final ISG. The NRC staff coordinated the UUTR LRA review using the guidance in the final ISG, dated October 15, 2009 (Ref. 10), and, since the licensed power level for the UUTR is less than 2 MW(t), the NRC staff performed a focused review of the licensee's LRA. Specifically, the review focused on reactor design and operation, accident analysis, TS, radiation protection, waste management programs, financial requirements, environmental assessment, and changes to the facility after submitting the application.

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

The NRC staff approved UUTR's physical security plan, "University of Utah Physical Security Plan for Protection of SNM of Low Strategic Significance under Licenses R-25 and R-126," dated January 1, 1980, submitted by a letter dated January 31, 1980, and revised by Revision 1, dated July 28, 1980, submitted by a letter dated August 11, 1980. The licensee maintains a program for providing for the physical protection of the facility and its special nuclear material (SNM) in accordance with the requirements of 10 CFR Part 73, "Physical Protection of Plants and Materials." All changes to the physical security plan have been made in accordance with 10 CFR 50.54(p) and therefore, according to the licensee, these changes will not decrease the effectiveness of the plan. In addition, the NRC staff performs routine inspections of the licensee's compliance with the requirements of the security plan, and the NRC staff's review of inspections for the past several years has identified no violations. Furthermore, in a letter dated June 24, 2011, "Submission of Emergency Plan" (Ref. 11), the

licensee indicated that no changes to the UUTR security plan were needed as a result of the LRA. For the reasons stated above, the NRC-approved plan remains in place.

As a result of the licensee's initial request for a power uprate, the NRC staff reviewed the UUTR EP and approved it by letter dated October 7, 2010 (Ref. 12). Subsequently, the licensee revised the UUTR EP in accordance with the requirements of 10 CFR 50.54(q) to incorporate the MEB evacuation provisions as a result of the maximum hypothetical accident (MHA) analysis described in the final SAR (Ref. 1). In addition, the NRC routinely inspects the licensee's compliance with the requirements of the EP, and the NRC staff's review of inspection reports for the past several years has identified no violations. The licensee maintains an EP in compliance with 10 CFR 50.54(q) and Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," to 10 CFR Part 50, which provides reasonable assurance that the licensee will continue to be prepared to assess and respond to emergency events.

The UUTR Operator Requalification Plan, dated February 1996, was identified in the LRA as the current plan (Ref. 6), and no changes were requested to the plan as a result of the license renewal (Ref. 11). However, the NRC staff reviewed and approved the plan by letter dated September 19, 2011 (ADAMS Accession No. ML112560604).

The purpose of this safety evaluation report (SER) is to summarize the findings of the UUTR safety review and to delineate the technical details considered in evaluating the radiological safety aspects for continued operation. This report provides the basis for renewing the UUTR license at a steady-state power level of 100 kW.

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

1.2 Summary and Conclusions on Principal Safety Considerations

The NRC staff's evaluation considered the information submitted by the licensee, including past operating history recorded in the licensee's annual reports to the NRC, as well as inspection reports prepared by the NRC staff. On the basis of this evaluation and resolution of the principal issues reviewed for the UUTR, the NRC staff made the following conclusions:

- The design and use of the reactor structures, systems, and components important to safety during normal operation, discussed in Chapter 4 of the final SAR (Ref. 1), in accordance with the TS, are safe, and safe operation can reasonably be expected to continue.
- The licensee considered the expected consequences of a broad spectrum of postulated credible accidents and an MHA, emphasizing those that could lead to a loss of integrity of fuel element cladding and a release of fission products. The licensee analyzed the most serious credible accidents and the MHA and determined that the calculated potential radiation doses outside the reactor room would not exceed doses in 10 CFR Part 20 for unrestricted areas.

- The licensee's management organization, conduct of training, and research activities, in accordance with the TS, are adequate to help ensure safe operation of the facility.
- The systems that provide for the control of radiological effluents, when operated in accordance with the TS, are adequate to help 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 TS, which provide limits controlling operation of the facility, are such that there is a high degree of assurance that the facility will be operated safely and reliably. There has been no significant degradation of the reactor, as discussed in Chapter 4 of the final SAR (Ref. 1), and the TS will continue to help ensure that there will be no significant degradation of safety-related equipment.
- The licensee has reasonable access to sufficient resources to cover operating costs and eventually to decommission the reactor facility.
- The licensee maintains a program for providing for the physical protection of the facility and its SNM, in accordance with the requirements of 10 CFR Part 73. All changes to the physical security plan have been made in accordance with 10 CFR 50.54(p).
- The licensee maintains an emergency plan in compliance with 10 CFR 50.54(q) and Appendix E to 10 CFR Part 50, which provides reasonable assurance that the licensee will continue to be prepared to assess and respond to emergency events. All changes to the emergency plan have been made in accordance with 10 CFR 50.54(q).
- The licensee's procedures for training its reactor operators and the operator requalification plan give reasonable assurance that the licensee will continue to have qualified personnel who can safely operate the reactor.

On the basis of these findings, the NRC staff concludes that the licensee can continue to operate UUTR in accordance with the Atomic Energy Act of 1954, as amended (AEA); NRC regulations; and Renewed Facility Operating License No. R-126, without endangering the health and safety of the public, facility personnel, or the environment. The NRC staff further concludes that the issuance of the renewed license will not be inimical to the common defense and security.

1.3 General Description

UUTR is located in the Utah Nuclear Engineering Facility (UNEF) on the ground floor of the MEB on the University of Utah campus, within the city limits of Salt Lake City. The 1,167 acre campus is situated east of the city center in the foothills of the Wasatch Mountains. This location places UUTR outside the city's primary residential areas.

The MEB is situated on high ground relative to other structures in the immediate area. The building conforms to seismic zone 3 requirements under the Uniform Building Code, as determined by the architects of Dean L. Gustavson Associates, and was constructed by Alder Child Construction Company. The UUTR SAR states that the MEB has approximately

254,778 square feet of floor space assigned as follows: classrooms (24,859); offices (25,547); teaching laboratories (59,399); research laboratories (97,847); other areas (56,126) (e.g., workshops, storerooms, corridors).

The areas immediately above the reactor room, on the second floor, are faculty and departmental offices. Radiation surveillances of the areas immediately above UUTR are regularly examined by the radiation safety officer at UU. The ceiling directly above UUTR has 4 inches of concrete and 3/16 inch of steel used as floor support. The third floor above UUTR comprises office and laboratory space.

The reactor is a heterogeneous pool-type nuclear reactor currently loaded with TRIGA fuel. The coolant is light water, which circulates through the core by natural convection. The core is reflected by water and by heavy-water and graphite elements. The maximum licensed steady-state power level is 100 kW. The fuel is nominally 8.5 weight percent (w%) uranium (U), enriched to nominally 19.75 percent (%) in the uranium isotope 235 (U-235).

The primary coolant system consists of a cylindrical pool in which the reactor core is submerged under a column of water. Heat generated from the reactor core is directly transferred to the pool water by natural convection. Reactor pool water is cooled by natural convection and optionally by a closed-loop cooling system; a pump takes water from a pipe connected to the reactor pool, passes it through a chiller, and returns it through a pipe to the reactor pool. Heat can be removed from the reactor through a heat exchanger that transfers the heat to the campus water system. This heat removal system is controlled remotely from the UUTR control room.

The reactor area consists of eight rooms, including the reactor control room, computational laboratory, radiation measurement laboratory, radiochemistry laboratory, microscope room, reactor chemistry laboratories, radioactive storage room, and reactor room. Entry to the reactor room from inside the building is restricted to a single door exiting the control room. The reactor room also has direct access to the outside loading area through a 12-foot-wide overhead door. This overhead door and the doors between the reactor and the control room form the confinement enclosure for UUTR. In this report and in all conclusions supporting the LRA, the reactor room and the laboratories are treated as a single area called the reactor bay.

The walls in the reactor bay are constructed of 1-hour (h) fire-resistant plaster and metal studs, with the exception of the west wall, which is an exterior reinforced concrete wall. Large windows provide visibility to the reactor room from the administrative offices, control room, and computational lab. Nonporous enamel paint finishes are used on all walls and ceilings of the reactor area. The reactor is located within the basemat structure of the MEB.

1.4 Shared Facilities and Equipment

UUTR is in a separate room within the MEB that contains minimal penetrations. Shared facilities include electrical power, heating, cooling, water, and sewerage. The electrical power, heating, cooling, and water distribution systems are separately controlled by UUTR from the distribution junctions.

The MEB electrical power system supplies the electrical power for UUTR. The electrical power provided for building lighting and reactor instrumentation is single phase, 60 hertz (Hz), 120/240 volts (V). The reactor room has its own independent circuit panel that is controlled and monitored by the UUTR staff. The design and safety equipment of UUTR does not require building electrical power to safely shut down the reactor, nor does UUTR require building electrical power to maintain acceptable shutdown conditions.

The water supplied to UUTR as primary circuit water is purified by equipment that is operated and maintained by the UUTR staff. The campus water system provides the ultimate heat sink for UUTR and is only needed when the reactor is at power. There is no safety function for this system credited in the safety analysis.

The reactor rooms (MEB 1205 D, E, F, and G) employ a heating, ventilating, and air conditioning (HVAC) system that is independent of all other areas of the MEB. Fresh air is supplied by the MEB building ventilation system and exhausted through two fume hoods in MEB 1205 F and G, as described in SAR Section 9.1.4.1. Each fume hood operates at 100 cubic feet per minute (CFM) and this maintains a slight negative pressure in the reactor bay. The fume hoods are operational at all times while the reactor is operating. On a high-radiation alarm, the damper on the system closes and reduces the intake of outside air. Air exhausted from the HVAC system is always passed through prefilters and high-efficiency particulate air (HEPA) filters. The system exhausts through a stack on the roof of the building having a height of approximately 40 feet above ground level. This system is depicted in SAR Figure 9.1-4.

1.5 Comparison with Similar Facilities

The TRIGA reactor designed by General Atomics (GA) is one of the most widely used research and training reactors in the United States. TRIGAs exist in a variety of configurations and capabilities. UUTR is similar to the other TRIGAs licensed in the United States by the NRC. SAR Section 1.5 provides general statements regarding the number of TRIGA reactors in service or being built. The licensed power levels for these reactors range from 100 kW (as in the case of UUTR) to 14 megawatts (MW). UUTR uses standard TRIGA reactor fuel with both aluminum and stainless-steel cladding. The fuel is arranged in a hexagonal grid similar to several other TRIGA reactors. Similarly, the pool size and complement of experimental facilities are similar to other TRIGAs. The TRIGA fuel has no established performance-related issues (Ref. 16).

1.6 Summary of Operations

UUTR provides a wide range of training, irradiation, and research services to educational and research institutions and industry. The reactor has experimental facilities to irradiate materials. These facilities support the irradiation services that include sample irradiation for medical and industrial clients, neutron activation analysis, neutron damage testing of electronic components, ultrasensitive detection of actinides, and educational and training support. UUTR typically operates about 5–8 hours per week at 90 kW. The operational workload is not expected to increase significantly from this level.

1.7 Compliance with the Nuclear Waste Policy Act of 1982

Section 302(b)(1)(B) of the Nuclear Waste Policy Act of 1982 specifies that the NRC may require, as a precondition to issuing or renewing an operating license for a research or test reactor, that the applicant have entered into an agreement with the U.S. Department of Energy (DOE) for the disposal of high-level radioactive wastes and spent nuclear fuel. In a letter dated May 3, 1983, R.L. Morgan, of DOE, informed H. Denton, of the NRC, that universities and other government agencies operating non-power reactors had entered into contracts with DOE providing that DOE retains title to the fuel and is obligated to take the spent fuel, or high-level waste, or both, for storage or reprocessing. An e-mail sent from James Wade of DOE to Paul Doyle (NRC) on May 3, 2010 (Ref. 13), reconfirms this obligation with respect to the fuel at UUTR (DOE Contract No. 73702, valid from February 25, 2008–December 31, 2012). By entering into such a contract with DOE, UUTR has satisfied the applicable requirements of the Nuclear Waste Policy Act of 1982.

1.8 Facility Modifications and History

This review considered changes made to the UUTR in the last 20 years. The UUTR SAR, Table 1.8-1, provides a comprehensive list of modifications and is restated in Table 1-1 below. Most of the modifications involved equipment replacement or improvements, or minor changes to the existing design that either enhanced capabilities or improved reactor operations. All of these modifications were subject to evaluation under 10 CFR 50.59, "Changes, Tests and Experiments," to ensure there was no impact on the safety of the UUTR. The most significant change, made in 2008, pertains to replacing a fume hood and moving the ventilation ducts and vent pipe. A review of NRC inspection reports documents this change as having been performed in accordance with 10 CFR 50.59, "Changes, Tests, and Experiments," and that it was acceptably accomplished.

Since the renewal of UUTR Facility Operating License R-126, dated April 17, 1985 (Amendment No. 5), there have been three license amendments. Amendment No. 6, dated August 8, 1992, corrected an omission in the TS. Amendment No. 7, dated December 3, 1998, increased the surveillance requirement testing period for the radiation monitoring system. Amendment No. 8, dated April 4, 2005, increased the licensed SNM possession limit. The NRC staff reviewed and approved these license amendments.

Table 1-1 Modifications to the UUTR Facility

Year of Activity	For all these activities, 10 CFR 50.59 reviews were performed.
October 1975	Construction completed, fuel loaded, initial criticality.
1987	Decommissioned 5W AGN 201 reactor, located close to UUTR.
1989	Added Fast Neutron Irradiation Facility (FNIF). (FNIF was installed on the west side of the reactor core; it provides 1 MeV equivalent fast neutron flux).
1991	Upgraded reactor control console. (A new control console was installed with digital equipment).
1997	Replaced continuous air monitoring recording system. (New recording system was installed).
August 1997	Added radiochemistry lab and class 100 clean room. (Two fume hoods in the radiochemistry lab and class 100 clean room were added).
October 2001	Remodeled facility office area. (Three offices for faculty and one student office area were added).
August 2005	Replaced hoist system in the reactor room. (New hoist system, which has 2 metric ton capacity, was installed).
May 2007	Pneumatic irradiator on D-4 position was added (with compressed He gas).
December 2008	Replaced fume hood in the radiochemistry laboratory and ventilation duct and pipe. (Two old fume hoods in the radiochemistry lab were replaced with new fume hoods).
March 2010	Remodeled reactor room, control room area, and associated labs.

1.9 Financial Considerations

1.9.1 Financial Ability to Operate the Reactor

10 CFR 50.33(f) states:

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

UU does not qualify as an “electric utility,” as defined in 10 CFR 50.2, “Definitions.” Under 10 CFR 50.33(f)(2), applicants 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. The NRC staff has determined that UU must meet the financial qualifications requirements of 10 CFR 50.33(f), and is subject to a full financial qualifications review. UU must demonstrate that it possesses or has reasonable assurance of obtaining the funds necessary to cover estimated operating costs for the period of the license. UU must submit estimates of the total annual operating costs for each of the first five years of facility operations from the expected license renewal date and indicate the source(s) of funds to cover those costs.

As supplemented by letters dated March 10, 2010, and May 13, 2010, UU submitted its projected operating costs for the UUTR for each of the fiscal years (FYs) 2012 through 2016, which are estimated to range from \$140,436 in FY 2012 to \$158,062 in FY 2016. Funds to cover the operating costs will be provided by UU, research or service contracts for which the work is performed, and the Utah Nuclear Engineering Program. UU expects that these funding sources will continue for FYs 2012 through 2016. The NRC staff reviewed UU's projected operating costs and projected sources of funds to cover these costs and finds them to be reasonable.

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

1.9.2 Financial Ability to Decommission the Facility

The NRC has determined that the requirements to provide reasonable assurance of decommissioning funding are necessary to help ensure the adequate protection of public health and safety. In 10 CFR 50.33(k), the NRC requires that an application for an operating license for a utilization facility provide information in the form of a report to demonstrate how reasonable assurance will be provided that funds will be available to decommission the facility. Under 10 CFR 50.75(d), each non-power reactor applicant for or holder of an operating license shall submit a decommissioning report that contains: (1) a cost estimate for decommissioning the facility; (2) an indication of the funding method(s) to be used to provide funding assurance for decommissioning; and (3) a description of the means of adjusting the cost estimate and associated funding level periodically over the life of the facility. The acceptable methods for providing financial assurance for decommissioning are specified in 10 CFR 50.75(e)(1).

The application for license renewal dated March 25, 2005 did not include a decommissioning cost estimate. In the March 10, 2010, and May 13, 2010, supplements, UU provided a decommissioning cost estimate for the UUTR of \$5,957,465 in 2010 dollars, which is based on the analysis done by the U.S. Department of Defense (DOD) for decommissioning the Armed Forces Radiobiology Research Institute TRIGA reactor facility. The decommissioning cost estimate summarized costs by labor, waste disposal, energy, other items (e.g., ancillary costs, such as spent fuel removal and shipment) and a 25 percent contingency factor. According to UU, the decommissioning cost estimate will be adjusted for future dollar values using the U.S. Bureau of Labor Statistics Consumer Price Index. In reviewing the decommissioning cost estimate for the UUTR (\$5,957,465 in 2010 dollars), the NRC staff reviewed the information provided as a response to an NRC RAI, in a UU letter dated March 10, 2010, and supplemented in a letter dated May 13, 2010, and concludes that the decommissioning approach and cost estimated submitted by UU are reasonable.

UU 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.

UU provided a SOI, dated May 7, 2010, stating that the signator, "...intend[s] to request that funds be made available when necessary in the amount of \$5,957,465 [for the DECON option], or other appropriate amount..." and the signator also states that he "...intend[s] to request and obtain these funds sufficiently in advance of decommissioning to prevent delay of required activities."

To support the SOI and UU's qualifications to use a SOI, the application states that UU is an agency of the State of Utah and a part of the state government of the State of Utah. The licensee included documentation which corroborates this statement. The application also provided information supporting the licensee's representation that the decommissioning funding obligations of UU are backed by the full faith and credit of the State of Utah. The licensee also provided documentation verifying that Michael K. Young, President of the University, the signator of the SOI, is authorized to execute contracts on behalf of UU.

The NRC staff reviewed UU's information on decommissioning funding assurance as described above and finds that UU is a State of Utah government licensee under 10 CFR 50.75(e)(1)(iv); the SOI is acceptable; the decommissioning cost estimate as well as the costs for the DECON option are reasonable; and UU's means of adjusting the cost estimate and associated funding level periodically over the life of the facility is reasonable. The NRC staff notes that any adjustment of the decommissioning cost estimate must incorporate, among other things, changes in costs due to the availability of disposal facilities, and that UU has an obligation under 10 CFR 50.9, "Completeness and Accuracy of Information," to update any changes in the projected cost, including changes in costs resulting from increased disposal options.

1.9.3 Foreign Ownership, Control, or Domination

Section 104d of the AEA, 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 NRC regulation 10 CFR 50.38, "Ineligibility of Certain Applicants," contains language to implement this prohibition. According to the application, UU is a State of Utah government licensee and is not owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government. The NRC staff does not know or have reason to believe otherwise.

1.9.4 Nuclear Indemnity

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

1.9.5 Conclusions

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

2. REACTOR DESCRIPTION

2.1 Summary Description

2.1.1 Introduction

UUTR is a GA TRIGA Mark I reactor that is licensed for a maximum power level of 100 kW and non-pulsed operation. It typically operates under administrative controls at a power level of 90 kW. UUTR is a standard design TRIGA providing a variety of irradiation facilities, including a central thimble, a pneumatic transfer system, a single-element replacement, and a fast neutron irradiation facility.

The reactor core is located near the bottom of a cylindrical water-filled aluminum tank that has a diameter of 8 feet and is about 24 feet deep. The tank is shielded radially by 2 feet of compacted sand, a $3/16$ -inch steel outer tank, and 3 feet of ordinary concrete. The approximately 22-foot column of water above the core provides axial shielding as well as coolant. The control rod drives are mounted above the tank on a bridge structure spanning the diameter of the tank.

UUTR uses solid uranium-zirconium hydride (U-ZrH) fuel containing nominally 8.5 wt.% U enriched to less than 20 wt% in U-235. UUTR contains a mixed core of stainless-steel clad and aluminum clad elements in a hexagonal array. The reactor power is regulated by inserting or withdrawing neutron-absorbing control rods. Many TRIGA reactors are designed and instrumented to operate in the pulse mode; however, UUTR has no pulsing capabilities.

The inherent safety of TRIGA reactors has been demonstrated by the extensive experience gained from similar designs used throughout the world. TRIGA fuel is characterized by a strongly negative prompt temperature coefficient characteristic of U-ZrH fuel moderator elements that contributes to safe operation. A series of GA and NRC reports discuss such features as reactor kinetic behavior (GA-7882, "Kinetic Behavior of TRIGA Reactors, dated March 31, 1967 (Ref. 14)); fission product retention (NUREG-1282, "Safety Evaluation Report on High-Uranium Content, Low-Enriched Uranium-Zirconium Hydride Fuels for TRIGA Reactors", issued August 1987 (Ref. 15)), and GA-4314, "The U-Zr_xH Alloy: Its Properties and Use in TRIGA Fuel," M.T. Simnad, 1980 (Ref. 16)); and accident analysis (NUREG/CR-2387, "Credible Accident Analyses for TRIGA and TRIGA-Fueled Reactors," issued April 1982 (Ref. 17)).

2.1.2 Summary of Reactor Data

Table 2-1 below contains a summary of pertinent reactor parameters, including thermal-hydraulic and neutronic design data, for the UUTR core.

Table 2-1 Reactor Parameters for the UUTR Core

UUTR Reactor Core Parameters	
Licensed Reactor Power (kW)	100
Number of Fuel Elements in Core	78
Number of Control Rods in Core	3
UUTR Fuel Temperature and Thermal-Hydraulic Parameters	
Maximum Fuel Temperature at 100 kW (°C)	129.67
Maximum Fuel Temperature at 110 kW (°C)	136.43
Fuel Temperature Coefficient, 293–1200 K (\$/K)	-0.01436
Coolant Void Coefficient, 0–75% (\$/% void)	-0.2702
Coolant Temperature Coefficient, 293–600 K (\$/K)	-0.0133
Peak-to-Average Fuel Element Power Ratio	1.577
Maximum Rod Power at 100 kW (kW)	2.022
Average Rod Power at 100 kW (kW)	1.282
DNBR at 100 kW	9.25
Prompt Neutron Lifetime (μs)	21.7
Effective Delayed Neutron Fraction	0.00768
UUTR Reactivity Parameters	
Safety Control Rod Worth, calculated (\$)	-1.924
Shim Control Rod Worth, calculated (\$)	-1.468
Regulating Control Rod Worth, calculated (\$)	-0.294
Excess Reactivity, calculated (\$)	+0.840
Shutdown Margin (\$ with Safety Control Rod out)	-0.922
UUTR Safety Parameters	
Linear Power Trip Setpoint (kW)	100
% Power Trip Setpoint (% power)	110
Fuel Temperature Trip Setpoint (°C)	200

2.1.3 Experimental Facilities

UUTR was designed with multiple in-core irradiation facilities to facilitate a broad range of potential experimental activities. These facilities consist of a central cavity, the pneumatic transfer tube, and individual fuel element locations.

The central irradiation facility is located in the central fuel element grid position. A special tube has been constructed to accommodate samples and can be placed in the central fuel pin position by means of a cable. The dimensions of this assembly are the same as a fuel pin. Because this facility has an in-core irradiator located in the center of the core, there are special restrictions on the reactivity of samples placed in it. Additionally, a cutout is available where both the A and B rings are removed to accommodate a larger irradiator that is capable of holding multiple samples. This irradiator has two special features associated with it. One is a sealed interior that holds heavy water. The other is a motor that rotates the sample holder to spatially average the neutron fluence in the assembly.

A pneumatic transfer system (PTS) is available at the UUTR facility. The PTS is installed within a 1.5-inch outside-diameter tube and is driven by the force of dry, compressed helium (He). The PTS has a slight curve in its tube to prevent direct streaming of neutrons from the core to the surface of the pool. The UUTR PTS is designed to transfer individual specimens quickly into and out of the reactor core. The specimens are placed in a small enclosed polyethylene holder (the rabbit), which in turn is placed into the receiver. The rabbit travels through aluminum and plastic tubing to the terminus at the reactor core centerline and returns along the same path to the receiver. Directional gas flow moves the rabbit between receiver and terminus. A compressed-gas system supplies He to the system, and a solenoid valve directs flow. Controls to operate the compressed gas and solenoid valve are on the console.

UUTR was designed with three beam tubes that were never fully installed and are not operational. The beam tubes are permanently filled with sand.

2.2 Reactor Core

The UUTR core is a hexagonal configuration of fuel elements, moderator-reflector elements, a central thimble, a neutron source, and control rods, all positioned between two grid plates. The control rods pass through guide tubes that are inserted through the top grid plate and are attached to the bottom grid plate by means of a locking device. The core is cooled by natural convection of water. Shielding above the core is provided by a column of water. The grid plates have 127 lattice positions arranged in six concentric rings around a central position.

UUTR uses solid fuel elements in which the zirconium-hydride (ZrH) moderator is homogeneously combined with low-enriched fuel (LEU) (U-ZrH_x). Two types of fuel elements are used in the UUTR core: (1) stainless-steel clad, high-hydride, uranium-zirconium (U-ZrH_{1.6}) fuel elements; and (2) aluminum clad, low-hydride, U-zirconium (U-ZrH_{1.0}) fuel elements. (Note that the H-to-Zr ratio is represented by the "x" in the U-ZrH_x nomenclature.) The H content is important because it influences many attributes of fuel behavior.

Neutron reflection in the radial direction is provided by 12 graphite and 12 heavy-water elements in an aluminum clad. The height of the graphite and heavy-water elements in the reflector is about 24 inches. Also, approximately 3.5 feet of water at the outer perimeter of the tank acts as a thermal shield to protect the aluminum tank from direct heating, and it also contributes to reducing the activation of the tank material.

The reactivity and the power level of the UUTR reactor are controlled with three control rods. Instrumentation channels monitor and indicate the reactor neutron flux and power level on the console. The UUTR reactor console displays the percent power, the log power, the reactor period, and the reactor count rate.

The bulk pool water temperature and the reactor tank outlet and inlet water temperatures are indicated on the console. The water conductivity, measured at the inlet and outlet of the demineralizer, is displayed on a panel near the console. In addition, primary reactor water is routinely monitored to identify changes in radioactivity.

The following sections will discuss the reactor core and fuel, the control elements, the neutron moderator and reflector, the neutron startup source, the core support structure, and the reactor pool.

Table 2-2 below lists the major components of the UUTR core, and is constructed from UUTR SAR Section 4.5, which describes the major components used to assemble the UUTR core. Also, UUTR SAR Section 4.5.2.3 describes the limiting core configuration (LCC).

Table 2-2 The UUTR Core Elements Used

UUTR Core Elements	Number
Fuel Elements—stainless-steel clad, high hydride (8.77% burnup)	17
Fuel Elements—stainless-steel clad, high hydride (0.61% burnup)	36
Fuel Elements—aluminum clad, low hydride	23
Instrumented Fuel Elements—stainless-steel clad	2
Safety Control Rod	1
Shim Control Rod	1
Regulating Rod	1
Graphite Reflector Elements	12
Heavy-Water Reflector Elements	12
Empty Core Locations	21
Central Irradiator	1
Total Grid Plate Positions	127

There are several design features that are important to the discussion of the core configuration. TS 5.3.1 presents the design feature requirements of the UUTR core as follows:

TS 5.3.1 Reactor Core

Applicability

This specification applies to the configuration of fuel and in-core experiments.

Objective

The objective is to assure that provisions are made to restrict the arrangement of fuel elements and experiments so as to provide assurance that excessive power densities shall not be produced.

Specifications

1. The core assembly shall consist of TRIGA fuel elements.
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, aluminum dummies, stainless steel dummies, control rods, heavy-water elements, startup sources, and vacant positions that are filled with water.
3. The reflector, excluding experiments and irradiation facilities, shall be water or a combination of graphite and heavy water elements and water.

