Enclosure 2

Safety Evaluation Report

University of Wisconsin

Docket No. 50-156

ADAMS Accession No. ML102370140

Safety Evaluation Report Related to the Renewal of

the Facility License for the TRIGA Reactor at the

University of Wisconsin

March 2011

Office of Nuclear Reactor Regulation

U.S. Nuclear Regulatory Commission

ABSTRACT

This safety evaluation report summarizes the findings of a safety review conducted by the staff of the U. S. Nuclear Regulatory Commission (NRC), Office of Nuclear Reactor Regulation. The NRC staff conducted this review in response to a timely application filed by the University of Wisconsin (the licensee) for a 20-year renewal of Facility License No. R-74 to continue to operate the University of Wisconsin 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, inspection reports prepared by NRC personnel, and on first-hand observations. On the basis of its review, the NRC staff concludes that the University of Wisconsin can continue to operate the facility for the term of the renewed facility license, in accordance with the license, without endangering public health and safety, facility personnel, or the environment.

CONTENTS

ABBREVIATIONS

A	ampere(s)
ac	alternating current
ADAMS	Agency-