TS 5.3.1, Specification 1 helps ensure that only TRIGA fuel elements are authorized to be used in the UUTR core. The fuel elements used in the UUTR core are typical of TRIGA reactors. This core design feature information is important to help ensure that the LCC for UUTR consists of those core elements that are approved for use. TS 5.3.1, Specification 1, only includes those elements identified and evaluated for use in UUTR SAR Section 4.5.

TS 5.3.1, Specification 2 helps ensure that the physical arrangement of fuel in order to limit empty fuel locations and thereby control peaking.

TS 5.3.1, Specification 3 helps ensure that reflectors are identified and evaluated for use in UUTR SAR Section 4.5.

The NRC staff reviewed the UUTR SAR and TS 5.3.1. The NRC staff finds that TS 5.3.1, Specifications 1 through 3 characterize the UUTR design features for the reactor core and help ensure that the core loading conforms and is limited to the analysis in SAR Chapter 4. The objective of the specification accurately states that the basic issue is to ensure that the excessive power densities will not be experienced by any core loading accomplished under the license. This satisfies the guidance in NUREG-1537, Section 4.5.1, requesting the applicant to identify the highest power density of any core arrangement used by the licensee. The accident analysis presented in Chapter 4 of this report used this configuration. On the basis of this review, the NRC staff concludes that TS 5.3.1 is acceptable.

TS 5.3.1 is supported by the following limiting condition for operation (LCO) TS describing the LCC.

TS 3.1.4 Core Configuration

Applicability

This specification applies to the configuration of fuel elements and in-core experiments.

Objective

The objective is to assure the provisions are made to restrict the arrangement of fuel elements and experiments so as to provide assurance that excessive power densities will not be produced.

Specifications

1. The reactor core shall be an arrangement of TRIGA LEU cylindrical stainless-steel clad, high-hydride fuel-moderator elements and aluminum clad, low-hydride fuel-moderator elements with neutron reflectors provided by up to 12 graphite and 12 heavy-water elements in aluminum cladding.
2. The reflector, excluding experiments and experimental facilities, shall be a combination of water, graphite, and heavy water.
3. Fuel shall not be removed from or inserted into the core unless the reactor is subcritical by more than the calculated worth of the most reactive fuel element.
4. Control rods shall not be removed manually from the core unless the core has been shown to be subcritical with all control rods fully withdrawn from the core.

TS 3.1.4, Specification 1 helps ensure the use of TRIGA fuel elements and neutron reflectors in compliance with the analysis provided in the UUTR SAR and TS 5.3.1.

TS 3.1.4, Specification 2 helps ensure that the reflector locations are filled with water or a combination of water and graphite, except experiments or experimental facilities.

TS 3.1.4, Specification 3 helps ensure that proper precautions are taken before inserting fuel elements into any core configuration.

TS 3.1.4, Specification 4 helps ensure that, if control rods are removed, the core is first made subcritical to the extent that no control rods are required to maintain the core subcritical. This specification allows the removal of one control rod, subject to the requirement that the core has been shown to be subcritical if all control rods were removed.

TS 5.3.1 and 3.1.4 help ensure that the core loading conforms and is limited to the analysis in SAR Chapter 4. The resulting core loading uses 78 fuel elements to configure the UUTR 100 kW core. The objective of the specification accurately states that the basic issue is to ensure that the excessive power densities will not be experienced by any core loading accomplished under the license. This satisfies the guidance in NUREG-1537, Section 4.5.1,

requesting the applicant to identify the highest power density of any core arrangement used by the licensee. The safety analysis presented in Chapter 4 of this report used this configuration. SAR Section 4.5.2.3 discusses the UUTR LCC. SAR Section 4.2.1 also discusses using Monte-Carlo N-Particle Transport Code 5 (MCNP5) to estimate the fuel burnup in the UUTR core. The licensee used MCNP5 to determine the information in Table 2-3. Results can be found in UUTR SAR Table 4.2-2.

Table 2-3 The UUTR Fuel Depletion History

UUTR Fuel Depletion History		
Period	1975–1998	1998–2010
Depletion history	7.985 megawatt days (MWd)	0.593 MWd
Burned U-235	239.554 grams (g)	17.776 g
Number of fuel elements	78	78
Average Burnup	3.07 g/element	0.228 g/element

The licensee used the LCC and depletion history to determine the burnup of the fuel reported in the SAR to be 8.77 percent (percent U^{235} consumed) for the stainless–steel-clad fuel and 8.91 percent for the aluminum-clad fuel present since 1975. The burnup of the fuel added in 1998 having stainless-steel clad was 0.61 percent. Since the MCNP5 code does not have an embedded burnup function, the licensee used an estimate of the power history of the UUTR core to calculate the average burnup of the fuel elements. The NRC staff reviewed the burnup results and the licensee’s analytical approach, and concludes that the burn-up results are consistent with other TRIGA reactors with similar operating history and therefore, acceptable.

Based upon a review of the information provided by the licensee in the SAR, the NRC staff concludes that the licensee has acceptably described the LCC used in UUTR, including design limits, and the technological and safety-related bases for these limits. The licensee has also acceptably discussed the constituents, materials, and components for the LCC.

Based on its review of the UUTR SAR, the NRC staff finds that the licensee has adequately analyzed the expected normal reactor operation during the period of the renewed facility operating license. The NRC staff further concludes that the TS 5.3.1 and TS 3.1.4 provide reasonable assurance that normal operation of the UUTR core will not pose a significant risk to public health and safety or the environment.

2.2.1 Reactor Fuel

UUTR uses aluminum-clad and stainless-steel-clad fuel elements in which the fuel is a solid homogeneous mixture of U-ZrH alloy containing nominally 8.5 w% U enriched to less than 20 w% in U-235. SAR Section 4.2.1.1 describes the design details of both fuel elements. Stainless-steel fuel elements have the same geometry as aluminum fuel elements except for the active fuel length, overall length, and cladding thickness. SAR Figure 4.2-2 shows the active part of the fuel element. SAR Section 4.2.1 describes the stainless-steel fuel as having a 0.19-inch hole in the center that is filled with a zirconium rod.

NUREG-1282 (Ref. 15) provides guidance for the use of the TRIGA fuel types listed in Table 2-4. UUTR uses only the “original” fuel type listed below, whereas the approval also includes other loadings of LEU fuel, which are included in the table for completeness.

Table 2-4 TRIGA Fuel Characteristics

Fuel Type	Uranium (w%)	Erbium (w%)	U-235 (w%)	$\alpha_F \times 10^5$ ($\Delta k/k$ -°C)	Uranium (% volume)
Original	8.5	0.0	20	9.5	2.6
LEU	20	0.5	20	10.5	6.8
LEU	30	0.9	20	8	11.2
LEU	45	1.8	20	5	19.5

UUTR has two instrumented fuel elements (IFE). An IFE has three thermocouples embedded in the fuel. The sensing tips of the IFE thermocouples are located halfway between the outer radius and the vertical centerline at the center of the fuel section and 1 inch above and below the horizontal center. The thermocouple wires pass through a seal contained in a stainless-steel tube welded to the upper end fixture. The watertight stainless-steel conduit carries the wires to the top of the reactor tank. The stainless-steel conduit is attached to a triangular support above the water surface. The IFE has the same dimension and shape except for the three thermocouples and stainless-steel conduit.

UUTR TS 5.3.1 and TS 3.1.4 specify the use of the reactor fuel, help ensure that the fuel is maintained in an acceptable state, and maintain important assumptions used in the UUTR SAR.

TS 5.3.3 Reactor Fuel

Applicability

This specification applies to the fuel elements used in the reactor core.

Objective

The objective is to assure that the fuel elements are of such a design and fabricated in such a manner as to permit their use with a high degree of reliability with respect to their physical and nuclear characteristics.

Specifications

The individual TRIGA fuel elements shall have the following characteristics:

1. Uranium content: maximum of 8.5 wt% enriched to less than 20% ²³⁵U,
2. Hydrogen-to-zirconium atom ratio (in the ZrH_x): between 1.0 and 1.60,
3. Cladding: 304 stainless steel or aluminum, nominal 0.02 and 0.03 inches thick respectively,
4. Identification: top pieces of fuel elements will have characteristic markings to allow visual identification of fuel elements, and

5. Burnable poisons: the fuel shall not include burnable poisons.

TS 5.3.3, Specification 1 specifies the maximum w% and enrichment of the TRIGA fuel and helps ensure that the fuel requirement is consistent with the analysis supplied in the UUTR SAR Section 4.5.

TS 5.3.3, Specification 2 helps ensure the H-to-Zr ratio is between 1.0 and 1.60. Clad stress is a function of the fuel rod internal pressure, which, in turn, is a strong function of the ratio of H to Zr. At the maximum upper limit of this ratio of 1.60, along with the conservative safety limit (SL) of 1,150 degrees C, the pressure is at least a factor of 5 lower than would be necessary for clad failure. The maximum value of the H-to-Zr ratio is adequate to account for uncertainties in clad strength and manufacturing tolerances.

TS 5.3.3, Specifications 2, 3, and 5 are consistent with the guidance provided in NUREG-1282 (Ref. 15).

TS 5.3.3, Specification 4 helps ensure that a reasonable system is provided to allow cognitive configuration of the reactor core.

Based on a review of the information provided in the UUTR SAR Section 4.2.1 and discussed above, the NRC staff concludes that TS 5.3.3 is acceptable.

TS 3.1.6 Fuel Parameters

Applicability

This specification applies to all fuel elements.

Objective

The objective of this specification is to maintain the integrity of the fuel element cladding.

Specifications

The reactor shall not operate with damaged fuel elements, except for the purpose of locating damaged fuel elements. A fuel element shall be considered damaged and must be removed from the core if any of the following conditions are met:

1. The transverse bend exceeds 0.0625 inches over the length of the cladding,
2. Its length exceeds its original length by 0.125 inches,
3. A cladding defect exists as indicated by release of fission products,
4. Visual inspection identifies bulges, gross pitting, or corrosion, or
5. Fuel burnup of Uranium-235 in the UZrH fuel matrix exceeds 50% of the initial content.

TS 3.1.6, Specifications 1 through 5, establish inspection requirements to detect gross failure or visual deterioration of the fuel. The fuel element attributes inspected include the fuel element

transverse bend and length, and a visual inspection is conducted for bulges or other cladding defects. TS 3.1.6 limits on transverse bend and length, and fuel burnup, are consistent with the values provided in NUREG-1537. Based on the NRC staff review of the information above, TS 3.1.6, Specifications 1 through 5, are acceptable to the NRC staff.

SAR Section 4.2.1.1 also describes fuel performance processes that are important to the UUTR fuel. This includes a discussion of how the dissociation of the H and Zr builds up a gas inventory in internal components and spaces of the fuel elements. The SAR stresses the importance of limiting the maximum fuel temperature to prevent an excessive internal pressure that could be generated by heating the gases. The temperature at which phase transitions may lead to cladding failure in aluminum-clad, low-hydride fuel elements is reported to be 530 degrees C.

The NRC staff reviewed the SAR information which described the fuel elements used in UUTR, their design limits, and the technological and safety-related bases for these limits. The licensee also acceptably discussed the constituents, materials, and components for the fuel elements. The NRC staff finds that compliance with the applicable TS will help ensure uniform core operating characteristics and adherence to the design bases and safety-related requirements. The NRC staff concludes that the UUTR fuel elements and the associated TS are acceptable.

2.2.2 Control Rods

UUTR uses boron carbide control rods that are characteristic of most TRIGA reactors. The rods are enclosed in aluminum tubes approximately 43 inches long and are 0.875, 0.875, and 0.25 inches in diameter (safety, shim, and regulator rods, respectively) with a powder boron carbide neutron absorber filling insight of the rods. The control rods are limited in their ability to fall through the core by a safety plate that is 1 inch below the fully inserted elevation.

UUTR uses three control rods to control reactivity. The rods are designated the safety, shim, and regulating rods. The regulating rod (sometimes referred to as the "Reg. rod") is used for fine control during the UUTR operation. The control rods pass through normal fuel positioning holes in the UUTR core on the top and the bottom of the grid plates. Guide tubes ensure that the control rods remain in the proper position during their use.

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

The control rod can be withdrawn from the reactor core when the electromagnet is energized. When the reactor is scrambled, the electromagnet is deenergized and the control rod falls by

gravity into the core. The withdrawal speed of the rods is adjustable, and the UUTR control rod drives are normally set to insert or withdraw the control rods at a nominal rate of 0.49 centimeters per second.

The safety, shim, and regulating control rods are located at core locations D-7, D-13, and D-1, respectively. TS 5.3.2 describes the design features that are applicable to the control rod design and operation.

TS 5.3.2 Control Rods

Applicability

This specification applies to the control rods used in the reactor core.

Objective

The objective is to assure that the control rods are of such a design as to permit their use with a high degree of reliability with respect to their physical and nuclear characteristics.

Specifications

The shim, safety, and regulating control rods shall have scram capability and contain borated graphite, B₄C powder or boron, with its compounds in solid form as a poison, in aluminum or stainless steel cladding.

The position of each control rod is displayed on the console as a percentage of the length that the rod is withdrawn from the reactor. The control rods are held in place by an electromagnet armature (UUTR SAR Section 7.3). When a scram is initiated, the current is deenergized to the electromagnets and the control rods drop by gravity into the core, shutting the reactor down. The control rods are designed to safely change the reactor power, or shut the reactor down, or both. All of the UUTR control rods are scrammable per TS 5.3.2. Similarly, if power is lost to the electromagnets from a loss of electrical power, the control rods electromagnets are disengaged from the armature.

TS 3.2.1 contains the following requirements for control rod operability:

TS 3.2.1 Control Rods

Applicability

This specification applies to the function of the control rods.

Objective

The objective is to determine that the control rods are operable.

Specifications

The reactor shall not be operated unless the control rods are operable. Control rods shall not be considered operable if:

1. Damage is apparent to the rod or rod drive assemblies,

2. The scram time exceeds 2 seconds, or
3. The rate of reactivity insertion by control rod motion exceeds \$0.30 per second.

TS 3.2.1, Specification 1 helps ensure that the control rods are free of any apparent damage.

TS 3.2.1, Specification 2 helps ensure that the scram time supports the analysis provided in UUTR SAR Section 4.2.2. A scram time less than 2 seconds is necessary to ensure that the reactor will be promptly shut down when a scram signal is initiated. Section 2.5.4 of this report provides the confirmatory analysis accepting this value.

TS 3.2.1, Specification 3 helps ensure that the rate of reactivity insertion is consistent with the analysis detailed in the UUTR SAR Section 4.2.2. Section 2.5.4 of this report evaluates the acceptability of the \$0.30 per second TS limit.

TS 3.2.1 helps ensure that, during the normal operation of the UUTR, the time required for the scrammable control rods to be fully inserted, from the instant that a safety channel variable reaches the safety system setting, is rapid enough to prevent fuel damage. This specification ensures that the reactor will be promptly shut down when a scram signal is initiated. Analysis has indicated that, for the range of transients anticipated for a TRIGA reactor, the specified scram time is adequate to ensure the safety of the reactor. The SAR assumed a 2 second value which is common to other TRIGA reactors, and discussed in Section 2.5.4 of this report. Based on the discussion above, TS 3.2.1 is acceptable to the NRC staff, as it supports the basic design requirements to prevent reactor fuel damage. TS 4.2.1 presents the surveillance requirements for the controls in Section 5.4.2 of this report.

The NRC staff reviewed the design and performance of the control rods and finds that UUTR has demonstrated that it provides adequate reactivity worth, structural rigidity, and reliability to help ensure reliable operation under all operating conditions. The scrammable rods have the ability to scram without challenging the integrity of the reactor fuel. The control rod materials have been used in many similar TRIGA reactors and have demonstrated reliable operation and long service life. The design of these control elements meets the UUTR design requirements.

Based on a review of the information provided by the licensee, the NRC staff concludes that the control rods conform to the applicable design bases and can shut down the UUTR from any operating condition. There is reasonable assurance that the scram features will perform as required during the renewal period to ensure fuel integrity and protect public health and safety. The control rod design for the UUTR includes reactivity worths that can control the excess reactivity planned for the UUTR, including the assurance of an acceptable shutdown reactivity and margin. The licensee has justified appropriate design limits, LCOs, and surveillance requirements for the control rods. Based on the above discussion, the NRC staff concludes that the requirements related to the UUTR control rods acceptable.

2.2.3 Neutron Moderator and Reflector

The UUTR pool water serves as the moderator, reflector, and coolant for the core. In addition, the U-ZrH_x fuel matrix provides significant moderation. The UUTR core includes an additional

row of graphite and heavy-water reflector elements. These are shown in Figure 2-1 below (SAR Figures 4.2-5 and 4.2-6 of the UTR SAR for heavy water and graphite, respectively).

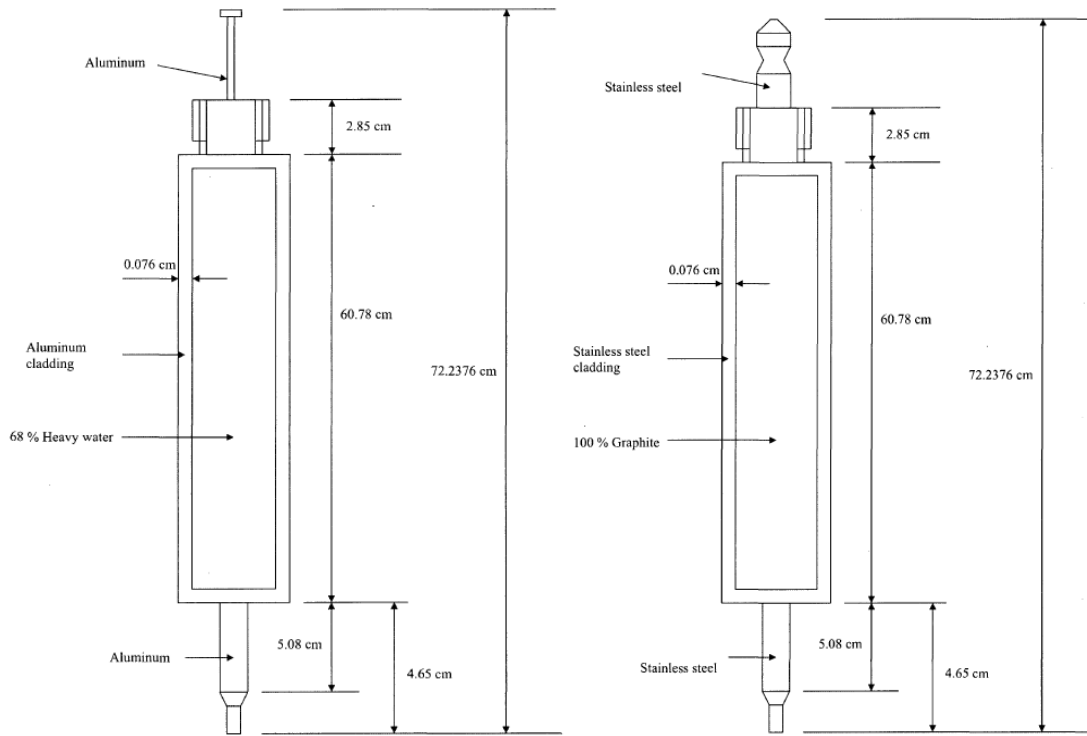


Figure 2-1 UTR heavy-water and graphite reflectors

Table 2-5 shows the specification of the heavy-water and graphite reflectors.

Table 2-5 Specification of the Heavy-Water and Graphite Reflectors in UUTR

UUTR Heavy-Water and Graphite Reflectors		
	Heavy-Water Reflector	Graphite Reflector
Number	12	12
Cladding material	Aluminum	Aluminum
Cladding thickness (cm)	0.076	0.076
Outside diameter (cm)	3.7465	3.7465
Overall length (cm)	72.2376	72.2376
Material	68% heavy-water; 32% light-water	100% graphite
Core location	G-8,9,10,12,14,16,18; F-21,22,23,24,25	F-1,26,27,28,29,30; G-2,3,4,5,6,7

The UUTR SAR indicates that water is kept from contacting the graphite reflector elements by welded aluminum cladding. The graphite reflector elements are located near the UUTR core thermal irradiator. Graphite elements have the same end structures at the top, so that the fuel-handling tool can be used for moving reflectors. The heavy-water elements have two different top-end fixtures: the elements in core locations G8 through G18 have the same top-end structure as a fuel element; a screw is attached on the top. The design and description of the moderator and reflector elements are typical for TRIGA reactors.

Based upon a review of the information in UUTR SAR Section 4.2.3, the NRC staff finds that the licensee has acceptably evaluated the use of the reflector elements in UUTR, including their design limits, and the bases for these limits. The licensee has also acceptably evaluated the constituents, materials, and components for the reflector elements. TS 3.1.4, Specification 2 (see Section 5.3.1.4 of this report) helps to ensure that the UUTR core will operate with uniform neutronic operating characteristics and in conformance with the LCC described in the SAR. Based on the discussion above, the NRC staff concludes that the UUTR reflector elements and their associated TS are acceptable.

2.2.4 Neutron Startup Source

UUTR utilizes a 5-curie (Ci) plutonium-beryllium (Pu-Be) startup source. The source is located in a special reflector element source holder in the outer ring of the core lattice. The source can be withdrawn from its in-core position manually by means of an attached steel cable that is connected to the top of the source holder cap. An indicator light coupled to the startup meter at the control console shows whether the source is in or out of the core. UUTR SAR Section 4.2.4 contains additional information on the UUTR startup source.

The primary function of a neutron source is to provide neutrons for reactor startup, and thus, sufficient counts for instrumentation to function properly during startup. TS 3.2.3, Table 3, contains the operational interlock on control rod withdrawal that requires a count rate of greater than 2 counts per second (cps) to allow control rod withdrawal. See Section 2.5.5 of this report for additional discussion on the TS interlock. A neutron-source cladding failure would be

detected during the routine analysis of pool water as required by TS 4.3 and discussed in Section 5.4.3 of this report.

The current source strength is approximately 6.74×10^6 neutrons per second. A fission counter is used with a transistorized linear amplifier to provide an indication of source neutrons to the operator. The meaningful count rates range from approximately 10^{-3} watts (W) to about 2 W (source level). These source levels are estimated to give count rates of about 5 cps and 10,000 cps, respectively.

The NRC staff finds that the startup source used at UUTR is similar to that used in other TRIGA reactors. Based on a review of the information in UUTR SAR Section 4.2.4, the NRC staff concludes that the UUTR source has sufficient strength to allow for controlled reactor startup, including providing source neutrons to support operation of the interlocks described in TS 3.2.3, and is therefore acceptable to the NRC staff.

2.2.5 Core Support Structure

The UUTR core support structure, including the reactor tank, consists of an inner aluminum liner welded to a sheet of aluminum (6061-T6), outer stainless-steel liner, a reinforced concrete pad, and tamped sand between the two tanks. This is described in detail in UUTR SAR Section 4.2.5.

The description, design information, and dimensions of the reactor core, core support structure, and bottom and top grid plates are provided in UUTR SAR Figures 4.2-11 through 4.2-13 and are displayed below as Figure 2-2.

The core components are contained between the top and bottom aluminum grid plates. The top grid plate has 126 positions for fuel elements and control rods arranged in six concentric rings around a central port (used for high-flux irradiations). The coolant flow is provided by holes in the bottom core plate.

The maximum hoop stresses at the bottom of the aluminum tank and stainless-steel tank have been evaluated under a hydrostatic head of 24 feet of water, and the hoop stresses are well below the yield stresses of 246 megapascals (MPa) and 240 MPa, respectively, for these materials.

The 2-foot annulus between the tanks is filled with sand and concrete. On the outside of the inner tank and the inside of the outer tank, where sand is placed, two vertical columns are dug into the sand all the way down to the bottom of the tank. Cameras are used to monitor the sand at the bottom to verify that no water leakage takes place. There is horizontal water shielding of at least 2 feet between the reactor core and the sides of the aluminum tank. The water level in the tank is maintained, in accordance with the TS, at a minimum of 5.5 meters (m) (18 feet) above the top of the core to provide adequate radiation shielding, as well as neutron moderation and fuel cooling.

UUTR is a natural-convection, water-cooled, pool-type reactor. Based on the size and low power of UUTR (100 kW), operation of the primary coolant system is not necessary as a safety system for the facility, but it is used to maintain efficient reactor operation and water quality. The system also removes any particulate and soluble impurities and is managed so as to maintain low conductivity and potential of hydrogen (pH) in the water and optical clarity of the tank water. The reactor pool is open to the atmosphere. The inner tank holds approximately 8,000 gallons of water.

The UUTR tank is placed on a 2-foot concrete block with a stainless-steel cover. The concrete is placed on a clay footing. The beam ports do not penetrate into the inner aluminum tank. The reactor beam ports are currently filled with sand; they were never completely installed and do not breach the reactor tank walls. As such, the licensee states that there is no potential loss of water through the beam ports.

The following TS design feature establishes the basic requirements for the reactor coolant system:

TS 5.2 Reactor Coolant System

Applicability

This specification applies to the tank containing the reactor and to the cooling of the core by the tank water.

Objective

The objective is to assure that coolant water shall be available to provide adequate cooling of the reactor core and adequate radiation shielding.

Specifications

1. The reactor core shall be cooled by natural convection water flow.
2. The reactor tank water inlet and outlet pipes to the heat exchanger and to the demineralizer shall be equipped with siphon breaks not less than 18 feet above the top of the core.
3. A reactor tank water level alarm shall be provided to indicate loss of coolant if the water level drops 15.5 inches from the top of the reactor tank.
4. A reactor tank water temperature shall be kept below 35 °C.

TS 5.2, Specification 1 helps ensure proper UUTR core cooling. Information provided in the UUTR SAR, Sections 4.6.1, 5.1, and 5.2, demonstrate that the UUTR core can be cooled by natural convection flow with the need for forced cooling. The UUTR reactor coolant system is consistent with typical TRIGA and GA design criteria.

TS 5.2, Specification 2 helps ensure that sufficient coolant inventory is available for UUTR cooling and radiation shielding and that coolant cannot be drained inadvertently by siphon action.

TS 5.2, Specification 3 helps ensure that operators will have sufficient awareness of reactor coolant levels.

TS 5.2, Specification 4 helps ensure that the assumptions used in UUTR SAR Sections 5.1 and 5.2, in the accident and DNBR analysis, are preserved and that the resin beds are protected against degradation.

Based upon a review of the information in UUTR SAR Sections 4.6.1, 4.6.3, 5.1, and 5.2, the NRC staff finds that the design of the UUTR tank will provide adequate cooling and shielding for the UUTR. Additionally, TS 5.2 helps ensure that the integrity of the reactor tank is maintained throughout the license renewal period and that water level monitoring will provide an indication of any potential tank leakage. Based on the discussion above, the NRC staff concludes that the design of the UUTR reactor tank, and requirements of UUTR TS 5.2, are acceptable.

2.4 Biological Shield

The UUTR biological shield is described in detail in UUTR SAR Section 4.3. The biological shield is the column of water that surrounds the reactor, the sand that fills the space between the inner and outer tanks, the tank materials, and the earth surrounding the below-grade portion of the UUTR pool structure. The UUTR core is shielded radially by approximately 3.5 feet of water, 5/16 inch of aluminum, 2 feet of sand, 3/16 inch of steel, and 3 feet of concrete and ground dirt, and axially by 22 feet of water above the core and 5/16 inch of aluminum, 2 feet of concrete, and clay under the core structure.

As discussed in SAR Section 11.1.7, environmental monitoring is required to help ensure compliance with 10 CFR Part 20, Subpart F, "Surveys and Monitoring," and with the UUTR TS. Installed monitoring systems include area radiation monitors (ARMs) and airborne contamination monitors. The UUTR SAR indicates that the facility has maintained a comprehensive environmental and facility monitoring program for the last 35 years, and that the program has been effective in quantifying the conclusion that the operation of the facility represents an insignificant impact on local environmental radiation levels and radiation exposure in and around the facility.

The NRC staff reviewed the UUTR annual reports from 2004 through 2010 and finds that the annual releases were below the allowable limits. The NRC staff also reviewed inspection reports from 2003 through 2010 and finds no contradictory information.

Based on a review of the information provided by the licensee in UUTR SAR Section 4.4 and UUTR operational experience, the NRC staff finds that the UUTR biological shield components

are typical of TRIGA reactors, were described accurately in the SAR, and are properly maintained by TS 5.2. In addition, the NRC staff concludes that there is reasonable assurance that the UUTR biological shield design will limit exposures from the reactor and reactor-related sources of radiation.

2.5 Nuclear Design

The information discussed in this section establishes the design bases for other chapters, especially the safety analyses and UUTR TS. The reactor design bases are established by the maximum operational capability for the fuel elements and fuel element configurations. The TRIGA reactor system has five major areas that define the reactor design bases:

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

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

2.5.1 Normal Operating Conditions

The current UUTR core consists of 78 fuel elements: 23 aluminum-clad elements with 8.91 percent burnup (% burnup), 36 stainless-steel elements with 0.61 % burnup, 17 stainless-steel elements with 8.77 % burnup, and 2 stainless-steel IFEs with 8.77 % burnup. The fuel elements are arranged from B-ring to G-ring. The A-ring is empty and is used as a central irradiator.

UUTR used the MCNP5 computer code to perform design confirmation, power distribution, and reactivity coefficient calculations. MCNP5 has been extensively benchmarked and widely used in the RTR community for neutronic evaluations. The UUTR core is consistent with the core loading described in UUTR SAR Section 5.4.2.1 and with geometry and core loading parameters obtained from the manufacturing drawings. SAR Tables 4.5-1 and 4.5-2 show the core power distribution calculated by MCNP5, including average power per ring, as well as maximum and minimum pin powers. The average power in the fuel rings varies from 1.979 kW/element in the B-ring to 0.7 kW/element in the G-ring, with core average power per fuel pin at 1.282 kW/element and a minimum and maximum pin power of 0.609 kW/pin and 2.022 kW/pin, respectively.

The NRC staff reviewed the licensee's use of MCNP5 for the UUTR core analysis and concludes that the analysis in UUTR SAR Section 4.5 met the TRIGA operational limits described in NUREG-1537 and, is therefore acceptable.

TS 3.1.1 Steady-State Operation

Applicability

This specification applies to the energy generated in the reactor during steady-state operation.

Objective

The objective is to assure that the fuel temperature safety limit shall not be exceeded during steady-state operation.

Specifications

The reactor power level shall not exceed 100 kW.

SAR Sections 4.6.1 and 4.6.3 discuss thermal-hydraulic calculations and design analysis and address the fuel temperature limits during steady-state operation of UUTR. The safety analysis discussed in Chapter 4 of this report assumes a maximum reactor power of 100 kW. The maximum fuel temperature at 100 kW is 129 degrees C, as provided in UUTR SAR Table 4.6-1. The TS operational limit of 100 kW is maintained through operator observation of reactor power instrumentation and limited by the reactor scram functions provided in TS 3.2.3.

The UUTR reactor is licensed to operate at measured powers up to 100 kW. The SLs require that the maximum temperature in the TRIGA aluminum- and stainless-steel-clad fuel not exceed those values set by ring location for each type of fuel and that the reactor power level not exceed 100 kW steady state under any conditions of operation. To comply with these SLs, limiting safety system settings (LSSs) are established for the fuel temperature, as measured by two IFEs. Therefore, steady-state operation of UUTR at 100 kW with an LSS based on fuel clad type and core location allows for a sufficient safety margin.

The NRC staff finds that TS 3.1.1 is appropriate and effective to ensure that UUTR maintains operational limits with the UUTR SAR design analysis and concludes that TS 3.1.1 is acceptable.

2.5.2 Reactor Core Physics Parameters

Calculational Methodology

The licensee modeled the UUTR core using the MCNP5 code. Since the UUTR power level is administratively limited to 90 kW, all measured data reported in the SAR was obtained at 90 kW. Several comparisons between measured and calculated values of parameters were performed using calculations at 90 kW to determine the validity of the MCNP results. However, the calculated values used for the purpose of licensing and demonstrating the adequacy of the TS and safety analysis were at the licensed power level of 100 kW.

Excess Reactivity, Shutdown Margin, and Control Rod Worth

The MCNP5 core model for UUTR was validated by comparing the calculated excess reactivity, shutdown margin (SDM), and control rod worth with the corresponding measured values for

these parameters in the UUTR core. Information from SAR Table 4.5-- is provided below as Table 2-6, which summarizes calculated and measured excess reactivity, SDM, and control rod worth. The variability in the comparisons of the control rod worths is approximately 10 percent and is considered acceptable to the NRC staff. The variability in the excess reactivity and shutdown margin are also acceptable to the NRC staff. Considering the measurement variability and MCNP5 simulation confidence interval, the comparison results provided below are acceptable to the NRC staff.

Table 2-6 UUTR Comparisons of Measured and Calculated Core Parameters

UUTR Comparisons of Measured and Calculated Core Parameters			
Component	Measurement average 2005 to 2009 at 100 kW (M-measured)	MCNP5 Calculation (C-calculated)	% Difference (\$)
Excess Reactivity (\$)	.819	.840 ±0.010	-2.53
Shutdown Margin (\$)	1.018	.980 ±0.023	3.80
Safety Control Rod (\$)	2.243	1.924 ±0.035	15.31
Shim Control Rod (\$)	1.550	1.468 ±0.031	5.43
Regulating Control Rod (\$)	.287	.294 ±0.022	-2.41

The NRC staff concludes that the control rod worths, both calculated and measured, are acceptable.

Figure 2-3 is provided below using information obtained from SAR Table 4.2-5. The rod worths for the safety, shim, and regulating control rods are based on the measured values performed every 6 months, and demonstrate consistent data.

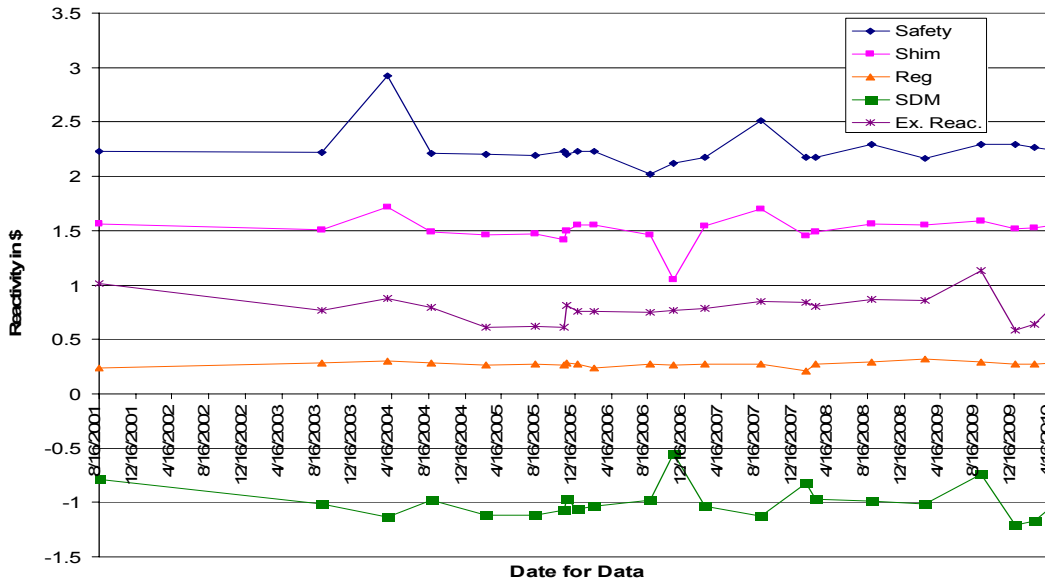


Figure 2-3 UTR control rod worths, shutdown reactivity, and excess reactivity

Excess Reactivity

The licensee states that the purpose for monitoring excess reactivity is two-fold. First, it is a component of the SDM calculation, which is a basic safety requirement. Second, the change in excess reactivity with burnup is predictable and consistent; this change is reviewed over time to monitor for reactivity anomalies.

TS 3.1.3 presents the requirements for core excess reactivity as follows:

TS 3.1.3 Core Excess Reactivity

Applicability

This specification applies to the reactivity condition of the reactor and the reactivity worths of control rods and experiments. It applies for all modes of operation.

Objective

The objective is to assure that the reactor can be shut down at all times and to assure that the fuel temperature safety limit shall not be exceeded.

Specifications

The maximum available excess reactivity based on the reference core configuration shall not exceed \$1.20.

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

Core excess reactivity includes the reactivity contribution from the fuel elements, control rods, and experimental components. Knowledge of these components is necessary to correctly determine the shutdown reactivity of UUTR, so that the SDM can be confirmed.

Since UUTR fuel has no burnable poisons, the reactivity of the fuel is reduced by reactor operation. The licensee has calculated (+\$0.840) and measured (+\$0.819) this parameter (SAR Table 4.5-5).

Based on a review of UUTR SAR Section 4.5, the NRC staff concludes that UUTR has selected the minimum excess reactivity which will allow the reactor to operate in accordance with the TS while allowing for operational flexibility. The NRC staff concludes that TS 3.1.3 is acceptable.

Shutdown Margin

SDM ensures that the reactor can be shut down under all operational conditions. The value often used by TRIGA research reactors and used by the licensee is -\$0.50. NUREG-1537 requests that the applicant define their LCC and then characterize the operating characteristics (see NUREG-1537, Section 4.5.1). UUTR has incorporated the SDM requirements into the TS as described below:

TS 3.1.2 Shutdown Margin

Applicability

These specifications apply to the reactivity condition of the reactor and the reactivity worths of control rods and experiments. They apply for all modes of operation.

Objective

The objective is to assure that the reactor can be shut down at all times.

Specifications

The reactor shall not be operated unless the shutdown margin provided by control rods shall be greater than \$0.50 with:

1. The irradiation facilities and experiments in place and the total worth of all non-secured experiments in their most reactive state,
2. The most reactive control rod fully-withdrawn, and,

3. The reactor in the reference core condition.

TS 3.1.2, Specification 1 helps ensure constraints on the core condition by considering the highest worth unsecured experiment to be in its most reactive state, to help ensure that the reactor remains subcritical, should a unsecured experiment move to its most reactive position.

TS 3.1.2, Specification 2 helps ensure that the reactor can be shut down even if the most reactive control rod becomes stuck out of the reactor core.

TS 3.1.2, Specification 3 helps ensure proper core reference conditions for deriving the SDM value. The reactivity state of a reactor can be affected by the fission product xenon, which is a neutron poison, and the temperature of the reactor. The purpose of defining a reference core condition is so that reactivity measurements can be adjusted to a fixed baseline. The reference core condition is the most limiting for determining the SDM.

SDM ensures that the reactor can be shut down under all operational conditions. The value generally used by research reactors and used by the licensee is $-\$0.50$. NUREG-1537 requests that the applicant define their LCC and then characterize the operating characteristics (see NUREG-1537, Section 4.5.1). The configuration described in the SAR Section 4.5.2 represents the LCC for UUTR. The maximum excess reactivity of the core allowed is $\$1.20$, in accordance with TS 3.1.3. The SDM requirement specifies that the reactivity of the core is at least $-\$0.50$ with the maximum worth control rod is stuck in the fully withdrawn position. The SDM is highly dependent upon the knowledge of the excess reactivity (ρ_x) and control rod worths. Maintenance of the SDM is assured by TS 3.1.4 and 4.1(4). The surveillance requirements in TS 4.1(2) determine control rod worths. TS 4.1(3) provides the determination of ρ_x which includes the fuel reactivity (ρ_F), rod position, and reactivity of experiments.

The technical justification for the SDM specifications provided is discussed in SAR Sections 4.2 and 4.5.3.9. From SAR Table 4.5-5, the maximum worth control rod is the safety rod having a value of $\$1.924$ (calculated) and $\$2.243$ (measured), and it represents the stuck rod for all SDM calculations (all rods in but with the maximum worth rod stuck out).

The control rods were discussed in Section 2.2.2 of this report.

Shutdown Margin—Confirmatory Analysis

The NRC staff performed a confirmatory analysis of the UUTR SDM using both measured and calculated control rod worths for various scenarios described in Table 2-7 below.

Table 2-7 UUTR Shutdown Margin Calculations

UUTR Shutdown Margin Calculations						
Calculation No. (Strongest Rod Withdrawn)	Initial excess reactivity (TS 3.1.3)	Shim Rod (ρ_{SH})	Safety Rod (ρ_S)	Reg. Rod (ρ_R)	Shutdown Reactivity	SDM Req. (ρ_{SDM})
		using calculated rod worths				
1	+\$1.20	-\$1.468	+\$0.000	-\$0.294	-\$0.562	-\$0.500
		using measured rod worths				
2	+\$1.20	-\$1.550	+\$0.000	-\$0.287	-\$0.637	-\$0.500

Calculation No. 1 – Calculated Rod Worths

In this calculation, the licensee has demonstrated that the TS 3.1.3 limit of the core excess reactivity (\$1.20) is offset by insertion of all control rods except the maximum worth control rod (-\$1.468 and -\$0.294). Calculated values of the control rod worths are used. As stated in TS 3.1.2 and TS 3.1.3, the value (+\$1.20) of the initial core reactivity includes all components, including the reactivity of the fuel and the experiments. The result is a shutdown reactivity (-0.562) that is more negative than the SDM requirement.

Calculation No. 2 – Measured Rod Worths

In this calculation, the licensee has demonstrated that the TS 3.1.3 limit of the core excess reactivity (\$1.20) is offset by insertion of all control rods except the maximum worth control rod (-\$1.55 and -\$0.287). Measured values of the control rod worths are used. As stated in TS 3.1.2 and TS 3.1.3, the value (+\$1.20) of the initial core reactivity includes all components, including the reactivity of the fuel and the experiments. The result is a shutdown reactivity (-0.637) that is more negative than the SDM requirement.

Based on the above discussion, the NRC staff concludes that the licensee’s values for measured and calculated excess reactivity, SDM, and control rod worths are acceptable.

Effective Delayed Neutron Fraction

The effective delayed neutron fraction, β_{eff} , calculated by MCNP5, is 0.00768 ± 0.00006 . The NRC staff finds that the calculated value of β_{eff} is comparable with other TRIGA reactor β_{eff} values. On this basis, the NRC staff concludes that the licensee’s calculated values for β_{eff} are acceptable.

On the basis of the discussion above, the NRC staff finds that UUTR has acceptable control rod reactivity worths to satisfy the SDM requirement and, therefore concludes that TS 3.1.2 is acceptable.

2.5.3 Reactivity Coefficients

Prompt Negative Fuel Temperature Coefficient

A significant safety feature of the TRIGA reactor is the large, prompt, negative fuel temperature coefficient (FTC) of reactivity, resulting from the intrinsic characteristics of the U-ZrH_x fuel material at elevated temperatures. The negative temperature coefficient results principally from the neutron hardening at elevated temperatures, which increases the leakage (and loss) of neutrons from the fuel-bearing material into the water moderator material. This reactivity decrease is a prompt effect and occurs more rapidly than any change to fuel, clad, or moderator temperature. An additional contribution to the prompt, negative temperature coefficient is the Doppler broadening of fuel resonances, which also increases neutron capture.

Because of the large, prompt, negative FTC, a step insertion of reactivity resulting in an increasing fuel temperature will be rapidly compensated for by the fuel material. This dampens any power excursion before the electronic or mechanical reactor safety systems or the RO can respond. Also, changes of reactivity resulting in a change in fuel temperature during steady-state operation can be rapidly compensated for by the fuel material, thus limiting the reactor steady-state power level, as discussed in GA-4314 (Ref.16).

The NRC staff finds that the FTC represents the change in reactivity per degree change in the fuel temperature and is calculated by varying the fuel temperature while keeping all other core parameters fixed. The effective delayed neutron fraction (SAR Table 4.5-8) is used to convert the multiplication factor to reactivity. The UUTR MCNP5 model was used to represent the core in an all-rods-out condition. UUTR SAR Section 4.5.3.3 provides the calculation of the FTC over a range of temperatures. The calculations have been examined in detail, compared with those of several other licensees, and compared with well established vendor calculations. The NRC staff concludes that the UUTR FTC is properly calculated and is correct.

Prompt Negative Fuel Temperature Coefficient—Confirmatory Analysis

The NRC staff performed a series of calculations of the UUTR FTC using a unit cell in an infinite lattice. The general model in Figure 2-4 below contains a central rod region for stainless-steel-clad fuel elements. This region contains only fuel in aluminum-clad fuel elements. The physical dimensions of the model were taken from the UUTR SAR.

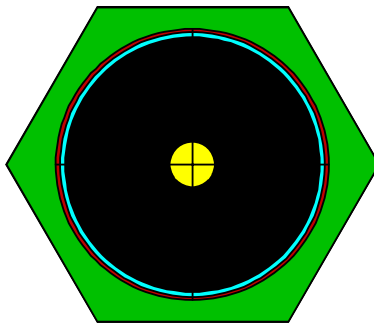


Figure 2-4 WIMS model of UUTR fuel elements

The NRC staff used the program WIMS-ANL (see “ANL/TD/TM99-07, “WIMS-ANL User Manual,” Revision 6, issued February 2004 (Ref. 18)), to perform the confirmatory analysis. This program uses a 69-group library that was specifically developed for RTR confirmatory analysis. This library has nuclear cross-section data covering a wide range of temperatures (300–1,600 kelvin (K)) and used representative spectra for TRIGA fuel. The confirmatory calculations were performed at seven temperatures of interest (31, 150, 300, 400, 600, 800, and 1,000 degrees C). At each temperature, a pair of eigenvalue calculations was performed (e.g., for 150 degrees C, confirmatory calculations were performed at 145 and 155 degrees C). Coefficients were calculated at each temperature of interest. Representative buckling values were used to model core average leakage.

The GA results (Ref. 14), the confirmatory results, and the UUTR results are displayed in Figure 2-5 below. The confirmatory results are consistent with the values and the trend of the GA results as shown in Figure 2-5 below. The UUTR results are core averaged with both stainless-steel and aluminum-clad fuel elements.

The NRC staff concludes that the calculation of the FTC by the licensee was acceptable.

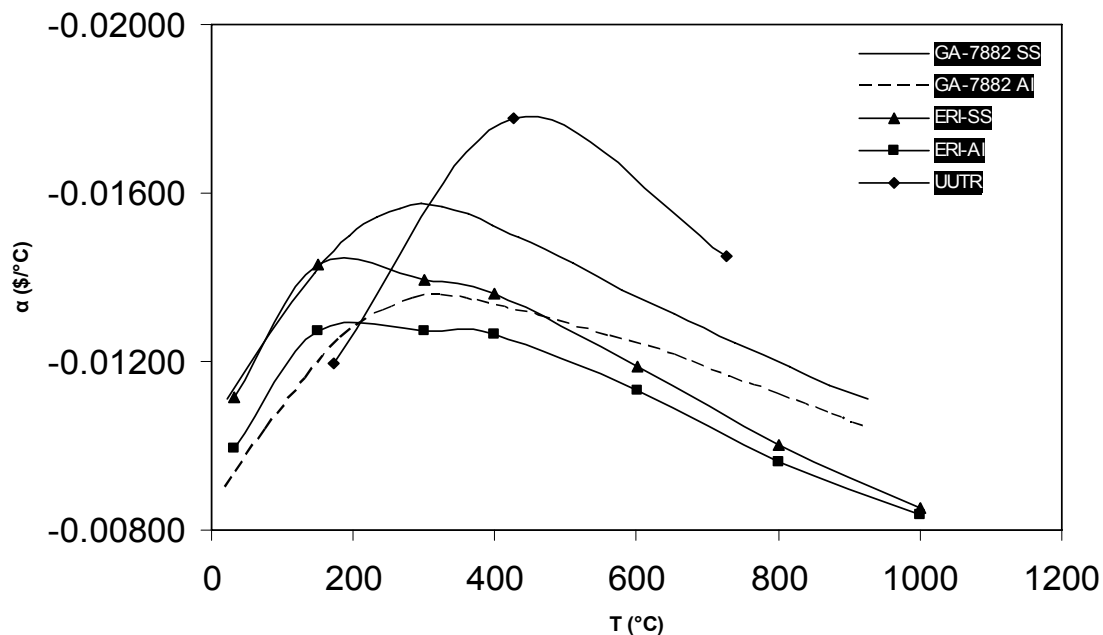


Figure 2-5 GA-7882, confirmatory, and UUTR fuel temperature coefficients

Moderator Temperature Coefficient

The moderator temperature coefficient represents the change in reactivity per degree change in the moderator temperature. The licensee indicated that the moderator temperature coefficient of reactivity was determined by varying the moderator temperature using the MCNP5 model of the 100 kW UUTR, as described in SAR Section 4.5.3.4. The moderator temperature coefficient provided by UUTR was calculated as $\alpha_M = -0.0133 \text{ \$/K}$. The NRC staff finds that α_M is comparable with other TRIGA reactor α_M values. On this basis, the NRC staff concludes that the licensee’s calculated values for α_M are acceptable.

Void Coefficient

The void coefficient of reactivity is defined as the change in reactivity per percent change in void volume and is included in this review because it validates the UUTR neutronics model. The void coefficient in the 100 kW UUTR core was evaluated by varying the void in the moderator from 0 to 75 percent in the MCNP5 core model and evaluating its impact on the reactivity. The calculated void coefficient $\alpha_V = -0.2702 \text{ \$/\%}$, and was provided in UUTR SAR Section 4.5.3.5. The NRC staff finds that α_V is comparable with other TRIGA reactor α_V values. On this basis, the NRC staff concludes that the licensee’s calculated values for α_V are acceptable.

2.5.4 Transient Analysis of an Uncontrolled Rod Withdrawal

UUTR SAR Section 4.5.3.10 analyzes the uncontrolled rod withdrawal scenario. UUTR used the Ramp-Input Response by Hypergeometric Functions technique (see “Dynamics of Nuclear Reactors,” by D. Hetrick, 1971 (Hetrick) (Ref. 19)) and determined that the maximum worth rod

was withdrawn at the nominal withdraw speed which would require 77 seconds for the rod to go from completely inserted to fully withdrawn. The results of the rod withdraw at this nominal speed was a reactivity rate of \$0.053/s until a linear power trip was attained at 22.74 seconds. A maximum reactivity insertion of \$0.69 resulted.

Uncontrolled Rod Withdrawal—Confirmatory Analysis

The NRC staff performed a confirmatory analysis on the uncontrolled rod withdrawal scenario. The ramp insertion model is based on the hyper-geometric function technique presented in the UUTR SAR Section 4.5.3.10. This technique was outlined in Hetrick (Ref. 19), and the results are provided in Figure 2-6 below. In this transient, the highest worth control rod was withdrawn at the maximum withdrawal rate allowed by the UUTR TS until a reactor scram occurs at 100 kW.

77 Second Ramp Rate

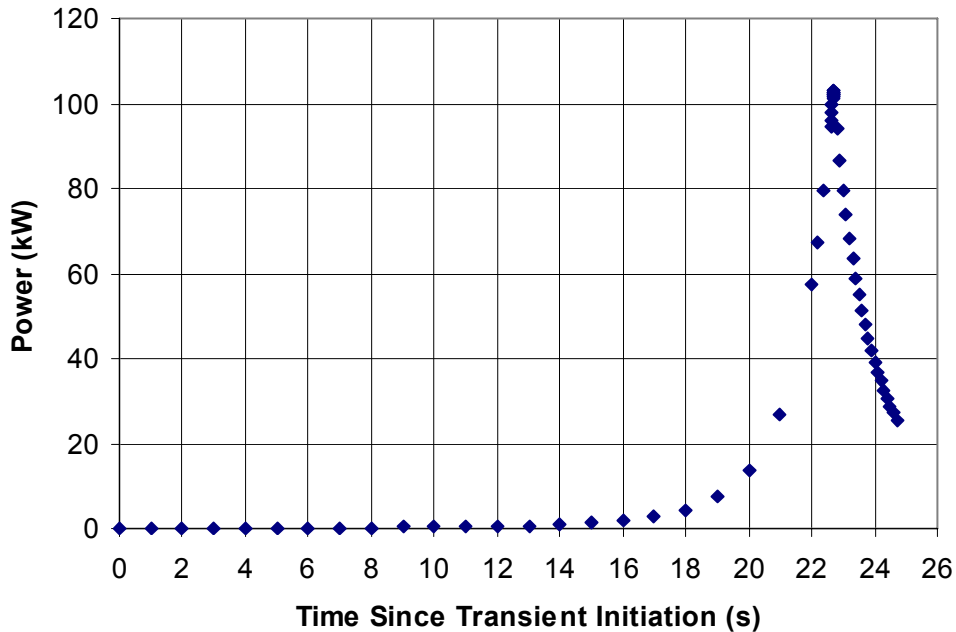


Figure 2-6 Uncontrolled rod withdrawal—77 seconds

In this confirmatory analysis, the trip signal is generated at 100.39 kW and occurs 22.68 seconds after initiation of the transient. Power increases to 103.18 kW because of the 0.02-second delay time before the control rod insertion. The control rods insert at the rate provided by TS 3.2.1, which is 2.0 seconds. A total reactivity of \$0.66 was inserted by the uncontrolled withdrawal of the control rod. The results are acceptable for two reasons: (1) the reactivity insertion limit is not reached (+\$1.20), and (2) the resulting power of 103 kW does not pose a DNBR or fuel temperature problem. Section 2.6 contains the DNBR analysis. This confirmatory analysis replicates the results provided in UUTR SAR Section 4.5.3.10.

The NRC staff performed a confirmatory calculation using the maximum reactivity insertion rate of $\$0.30/s$, as specified in TS 3.2.1, Specification 3. This reactivity insertion rate ($\$0.30/s$) corresponds to a control rod withdraw from fully inserted to fully withdrawn in 7.5 seconds. The results of the 7.5s control rod withdrawal time are provided in Figure 2-7 below.

7.5 Second Ramp Rate

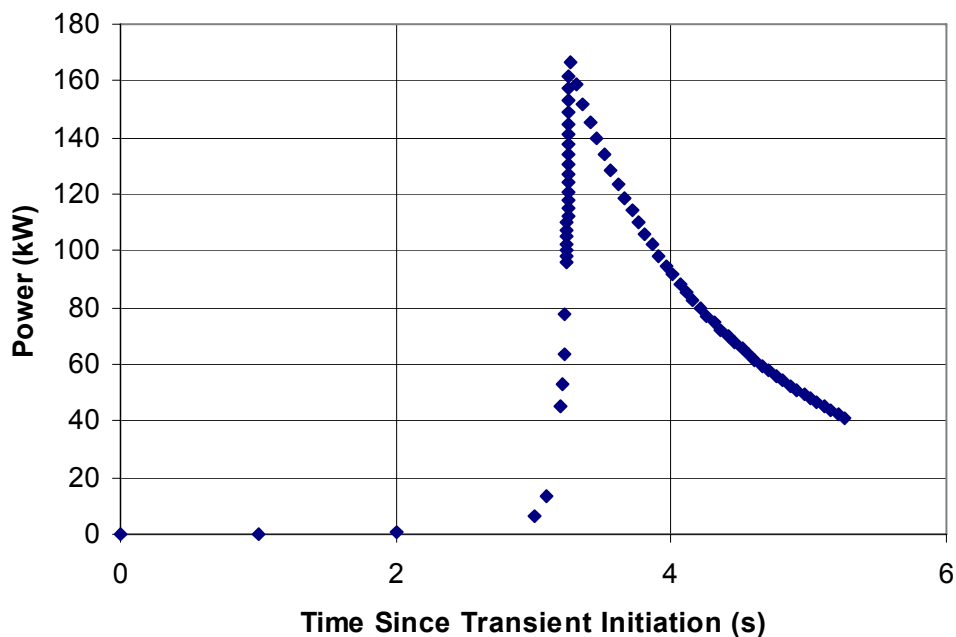


Figure 2-7 Uncontrolled rod withdrawal—7.5 seconds

In this confirmatory analysis, the trip signal is generated at 100.12 kW and occurs 3.242 seconds after initiation of the transient. Reactor power increases to 166.27 kW as a result of the 0.02-second control rod insertion delay. The control rods insert within the TS 3.2.1 time of 2.0 seconds. A total of $\$0.98$ of reactivity was inserted by the uncontrolled withdrawal of the control rod. The results are acceptable because the reactivity insertion limit is not reached ($+\$1.20$), and the resulting reactor power of 166 kW does not pose a DNBR or a fuel temperature challenge.

On the basis of the discussion above, the NRC staff finds that the consequences of this event are acceptable. The NRC staff concludes that TS 3.2.1, Specification 3, is acceptable.

2.5.5 Operating Limits

The regulations in 10 CFR 50.36(d)(1) require reactors to specify SLs and LSSs. SLs are limits upon important process variables necessary to reasonably protect the integrity of the physical barriers that guard against the uncontrolled release of radioactivity.

The principal physical barrier for TRIGA reactors is the fuel element cladding, and the most important parameter to maintain the fuel cladding integrity in a TRIGA reactor is the fuel element temperature. A loss in the integrity of the fuel rod cladding may occur if there is a buildup of excessive pressure between the fuel moderator and the cladding and if the fuel temperature then exceeds the SL. Such pressure is caused by the presence of air, fission product gases, and H from the dissociation of the H and Zr in the fuel moderator. The fuel moderator temperature and the ratio of H to Zr in the alloy determine the magnitude of this pressure.

The SL for the stainless-steel-clad, high-hydride TRIGA fuel is based primarily on experimental evidence obtained during high performance reactor tests on this fuel. These data indicate that the stress in the cladding caused by H pressure from the disassociation of ZrH will remain below the stress limit, provided that the temperature of the fuel does not exceed 1,150 degrees C and the fuel cladding is water cooled. The SL for the aluminum-clad, low-hydride TRIGA fuel elements depends upon avoiding the phase change in the ZrH that might cause excessive distortion of a fuel element. This phase change takes place at 530 degrees C (Ref. 7).

The temperature in a standard TRIGA fuel element in the UUTR core is limited by SL recommendations to a maximum of 1,000 degrees C for stainless-steel-clad, high-hydride fuel elements and to 500 degrees C for aluminum-clad, low-hydride fuel elements under any reactor operating conditions. These SLs are imposed to prevent excessive stress on the cladding because of the H pressure caused by phase change of the U-ZrH_x fuel. Based on the theoretical and experimental evidence (Ref. 16), these limits represent conservative values to provide confidence that the integrity of the fuel elements will be maintained and that no damage to cladding will occur.

LSSSs for nuclear reactors are defined as settings for automatic protective devices related to those variables having significant safety functions. Where an LSSS is specified for a variable on which a SL is placed, the setting must be so chosen that automatic protective actions will correct the abnormal situation before a SL is exceeded.

TS 2.2 provides LSSSs to ensure that there is a considerable margin of safety before the SLs specified above are reached. The limiting safety system temperature settings depend on the location of both the IFE and the aluminum-clad fuel elements. For this reason, the licensee's LSSS contains several limiting safety system temperature settings that vary with the location of the IFE and the remaining core configuration.

The UUTR TS SLs and LSSSs are discussed below.

TS 2.1 Safety Limits

Applicability

This specification applies to the maximum temperature of the reactor fuel.

Objective

The objective is to define the maximum fuel temperature that can be permitted with confidence that a fuel cladding failure will not occur.

Specifications

1. The temperature in a stainless-steel clad, high hydride fuel element shall not exceed 1,000 °C (1,273.15 °K) under any conditions of operation, and
2. The temperature in an aluminum clad, low hydride fuel element shall not exceed 500 °C (773.15 °K) under any conditions of operation.

UUTR SAR Section 7.2.3.2 references NUREG-1282 (Ref. 15), which identifies the SL for TRIGA fuel elements with stainless-steel clad based on the clad stress resulting from H pressure from the dissociation of the ZrH. This stress will remain below the yield strength of the stainless-steel clad if the fuel temperature is below 1,150 degrees C. During operation, fission product gases and dissociation of the H and Zr builds up a gas inventory in internal components and spaces of the fuel elements. Limiting the maximum fuel temperature prevents an excessive internal pressure that could be generated by heating the gases. The temperature at which phase transitions may lead to clad failure in aluminum-clad low-hydride fuel elements is reported to be 530 degrees C. Fuel growth and deformation can occur during normal operation, as described in the GA technical report (Ref. 16).

TS 2.1, Specifications 1 and 2 provide the SL for the stainless-steel- and aluminum-clad fuel in UUTR. They follow the guidance provided in NUREG-1537, Appendix 14.1, Section 2.1, and are therefore, acceptable to the NRC staff.

LSSSs are required for the operation of the reactor in accordance with 10 CFR 50.36, "Technical Specifications." LSSSs are those limiting values for settings of the safety channels by which point protective action must be initiated. The LSSSs need to be chosen so that automatic protective action will terminate the abnormal situation before an SL is reached.

UUTR SAR Section 4.2.1.4 provides that the PARET-ANL (Program for the Analysis of Reactor Transients—Argonne National Laboratory (ANL)) code was used to calculate the fuel average centerline, fuel average surface, and maximum fuel centerline temperature for the UUTR reactor at a power level of 90 kW and 100 kW UUTR. UUTR SAR Table 4.2-3 provides the resulting values. The core inlet pool water temperature for the PARET-ANL calculations was the bulk average temperature of the reactor pool before reactor startup was measured to be 20 degrees C. The maximum fuel centerline temperature in the hottest fuel element (ring B) was calculated to be 121 degrees C at 100 kW, with an average fuel temperature (centerline) of 74 degrees C and average fuel surface temperature of 59 degrees C.

The SAR states the following:

TS 2.2 Limiting Safety System Settings

Applicability

This specification applies to the settings that prevent the safety limit from being reached.

Objective

The objective is to prevent the safety limits from being exceeded.

Specifications

1. For a core composed entirely of stainless steel clad, high hydride fuel elements, or a core composed of stainless steel clad, high hydride fuel elements, with aluminum clad, low hydride fuel elements in the F or G hexagonal ring only, limiting safety system settings apply according to the location of the instrumented fuel as indicated in the following table:

Location of Instrumented Fuel Element	Limiting Safety System Setting for SS Cladding
B-hexagonal ring	800 °C (1,073.15 °K)
C-hexagonal ring	755 °C (1,028.15 °K)
D-hexagonal ring	680 °C (953.15 °K)
E-hexagonal ring	580 °C (853.15 °K)

2. For a core composed of aluminum clad, low hydride fuel elements installed in other than the F or G hexagonal ring, limiting safety system settings apply according to the location of the instrumented fuel element, as indicated in the following table:

Location of Instrumented Fuel Element	Limiting Safety System Setting for Al Cladding
B-hexagonal ring	460 °C (733.15 °K)
C-hexagonal ring	435 °C (700.15 °K)
D-hexagonal ring	390 °C (663.15 °K)
E-hexagonal ring	340 °C (613.15 °K)

As discussed in UUTR SAR Section 4.2.1.4, the PARET-ANL code was used to calculate the fuel average centerline, fuel average surface, and maximum fuel centerline temperature for 90 kW and 100 kW. In UUTR SAR Section 4.6.1, the calculated maximum fuel temperature is 129 degrees C at 100 kW, assuming an inlet temperature of 20 degrees C. The NRC staff finds that, based on confirmatory calculations, if the inlet temperature was increased to the TS limit of 35 degrees C, the resulting affect on the maximum fuel temperature would not increase appreciably. These temperatures are significantly less than the safety limits established in TS 2.1.

UUTR uses the IFE scram setpoint as the LSSS. The IFE scram setpoint is provided in TS 3.2.3 below. The NRC staff finds that the selection of the IFE scram as the LSSS provided adequate protection for UUTR from exceeding the SL, and the setpoint protects the SL. The UUTR reactor is licensed to operate at a power of 100 kW. The SLs require that the maximum temperature in the TRIGA aluminum- and stainless-steel-clad fuel not exceed those values set by ring location for each type of fuel and that the reactor power level not exceed 100 kW steady

state under any conditions of operation. To comply with these SLs, LSSSs are established for the fuel temperature, as measured by two IFEs. Therefore, steady-state operation of UUTR at 100 kW with an LSSS based on fuel clad type and core location allows for a sufficient safety margin.

SAR Sections 4.6.1 and 4.6.3 describe the thermal-hydraulic calculations and design analysis and address the fuel temperature limits during steady-state operation of UUTR. The accident analysis previously presented assumes a maximum reactor power of 100 kW.

On the basis of the discussion above, the NRC staff concludes that TS 2.2, Specifications 1 and 2 are adequate to help ensure safe operation of the facility and are acceptable to the NRC staff.

The UUTR TS 3.2.3, Reactor Safety Systems, is presented below:

TS 3.2.3 Reactor Safety System

Applicability

This specification applies to the reactor safety system channels.

Objective

The objective is to specify the minimum number of reactor safety system channels that shall be available to the operator to assure safe operation of the reactor.

Specifications

The reactor shall not be operated unless the minimum number of safety channels described in Table 1 and 2 are operable¹.

Table 1. Minimum Reactor Safety Channels

Safety Channel	Minimum Number Operable	Function
Fuel element temperature	1	Scram at 200 °C (473.15 °K)
Linear power level ²	1	Scram at 100 kW
Percent power level ²	1	Scram at 110% of full licensed power
Manual Console scram	1	Manual scram
Magnet current key switch	1	Manual scram
Console power supply	1	Scram on loss of electrical power
Reactor tank water level	1	Scram at 15.5 inches below the top of the UUTR tank

Table 2. Minimum Interlocks

Safety System Interlock	Minimum Number Operable	Function
Startup count rate interlock	1	Prevent control rod withdrawal when neutron count rate is less than 2 counts per second
Control rod withdrawal interlocks	All control rods	Prevent manual withdrawal of more than one control rod simultaneously

¹If any required safety channel or interlock becomes inoperable while the reactor is operating for reasons other than identified in this TS, the channel shall be restored to operation within 5 minutes or the reactor shall be immediately shut down.

²Any single linear or percent power level channel or interlock may be inoperable while the reactor is operating for the purpose of performing a channel check, channel test, or channel calibration.

Linear Power Setpoint

The linear power setpoint activates a scram at 100 kW. The calculated maximum centerline fuel temperature provided in UUTR SAR Section 4.6.1 is 129 degrees C at 100 kW. This temperature is below the limits specified in TS 2.2. The NRC staff finds that the linear power setpoint of 100 kW ensures that UUTR will operate in compliance with NRC license conditions and provides protection for the safety limit. On this basis, the NRC staff concludes that the linear power setpoint is acceptable.

Percent Linear Power Setpoint

The setpoint for the percent linear power setpoint automatic trip is 110 percent of licensed reactor power. This scram channel provides a backup to the linear power setpoint, which activates a scram at 100 kW. The 110 percent power scram also protects the SL, as the calculated fuel temperature provided in UUTR SAR Section 4.6.1 is 136.43 degrees C. This temperature is below the limits specified in TS 2.2. The NRC staff finds that the percent linear power setpoint ensures protection for the safety limit. On this basis, the NRC staff concludes that the percent linear power setpoint is acceptable.

Fuel Temperature Setpoint

The fuel temperature setpoint is conservatively established at 200 degrees C to protect the SLs specified in TS 2.1 and activate a scram before the TS LSSs specified in TS 2.2. The NRC staff finds that the fuel temperature setpoint ensures protection for the safety limits. On this basis, the NRC staff concludes that the fuel temperature setpoint is acceptable.

Reactor Tank Level Setpoint

The reactor tank level setpoint provides an alarm and scram signal if the level decreases below 15.5 inches from the top of the tank. The NRC staff finds that this setpoint provides protection to the reactor core heat sink inventory and radiation shielding by having a sufficient inventory of water in the reactor tank. This setpoint also alerts the UUTR staff to a potential tank leak as

described in Section 2.3 of this report. The NRC staff concludes that the reactor tank level trip at 15.5 inches is acceptable.

Interlocks

The startup count rate and control rod withdrawal interlocks are typical of TRIGA facilities. The NRC staff finds that the interlocks are consistent with the guidance in NUREG-1537, Appendix 14.1, Section 3.2(5). The NRC staff also reviewed the footnotes to the cited portions of the TS and concludes that they are acceptable.

The NRC staff finds that the interlocks provided in TS 3.2.3 are consistent with other TRIGA reactor facilities, are appropriate to UUTR operation, have been properly considered in the UUTR SAR, and are acceptable.

2.6 Thermal-Hydraulic Design

The important parameter in the thermal-hydraulic design of any reactor is the critical heat flux (CHF), which describes the heat flux associated with the departure from nucleate boiling (DNB). The parameter of interest is the DNB ratio (DNBR), which is the ratio of the CHF to the maximum heat flux at full power.

The licensee presented a detailed analysis of the UUTR DNBR using a PARET-ANL model of the UUTR core (see "PARET-A Program for the Analysis of Reactor Transients," IDO-17282, Idaho National Laboratory, 1969 (Ref. 20)) in UUTR SAR Section 4.6. The evaluation of the safety margin that exists during the operation of UUTR at the licensed power level is based on this analysis.

PARET-ANL was primarily used for the design and analysis of thermal-hydraulics of the test and research reactors with pin and plate fuel types. The PARET-ANL code has been extensively compared to the SPERT I, and SPERT II experiments and has been validated extensively. PARET-ANL has also been used to analyze pulsing TRIGA reactors. PARET-ANL has been benchmarked against the RELAP5/MOD3 codes used to analyze a series of benchmark transients specified in "IAEA Research Reactor Core Conversion Guidebook" (Ref. 21).

The licensee performed the UUTR thermal-hydraulics analysis using a two-channel model with the hottest channel and the average channel representing the rest of the core. Both the radial power distribution and axial power distribution in the core were calculated using MCNP5. In the case of the UUTR core, the hottest channel is located in the B-ring of the UUTR core and is shown in SAR Figure 4.6-1. The channel is divided into 19 axial regions, and SAR Figure 4.6-2 shows the peaking factors for both the hottest channel and the average channel.

To demonstrate that the PARET-ANL correctly models the thermal-hydraulic condition of UUTR, the licensee evaluated the core average exit temperature and compared it against measured values. By measuring exit temperatures in each ring of the core at several locations and averaging these temperature distributions across the core, the licensee obtained the core average moderator exit temperature. The calculated core average moderator exit temperature was 21.75 degrees C compared to a measured value of 23.1 degrees C—a difference of about

5.66 percent. Such a difference is in the range of measurement error. This comparison validates the accuracy of PARET-ANL for calculating the thermal-hydraulic conditions of UUTR.

In a fully developed nucleate boiling regime, the heat flux can be increased without a significant increase in the surface temperature of the fuel element cladding up until the point of DNB. In the subcooled boiling regime, the CHF is a function of the following parameters:

- coolant velocity
- degree of subcooling
- pressure

To evaluate the DNBR for the UUTR core, the licensee used the Bernath correlation in PARET-ANL, which GA has historically used for that purpose. The UUTR DNBR evaluation also presents DNBR analyses for other TRIGA reactors, using both Bernath and Groeneveld correlations that clearly demonstrate that the Bernath correlation consistently predicts a more conservative DNBR.

Using PARET-ANL and the Bernath correlation, the licensee has evaluated the UUTR DNBR. The results for DNBR and CHF are given in UUTR SAR Figures 4.6-9 and 4.6-10, respectively, and are reproduced below in Figure 2-8. It is clear from this evaluation that the DNBR stays above 8.0 for various coolant inlet temperatures in the hottest channel. This DNBR value represents a safe region for the reactor operation in terms of fuel and clad integrity.

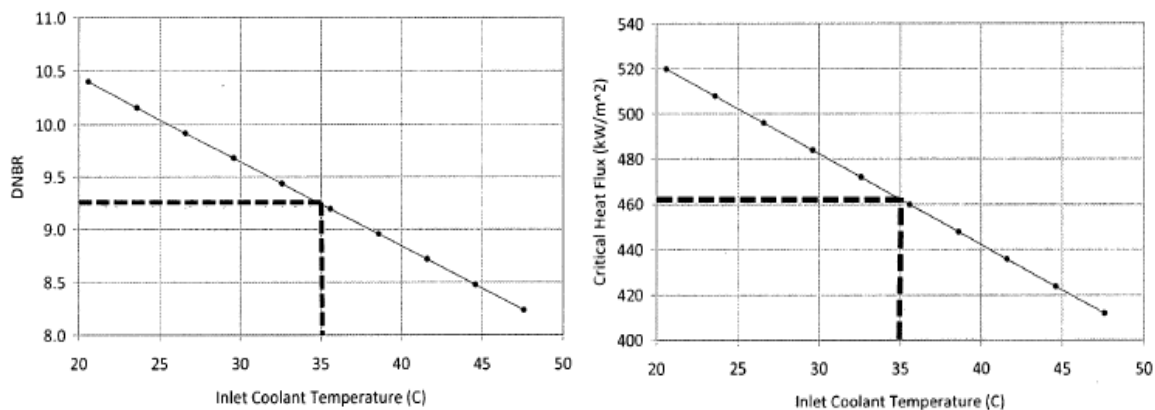


Figure 2-8 DNBR and CHF for UUTR

The licensee also provided calculations for a pool temperature up to 90 degrees C, even though the UUTR TS limit the pool temperature to 35 degrees C. The calculations for a pool temperature of 90 degrees C results in a DNBR of 5.11, and is still acceptable for the safe operation of UUTR. The NRC staff finds that, within the valid pool water temperatures at atmospheric pressure, the DNBR will always stay in an acceptable range at the licensed power level. The NRC staff concludes that the calculated minimum DNBR of 8.0 was expected and is consistent with the DNBR results provided by other TRIGA reactors. The NRC staff concludes that the DNBR evaluation for UUTR is acceptable.

The UUTR coolant circulates by natural convection, resulting in the flow velocity in a given channel increasing as the power in the reactor, and subsequent temperature, rises. Higher flow velocity results in better heat transfer, thereby increasing the CHF. The NRC staff reviewed the results of the CHF calculations, to evaluate the impact of flow velocity in the hot channel as a function of inlet temperature for various flow velocities from 0.1 ft/s to 3.0 ft/s.

In UUTR SAR Section 4.6.2, the hot channel flow rate is 223 kilograms (kg)/square meters (m²)-s. In UUTR SAR Table 4.6-1, the core average flow rate at 100 kW is stated to be 115 kg/m²-s. The SAR states that the DNBR analysis was performed using hot channel power conditions with the core average flow rate. The resulting DNBR at 100 kW is cited as 9.25. Since the CHF is directly related to the flow rate, the use of the hot channel flow rate would make the DNBR even larger. The NRC staff finds that the UUTR assumption regarding the use of the core average flow rate is conservative.

DNBR - Confirmatory Analysis

The NRC staff performed a confirmatory calculation of the UUTR DNBR using the UUTR SAR data and the methods in "Fundamental Approach to TRIGA Steady-State Thermal-Hydraulic CHF Analysis," published by ANL in December 2007 (Ref. 22). Given this method, the NRC staff calculated a hot rod power of 2.022 kW without uncertainties. This power gives a natural circulation flow rate of 0.043 kg/s. At this flow rate, the critical heat flux using the Bernath correlation was 27.22 kW. Under these conditions, the DNBR was estimated to be approximately 13.5. This value is consistent with the UUTR value of 10.7 as provided in SAR Table 4.6-1. Therefore, the NRC staff concludes that the DNBR values provided by UUTR are acceptable.

The licensee also developed a model to calculate pool temperature and compared it with the measurement. For the range of operation of UUTR, the highest error obtained is 3.65 percent, which is well within an acceptable range. The NRC staff reviewed the licensee's model and calculations and concludes that the methodology used and results obtained are correct. The NRC staff concludes that the thermal-hydraulic analysis for UUTR demonstrates that the reactor can operate at its licensed power level with sufficient safety margin.

Based on its review of the licensee's submission, the NRC staff finds that the UUTR thermal-hydraulic design analysis was typical of TRIGA reactors, was described in the SAR appropriately, and was properly controlled and implemented in the TS. On this basis, the NRC staff concludes that the information submitted regarding thermal-hydraulic design is acceptable.

UUTR has provided TS pertaining to the conditions of the coolant system as follows:

TS 3.3 Coolant System

Applicability

This specification applies to the primary water of the reactor tank.

Objective

The objective is to assure that there is an adequate amount of water in the reactor tank for fuel cooling and shielding purposes, and that the bulk temperature of the reactor tank water remains sufficiently low to guarantee reactor tank integrity.

Specifications

The reactor primary water shall exhibit the following parameters:

1. The reactor tank water level alarm shall indicate loss of coolant if the tank water level decreases greater than 15.5 inches from the top of the UUTR water tank,
2. The reactor tank water temperature shall be less than 35 °C (308.15 °K),
3. The conductivity of the reactor tank water shall be less than 5 μmhos/cm,
4. The pH shall be between 5.5 and 7.5, and,
5. The reactor shall not be operated if the radioactivity of reactor pool water exceeds the limits of 10 CFR 20 Appendix B Table 3 for radioisotopes with half-lives > 24 hours.

TS 3.3, Specification 1 helps ensure that an adequate inventory of reactor water is available for cooling and radiation shielding purposes. The TS basis is supported by information presented in UUTR SAR Sections 4.3, 4.5.3, and 5.2. As stated in the UUTR SAR, the minimum height of 18 feet of water above the top of the core provides sufficient water for effective cooling of the fuel and limits radiation levels at the top of the reactor.

TS 3.3, Specification 2 provides a bulk water temperature limit to help ensure that the aluminum reactor tank maintains its structural integrity, in accordance with the reactor manufacturer's recommendation, and to protect the reactor pool cleanup system resin from overheating. This specification is provided in UUTR SAR Sections 4.3 and 5.2.

TS 3.3, Specifications 3 and 4 help ensure that the conductivity of the tank water is maintained at or below 5 mhos/cm and that the pH level is kept between 5.5 and 7.5 to control corrosion. The licensee states that a small rate of corrosion continuously occurs in a water-metal system. Limiting this rate extends the longevity and integrity of the fuel cladding. It also ensures that the heat transfer between the cladding and coolant will not degrade because of oxide buildup on the cladding. A pH limit between 5.5 and 7.5 is consistent with other TRIGA reactors and the guidance provided in NUREG-1537.

TS 3.3, Specification 5 helps ensure that the radioactive content of the primary cooling water will be low and known in the event of pool leakage. TS 3.3, Specification 5, is consistent with the guidance in NUREG-1537, Section 5.2, and requires prudent oversight of radiological conditions

in the coolant. Such monitoring will likely detect fuel failure long before continuous air monitors (CAMs) or ARMs would.

The NRC staff reviewed SAR Chapter 13.2.3. The NRC staff finds that if the UUTR reactor is operated in accordance with the TSs, the radiological consequences of a loss of primary coolant are acceptable. The licensee has the ability to detect and contain potential leakage from the tank. On this basis, the NRC staff concludes that TS 3.3, Specifications 1 through 5, are acceptable.

2.7 Reactor Description Conclusions

Based on the above considerations, the NRC staff concludes that the licensee has presented adequate information and analyses to demonstrate the technical ability to configure and operate the UUTR core without undue risk to public health and safety or the environment. The NRC staff review of the facility included studying its design and installation, its controls and safety instrumentation, its operating procedures, and its operational limitations, as identified in the TS. The NRC staff concludes that the thermal-hydraulic analysis in the UUTR SAR demonstrates that the UUTR core results in acceptable safety margins with regard to thermal-hydraulic conditions.

The licensee's analyses used qualified calculation methods and conservative or justifiable assumptions. The applicability of the analytical methodology is demonstrated by comparing analytical results with measurements obtained from the UUTR core. The NRC staff reviewed the analysis of the steady-state operation of the UUTR core at a power level of 100 kW and finds that the maximum core fuel temperature remains below the limit set by the known mechanical and thermal properties of the fuel. The NRC staff concludes that the UUTR TS regarding the reactor design, reactor core components, reactivity limits, and related surveillance requirements provide reasonable assurance that the reactor will be operated safely in accordance with the TS. The NRC staff concludes that there is reasonable assurance that UUTR is capable of safe operation up to 100 kW, as limited by the TS, for the period of the requested license renewal.

3. RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

3.1 Radiation Protection

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

The NRC inspection program routinely reviews radiation protection and radioactive waste management at UUTR. The licensee's performance history in this area is acceptable, and the SAR provides acceptable documentation of the licensee's management of radiation protection.

3.1.1 Radiation Sources

The NRC staff reviewed the descriptions of potential radiation sources, including the inventories of each physical form and their locations. The radiation sources at UUTR can be categorized as airborne, liquid, and solids, as presented in Chapter 11 of the UUTR SAR.

Airborne Radiation Sources

During normal operations of UUTR, the primary airborne sources of radiation are argon (Ar)-41 and nitrogen (N)-16. Ar-41 results from irradiation of the air in experimental facilities and dissolved air in the reactor pool water. The primary means of Ar-41 production is by thermal neutron capture by natural Ar-40. N-16 is produced when oxygen in the pool water is irradiated by the reactor core. The NRC staff's review considered the licensee's calculations of the production and release of routine airborne radioactive effluents and the resultant doses to the UUTR staff and members of the public.

Ar-41 is produced in the UUTR reactor core in the FNIF, in the pneumatic irradiator (PI), and in the Ar gas from the atmosphere dissolved in the reactor primary coolant. Licensee calculations show that the production of Ar-41 in the FNIF and the PI provide a trivial source in the reactor room because the activation volumes are small, with the combined concentrations expected to be about 3 orders of magnitude less than the Ar-41 DAC from 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, which is 3×10^{-6} microcuries per cubic centimeter ($\mu\text{Ci}/\text{cm}^3$) for a semi-infinite cloud.

Since the core is cooled by natural convection of the pool water, the heated water rises to the surface of the pool, along with the air dissolved in the coolant water and Ar-41 produced by activation. Some of the Ar-41 escapes into the air in the reactor room, where it is exhausted by the building ventilation system and released through the ventilation stack. The licensee determined compliance with the DAC in the reactor room by conservatively estimating the concentration in the reactor room (not crediting the reactor room exhaust), assuming steady-state operations at 100 kW, and then comparing this with operational measurements, as indicated in SAR Section 11.1.1.1.6. For steady-state operation at 100 kW, the resulting concentration was estimated to be $6.4 \times 10^{-7} \mu\text{Ci}/\text{cm}^3$. The licensee compared the calculated values with operational measurements of Ar-41 in the reactor room after about 4 hours at 90 kW of reactor operation (with the reactor room exhaust system operating), which averaged $2.67 \times 10^{-8} \mu\text{Ci}/\text{cm}^3$. This comparison shows acceptable agreement.

The licensee estimated the radiation dose limits to the maximally exposed member of the public using the steady-state Ar-41 releases (assuming a full year of continuous reactor operation), the flow rate of the reactor room exhaust system, and the steady-state Ar-41 concentration for operations at 100 kW, and applying Sutton's formula for estimating maximum ground level concentrations for a variety of atmospheric stability classes. Appendix B to 10 CFR Part 20 lists the allowable effluent concentration of Ar-41 as $1 \times 10^{-8} \mu\text{Ci}/\text{cm}^3$, which results in 50 millirem per year (mrem/yr) for continuous exposure. The licensee calculated the peak downwind ground level concentration of Ar-41 in the air for this conservative case to be $1.97 \times 10^{-9} \mu\text{Ci}/\text{cm}^3$, which is less than the Ar-41 effluent concentration. The licensee further stated that the reactor operates for about 50 hours per year, which would result in a source term that is less than 1 percent of the continuous operation estimate, underscoring the conservative nature of this calculation.

Two sources of N-16 production were assumed: oxygen in the FNIF and oxygen dissolved in the reactor coolant. The FNIF typically has a 30-minute decay time before it is retrieved from the reactor pool and, with a half-life of 7.13 seconds, N-16 potentially released from the container would be negligible. For the reactor coolant, the concentrations produced from continuous steady-state operations at 100 kW would produce a dose rate of about 0.077 mrem/h. This dose rate is comparable to the dose rate observed during normal reactor operations, which is typically no more than 0.05 mrem/h, and is within the 10 CFR Part 20 occupational dose limit of 5,000 mrem/yr.

The TS 3.7.2 states the following:

TS 3.7.2 Effluents

Applicability

This specification applies to the release rate of ⁴¹Ar.

Objective

The objective is to ensure that the concentration of the ⁴¹Ar, in the unrestricted areas shall be below the applicable effluent concentration value in 10 CFR 20.

Specification

The annual average concentration of ⁴¹Ar discharged into the unrestricted area shall not exceed 1×10^{-8} $\mu\text{Ci/ml}$ at the point of discharge averaged over one year.

The NRC staff reviewed SAR Sections 11.1.1.1.5 through 11.1.1.1.8. The NRC staff finds that the production and control of the UUTR routine airborne radiation sources and atmospheric effluent releases of Ar-41 and N-16 are within the limitations of TS 3.7.2 and 10 CFR Part 20 criteria. The NRC staff concludes that TS 3.7.2 and the information provided by the licensee in the SAR provided reasonable assurance that, during continued normal operation of UUTR, airborne radioactive releases will result in doses to the maximally exposed member of the public on the order of 1 mrem/yr or less, in compliance with 10 CFR Part 20, and will not pose a significant risk to public health and safety or the environment. The NRC staff concludes that TS 3.7.2 is acceptable.

Liquid Radiation Sources

During normal operations of UUTR, SAR Section 11.1.1.2 states that there is no leakage of coolant from the primary or secondary cooling system; thus, there is no liquid radioactive material release to the environment. However, this section does recognize that there is the potential for an evaporative release of tritium produced by neutron activation of deuterium in the reactor pool water, in the heavy-water elements, and in the thermal irradiator (TI). Impurities in the primary coolant become activated by operation of the reactor. Most of this material is captured in mechanical filtration or ion exchange resins. The licensee provided neutron activation calculations to estimate the magnitude of tritium produced and released into the reactor room from evaporation of the reactor coolant water.

The licensee's calculations, described in SAR Section 11.1.1.2.2, indicated that the tritium concentration would be 4.12×10^{-10} Ci/liter, assuming 50 hours/year of reactor operation. The evaporation rate from the reactor pool is estimated to be about 11 liters/day, which would result in a reactor room discharge concentration of 8.60×10^{-14} $\mu\text{Ci/cm}^3$, assuming an air discharge rate of 5.27×10^{10} cm^3/day . In Appendix B to 10 CFR Part 20, the effluent concentration for tritium is 1×10^{-7} $\mu\text{Ci/cm}^3$, and the U.S. Environmental Protection Agency (EPA) discharge limit for tritium is 1.5×10^{-9} $\mu\text{Ci/cm}^3$. The licensee concluded that the tritium generation from the reactor pool water was significantly below both the NRC and the EPA discharge limits. The

NRC staff reviewed the licensee's assessment and finds that the calculations are acceptable.

The UUTR core contains 12 heavy-water reflector elements and a TI that contains deuterium in the form of deuterium oxide (D₂O) as described in SAR Section 11.1.1.2.3. Each heavy-water element contains about 454 grams of heavy water, and the TI contains about 1.54×10^4 grams of heavy water. Although both the heavy-water elements and the TI have been leak tested, the licensee provided a conservative calculation, assuming reactor operation of 50 hours/year, with a release of the generated tritium into the 8,000 gallons of UUTR reactor pool water. Assuming that the reactor pool water evaporates at a rate of 11 liters/day, the licensee estimated that the maximum concentration of the tritium in the stack exhaust would be 7.14×10^{-12} $\mu\text{Ci}/\text{cm}^3$, which is below the NRC and EPA discharge limits. The NRC staff reviewed the licensee's calculations and finds that this estimate is accurate and acceptable.

Liquid radioactive sources from continued normal operation of UUTR are controlled, and airborne releases of tritium from the evaporation of reactor pool water are within the limitations of 10 CFR Part 20 and EPA discharge limit criteria. On the basis of this review, the NRC staff concludes that these sources do not pose a significant hazard to the public or operating personnel.

Solid Radiation Sources

The fission products in the reactor fuel constitute the most significant solid radiation source. Water shielding helps to control this source of radiation. Nonfuel sources include activated reactor components; the Pu-Be reactor startup source; a 1.8 Ci americium-beryllium (Am-Be) neutron source; ion-exchange resins; irradiated samples; lab ware; contaminated clothing from reactor experiments or maintenance; and fixed sources, such as those used for instrumentation calibration.

The main solid radiation source at UUTR is the reactor fuel. Since the fuel elements are stored under 22 feet of water in the UUTR reactor pool, they do not present a hazard to personnel at UUTR or to the public. For experiments involving neutron activation, the expected activity of a sample is calculated before its irradiation and is generally stored in the reactor pool for the decay of short-lived radionuclides until the samples can be safely handled. The main radionuclides reported in activated material are the isotopes of aluminum and sodium. Typical gamma activity associated with irradiated samples is less than 1 μCi .

The NRC staff reviewed the radioactive solid waste at UUTR provided in the annual reports from 2004 through 2010 and finds that the solid radioactive waste has historically been a small quantity and consisted mostly of consumables, such as absorbent materials or protective clothing. The Radiological Health Department (RHD) at UUTR is responsible for the administration of radioactive waste disposal for the UUTR facility. When possible, solid radioactive waste is initially segregated at the point of origin and screened, based on the presence of detectable radioactivity.

The NRC staff finds that solid radioactive sources from normal operation of UUTR are controlled and have resulted in no significant exposures. Based on its review of the history of solid radioactive waste at the UUTR, the NRC staff concludes that the control of solid radioactive sources at UUTR is acceptable.

Direct External Sources

Operation of UUTR creates a source of direct radiation from the core when the reactor is at power. The gamma rays that are produced include prompt gammas from the fission of U-235 and gammas from the fission and activation product inventory in the reactor core. The intensity of the gamma radiation is proportional to the reactor power, while the intensity of the delayed gamma radiation is a function of the operational history of the core and the elapsed time after the reactor is shut down.

In addition to personnel monitoring, UUTR maintains a program of area monitoring in controlled and uncontrolled locations. Average and maximum personnel doses were reported for the period from 2004 through 2010 in the UUTR annual reports. During that period, the highest average dose to the UUTR personnel was 6.9 mrem/yr, with the maximum individual dose reported as 24 mrem/yr. These personnel doses are in compliance with the dose limits for radiation workers required by 10 CFR Part 20.

The NRC staff reviewed annual reports from 2004 through 2010, in which radiation dosimeters in controlled and uncontrolled locations showed that the highest recorded doses were made by a dosimeter attached directly to the reactor tank, which read 119 mrem/yr during 2006. The highest recorded dose in an uncontrolled area was for a dosimeter in an uncontrolled hallway directly above the reactor room. The highest value at this location was recorded as 8 mrem/yr. These results further demonstrate that the annual external doses from reactor operations are within the limits provided by 10 CFR Part 20.

MCNP has been used to model exposure at certain locations using a point source geometry. Although some generalizations about the reactor were made regarding spatial dimensions and dose locations, a sample application of the MCNP model is provided. For the period from August 2008 through June 2009, the reactor operated 13.45 hours (with a thermal energy of 891.2 kW-h). The dosimeter on the reactor tank read 55 mrem, while the MCNP modeling results indicated 65 mrem. The NRC staff finds that the agreement between the dosimeter and MCNP model demonstrates the ability of the UUTR staff to produce satisfactory code-based results for reactor operations.

Based on a review of UUTR historical radiation dose information, the NRC staff finds that direct radiation from normal operation of UUTR is controlled, and radiation doses to personnel and in unrestricted areas are in compliance with the 10 CFR Part 20 requirements. Therefore, the NRC staff concludes that the control of direct radiation sources at UUTR was acceptable.

3.1.2 Radiation Protection Program

Based on 10 CFR 20.1101(a), each licensee is required to develop, document, and implement a radiation protection program commensurate with the scope and extent of licensed activities. Production and use of radioactive materials within the reactor laboratory are subject to the guidelines issued by the UU RHD. The reactor staff accomplishes health physics functions following approved procedures. The NRC regularly inspects the UUTR radiation protection program and finds that, as implemented, it meets the requirements of the regulations. UUTR is operated following internal procedures that fall within the guidelines of UU; the Utah State Division of Radiation Control; Federal regulations; and ANSI/ANS15-11, "Radiation Protection at Research Reactor Facilities," 1993 (R2004) (Ref. 23).

TS 6.3 states the following:

TS 6.3 Radiation Safety

The Radiation Health Physicist from the Radiological Health Department shall be responsible for implementation of the radiation safety program. The requirements of the radiation safety program are established in 10 CFR 20. The program shall use the guidelines of the ANSI/ANS 15.11-1993; R2004, "Radiation Protection at Research Reactor Facilities."

The management of the radiation protection program are the responsibility of the Director of the UUTR facility (Utah Nuclear Engineering Facilities inclusive of UUTR and other associated facilities and laboratories). The Director is responsible for the preparation, audit, and review of the program. The Reactor Safety Committee (RSC) reviews the activities of the Director and audits the program. Surveillance and recordkeeping are the responsibility of the reactor supervisor (RS), who reports to the Director. ALARA activities are the responsibility of the RS and are incumbent on all radiation workers associated with the UUTR facility. Substantive changes in the radiation protection program require the approval of the RSC.

The RS is responsible for radiation protection training. Non-reactor staff are escorted by trained personnel or provided training to access the facility. Radiation training for licensed operators and reactor staff are integrated with the training and requalification program. The goal of facility access training is to provide the knowledge and skills necessary to control personnel exposure. The training includes specific training requirements in 10 CFR Part 19, "Notices, Instructions, and Reports to Workers: Inspection and Investigations," and 10 CFR Part 20, and additional requirements in the UUTR Radiation Protection Plan and the EP.

Operation of the radiation protection program is carried out under the authority of the Director of UUTR and the senior RS, using formal RHD procedures. The procedures cover all aspects of the radiation protection program, including testing and calibration of monitors, working with radioactive materials, facility and environmental monitoring,

worker radiation protection, training, receipt and transfer of radioactive materials, decontamination, personnel access, spill recovery, and ALARA.

The RSC is responsible for auditing all procedures, personnel radiation doses, radioactive material shipments, radiation surveys, and radioactive effluents released to unrestricted areas. In addition, the RSC provides independent reviews, evaluations, advice, and recommendations on items affecting nuclear safety at UUTR.

UUTR and the UU RHD personnel prepared the Radiation Protection Plan. The NRC staff reviewed the Radiation Protection Plan and finds that the UUTR Radiation Protection Plan complies with NRC and State regulations and follows the guidelines described in ANSI/ANS-15.11-1993 (R2004) (Ref. 23). The NRC staff concludes that TS 6.3 is acceptable.

3.1.3 ALARA Program

UUTR established a program designed to keep radiation exposures to personnel ALARA, so as to comply with 10 CFR 20.1101. This includes using methods and procedures that shield radiation sources and personnel; increase the distance between an exposure point and a radiation source; reduce the time a person might be exposed to a given dose rate; contain sources; and use careful, thoughtful, advanced planning when working in an area that might contain a radiation field. The UUTR ALARA program provides various administrative controls to accomplish the ALARA goals. A senior RO (SRO), licensed for UUTR, reviews all experiments involving the reactor.

As part of its commitment to ALARA, the licensee establishes specific goals to ensure that actual exposures are no greater than 10 percent of the occupational limits and no greater than 50 percent of the public limits in 10 CFR Part 20. The RS or the Director of UUTR, or both, are responsible for planning and scheduling operations, experiments, and personnel training. The ALARA policy is consistent with TS 6.3 on radiation safety and the guidance in ANSI/ANS-15.11-1993 (R2004). In addition, UUTR uses contamination control procedures to further minimize radiation exposures. UUTR applies the radiation exposure limits in 10 CFR Part 20 for occupational workers, members of the public, minors, and pregnant women. The UUTR ALARA program also defines and requires surveys, monitoring, radiation records, and personnel dosimetry. The NRC staff reviewed the UUTR ALARA program and finds that the UUTR ALARA program complies with the regulations in 10 CFR 20.1101, and is consistent with the guidance of ANSI/ANS-15.11-1993 (R2004), and provides reasonable assurance that radiation exposures will be maintained ALARA for all UUTR activities.

3.1.4 Radiation Monitoring and Surveying

Radiation levels at UUTR are measured in unrestricted areas at locations in closest proximity to the strongest radiation sources to ensure that acceptably low dose rates exist in those areas. In addition to radiation levels, swipes are taken at the same locations and counted for alpha and beta radiation for contamination control determinations. SAR Section 10.2 states that film and thermoluminescent dosimeters (TLDs) are used to monitor long-term average gamma and neutron doses throughout the

facility, including the main reactor room, the control room, the reactor lab, and the office space directly above the reactor room.

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

- (1) may be necessary for the licensee to comply with the regulations;
- (2) are reasonable under the circumstances to evaluate the following:
 - (i) the magnitude and extent of radiation levels,
 - (ii) concentrations or quantities of radioactive material, and
 - (iii) potential radiological hazards.

The regulations of 10 CFR 20.1501(b) 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.

TS 3.7.1 states the following:

TS 3.7.1 Radiation Monitoring Systems and Effluents

Applicability

This specification applies to the radiation monitoring information, which must be available to the RO during reactor operation.

Objective

The objective is to specify the minimum radiation monitoring channels that shall be available to the operator to assure safe operation of the reactor.

Specifications

The reactor shall not be operated unless the minimum number of radiation monitoring channels is operating as in the accompanying table.

Radiation Monitoring Channels	Number
Area Radiation Monitor (ARM)	1
Continuous Air Monitor (CAM) (particulate, noble gas, and iodine) ¹	1

¹ The reactor can be operable for 48 hours without the CAM system (SAR 5.6) but with the operable ARM system.

SAR Table 11.1-7 lists all radiation monitoring equipment used in the UUTR radiation protection program. This includes four ARMs and three CAMs. This equipment provides a comprehensive set of radiation survey instrumentation that covers, with sufficient ranges, the various types of radiation that may be encountered at UUTR.

TS 3.7.1 helps ensure that at least one reactor room ARM and one CAM for particulate, noble gas, and radioiodine are operable to support reactor operations.

The ARM provides the UUTR operator with an indication of airborne radioactivity. The UUTR operator will scram the reactor if the ARM radiation level is higher than 10 mrem/h. The CAM monitors provide an indication of particulate, noble gas, and iodine releases. TS 3.7.1, Footnote 1, allows the reactor to be operable for 48 hours without the CAM system, as long as portable equipment is used to serve an equivalent purpose. If the CAM setpoint of 10 mrem/h is exceeded, an alarm is sounded. The CAM is located in the radiochemistry laboratory, and the readings display on the reactor console. The CAM draws air from the facility ventilation system and tests it for radioactive noble gases, radioactive iodine, and radioactive airborne particulates. The CAM uses a combination of Geiger-Mueller tubes for ambient radiation levels and NaI detectors for radioactive iodine detection. The readout at the console can be verified against the readout on the unit by holding the CAM module in the calibration mode and visually verifying the agreement of the responses. The CAM provides both visual and audible alarms at both the reactor console and the CAM unit in the event of high readings above the setpoints.

The calculations presented in SAR Section 13.2.1.1 show that, for routine operations, and under the accident scenarios identified in the UUTR SAR Chapter 13, predicted occupational and general public doses are below the applicable annual limits specified in 10 CFR Part 20. The MHA scenario requires an evacuation of the MEB which is described in detail in Section 4.1.1 of this report.

SAR Section 7.7.2, states that the following:

Experience has shown that monthly verification of area radiation and air-monitoring setpoints in conjunction with annual calibration is adequate to correct for any variation in the system caused by a change of operating characteristics over a long time span.

The NRC staff reviewed the information as discussed above and finds that TS 3.7.1 helps ensure that radiation monitoring systems are required to support reactor operation and will actuate confinement mode of ventilation if alarm setpoints are exceeded during reactor operation. The NRC staff finds that TS 3.7.1 is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007 and, therefore is acceptable.

The NRC staff finds that the UUTR Director and RS are responsible for calibrating the instruments onsite using written approved procedures. Calibration is indicated through stickers on each instrument, and the records are maintained by the reactor staff and audited annually by the RSC. The information provided in SAR Table 11.1-7 lists the radiation monitoring and surveillance equipment available for routine monitoring and surveys. In addition to the monitors required by TS 3.7.1, the licensee has a comprehensive set of ARMs, CAMs for the stack, and portable radiation survey instruments that covers, with sufficient ranges, the various types of radiation that may be encountered at UUTR. The licensee also has other specialized radiation monitoring

equipment, such as a high-purity germanium gamma spectroscopy system, a liquid scintillation detector, and a portable sodium iodine (NaI) gamma spectroscopy system.

The NRC staff concludes that the equipment used by the licensee 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).

3.1.5 Radiation Exposure Control and Dosimetry

Radiation exposure control depends on factors such as the facility design features, operating procedures, training, and equipment. Design features include shielding, ventilation, containment of the inventory within the fuel, entry control, protective equipment, personnel dosimetry, and annual dose versus location estimates at various locations in the facility.

The NRC staff finds that the shielding for UUTR is similar to shield designs used successfully at many similar reactors. The principal design feature for control of radiation exposure during operation is the column of water around and above the reactor, plus the location of the reactor tank partially below ground level. UUTR is designed so that the radiation from the core can be accessed through vertical ports for research and educational purposes. The radiation exposure is controlled by restricting access to areas of elevated radiation fields.

10 CFR 20.1502, "Conditions Requiring Individual Monitoring of External and Internal Occupational Dose," requires monitoring of workers likely to receive, in 1 year, from sources external to the body, a dose in excess of 10 percent of the limits described in 10 CFR 20.1501, "General." The regulation requires monitoring individuals entering a high- or very high-radiation field in which an individual could receive a dose equivalent of 0.1 rem in 1 hour. From a list of the average occupational exposures for the years 2004 through 2009, and provided in SAR Table 11.1-5, the occupational doses were maintained below NRC regulatory limits. The licensee stated that there have been no instances of any exposures in excess of 10 percent of the applicable limits. However, UUTR maintains a radiation dosimetry program and restricts access to areas of elevated radiation fields, to demonstrate compliance with 10 CFR 20.1502, and to help ensure personnel exposure are ALARA.

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

The NRC staff finds that the licensee collects and maintains records of occupational exposure information, using the appropriate NRC forms. Records of self-reading dosimeters are kept in a logbook maintained by the UUTR staff as permanent records, as are measurement results of accidental releases to the environment. The licensee states that the environmental monitoring records over 35 years of operation demonstrate the fact that the operation of the UUTR facility has had an insignificant impact on the local environment and with no accidental radioactive material releases.

The NRC staff reviewed the information provided in UUTR SAR Section 11.1.5 and concludes that the UUTR radiation exposure and control program is acceptable and that, as evidenced by the historically low radiation doses and the application of the equipment and procedures used, the personnel exposures at UUTR are controlled through satisfactory radiation protection and ALARA programs. As described in Section 2.4 of this report, the UUTR annual reports from 2004–2010 were reviewed by the NRC staff, and the annual releases reported were below the allowable limits of 10 CFR Part 20. Additionally, NRC inspection reports from 2003 through 2009 contain no contradictory findings. On this basis, the NRC staff concludes that the licensee's program for control of personnel exposures and dosimetry are acceptable.

3.1.6 Contamination Control

The licensee controls radioactive contamination at UUTR by using written approved procedures for radioactive material handling, trained personnel, and a monitoring program designed to detect contamination in a timely manner.

The licensee has identified the locations most likely to have radioactive contamination and has developed control methods, including procedures and equipment. When working in potentially contaminated areas, UUTR workers are required to wear protective gloves and other appropriate protective clothing and are required to perform radiation surveys to help ensure that no contamination is present on hands, clothing, or shoes before leaving the work location. If contamination is detected, a survey is required to isolate the contamination. Materials and tools are monitored for contamination before removal from contaminated areas or from restricted areas likely to be contaminated. When individuals exit the reactor room, their hands and feet are surveyed for removable contamination. On a biweekly basis, swipe tests are performed and analyzed for contamination. The licensee states that acceptable surface contamination levels for unconditional release are no more than 1,000 disintegrations per minute per 100 square centimeters (dpm/100 cm²) for beta-gamma radiation (SAR Section 11.1.6). From its review of information provide in SAR Section 11.1.6 and the performance history of the UUTR contamination control program, the NRC staff concludes that acceptable controls exist to prevent the spread of contamination within the facility.

3.1.7 Environmental Monitoring

Environmental monitoring is conducted at UUTR to ensure compliance with Subpart F of 10 CFR Part 20 and the UUTR TS. Installed monitoring systems include ARMs and CAMs, which the licensee has managed and maintained in a comprehensive program

for the last 35 years. Based on a review of the UUTR annual reports from 2004 to 2010, the NRC staff finds that the operation of UUTR has an insignificant impact on local environmental radiation levels and radiation exposure in and around the facility.

TS 3.7.1 requires an ARM in the reactor room; however, as noted in SAR Table 11.1-7 of the SAR, UUTR maintains four monitors in additional locations, including the stack, the reactor room tank, and the counting room (a laboratory). With the exception of Ar-41, which is discussed earlier in Section 3.1.1 of this report, there are no pathways for radioactive materials from UUTR to enter the unrestricted environment during normal operations.

TS 3.7.1 requires one CAM in the reactor room to monitor airborne radioactive particulates, noble gases, and iodine. The monitoring program includes two additional CAM systems to monitor stack effluents, which alarm in the reactor control room. Calibration of the CAMs is accomplished as required by the TS and in accordance with facility procedures.

The licensee monitors the average radiation doses at specific locations quarterly using TLDs. The monitoring locations include unrestricted areas adjacent to the UUTR facility, the closest offsite point of continuous occupancy, and other offsite locations. The monitoring program includes at least 20 sampling locations, and the exposure data are analyzed to help ensure compliance with 10 CFR 20.1301, "Radiation Dose Limits for Individual Members of the Public." Three control dosimeters measure background radiation levels. The licensee analyzes annual exposures to the closest offsite occupancy location to help ensure compliance with the ALARA criteria in ANSI/ANS-15.11-1993 (R2004).

The licensee states in UUTR SAR Section 11.1.4, that as required by 10 CFR 20.1501, contamination surveys are performed which document the extent of contamination within the facility. Quarterly environmental monitoring surveys use fixed area dosimeters in both restricted and unrestricted areas. For the past 5 years, the largest recorded dose inside UUTR was 119 mrem/yr from a dosimeter attached directly to the reactor tank, and the highest dose recorded in the hallway directly above the reactor room was 8 mrem/yr. Both of these doses, and all other measured doses, are well within the appropriate limits cited in 10 CFR Part 20. The NRC staff reviewed the information in UUTR SAR Section 11.1.7, and concludes that the environmental monitoring program is acceptable to assess the radiological impact of UUTR on the environment.

3.2 Radioactive Waste Management

The purpose of the radioactive waste management program is to help ensure that radioactive waste materials are identified, assessed, controlled, and disposed of in conformance with all applicable regulations and in a manner that protects the health and safety of the public and the environment. UUTR SAR Section 11.2 provides a detailed description of the UUTR radioactive waste management program.

3.2.1 Radioactive Waste Management Program

The objectives of the UUTR radioactive waste management program are to minimize, properly handle, store, and dispose of the waste. The NRC staff reviewed the UUTR radioactive waste management program during a site visit on December 8, 2009. During that review, it was determined that radioactive waste is disposed of in accordance with the practices and procedures established and enforced by the UU RHD, which administers the radioactive waste management program. The UU RHD coordinates disposal of all UU-generated radioactive waste. The onsite review, and the review of the SAR, confirmed that acceptable controls are in place to prevent uncontrolled personnel exposures from radioactive waste operations, and if they occur, provide the necessary accountability to prevent unauthorized release of radioactive waste. Therefore, the NRC staff concludes that the radioactive waste management program is acceptable.

3.2.2 Radioactive Waste Controls

The UUTR radioactive waste control program defines radioactive waste as any item or substance that is no longer of any use to the facility and that contains or is suspected of containing radioactivity above the established natural background radioactivity. The UUTR SAR makes a distinction between radioactive waste and radioactive effluents, notably Ar-41. Waste volumes at UUTR have historically been small and of known characterization. When possible, radioactive waste is segregated at the point of origin from items that are not considered to be radioactive waste. Screening is based on the presence of detectable radioactivity, using appropriate monitoring and detection techniques, and on the projected future need for the materials involved. Solid wastes are either allowed to decay in storage to background levels or are transferred to the UU RHD for offsite disposal.

UU RHD imposes standardized packaging and labeling requirements, consistent with current low-level radioactive waste shipment and disposal requirements. Radioactive waste is not released into the environment as an effluent, which means that, if contaminated liquids are produced (such as liquid scintillation fluids), they are typically contained, added to an absorbent, and transferred to a solid radioactive waste disposal container in preparation for transfer to UU RHD for disposal. All waste is properly labeled in the appropriate waste container and an accurate estimate of the specific radionuclide content is made. Tagged waste containers are then stored before transfer to UU RHD.

Although disposal of liquids to the sanitary sewer system is permitted under 10 CFR 20.2003, "Disposal by Release into Sanitary Sewerage," RHD must approve such disposal to help ensure that the total UU waste released complies with the regulations. The RHD has set apportioned limits for individual users for key radionuclides often encountered in liquid wastes. In general, UUTR does not dispose of its liquid waste into the sanitary sewer. Radioactive waste that contains hazardous chemical wastes is termed mixed waste, and such waste is subject to additional regulatory control.

The UU license and State regulations require that inventory and control methods cover all aspects of work with radioactive materials. All packages and containers of radioactive waste must be labeled with a radiation symbol and a description of the contents. As waste is accumulated in a container, RHD requires a record to be made of each addition, so that a summary sheet can be prepared and an appropriate label can be applied when the container is full.

The NRC staff reviewed the information in UUTR SAR Sections 11.2.2 through 11.2.8, and concludes that acceptable procedures are in place to monitor the radiation exposure from radioactive waste, perform required handling operations, and prepare proper documentation for transfer to the RHD for offsite disposal.

3.2.3 Release of Radioactive Waste

UUTR radiation protection policy does not allow the release of radioactive waste into the environment. The only exception is gaseous radioactive effluents, notably Ar-41, which is regulated under 10 CFR Part 20 and EPA discharge limits. A CAM in the stack monitors gaseous effluents to help ensure compliance with the regulatory limits (i.e., the allowable effluent concentration for Ar-41 is 10^{-8} $\mu\text{Ci}/\text{cm}^3$).

When contaminated liquids are produced, such as liquid scintillation fluids, they are contained, added to an absorbent, and transferred to a solid radioactive waste disposal container. Then they are transferred to RHD for disposal. UUTR does not dispose of liquid waste into the sanitary sewer.

The NRC staff reviewed the information in UUTR SAR Section 11.2.9, and concludes that controls are available to eliminate or control potential releases of radioactive material into the sanitary sewer system. Therefore, the NRC staff concludes that the UUTR liquid releases do not pose a significant risk to public health and safety.

3.3 Radiation Protection Program and Waste Management Conclusions

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

- The UUTR radiation protection program complies with the requirements in 10 CFR 20.1101(a). The program is acceptably staffed and implemented and provides reasonable assurance that the facility staff, the environment, and the public are protected from unacceptable radiation exposures.
- Radiation sources and effluents are acceptably characterized and controlled. The radiation protection organization has acceptable lines of authority and communication to carry out the program.
- The systems provided for the control of radiological effluents, when operated in accordance with the TS, are acceptable to help ensure that releases of

radioactive materials from the facility are within the limits of the NRC regulations and are ALARA.

- The UUTR ALARA radiation protection program complies with the requirements of 10 CFR 20.1101(b) and uses the guidelines of ANSI/ANS-15.11-1993 (R2004) implementing time, distance, and shielding to reduce radiation exposures. A review of historical radiation doses and current controls for radioactive material in UUTR provides reasonable assurance that radiation doses to the environment, the public, and facility personnel will be ALARA.
- The results of radiation surveys carried out at UUTR, doses to the persons issued dosimetry, and results of the environmental monitoring program help verify that the radiation protection and ALARA programs are effective.
- The licensee acceptably identifies and describes potential radiation sources and controls them.
- Facility design and operational procedures limit the production and release of Ar-41 and N-16 and control the potential for facility staff and public radiation exposures. Conservative calculations of the quantities of these gases released into restricted and unrestricted areas give reasonable assurance that doses to the UUTR staff and public will be below applicable 10 CFR Part 20 limits.
- The radioactive waste management program provides reasonable assurance that radioactive waste released from the facility will neither exceed applicable regulations nor pose an unacceptable radiation risk to the environment and the public.

The NRC staff reviewed the UUTR radiation protection program and waste management summary as described in SAR Chapter 11. The NRC staff concludes that UUTR has implemented adequate and sufficient measures to minimize radiation exposure to workers and the public and has provided acceptable protection against operational releases of radioactivity to the environment.

4. ACCIDENT ANALYSES

The UUTR SAR provided accident analyses to demonstrate that the health and safety of the public and workers were protected during analyzed reactor transients and other hypothetical accident scenarios. The accident analyses presented in the UUTR SAR provided the basis to establish the UUTR TSs described in this report. The accident analysis presented in this chapter ensured that no credible accident could lead to unacceptable radiological consequences to the UUTR staff, the public, or the environment. Additionally, the licensee,, consistent with the guidance in NUREG-1537, analyzed the consequences of the MHA, which is an event involving the rupture of the cladding of an irradiated fuel element in air. The MHA is considered the worst-case fuel failure scenario for a TRIGA reactor which would lead to the maximum potential radiation hazard to facility personnel and members of the public. The results of the MHA are used to evaluate the ability of the licensee to respond and mitigate the consequences of this postulated radioactive release.

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

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

4.1 Accident Analysis Initiating Events and Determination of Consequences

4.1.1 Maximum Hypothetical Accident

For UUTR, the MHA is defined as the rupture of the cladding of one fuel element in air. The scenario assumes that such an accident occurs after UUTR operation at full licensed power long enough for the inventories of the radionuclides in the scenario to be at their maximum concentration. The analysis assumes that, at the time of clad failure, the volatile fission products have accumulated in the gap and are released abruptly into the air with no radioactive decay; this includes the release of noble gases, halogens, and other volatile fission products.

Because there are no specific accident-related regulations for research reactors, the NRC staff compared calculated dose values for accidents with related standards in 10 CFR Part 20. Amendments to 10 CFR Part 20 (Sections 20.1001 through 20.2402 and the appendices) became effective January 1, 1994. Among other things, these amendments changed the dose limits for occupationally exposed persons and members of the public, as well as the concentrations of radioactive material that are allowed in effluents released from licensed facilities. The licensee must follow the requirements of 10 CFR Part 20, as amended, for all

aspects of facility operation. However, because the reactor was initially licensed before January 1, 1994, in conducting the accident evaluation, the NRC staff used the dose limits in 10 CFR Part 20 that have been historically applied to accidents in this reactor (10 CFR 20.1 through 10 CFR 20.602 and appendices, referred to as the “old” Part 20). See NUREG-1537, Chapter 13, for an additional discussion of accident dose limits. As shown below, the doses presented are also within the limits of the current version of 10 CFR Part 20.

For determining the radionuclide inventories, the licensee assumed two different thermal powers:

- For long-lived radionuclides, the fuel inventory is based on the historical operation of the reactor for the last 35 years at an average of 70 hours of annual reactor operation.
- For short-lived radionuclides, the fuel inventory is based on the continuous operation for 100 hours.

For both cases, the licensee increased the average thermal power by a factor of 2.0 to adjust for the “worst case” fuel element power (i.e., the ratio of the maximum to the average fuel rod power based on an MCNP modeling in SAR Section 4.5). Given these adjusted thermal powers and the fuel enrichment, the licensee used a combination of SCALE6-TRITON (Ref. 24) and SCALE6-KENO6 (Ref. 25) computer codes to calculate the radionuclide inventories. The licensee compared the UUTR projected radionuclide inventories to another TRIGA reactor that used the ORIGIN 2.1 computer code and concluded that the UUTR inventory values were within an acceptable range, given the thermal power ratio between the two reactors. The computer codes are used extensively throughout the nuclear industry to calculate core fission product inventories. The licensee provided the radionuclide inventories used in dose calculations in UUTR SAR Tables 13.2-5 and 13.2-6.

Based on the power history assumptions provided in the UUTR SAR, the NRC staff determined that the inventories of isotopes that are dominant to the MHA dose calculations (halogens and noble gases) are at the saturation (maximum) concentration for continuous full power operation of the UUTR with the exception of Kr^{85} , which is a long-lived isotope. However, the NRC staff noted that the Kr^{85} contribution was small when compared with other more dominant contributors such as the iodine isotopes. The NRC staff performed an independent calculation using the saturation concentration of Kr^{85} (at continuous full power operation) to confirm that the Kr^{85} was a negligible contributor to the MHA dose as it only accounted for approximately 10^{-4} mrem in the MHA scenarios. The NRC staff concluded that the UUTR inventory estimates are acceptable for the MHA dose calculations.

The licensee calculated the releases of noble gases and halogens from the fuel gap using a release fraction of 1×10^{-4} . GA has developed a correlation for the fission product release fraction based on fuel temperature (see “Fission Product Release from TRIGA-LEU Reactor Fuels,” issued October 1980 (Ref. 26)). When the fuel specimen was irradiated at temperatures below about 350 degrees C, the fraction of the total inventory that is released could be summarized as a constant, independent of operating temperature (i.e., a value of 1.5×10^{-5}). The release fraction increased as temperature increased above 350 degrees C. The licensee’s assumed release fraction corresponds to an average fuel temperature of 490 degrees C. Since this temperature was conservative, with the maximum operating temperature of the UUTR fuel

(129.67 degrees C) at the licensed limit of 100 kW, the NRC staff concluded that it was conservative and acceptable.

The licensee used RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1, to determine the atmospheric dispersion factors (i.e., atmospheric relative concentration values, X/Q) at select distances from the reactor building for both the elevated and ground release (Ref. 38). The elevated release is based on an effective stack height of 40 feet with the reactor room ventilation system in operation. TS 5.1 identifies the point of release from the reactor room ventilation stack at a minimum height of 40 feet from the ground. The NRC staff reviewed the method and data used in the atmospheric dispersion factor calculations and concludes that the results presented in SAR Table 13.2-7 are acceptable.

The licensee calculated the occupational dose for an individual in the reactor room. Boundary conditions for these calculations included assuming the failure of the hottest fuel element, incorporating the calculated release fractions, and assuming the reactor room has a volume of 459 m³ (an added conservatism, since this is at least 20 percent smaller than the actual volume). Other parameters used in the dose calculations include a breathing rate of 0.02 m³/min (consistent with the value given in Appendix B to 10 CFR Part 20), and a ventilation rate of 6.1×10⁵ cm³/s, in accordance with UUTR SAR Section 13.2.1.2. In addition, the licensee used dose conversion factors (DCF) for the inhalation and external exposure pathways from the DOE reports (DOE/EH-0071, "Internal Dose Conversion Factors for Calculation of Dose to Public," issued July 1988 (Ref.27), and DOE/EH-0070, "External Dose-Rate Conversion Factors for Calculation of Dose to the Public," issued July 1988 (Ref. 28).

Based on the above considerations, the licensee calculated the potential doses to members of the public within the MEB and at specified distances from the reactor outside the MEB, as well as to workers within the reactor facility. The NRC staff reviewed the licensee's methods for computing the dose within and beyond the confines of the reactor facility caused by an MHA release and finds that all dose estimates provided by the licensee were acceptable.

In addition, the NRC staff performed confirmatory calculations which demonstrated the adequacy of the submitted information and dose results. These confirmatory dose calculations used the same assumptions, geometry, and source term as those provided by the licensee. The confirmatory dose calculations were used to verify the accuracy of the final dose estimates for each scenario provided in the UUTR SAR.

UUTR MHA Dose Scenarios

UUTR provided six scenarios, as described in SAR Section 13.2.1.2. In the accompanying SAR dose calculations, the licensee employed the methodology of the DCFs in DOE/EH-0070 and DOE/EH-0071. The NRC confirmatory analysis used DCFs from Federal Guidance Report (FGR) No. 11, "Limiting Values of Radionuclide Intake and Air Concentration, and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," EPA-502/1-88-020, issued September 1988 (Ref. 29), and FGR No. 12, "External Exposure to Radionuclides in Air, Water, and Soil, EPA-402-R-93-081, issued September 1993 (Ref. 30). The use of two independent methods provided additional assurance of the validity of the licensee's dose results.

Scenario A—In this scenario, the licensee postulated a sudden opening of the reactor bay west

wall with an effective surface area of 100 m² with a wind speed of 1 m/s, which would cause the reactor room air to exit to the environment at a rate of 100 m³/s. The UUTR SAR Table 13.2-8 summarizes the total dose to an individual at select distances from the reactor room in terms of total effective dose equivalent (TEDE). Table 4-1 below presents the licensee's results and the results from the NRC's confirmatory dose calculations.

Table 4-1 MHA Dose Estimates for Scenario A

MHA Dose Estimates for Scenario A			
Downwind distance (m)	Total Effective Dose Equivalent (mrem)		
	UUTR Result	Confirmatory Calculation	Dose limit
10	8.7	8.3	100
50	3.9	3.8	
100	1.3	1.3	
150	0.6	0.6	
200	0.3	0.3	
250	0.2	0.2	

Scenario B—In this scenario, the licensee postulated that the MHA occurs with the UUTR reactor room intact and the ventilation system in normal operation. The licensee assumed the entire reactor room air inventory would be exhausted to the environment through the ventilation stack (elevated release), with one complete air exchange taking about 12.6 minutes. The licensee calculated doses to members of the public from both the elevated (stack) and ground release.

The elevated release results in potential doses to members of the public at distances below 100 m that are very small. In addition, the elevated release doses in the NRC staff's confirmatory calculations indicate that the maximum dose will occur at a distance of 300 m from the reactor room with a TEDE of 0.05 mrem, which is below the 10 CFR Part 20 dose limit. Table 4-2 below presents the results from the licensee and the NRC staff's confirmatory calculations.

Table 4-2 MHA Dose Estimates for Scenario B

MHA Dose Estimates for Scenario B					
Downwind distance (m)	Total Effective Dose Equivalent (mrem)				Dose limit
	UUTR Result Ground Release	Confirmatory Calculation Ground Release	UUTR Result Elevated Release	Confirmatory Calculation Elevated Release	
10	7.1	7.5	4×10^{-29}	4×10^{-29}	100
50	3.2	3.4	5×10^{-22}	5×10^{-22}	
100	1.1	1.2	9×10^{-7}	1×10^{-6}	
150	0.5	0.5	1×10^{-3}	2×10^{-3}	
200	0.3	0.3	1×10^{-2}	2×10^{-2}	
250	0.2	0.2	4×10^{-2}	4×10^{-2}	

Scenario C—In this scenario, the licensee postulated two different leakage rates through the reactor room wall for the postulated ground release. The reactor room air leakage rates to the outside environment were 1.69×10^4 and 6.1×10^3 cm³/s, resulting in 7.54 and 20.7 hours, respectively, for one complete reactor room air change. Table 4-3 below presents the licensee’s results, as well as those of the NRC staff’s confirmatory calculations.

Table 4-3 MHA Dose Estimates for Scenario C

MHA Dose Estimates for Scenario C					
Downwind distance (m)	Total Effective Dose Equivalent (mrem)				Dose limit
	UUTR Result (7.54 h)	Confirmatory Calculation (7.54 h)	UUTR Result (20.7 h)	Confirmatory Calculation (20.7 h)	
10	5.9	6.2	5.2	5.5	100
50	2.7	2.9	2.4	2.5	
100	0.9	1.0	0.8	0.9	
150	0.4	0.4	0.4	0.4	
200	0.2	0.2	0.2	0.2	
250	0.2	0.2	0.1	0.2	

Scenario D—In this scenario, occupational doses to the workers in the reactor room were postulated, assuming that the ventilation system was shut down and there is no air leakage from

the reactor room. This condition maximized the potential doses to the workers. For analysis purposes, the licensee assumed the individuals would be exposed to the reactor room concentrations for 2 to 5 minutes, based on the expected worker evacuation times from the reactor facility. Table 4-4 provides the licensee’s results, along with the results of the NRC staff’s confirmatory calculations.

Table 4-4 MHA Dose Estimates for Scenario D

MHA Dose Estimates for Scenario D			
Duration of Exposure (minutes)	Total Effective Dose Equivalent (mrem)		
	UUTR Result	Confirmatory Calculation	Dose limit
2	32	33	5,000
5	80	83	

Scenarios E and F—The licensee used the MHA source term to evaluate potential exposure to individuals in the MEB public workspaces and passageways that adjoin the UUTR reactor bay area. The areas of the MEB that were evaluated were those in direct proximity to the reactor facility, including: (1) the Mechanical Engineering Heat Laboratory (MEHL), (2) the MEB hallway east of the reactor facility, (3) the classroom next to the UNEF office area, and (4) the office area directly above the reactor. The intent of the licensee’s methodology was to evaluate direct exposure through the walls and to account for contamination through the air.

Scenario E involves an analysis of the four areas described above with a complete shutdown of the ventilation system.

Scenario F involves the same areas described above with the ventilation operating in the limited intake mode.

For a more detailed description, see UUTR SAR Section 13.2.1.2.

The NRC staff reviewed the results of Scenarios E and F for the exposure of members of the public occupying adjacent rooms or spaces. The NRC staff’s confirmatory dose calculations included direct exposure and mass transport, and used the verified parameters and assumptions as specified in the UUTR SAR, such as room volumes, geometry, and leakage rates. The MEHL doses were limiting and, thus, were most applicable to UUTR SAR MEB exposure analyses.

Table 4-5 and Figure 4-1 below present the calculations for the UUTR SAR MEB and MEHL and the confirmatory dose calculations.

TEDE to Individuals in the MEHL

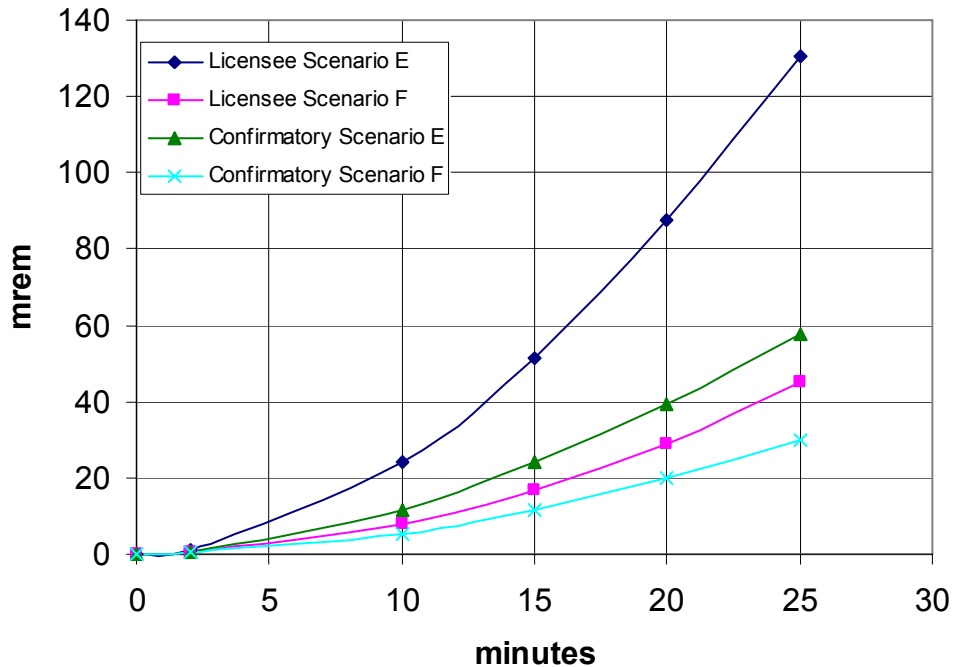


Figure 4-1 Comparison of TEDE calculations for individuals in the MEHL

Table 4-5 MHA Dose Estimates for Scenarios E and F in the MEB

MHA Dose Estimates for Scenarios E and F in the MEB				
Exposure time (minutes)	UUTR Result (mrem)		Confirmatory Calculation (mrem)	
	Scenario E	Scenario F	Scenario E	Scenario F
0	0	0	0	0
2	1	0.3	0.6	0.3
10	24	8	11.5	5.4
15	52	17	23.9	11.6
20	88	29	39.5	19.8
25	>100	45	57.5	29.7

As a result of the analysis in Scenarios E and F, the licensee determined that the dose in the MEB MEHL would exceed the 100 mrem limit specified in 10 CFR 20.1301 in approximately 20 minutes. As such, the licensee implemented actions in the UUTR EP to evacuate the MEB in case of an emergency, accident, or major radiological release. UUTR SAR Section 13.2.1.2 indicates that, in past evacuation drills performed during annual fire alarm testing with the Salt Lake City Fire Department, the average evacuation time was approximately 5 minutes. Assuming the ROs take 5 minutes to evaluate and respond to the event before activating the fire alarm and another 5 minutes to evacuate the MEB, the resulting 10 minutes is still sufficient to help ensure that no member of the public would exceed the dose limit specified in 10 CFR Part 20.

The NRC staff's confirmatory calculations demonstrate that the maximum exposure to members of the public under those circumstances is less than the 100 mrem dose limit from 10 CFR Part 20 for any condition of the ventilation system, and thus, the NRC staff finds that the evacuation time cited by the licensee is acceptable.

MHA Dose Calculation Conclusions

The NRC staff reviewed the MHA analyses for all the scenarios, as well as the dose calculational results, and concluded that the licensee used appropriate assumptions and analytical techniques and that their conclusions were appropriate and acceptable. The independent confirmatory dose calculations performed by the NRC staff demonstrates that the licensee properly evaluated the postulated doses from the MHA scenarios. The results of the NRC staff's confirmatory dose calculations are consistent with the dose results provided by the licensee. In addition, the doses from the postulated scenarios provided above demonstrate that the maximum TEDE doses were below the occupational limits in 10 CFR 20.1201 and the public exposure limits in 10 CFR 20.1301. Because the dose results were within the requirements of 10 CFR Part 20, the NRC staff concludes that the results of the MHA analysis are acceptable.

4.1.2 Insertion of Excess Reactivity

In UUTR SAR Section 13.2.2, the licensee analyzed the excess reactivity insertion scenario in which the maximum reactivity insertion selected was based on the TS 3.8.1, which limits the maximum reactivity insertion of an unsecured experiment to \$1.0, with an additional \$0.20 added for conservatism. The NRC staff finds that the additional \$0.20 (or 20 percent) is appropriate as the maximum calculational variance between the measured and calculated reactivity is approximately 15 percent (reference Table 2.6 in this report). The NRC staff thus concludes that the licensee selected a reasonable initiating event and appropriate conservatism.

The licensee modeled the reactivity insertion using a combination of PARET-ANL and MCNP computer codes. The analyses assumed that the reactor operated in accordance with administrative limits for power at 90 kW and with a reactor trip setpoint at 100 kW, but because of modeling limitations and a rapid prompt jump caused by a sudden reactivity insertion, the scram occurs at 1,100 kW. The licensee calculated the maximum centerline fuel temperature and coolant as a function of time after the \$1.2 reactivity insertion. The peak fuel centerline temperature for this insertion is about 110 degrees C, which is far below the limiting temperature of 530 degrees C. Based on the information provided above, the NRC staff concludes that the methods used by the licensee are acceptable.

Insertion of Excess Reactivity—Confirmatory Analysis

The NRC staff performed independent confirmatory calculations using a modified Fuchs-Nordheim model. Using a reactivity insertion of \$1.2 and starting at a power level of 100 kW, the peak fuel temperature increased to 88 degrees C. The results obtained using this method were less exact than the method used by the licensee in that it does not model the effect of delayed neutrons, control system response, or control rod scrams. However, it is used because it provides an independent method to confirm the UUTR calculations and allow comparison with historical TRIGA analyses. Table 4-6 below provides the results of the confirmatory analysis.

Table 4-6 Fuel Temperature after a Step Reactivity Insertion

Fuel Temperature Following a Step Reactivity Insertion	
Parameter	Confirmatory Result
prompt neutron lifetime (s)	2.17×10^{-5}
fuel temperature coefficient ($\rho/^\circ\text{C}$)	1.1×10^{-4}
β_{eff}	7.68×10^{-3}
initial reactor power (MW)	0.1
initial reactor temperature ($^\circ\text{C}$)	35.0
amount of reactivity pulse (\$)	1.20
total peaking factor	1.955
reactivity inserted (\$)	0.00922
reactor period (1/s)	0.0024
amount of reactivity above \$1	0.00154
temperature rise during pulse ($^\circ\text{C}$)	27.62
energy released during pulse (MW-s)	2.27
peak power attained (MW)	39.88
peak fuel temperature ($^\circ\text{C}$)	89.00

On the basis of this review, the NRC staff concludes that a rapid insertion into the UUTR core of the \$1.2 reactivity will not result in fuel melting or cladding failure as a result of high temperature or high internal gas pressure.

TS 3.8.1, Reactivity Limits, states the following:

TS 3.8.1 Reactivity Limits

Applicability

These specifications apply to experiments installed in the reactor and its irradiation facilities.

Objective

The objective is to prevent damage to the reactor excess release of radioactive materials in the event of an experiment failure.

Specifications

The reactor shall not be operated unless the following conditions exist:

1. The absolute value of the reactivity worth of any single secured or unsecured experiment shall be less than \$1.00, and,
2. The sum of the absolute values of the reactivity worths of all experiments shall be less than \$1.20.

The NRC staff reviewed the reactivity limits established in Specifications 1 and 2 above include the determination of SDM and excess reactivity, as provided in TS 3.1.2, and 3.1.3, respectively. The NRC staff finds that TS 3.8.1, Specifications 1 and 2, are justified by the analysis presented in this section, and the NRC staff concludes that TS 3.8.1 is acceptable.

Based on the information provided above, including a confirmatory calculation performed by the NRC staff, the NRC staff concludes that the analysis for the insertion of excess reactivity scenario indicated used appropriate assumptions and analytical techniques. Additionally, TS 3.8.1 helps ensure that an excess reactivity scenario is properly controlled by the UUTR staff. The NRC staff concludes that the licensee's insertion of excess reactivity scenario and TS 3.8.1 provide reasonable assurance that the UUTR will continue to operate safely.

4.1.3 Loss-of-Coolant Accident

In UUTR SAR Section 13.2.3, the licensee analyzed a loss-of-coolant accident (LOCA) scenario by assuming a loss of the reactor pool water. Because the UUTR reactor tank has no beam port penetrations, and the reactor cooling system return lines contain antisiphon holes located 1 foot below the normal tank water level, the licensee postulated that an earthquake was the only potential scenario that could lead to a loss of coolant. The design and construction of the UUTR reactor tank was intended to help ensure that the probability of an earthquake-initiated failure was low. The tank design employs the "tank within a tank" concept with a 2-foot annular region between the two tanks filled with energy-absorbing sand. Additionally, 16 feet of the 24-foot tank is located below ground. UUTR SAR Sections 2.5.2 and 3.4 contain additional details on the UUTR reactor tank and the seismic design considerations.

Although unlikely to occur for the reasons stated above, the licensee postulated a tank failure caused by an earthquake, where the bottom of the reactor tank was breached and allowed water to drain out. The accident scenario assumed a complete opening of the bottom of the reactor tank, allowing water to drain through the soil and sands underground. The licensee

used Darcy's law to determine water flow through porous media. Considering a pool water level of 6.7 meters and a hydraulic conductivity of 3,000 m/yr for sand, the licensee estimated that it would take 19.3 hours for the pool to completely drain. The licensee concluded that this was sufficient time to provide emergency makeup water and keep the core covered.

The NRC staff finds that the assumption that the pool water percolates through sand was conservative, because, as described in UUTR SAR Section 4.3, the licensee indicated that the reactor tank sits on a concrete block that is within clay, and less porous than sand. Additionally, the NRC staff finds that the use of Darcy's Law was an acceptable approach for viscous flow through porous media. The NRC staff also finds that the licensee's time estimate for coolant loss from the reactor tank caused by an earthquake-induced breach was accurate and allowed ample time for reactor tank inventory makeup to keep the reactor core covered.

Fuel Integrity Following a LOCA

In UUTR SAR Section 13.2.3, the licensee analyzed the fuel temperature following a postulated LOCA event, based on an operating power level of 100 kW. The estimated maximum fuel temperature calculated was 61 degrees C, which was well below the SL in TS 2.1. The corresponding estimated pressure inside the fuel element from the fission product gases was 19.6 pounds per square inch (psi) and was also well below the yield stress of 8,000 psi for aluminum at 150 degrees C.

The UUTR analysis used natural convection for cooling the core and ignored conduction to the core grid and the radiation heat losses. The licensee also used the specific heat capacity and thermal conductivity of U-ZrH_{1.0} in the analysis, whereas the fuel of interest was U-ZrH_{1.6}, which has a higher specific heat and results in a lower temperature rise for the same heat load. The NRC staff finds that the results presented are conservative and acceptable.

Based on its review of the information provided above, the NRC staff concludes that the licensee's analysis was correctly performed, consistent with the results of SAR information presented for other TRIGA reactors following a LOCA event, and is acceptable.

Radiation Levels after Loss of Pool Water

In UUTR SAR Section 13.2.4, the licensee analyzed radiation levels on the reactor pool platform and the laboratory floor during a postulated loss of pool water. Using the estimated reactor tank water leak rate of 0.4 m/h, the licensee determined the water column above the core at selected times following the start of the leak. The main consequence was the increased gamma ray dose from the exposed core. The licensee provided calculated potential direct gamma dose rates to an individual above the core and on the laboratory floor. The analysis indicated that the dose rates within the first 10 hours were less than 20 mrem/h because of the shielding, which the remaining water column continued to provide. Without any action to replenish the water inventory, the core would uncover in approximately 15 hours.

The NRC staff concludes, based on the review of the licensee's LOCA analysis, that appropriate assumptions and analytical techniques were used, and the licensee's calculated doses were appropriate. In addition, the NRC staff noted that the licensee's gamma dose rates are very low for the first 10 hours, providing sufficient time for the licensee to take action to replenish the reactor coolant inventory and to help ensure that the 10 CFR Part 20 dose guidelines to the

workers, building occupants, and the public are satisfied. On this basis, the NRC staff finds the licensee's conclusions acceptable.

4.1.4 Loss-of-Coolant Flow

UUTR SAR Section 13.2.5 analyzed the postulated loss-of-coolant accident scenario. Since UUTR uses natural convection cooling without the need for any forced cooling, a loss-of-coolant flow is not considered a credible event. Additionally, the licensee indicated that the loss of secondary cooling flow would also not affect this scenario, as the reactor can operate with secondary cooling as long as the reactor pool temperature limit (35 degrees C) is maintained. The NRC staff reviewed the licensee's loss-of-coolant flow accident scenario and core grip plate configuration as provided in Figure 2-2 of this report. The NRC staff finds, that in the event of a possible blockage of a coolant channel created by a foreign object lodged in the grid plate, the open fuel element lattice would ensure sufficient continuing cooling of all fuel elements as a result of cross flow.

The NRC staff reviewed the licensee's postulated loss-of-coolant flow accident scenario and concludes that the licensee's results are acceptable.

4.1.5 Mishandling or Malfunction of Fuel

UUTR SAR Section 13.2.6 evaluated events that could cause a postulated accident involving the mishandling or malfunction of fuel. These included: (1) a fuel handling accident, (2) the failure of the fuel clad caused by a manufacturing defect or corrosion, and (3) the overheating of a fuel element causing cladding failure. In the unlikely event of such a failure in air, the event consequences would be bounded by the results of the MHA scenario, which have been discussed and accepted in Section 4.1.1 above.

The NRC staff reviewed the licensee's postulated mishandling or malfunction of fuel accident scenario and concludes that the licensee's results are acceptable.

4.1.6 Experiment Malfunction

UUTR SAR Section 13.2.7 evaluated the scenario for a postulated experimental malfunction accident. TS 3.8 and 4.8 establish controls and limits for experiments. Specifically, TS 3.8.1 limits a step change in reactivity greater than \$1.0 for any single experiment and \$1.2 for all experiments. TS 3.8.2 requires UUTR to evaluate experiments that contain chemical and explosive hazards during the experiment review process. Limits are placed on reactivity worths and on the mass of explosive materials and other experiment materials to avoid accidental reactivity insertions, damage to reactor components, and a release of radioactivity. TS 6.5 requires RSC review and approval of all new experiments.

The NRC staff reviewed the licensee's postulated experimental malfunction accident scenario and concludes that the licensee's results are acceptable.

4.1.7 Loss of Normal Electrical Power

UUTR SAR Section 13.2.8 evaluated the scenario for a postulated UUTR accident involving a loss of normal electrical power. UUTR does not require emergency backup electrical power to

safely shut down the reactor or maintain core cooling. The loss of normal electrical power will result in a reactor shutdown through loss of voltage to the control rod drive mechanism and a reactor scram, as required in TS 3.2.3. The loss of normal electrical power and resulting scram will not result in a release of radioactive material. Additionally, the loss of electrical power does not affect the radiation safety and alarm equipment in the reactor room, as a backup power system is used. Loss of electrical power would result in the loss of primary and secondary cooling. However, reactor decay heat would dissipate through natural circulation in the reactor pool.

The NRC staff reviewed the licensee's postulated loss of normal electrical power accident scenario and concludes that the licensee's results are acceptable.

4.1.8 External Events

In UUTR SAR Section 13.2.9, the licensee analyzed the potential impact to UUTR from external events. Floods, hurricanes, and tornados are rare in the Salt Lake City area and are not considered to pose a threat to UUTR. Seismic activity in the State of Utah and adjacent areas are typically moderate with minor consequences. In recent history, seismic occurrences have been low-intensity events with little or no damage. In an earthquake with significant severity, the consequences to the UUTR facility are not expected to cause events more severe than the MHA event analyzed. The UUTR tank was designed to mitigate the consequence of an earthquake (see Section 4.1.3 above for additional details). A significant earthquake would result in a reactor trip and potentially result in the loss of coolant from the reactor tank. However, the control rods travel in guide tubes and would successfully shutdown the reactor. The results of the LOCA, analyzed in Section 4.1.3 above, indicate that the fuel integrity would be maintained.

The NRC staff finds that severe storms, floods, and tornadoes were very unlikely for the area around the UUTR site. The UUTR building, reactor foundation, shielding structure, reactor tank, and core support structure were designed in accordance with Uniform Building Code Zone 3 requirements. Meeting these requirements helps ensure that the reactor can be safely shutdown following an earthquake likely to occur during the facility's lifetime. On this basis, the NRC staff concludes that the consequences of external events are bounded by the MHA analysis and are acceptable.

4.1.9 Mishandling or Malfunction of Equipment

The licensee analyzed the postulated mishandling or malfunction of equipment in UUTR SAR Section 13.2.10. The SAR indicated that the UUTR reactor design includes appropriate control system interlocks and automatic protective circuits. The UUTR is designed to shutdown without fuel damage following a large positive reactivity insertion. TRIGA fuel is designed to accept large step reactivity insertion events without the loss of cladding integrity. Events caused by operator errors during reactor operation would also most likely result in a reactor shutdown.

The NRC staff reviewed the licensee's postulated mishandling or malfunction of equipment accident scenario and concludes that the licensee's results are acceptable.

4.2 Accident Analyses and Determination of Consequences

The NRC staff reviewed the licensee's postulated and analyzed accident scenarios. On the basis of its evaluation of the information presented in the licensee's SAR, 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 cladding and a release of fission products.
- The licensee analyzed the most serious credible accidents and the MHA and determined that the calculated potential radiation doses outside the reactor room would not exceed dose limits in 10 CFR Part 20 for unrestricted areas.
- The licensee has employed appropriate methods for accident analysis and consequence analysis.
- The licensee used conservative assumptions in evaluating occupational and public exposure from releases in an MHA. As a result, the MHA analysis identified the need to implement an evacuation of the MEB in order to ensure the MHA will not result in an occupational radiation exposure to the facility staff or radiation exposure to the public in excess of the applicable NRC limits in 10 CFR Part 20.
- For accidents involving insertions of excess reactivity, the licensee has demonstrated that a reactivity insertion of \$1.20 will result in a peak fuel temperature far below the SL. An insertion of excess reactivity resulting from the uncontrolled withdrawal of an experiment is limited to \$1.0 by TS 3.8 and hence, does not pose a threat to fuel integrity. The licensee did not identify any other accidents involving a reactivity addition that are not bounded by the supplied analysis.
- The review of the calculations, including assumptions, demonstrated that a LOCA would not result in unacceptable fuel element temperatures.
- Doses from the MHA and all credible accidents are below the limits of 10 CFR Part 20.
- The accident analysis for UUTR establishes the acceptability of the limiting core configuration defined and analyzed in UUTR SAR Sections 4.5 and 4.6.
- The accident analysis confirms the acceptability of the licensed power of 100 kW.
- The accident analysis confirms the acceptability of assumptions regarding excess reactivity limits (\$1.20).
- The accident analysis confirms the acceptability of the assumptions stated in the individual analyses provided in the SAR.

4.3 Accident Analyses Conclusions

The NRC staff reviewed the radiation source term and MHA calculation for UUTR. The NRC staff finds the calculations, assumptions, source term assumed and other boundary conditions, acceptable. The radiological consequences to the public and occupational workers at UUTR are in conformance with the requirements in 10 CFR Part 20. The licensee did not identify any other accidents that are not bounded by those analyzed by the MHA. The UUTR design features and administrative restrictions in the TS prevent the initiation of accidents and mitigate any consequences. Therefore, on the basis of this review, the NRC staff concludes that there is reasonable assurance that no credible accident would cause a significant radiological risk, and the continued operation of UUTR poses no undue risk to the facility staff, the environment, or the public.

5. TECHNICAL SPECIFICATIONS

In this section of the report, the NRC staff evaluated the licensee's TS. The UUTR TS define specific features, characteristics, and conditions that are required for the safe operation of the UUTR facility. NUREG-1537, Part 1, Chapter 14 and Appendix 14.1 (Ref. 7), and ANSI/ANS-15.1-2007 (Ref. 32) provided guidance, including an accepted style, format, and content for RTR TS. The NRC staff also relied on the references provided in NUREG-1537 and the ISG (Ref. 10) to perform this review.

5.1 Technical Specification Definitions

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

Audit: An audit is a quantitative examination of records, procedures, or other documents after implementation from which appropriate recommendations are made.

Channel: A channel is the combination of sensor, line, amplifier, 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 is an enclosure of the overall facility that is designed to limit the release of effluents between the enclosure and its external environment through controlled or defined pathways. These are rooms MEB 1205 (A through K) and 1206 in Merrill Engineering Building.

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

1. **Regulating Rod (Reg Rod):** The regulating rod is a control rod having an electric motor drive and scram capabilities. Its position may be varied manually or by the servo-controller.
2. **Shim/Safety Rod:** A shim safety rod is a control rod having an electric motor drive and scram capabilities.

Core Lattice Position: The core lattice position is defined by a particular hole in the top grid plate of the core. It is specified by a letter indicating the specific ring in the grid plate and a number indicating a particular position within that ring.

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

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

Specific experiments shall include:

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

Experimental Facilities: Experimental facilities shall mean vertical in-pool irradiation facilities, vertical tubes, in-core irradiation ports such as the A fuel ring (central ring) or other empty fuel element positions, rotating specimen rack, pneumatic transfer system, sample holding dummy fuel elements and any other in-tank irradiation facilities.

Fuel Element: A fuel element is a single TRIGA[®] fuel element.

Instrumented Element: An instrumented element is a special fuel element in which one or more thermocouples have been embedded for the purpose of measuring the fuel temperatures during reactor operation.

Irradiation: Irradiation shall mean the insertion of any device or material that is not a part of the existing core or experimental facilities into an experimental facility so that the device or material is exposed to radiation available in that experimental facility.

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

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

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

Operational Core: An operational core shall be a fuel element core, which operates within the licensed power level and satisfies all the requirements of the Technical Specifications.

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

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

Reactor Operator (RO): An individual who is licensed to manipulate the controls of a reactor.

Reactor Safety Systems: Reactor safety systems are those systems, including their associated input channels, which are designed to initiate, automatically or manually, a reactor scram for the primary purpose of protecting the reactor.

Reactor Secured: The reactor is secured when:

1. Either there is insufficient moderator available in the reactor to attain criticality or there is insufficient fissile material in the reactor to attain criticality under optimum available conditions of moderation and reflection; or
2. All of the following exist:
 - 2.1. The three (3) neutron absorbing control rods are fully inserted as required by technical specifications,
 - 2.2. The reactor is shutdown,
 - 2.3. The console key switch is in the "off" position and the key is removed from the console,
 - 2.4. No experiments are being moved or serviced that have, on movement, reactivity worth exceeding the maximum value allowed for a single experiment, or of one dollar, whichever is smaller, and
 - 2.5. No work is in progress involving core fuel, core structure, installed control rods, or control rod drives unless they are physically decoupled from the control rods.

Reactor Shutdown: The reactor is shut down when it is subcritical by at least one dollar both in the reference core condition and for all allowed ambient conditions, with the reactivity worth of all installed experiments and irradiation facilities included.

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

Review: A review is a qualitative examination of records, procedures, or other documents prior to implementation from which appropriate recommendations are made.

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

Scram time: Scram time is the elapsed time from the initiation of a scram signal to the time the slowest scrammable control rod is fully inserted.

Senior Reactor Operator (SRO): An individual who is licensed to direct the activities of ROs. Such an individual is also an RO.

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

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

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

1. Biennial—interval not to exceed 30 months
2. Annual—interval not to exceed 15 months
3. Semiannual—interval not to exceed 7.5 months
4. Quarterly—interval not to exceed 4 months
5. Monthly—interval not to exceed 6 weeks
6. Weekly—interval not to exceed 10 days

5.2 Safety Limits and Limiting Safety System Settings

5.2.1 TS 2.1 Safety Limits

See Section 2.5.5 of this report.

5.2.2 TS 2.2 Limiting Safety System Settings

See Section 2.5.5 of this report.

5.3 Limiting Conditions for Operation

5.3.1 TS 3.1 Reactor Core Parameters

5.3.1.1 TS 3.1.1 Steady State Operation

See Section 2.5.1 of this report.

5.3.1.2 TS 3.1.2 Shutdown Margin

See Section 2.5.2 of this report.

5.3.1.3 TS 3.1.3 Core Excess Reactivity

See Section 2.5.2 of this report.

5.3.1.4 TS 3.1.4 Core Configuration

See Section 2.2 of this report.

5.3.1.5 TS 3.1.5 Reactivity Coefficients

UUTR limitation on reactivity coefficients are not required by TSs.

The NRC guidance in NUREG-1537, Appendix 14.1, Section 3.1, "Core Parameters," item (5) (Ref. 7) states that TS are needed only if the coefficients "could vary unacceptably with reactor operation." The accident analysis in SAR Section 13.2.2 relies on the fuel temperature coefficient (α_F) calculated in SAR Section 4.5.3.3. This coefficient has been reviewed from a variety of published reports (see "University of Wisconsin Nuclear Reactor LEU Conversion Report," issued August 2008 (Ref. 34)), and, although the values vary slightly from one TRIGA reactor to another, the values are characteristic of the TRIGA fuel, operating conditions, and methods employed, and are not sensitive to operating history. As such, the NRC staff concludes that it is acceptable that the licensee provides no TS on limitations on the reactivity coefficients.

5.3.1.6 TS 3.1.6 Fuel Parameters

See Section 2.2.1 of this report.

5.3.2 TS 3.2 Reactor Control and Safety System

5.3.2.1 TS 3.2.1 Control Rods

See Section 2.2.2 of this report.

5.3.2.2 TS 3.2.2 Reactor Measuring Channels

TS 3.2.2 describes those reactor measuring channels that provide required information to the operators, alarms, and interlocks, or scram the reactor:

Applicability

This specification applies to the information, which shall be available to the RO during reactor operation.

Objective

The objective is to specify the minimum number of measuring channels that shall be available to the operator to assure safe operation of the reactor.

Specifications

The reactor shall not be operated in the specified mode unless the minimum number of measuring channels listed in this table is operable:

Measuring Channel	Minimum Number Operable
Start-up Count Rate	1
Fuel element temperature	1
Linear power level	1
Percent power level	1

TS 3.2.2 establishes which measuring channels must be operable during reactor operation.

- The startup count rate provides the interlock in TS 3.2.3, Table 2.
- The fuel element temperature provides the fuel element temperature scram in TS 3.2.3, Table 1.
- The linear power level provides the linear power level scram in TS 3.2.3, Table 1.
- The percent power level provides the percent power level scram in TS 3.2.3, Table 1.

TS 3.2.2 helps ensure that, during the normal operation of UUTR, sufficient information is available to the operator to help ensure safe operation of the reactor. The minimum number of operable measuring channels shown in the table in TS 3.2.2 provides the operator with the startup count rate, fuel temperature, linear power, and percent power, and will be displayed at the control console providing continuous information.

The NRC staff finds that TS 3.2.2 helps ensure that the reactor will not be operated unless the required minimum number of measuring channels is available to the operator to help ensure safe operation of the reactor. On this basis, the NRC staff concludes that TS 3.2.2 is acceptable.

5.3.2.3 TS 3.2.3 Reactor Safety System

See Section 2.5.5 of this report.

5.3.3 TS 3.3 Coolant System

See Section 2.6 of this report.

5.3.4 TS 3.4 Confinement

TS 3.4 states the following:

Applicability

These specifications apply to the area housing the reactor and the ventilation system controlling that area.

Objective

The objective is to provide restrictions on radioactive airborne materials releases into environment.

Specifications

1. Confinement is required for reactor operation and/or any movement of irradiated fuel, and,
2. To achieve confinement, the ventilation system shall be operating in accordance with TS 3.5.

TS 3.4, Specifications 1 and 2 help ensure that, during reactor operation or any movement of irradiated fuel, the potential consequences from the release of radioactivity will be minimized. TS 3.4 helps ensure that the ventilation system is fully operable and that an adequate pressure differential exists between the reactor room and the outside of the MEB. The NRC staff finds that the TS 3.4, Specifications 1 and 2 are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007 and, therefore is acceptable to the NRC staff.

5.3.5 TS 3.5 Ventilation System

TS 3.5 states the following:

Applicability

This specification applies to the operation of the reactor area ventilation system.

Objective

The objective is to assure that the ventilation system shall be in operation to mitigate the consequences of possible releases of radioactive materials resulting from reactor operation.

Specifications

1. The pressure difference between the reactor room and outside of the Merrill Engineering Building is larger than 0.1 inches-of-water.
2. In the event of a substantial release of airborne radioactivity within the reactor area, the ventilation system will be secured or operated in the limited intake mode to prevent the release of a significant quantity of airborne radioactivity from the reactor area.

TS 3.5 helps ensure that the HVAC system is providing the necessary ventilation conditions analyzed in the MHA, as described in UUTR SAR Section 13.2.1. UUTR SAR Section 9.1.4.1 discusses the operational aspects and modes of the HVAC system. Operation of the UUTR in conformance with the requirements of TS 3.5 helps ensure that postulated radiation doses to the UUTR staff or to members of the public will be minimized during normal operation or the MHA. On the basis provided above, the NRC staff finds that TS 3.5 is appropriate and acceptable.

5.3.6 TS 3.6 Emergency Power

In the UUTR SAR Sections 1.3.5.3 and 8.1, the licensee states that emergency power is not required. Loss of electrical power will initiate a reactor scram and will not result in the release of any radioactive material or increase the dose to the public. The MHA includes a loss of ventilation and therefore provides a bounding scenario for the loss of power event. The NRC staff reviewed the UUTR SAR and concludes that emergency power is not required for UUTR.

5.3.7 TS 3.7 Radiation Monitoring Systems and Effluents

5.3.7.1 TS 3.7.1 Radiation Monitoring Systems

See Section 3.1.4 of this report.

5.3.7.2 TS 3.7.2 Effluents

See Section 3.1.1 of this report.

5.3.8 TS 3.8 Experiments

5.3.8.1 TS 3.8.1 Reactivity Limits

See Section 4.1.2 of this report.

5.3.8.2 TS 3.8.2 Materials

TS 3.8.2 states the following:

Applicability

This specification applies to experiments installed in the reactor and its irradiation facilities.

Objective

The objective is to prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure.

Specifications

The reactor shall not be operated unless the following conditions governing experiments exist:

1. Explosive materials, such as gunpowder, TNT, nitroglycerin, or PETN, in quantities greater than 25 milligrams TNT equivalent shall not be irradiated in the reactor or irradiation facilities. Explosive materials in quantities less than 25 milligrams TNT equivalent may be irradiated provided the pressure produced upon detonation of the explosive has been calculated and/or experimentally demonstrated to be less than half the design pressure of the container, and,

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

TS 3.8.2, Specification 1 limits the quantity of explosive material to 25 milligrams or less. Explosive material up to 25 milligrams may be irradiated, provided the pressure produced on detonation of the explosive has been calculated or experimentally demonstrated to be less than half the design pressure of the irradiation container. This specification helps ensure that no damage to the fuel cladding will result because of an experiment containing explosive material. This specification is consistent with the recommendations of RG 2.2, "Development of Technical Specifications for Experiments in Research Reactors," issued November 1973 (Ref. 36).

TS 3.8.2, Specification 2 requires double encapsulation for corrosive materials to reduce the likelihood that the encapsulation will fail and the corrosive material could damage the fuel cladding.

The NRC staff finds that the TS 3.8.2, Specifications 1 and 2, are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007 and therefore, is acceptable to the NRC staff.

5.3.8.3 TS 3.8.3 Failures and Malfunctions

TS 3.8.3 states the following:

Applicability

This specification applies to experiments installed in the reactor and its irradiation facilities.

Objective

The objective is to prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure.

Specifications

Where the possibility exists that the failure of an experiment 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 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 54.4 °C (130 °F or 327.6 °K) and where vapors formed by boiling this material can escape only through an undisturbed column of water above the core, 10% of these vapors can escape.

TS 3.8.3, Specifications 1 through 4 contain standard research reactor assumptions for experiments and help ensure that the source term calculations are conservative. The NRC staff finds that the TS 3.8.3, Specifications 1 through 4 are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007 and therefore, is acceptable to the NRC staff.

5.3.9 TS 3.9 Facility-Specific Limited Conditions for Operation

The UUTR SAR indicates that there are no facility-specific TS LCOs applicable to UUTR. The NRC staff reviewed the UUTR SAR and concludes that UUTR has no facility-specific TS.

5.4 Surveillance Requirements

5.4.0 TS 4.0 Surveillance Requirements

NUREG-1537 and ANSI/ANS-15.1-2007 recommend surveillance requirements that prescribe the frequency and scope of the surveillance activity to help ensure that the LCOs are acceptably maintained.

TS 4.0 required that changes to certain important systems be controlled to their original design and fabrication specifications, or, if to new specifications, that those specifications be reviewed. TS 4.0 also governed the scheduling of required surveillance testing to allow operational flexibility that does not affect safety. Since TS 4.0 maintained acceptable control over the design change process, it is acceptable to the NRC staff.

Applicability

This specification applies to the surveillance requirements of any system related to reactor safety.

Objective

The objective is to verify the proper operation of any system related to reactor safety.

Specifications

1. Surveillance requirements may be deferred during reactor shutdown (except TS 4.3 (1) and (5)); however, they shall be completed prior to reactor startup unless reactor operation is required for performance of the surveillance. Such surveillance shall be performed as soon as practicable after reactor startup. Scheduled surveillance, which cannot be performed with the reactor operating, may be deferred until a planned reactor shutdown.

2. Any additions, modifications, or maintenance to the ventilation system, the core and its associated support structure, the pool or its penetrations, the pool coolant system, the rod drive mechanism or the reactor safety system shall be made and tested in accordance with the specifications to which the systems were originally designed and fabricated or to specifications reviewed by the Reactor Safety Committee (RSC). A system shall not be considered operable until after it is successfully tested.

TS 4.0 helps to ensure that the quality of systems and components will be maintained to their original design and fabrication specifications. TS 4.0, described above, follows the guidance provided in NUREG-1537, Appendix 14.1, Section 4.0. Accordingly, the NRC staff finds that TS 4.0, Specifications 1 and 2, provide appropriate UUTR surveillance practices, are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007, and therefore, are acceptable to the NRC staff.

5.4.1 TS 4.1 Reactor Core Parameters

TS 4.1 states the following:

Applicability

This specification applies to the surveillance requirements for reactor core parameters.

Objective

The objective is to verify that the reactor does not exceed the authorized limits for power, shutdown margin, core excess reactivity, specifications for fuel element condition and verification of the total reactivity worth of each control rod.

Specifications

1. The shutdown margin shall be determined prior to each day's operation, prior to each operation extending more than one day, or following any change ($> \$0.25$) from a reference core.
2. The total reactivity worth of each control rod shall be measured semi-annually or following any change ($> \$0.25$) from a reference core.
3. The core excess reactivity shall be determined semi-annually or following any reactivity change ($> \$0.25$) from a reference core.
4. Each planned change in core configuration shall be determined to meet the requirements of TS 3.1.4 of these specifications before the core is loaded.
5. Inspection for transverse bend and length exceeding for fuel elements, cladding defect, overall visual inspection shall be performed biennially.
6. Fuel burnup of Uranium-235 in the UZrH fuel matrix shall not exceed 50% of initial content. Fuel burnup calculation shall be performed biennially.

TS 4.1, Specification 1 helps ensure the determination of SDM as required to support TS 3.1.2.

TS 4.1, Specification 2 helps ensure the determination of the control rod worths as required to support TS 3.1.2.

TS 4.1, Specification 3 helps ensure the determination of core excess reactivity as required to support TS 3.1.3.

TS 4.1, Specification 4 helps ensure that changes to the core configuration are limited to the conditions specified in TS 3.1.4.

TS 4.1, Specification 5 helps ensure that the fuel inspection requirements of TS 3.1.6, Specifications 1 through 4, are accomplished.

TS 4.1, Specification 6 helps ensure that the generally accepted lifetime for TRIGA fuel is not exceeded as required to support TS 3.1.6, Specification 5.

The NRC staff reviewed TS 4.1, Specifications 1 through 6, for the reactor core components. The NRC staff finds that TS 4.1 is consistent with guidance in NUREG-1537 and ANSI/ANS-15.1-2007 and therefore, acceptable to the NRC staff.

5.4.2 TS 4.2 Reactor Control and Safety Systems

TS 4.2 states the following:

Applicability

This specification applies to the surveillance requirements of reactor control and safety systems.

Objective

The objective is to verify performance and operability of those systems and components, which are directly related to reactor safety.

Specifications

1. Control rod inspection: The control rods and drives shall be visually inspected for damage or deterioration biennially.
2. SCRAM time: The scram time shall be measured annually and following maintenance to the control element or their drives.
3. Control rod movement: The speed of the control rod movement shall be measured annually.
4. Fuel element temperature (channel calibration, channel test, and channel check): The fuel element temperature measuring channel shall be calibrated semi-annually. The channel test shall be performed annually. The channel check shall be performed prior and during start-up and during every operation of the reactor.
5. Linear power level (channel check and channel calibration): Channel check shall be performed for every operation of the reactor. Channel calibration of the linear power channel shall be performed semi-annually.

6. Percent power level (channel check and channel calibration): Channel check shall be performed for every operation of the reactor and channel calibration shall be performed semi-annually.
7. Manual console scram (channel test): Manual console scram function channel test shall be performed prior to every reactor operation.
8. Magnet key current switch (channel test): The magnet key current channel test switch shall be performed prior to every reactor operation.
9. Console power supply (channel test): Console power supply system shall be channel tested prior to every reactor operation.
10. Reactor tank water level (channel check and channel test): Reactor tank water level shall be channel checked and channel tested prior to every reactor operation.
11. Startup count rate interlock (channel test): Startup count rate interlock system shall be channel tested prior to every reactor operation.
12. Control rod withdrawal interlocks (channel check and channel test): Control rod interlock function shall be channel checked prior to every reactor operation. Control rod interlock function shall be channel tested prior to every reactor operation and semi-annually.

TS 4.2, Specification 1 helps ensure the determination of the control rod inspection as required to support TS 3.2.1, Specification 1.

TS 4.2, Specification 2 helps ensure the determination of the scram time as required to support TS 3.2.1, Specification 2.

TS 4.2, Specification 3 helps ensure the determination of control rod movement as required to support TS 3.2.1, Specification 3.

TS 4.2, Specification 4 helps ensure the fuel element temperature channel calibration, channel test, and channel check are performed to support TS 3.2.2 channel operability and the corresponding TS 3.2.3 setpoint.

TS 4.2, Specification 5 helps ensure that the linear power level channel check and channel calibration are performed as required to support TS 3.2.2 channel operability and the corresponding TS 3.2.3 setpoint.

TS 4.2, Specification 6 helps ensure that the percent power level channel check and channel calibration are performed as required to support TS 3.2.2 channel operability and the corresponding TS 3.2.3 setpoint.

TS 4.2, Specification 7 helps ensure that the manual console scram channel test is performed as required to support TS 3.2.2 channel operability and the corresponding TS 3.2.3 setpoint.

TS 4.2, Specification 8 helps ensure that the magnet key current switch channel test is performed as required to support TS 3.2.2 channel operability and the corresponding TS 3.2.3 function.

TS 4.2, Specification 9 helps ensure that the console power supply channel test is performed as required to support TS 3.2.2 channel operability and the corresponding TS 3.2.3 function.

TS 4.2, Specification 10 helps ensure that the reactor tank water level channel check and channel test are performed as required to support TS 3.2.2 channel operability and the corresponding TS 3.2.3 setpoint.

TS 4.2, Specification 11 helps ensure that the startup count rate interlock channel test is performed as required to support TS 3.2.2 channel operability and the corresponding TS 3.2.3 setpoint.

TS 4.2, Specification 12 helps ensure that the control rod withdrawal interlocks channel check and channel test are performed as required to support TS 3.2.2 channel operability and the corresponding TS 3.2.3 function

The NRC staff reviewed TS 4.2, Specifications 1 through 12 for reactor control and safety systems. The NRC staff finds that TS 4.2 is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007 and therefore, acceptable the NRC staff.

5.4.3 TS 4.3 Coolant System

TS 4.3 states the following:

Applicability

This specification applies to the surveillance requirements for the reactor tank water.

Objective

The objective is to assure that the reactor tank water level and the bulk water temperature monitoring systems are operating, and to verify appropriate alarm settings.

Specifications

1. A channel check of the reactor tank water level monitor shall be performed monthly.
2. A channel test of the reactor tank water temperature system shall be performed prior to each day's operation or prior to each operation extending more than one day.
3. A channel calibration of the reactor tank water temperature system shall be performed semi-annually.
4. The reactor tank water conductivity and pH shall be measured monthly.
5. The reactor tank water radioactivity shall be measured monthly.

TS 4.3, Specification 1 helps ensure that a channel check of the reactor tank water level monitor is performed as required to support TS 3.3, Specification 1; TS 3.2.2 channel operability; and the corresponding TS 3.2.3 setpoint.

TS 4.3, Specification 2 helps ensure that a channel test of the reactor tank water temperature system is performed as required to support TS 3.3, Specification 2; TS 3.2.2 channel operability; and the corresponding TS 3.2.3 setpoint.

TS 4.3, Specification 3 helps ensure that the channel calibration of the reactor tank water temperature system is performed as required to support TS 3.3, Specification 2; TS 3.2.2 channel operability; and the corresponding TS 3.2.3 setpoint.

TS 4.3, Specification 4 helps ensure that the determination of reactor tank water conductivity and pH is performed as required to support TS 3.3, Specifications 3 and 4.

TS 4.3, Specification 5 helps ensure that the determination of reactor tank water radioactivity is performed as required to support TS 3.3, Specification 5.

The NRC staff reviewed TS 4.3, Specifications 1 through 5 for coolant system. The NRC staff finds that TS 4.3 is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007 and therefore, acceptable the NRC staff.

5.4.4 TS 4.4 Confinement

TS 4.4 states the following:

Applicability

This specification applies to the reactor confinement.

Objective

The objective is to assure that air is swept out of confinement and exhausted through a monitored release point (two fume hood systems located at Fuel Inspection area).

Specification

The ventilation system shall be verified operable in accordance with TS 4.5 monthly.

TS 4.4 helps ensure that the UUTR confinement satisfies the analysis assumptions of the accident analysis and TS 4.4. The NRC staff reviewed TS 4.4 for reactor confinement. The NRC staff finds that TS 4.4 is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007 and therefore, acceptable the NRC staff.

5.4.5 TS 4.5 Ventilation System

TS 4.5 states the following:

Applicability

This specification applies to the reactor area confinement ventilation system.

Objective

The objective is to assure the proper operation of the ventilation system in controlling releases of radioactive material to the unrestricted area.

Specifications

1. A channel check of the reactor area confinement ventilation system's ability to maintain a negative pressure in the reactor room with respect to surrounding areas shall be performed prior to each day's operation or prior to each operation extending more than one day.
2. A channel test of the reactor area confinement ventilation system's ability to be secured shall be performed monthly.
3. A channel test of the ventilation system's ability to operate in the limited intake mode shall be performed monthly.

TS 4.5, Specification 1 helps ensure that the conditions of TS 3.5 regarding maintenance of negative pressure are satisfied.

TS 4.5, Specification 2 helps ensure the ability of the HVAC system to be secured.

TS 4.5, Specification 3 helps ensure the ability of the HVAC system to operate in the limited intake mode.

TS 4.5 helps ensure that the HVAC system is operable and that it satisfies the analysis assumptions of the accident analysis and TS 3.5. The NRC staff reviewed TS 4.5, Specifications 1 through 3, for the facility ventilation system. The NRC staff finds that TS 4.5 is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007 and, therefore acceptable the NRC staff.

5.4.6 TS 4.6 Emergency Power System

In the UUTR SAR Sections 1.3.5.3 and 8.1, the licensee states that emergency power is not required. Loss of electrical power will initiate a reactor scram and will not result in the release of any radioactive material or increase the dose to the public. The MHA includes a loss of ventilation and therefore provides a bounding scenario for the loss of power event. The NRC staff reviewed the UUTR SAR and concludes that emergency power is not required for UUTR.

5.4.7 TS 4.7 Radiation Monitoring System

TS 4.7 states the following:

Applicability

This specification applies to the surveillance requirements for the area radiation monitoring equipment and the air monitoring systems.

Objective

The objective is to assure that the radiation monitoring equipment is operating properly and to verify the appropriate alarm settings.

Specifications

1. A channel test of the ARM system, as in TS 3.7.1, shall be performed prior to each day's operation or prior to each operation extending more than one day.
2. A channel test of the CAM system, as in TS 3.7.1, shall be performed monthly.
3. A channel calibration of the radiation monitoring systems, as in TS 3.7.1, shall be performed annually.

TS 4.7, Specification 1 helps ensure that a channel test of the ARM system is performed to support TS 3.7.1.

TS 4.7, Specification 2 helps ensure that a channel test of the CAM system is performed as required to support TS 3.7.1.

TS 4.7, Specification 3 helps ensure that calibrations of the ARM and CAM systems are performed as required to support TS 3.7.1.

TS 4.7 helps ensure the availability of the radiation monitoring system. The NRC staff reviewed TS 4.7, Specifications 1 through 3, for the reactor radiation monitoring system. The NRC staff finds that TS 4.7 is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007 and therefore, acceptable the NRC staff.

5.4.8 TS 4.8 Experiments

TS 4.8 states the following:

Applicability

This specification applies to the surveillance requirements for experiments installed in the reactor and its irradiation facilities.

Objective

The objective is to prevent the conduct of experiments, which may damage the reactor or release excessive amounts of radioactive materials as a result of experiment failure.

Specifications

1. The reactivity worth of an experiment shall be estimated or measured, as appropriate, before reactor operation with said experiment.
2. An experiment shall not be installed in the reactor or its irradiation facilities unless a safety analysis has been performed and reviewed for compliance with TS 3.8 by the RSC in full accord with TS 6.2.3, and the procedures, which are established for this purpose.

TS 4.8, Specification 1 helps ensure that the reactivity worth of an experiment is determined before the performance in UUTR, as required to support TS 3.8.1, TS 3.1.3, and TS 3.1.2.

TS 4.8, Specification 2 helps ensure that experiments are not inserted into the reactor unless a valid safety analysis has been performed and reviewed, as required to support TS 3.8.1, TS 3.8.2, and TS 3.8.3.

The NRC staff reviewed TS 4.8, Specifications 1 and 2 for controlling experiments. The NRC staff finds that TS 4.8 is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007 and therefore, acceptable the NRC staff.

5.4.9 TS 4.9 Facility-Specific Surveillance

The UUTR SAR indicates that there are no facility-specific TS LCOs applicable to UUTR. The NRC staff reviewed the UUTR SAR and concludes that UUTR has no facility-specific TS. Therefore, TSs for facility-specific surveillances are not required.

5.5 Design Features

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

5.5.1 TS 5.1 Site and Facility Description

TS 5.1 states the following:

Applicability

This specification applies to the University of Utah TRIGA Reactor site location and specific facility design features.

Objective

The objective is to specify the location of specific facility design features.

Specifications

1. The restricted area is that area inside the MEB 1205 A room through 1205 G room. The unrestricted area is that area outside the MEB 1205 A room through 1205 G room, and MEB 1206.
2. The Merrill Engineering Building houses the TRIGA reactor.
3. The reactor room shall be equipped with ventilation systems designed to exhaust air or other gases from the reactor room and release them from a stack at a minimum of 40 feet from ground level.
4. Emergency shutdown controls for the ventilation systems shall be located in the reactor control room.
5. Free volume of the reactor area shall be $5.65 \times 10^8 \text{ cm}^3$.

The NRC staff finds that TS 5.1, Specifications 1 through 5 provide important features of the physical design of the facility used to house UUTR and define the boundaries of the facility that is being licensed. These specifications support the accident analysis fundamental to acceptably

meeting 10 CFR Part 20 requirements and define the operational and site area boundaries for the facilities, and are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On this basis, the NRC staff concludes that TS 5.1, Specifications 1 through 5 are acceptable.

5.5.2 TS 5.2 Reactor Coolant System

See Section 2.3 of this report.

5.5.3 TS 5.3 Reactor Core and Fuel

5.5.3.1 TS 5.3.1 Reactor Core

See Section 2.2 of this report.

5.5.3.2 TS 5.3.2 Control Rods

See Section 2.2.2 of this report.

5.5.3.3 TS 5.3.3 Reactor Fuel

See Section 2.2.1 of this report.

5.5.4 TS 5.4 Fuel Storage

TS 5.4 presents the reactor fuel storage design features as follows:

Applicability

This specification applies to the storage of reactor fuel at times when it is not in the reactor core.

Objective

The objective is to assure that fuel, which is being stored shall not become critical and shall not reach an unsafe temperature.

Specification

1. All fuel elements shall be stored in a geometrical array where the k-effective is less than 0.9 for all conditions of moderation.
2. Irradiated fuel elements and fuel devices shall be stored in an array, which will permit sufficient natural convection cooling by water or air such that the temperature of the fuel element or fueled device will not exceed the safety limit.

TS 5.4, Specification 1 limits the k_{eff} value to 0.9, which is recommended in NUREG-1537 and ANSI/ANS-15.1-2007. UUTR SAR, Section 9.2.4, describes a comprehensive analysis of the fuel element criticality for the in-tank storage racks, which demonstrates that the fuel cannot exceed the subcritical value cited in TS 5.4, Specification 1 under normal or accident conditions. On this basis, the NRC staff finds that fuel cooling is assured and acceptable.

TS 5.4, Specification 2 provides the basic design requirement to help ensure adequate cooling by natural convection cooling, either by water or air, of stored irradiated fuel rods and fueled devices. UUTR SAR Section 9.2.5 describes a comprehensive analysis of fuel element criticality for the fuel storage pits. The analysis included both air- and water-moderated conditions and demonstrated acceptable subcriticality results. Cooling and exposure of unirradiated fuel would not be a concern under either air- or water-immersed situations.

The NRC staff finds that TS 5.4, Specifications 1 and 2 for fuel storage, is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007 and, therefore, acceptable to the NRC staff.

5.6 Administrative Controls

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

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

5.6.1 TS 6.1 Organization

Responsibility for the safe operation of the reactor facility is vested within the chain of command shown in TS Figure 6-1, which depicts the licensee's organization. The organizational responsibilities delineated in TS Figure 6-1 are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007, and are acceptable to the NRC staff.

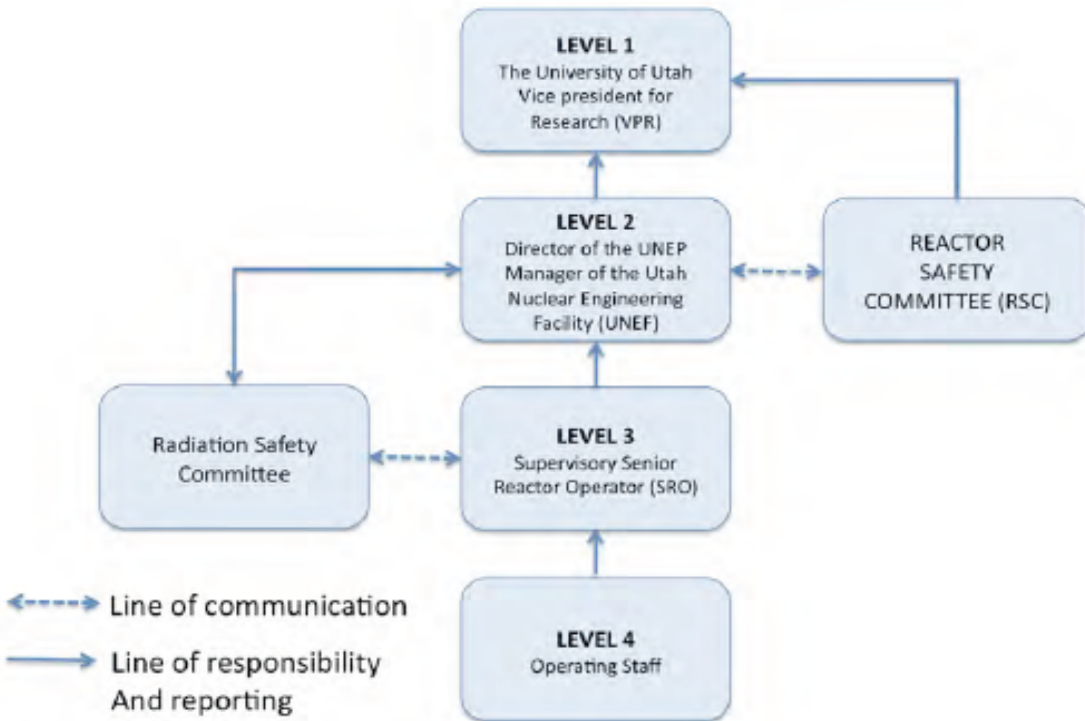


Figure 5-1 University of Utah Administrative Organization for Nuclear Reactor Operations

5.6.1.1 TS 6.1.1 Structure

TS 6.1.1 states the following:

The reactor administration shall be related to the University as shown in Fig. 6-1.

The organizational structure described in TS 6.1.1 is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007 and therefore, acceptable to the NRC staff.

5.6.1.2 TS 6.1.2 Responsibilities

TS 6.1.2 states the following:

The following specific organizational levels and responsibilities shall exist:

1. The UUTR is an integral part of the Utah Nuclear Engineering Facilities (UNEF) of the University of Utah Nuclear Engineering Program (UNEP) at the University of Utah. The organization of the facility management and operation is illustrated in Fig. 6-1. The responsibilities and authority of each member of the operating staff shall be defined in writing, and
2. As indicated in Fig. 6.1, the RSC shall report to Level 1. Radiation safety personnel shall report to level 2. Additional description of levels follows:

- 2.1 Level 1: Individual responsible for the reactor facility's licenses, i.e., the Associate Vice President for Research in the Office of Vice President for Research; The Vice President for Research will assign which of the Associate Vice Presidents for Research will be the responsible Level 1 individual.
- 2.2 Level 2: Individual responsible for reactor facility operation, i.e., the Utah Nuclear Engineering Facility (UNEF) Manager shall be the Director of the Utah Nuclear Engineering Program (UNEP).
- 2.3 Level 3: Individual responsible for day-to-day operation or shift shall be the Reactor Supervisor (RS). This person shall be an SRO.
- 2.4 Level 4: Operating staff shall be SROs, ROs, and trainees.

The organizational responsibilities described in TS 6.1.2 are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007 and therefore, acceptable to the NRC staff.

5.6.1.3 TS 6.1.3 Staffing

TS 6.1.3 states the following:

1. The minimum staffing when the reactor is operating shall be:
 - 1.1 A licensed RO or the RS in the control room,
 - 1.2 A second person present in the UNEF able to carry out prescribed instructions, and,
 - 1.3 If neither of these two individuals is the RS, the RS shall be readily available on call. Readily available on call means an individual who:
 - i. Has been specifically designated and the designation is known to the operator on duty,
 - ii. Can be rapidly contacted by phone by the operator on duty, and,
 - iii. Is capable of getting to the reactor facility within a reasonable time under normal conditions (e.g., 30 minutes or within a 15- mile radius).
2. A list of reactor facility personnel by name and telephone number shall be readily available in the control room for use by the operator. The list shall include:
 - 2.1 UNEP Director and/or UNEF Manager
 - 2.2 RS
 - 2.3 Radiation Safety Officer
 - 2.4 Any Licensed RO or SRO

3. Events requiring the direction of the RS:
 - 3.1 Initial startup and approach to power of the day,
 - 3.2 All fuel or control-rod relocations within the reactor core region,
 - 3.3 Relocation of any in-core experiment or irradiation facility with a reactivity worth greater than one dollar, and,
 - 3.4 Recovery from unplanned or unscheduled shutdown or significant power reduction.

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

TS 6.1.3, Specification 2 describes the organization of the facility and the requirement for establishing formal responsibilities and authorities for the operating staff, which is consistent with the guidance in ANSI/ANS-15.1-2007 and NUREG-1537.

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

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

5.6.1.4 TS 6.1.4 Selection and Training of Personnel

TS 6.1.4 states the following:

The selection, training and requalification of operations personnel shall be in accordance with ANSI/ANS 15.4-1988; R1999, "Standard for the Selection and Training of Personnel for Research Reactors."

TS 6.1.4 established the criteria for the training and requalification program for operations personnel. The licensee used ANSI/ANS-15.4, "Selection and Training of Personnel for Research Reactors," 1988 (Ref. 37), as guidance for the selection and training of personnel. The NRC staff finds that TS 6.1.4 is consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.4-1988, and therefore, is acceptable.

5.6.2 TS 6.2 Review and Audit

TS 6.2 states the following:

The RSC shall have primary responsibility for review and audit of the safety aspects of reactor facility operations. The RSC or a subcommittee thereof shall audit reactor operations semiannually. Minutes, findings or reports of the RSC shall be presented to Level 1 and Level 2 management within ninety (90) days of completion.

The function of the RSC, as outlined in TS 6.2, is consistent with the guidance of NUREG-1537 and ANSI/ANS-15.1-2007. The NRC staff concludes that TS 6.2 is acceptable.

5.6.2.1 TS 6.2.1 Composition and Qualifications

TS 6.2.1 states the following:

An RSC of at least five (5) members knowledgeable in fields, which relate to reactor engineering and nuclear safety, shall review and evaluate the safety aspects associated with the operation and use of the facility. Level 1 management shall appoint the RSC members and RSC chair. Individuals may be either from within or outside the University of Utah. Qualified and approved alternates may serve in the absence of regular members. The Level 2 and Level 3 should be the members of the RSC but they would not comprise a majority of voting RSC members.

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

5.6.2.2 TS 6.2.2 RSC Rules

TS 6.2.2 states the following:

The operations of the RSC shall be in accordance with written procedures including provisions for:

1. Meeting frequency (at least annually),
2. Voting rules,
3. Quorums (5 members, no more than two voting members may be of the operating staff at any time),
4. Method of submission and content of presentation to the committee,
5. Use of subcommittees, and,
6. Review, approval, and dissemination of minutes.

TS 6.2.2 establishes the RSC meeting frequency, rules, and the committee charter. The NRC staff finds that TS 6.2.2 consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007 and therefore, is acceptable.

5.6.2.3 TS 6.2.3 RSC Review Function

TS 6.2.3 states the following:

The responsibilities of the RSC, or designated Subcommittee thereof, include, but are not limited to, the following:

1. Review all changes made under 10 CFR 50.59,
2. Review of all new procedures and substantive changes to existing procedures,
3. Review of proposed changes to the technical specifications, license or charter,
4. Review of violations of technical specifications, license, or violations of internal procedures or instructions having safety significance,
5. Review of operating abnormalities having safety significance,
6. Review of all events from reports required in TS 6.6.1 and 6.7.2 of these Technical Specifications,
7. Review of audit reports, and,
8. Review of the experiments and classes of the experiments.

TS 6.2.3 establishes the RSC review functions. The NRC staff finds TS 6.2.3 consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007 and therefore, is acceptable.

5.6.2.4 TS 6.2.4 RSC Audit Function

TS 6.2.4 states the following:

The RSC or a Subcommittee thereof shall audit reactor operations at least annually. The annual audit shall include at least the following:

1. Facility operations for conformance to the technical specifications and applicable license or charter conditions,
2. The retraining and requalification program for the operating staff,
3. The results of action taken to correct those deficiencies that may occur in the reactor facility equipment, systems, structures, or methods of operation that affect reactor safety, and
4. The Emergency Response Plan and implementing procedures.

TS 6.2.4 establishes the RSC audit functions. The NRC staff finds TS 6.2.4 consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007 and therefore, is acceptable.

5.6.3 TS 6.3 Radiation Safety

See Section 3.1.2 of this report.

5.6.4 TS 6.4 Procedures

TS 6.4 states the following:

Written operating procedures shall be adequate to assure the safety of operation of the reactor, but shall not preclude the use of independent judgment and action should the situation require such. Operating procedures shall be in effect 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 have an effect on reactor safety,
5. Radiation protection,
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 required plans such as emergency or security plans, and,
8. Use receipt, and transfer of by-product material held under the reactor license.

Substantive changes to the above procedures shall be made only after review by the RSC. Except for radiation protection procedures, unsubstantive changes shall be approved prior to implementation by the UNEP Director and documented by the UNEP Director within 120 days of implementation. Unsubstantive changes to radiation protection procedures shall be approved prior to implementation by the Radiation Safety Officer (RSO), and documented by the RSO within 120 days of implementation.

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

TS 6.4, Specifications 1 through 8, establish operational procedures for the UUTR. The NRC staff finds that TS 6.4 is consistent with the guidance of ANSI/ANS-15.1-2007 and NUREG-1537 and therefore, is acceptable.

5.6.5 TS 6.5 Experiments Review and Approval

TS 6.5 states the following:

Approved experiments shall be carried out in accordance with established and approved procedures. Procedures related to experiment review and approval shall include:

1. All new experiments or class of experiments shall be reviewed by the RSC and approved in writing by the Level 2 or designated alternates prior to initiation, and,
2. Substantive changes to previously approved experiments shall be made only after review by the RSC and approved in writing by the Level 2 or designated alternates. Minor changes that do not significantly alter the experiment may be approved by Level 3 or higher.

TS 6.5, Specifications 1 and 2 require review and approval of different types of experiments before being performed at UUTR and specify the extent of the analysis that is submitted for review. TS 6.5 helps ensure acceptable management control over experiments. The NRC staff finds that TS 6.5 is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007 and, therefore is acceptable.

5.6.6 TS 6.6 Required Actions

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

TS 6.6.1 states the following:

In the event a safety limit (fuel temperature) is exceeded:

1. The reactor shall be shut down and reactor operation shall not be resumed until authorized by the NRC,
2. An immediate notification of the occurrence shall be made to the UNEP Director, and Chairperson of the RSC, NRC, and
3. A report, and any applicable follow-up report, shall be prepared and reviewed by the RSC. The report shall describe the following:
 - 3.1 Applicable circumstances leading to the violation including, when known, the cause and contributing factors,
 - 3.2 Effects of the violation upon reactor facility components, systems, or structures and on the health and safety of personnel and the public, and
 - 3.3 Corrective action to be taken to prevent recurrence.

TS 6.6.1, Specifications 1 through 3 require the facility to shut down in the event that an SL is exceeded. The facility may not resume operation without authorization from the NRC. The violation must also be reported to the RSC and the NRC. The reporting requirement is detailed in TS 6.7.2, specifying that the NRC must be notified within 24 hours by telephone and requiring a report to be submitted to the NRC within 14 days. TS 6.6.1(3) specifies the content of the report and the appropriate evaluations and corrective actions to be taken. The NRC staff finds that these TS actions are consistent with the guidance of NUREG-1537 and ANSI/ANS-15.1-2007 and therefore, are acceptable.

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

TS 6.6.2 states the following:

For all events, which are required by regulations or Technical Specifications to be reported to the NRC within 24 hours under Section 6.7.2 except a safety limit violation, the following actions shall be taken:

1. The reactor shall be secured and UNEP Director notified,
2. Operations shall not resume unless authorized by the UNEP Director,
3. The RSC shall review the occurrence at their next scheduled meeting, and,
4. A report shall be submitted to the NRC in accordance with Section 6.7.2 of these Technical Specifications.

TS 6.6.2, Specifications 1 through 4 require the facility to shut down in case of a reportable occurrence. The event and corrective actions taken must also be reported to the facility director, who notifies the RSC Chairman. The reporting requirement is also detailed in TS 6.7.2, specifying that the NRC must be notified within 24 hours by telephone and a report submitted to the NRC within 14 days. TS 6.6.2 specifies the content of the report and the appropriate evaluations and corrective actions to be taken. The NRC staff finds that these TS actions are consistent with the guidance of NUREG-1537 and ANSI/ANS-15.1-2007 and therefore, are acceptable.

5.6.7 TS 6.7 Reports

5.6.7.1 TS 6.7.1 Annual Operating Report

TS 6.7.1 states the following:

An annual report shall be created and submitted by the UNEP Director to the U.S. NRC by the end of July of each year 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 SCRAMs, including reasons therefore,
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 finds that TS 6.7.1, Specifications 1 through 7 annual operating report requirement, are consistent with guidance in NUREG-1537 and ANSI/ANS-15.1-2007 and therefore, are acceptable.

5.6.7.2 TS 6.7.2 Special Reports

TS 6.7.2 states the following:

In addition to the requirements of applicable regulations, and in no way substituting therefore, reports shall be made by the UNEP Director to the NRC as follows:

1. A report not later than the following working day by telephone and confirmed in writing by facsimile to the NRC Headquarters Operation Center, and followed by a written report that describes the circumstances of the event within 14 days to the U.S. NRC, Attn: Document Control Desk, Washington, D.C. 20555, of any of the following:
 - 1.1 Violation of the safety limit,
 - 1.2 Release of radioactivity from the site above allowed limits,
 - 1.3 Operation with actual safety system settings from required systems less conservative than the limiting safety system setting,
 - 1.4 Operation in violation of limiting conditions for operation,
 - 1.5 A reactor safety system component malfunction that renders or could render the reactor safety system incapable of performing its intended safety function. If the malfunction or condition is caused by maintenance, then no report is required,
 - 1.6 An unanticipated or uncontrolled change in reactivity greater than one dollar. Reactor trips resulting from a known cause are excluded;
 - 1.7 Abnormal and significant degradation in reactor fuel or cladding, or both, coolant boundary, or confinement boundary (excluding minor leaks) where applicable, or
 - 1.8 An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence or development of an unsafe condition with regard to reactor operations, and
2. A report within 30 days in writing to the U. S. NRC, Attn: Document Control Desk, Washington, D.C. 20555 of:
 - 2.1 Permanent changes in the facility organization involving Level 1–2 personnel, and

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

The NRC staff finds that TS 6.7.2, Specifications 1 and 2 special report requirements, are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007 and therefore, are acceptable.

5.6.8 TS 6.8 Records

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

TS 6.8.1 states the following:

1. Normal reactor operation (but not including supporting documents such as checklists, log sheets, etc., which shall be maintained for a period of at least one year),
2. Principal maintenance activities,
3. Reportable occurrences,
4. 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. RSC meetings and audit reports.

The NRC staff finds TS 6.8.1, Specifications 1 through 9 record requirements, are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007 and therefore, are acceptable.

5.6.8.2 TS 6.8.2 Records To Be Retained for at Least One Certification Cycle

TS 6.8.2 states the following:

Records of retraining and requalification of licensed ROs and SROs shall be retained at all times the individual is employed or until the certification is renewed. For the purpose of this technical specification, a certification is an NRC issued operator license.

The NRC staff finds TS 6.8.2, records to be retention requirements, consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007 and therefore, are acceptable.

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

TS 6.8.3 states the following:

1. Gaseous and liquid radioactive effluents released to the environs,
2. Offsite environmental monitoring surveys,
3. Radiation exposures for all personnel monitored,
4. Drawings of the reactor facility, and
5. Reviews and reports pertaining to a violation of the safety limit, the limiting safety system setting, or a limiting condition of operation.

The NRC staff finds TS 6.8.3, Specifications 1 through 5 lifetime record retention requirements are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007 and therefore, are acceptable.

5.7 Technical Specifications Conclusions

The NRC staff reviewed and evaluated the TSs as part of its review of the application for license renewal. The UUTR TS defined certain features, characteristics, and conditions governing the operation of UUTR. The NRC staff specifically evaluated the content of the TS to determine if they meet the requirements in 10 CFR 50.36. The NRC staff concluded that UUTR TS were acceptable for the following reasons:

- To satisfy the requirements of 10 CFR 50.36(a), UUTR provided proposed TS with the application for license renewal. As required by the regulations, the proposed TS included the appropriate summary bases.
- UUTR is a facility of the type described in 10 CFR 50.21(c), and, therefore, as required by 10 CFR 50.36(b), the facility license will include the TS. To satisfy the requirements of 10 CFR 50.36(b), UUTR provided TS derived from analyses in the SAR.
- To satisfy the requirements of 10 CFR 50.36(c)(1), UUTR provided TS specifying an SL on the fuel temperature and an LSSS for the reactor protection system to preclude reaching the SL.
- The TS acceptably implement the recommendations of NUREG-1537, Part 1, and ANSI/ANS-15.1-2007 by using definitions that are acceptable.
- The TS contain LCOs on each item that meets one or more of the criteria specified in 10 CFR 50.36(c)(2)(ii).
- The TS contain surveillance requirements that satisfy the requirements of 10 CFR 50.36(c)(3).
- The TS contain design features that satisfy the requirements of 10 CFR 50.36(c)(4).

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

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

6. CONCLUSIONS

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

- The application for license renewal dated March 25, 2005, supplemented in its entirety by an updated SAR dated June 8, 2011, complies with the standards and requirements of the Atomic Energy Act and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations*.
- The facility can operate in conformity with the application, as well as the provisions of the Atomic Energy Act and the rules and regulations of the Commission.
- There is reasonable assurance that (1) the activities authorized by the renewed facility operating license can be conducted at the designated location without endangering public health and safety, and (2) such activities will be conducted in compliance with the rules and regulations of the Commission.
- The licensee is technically and financially qualified to engage in the activities authorized by the renewed facility operating license, in accordance with the rules and regulations of the Commission.
- The issuance of the renewed facility operating license will not be inimical to the common defense and security or to health and safety of the public.

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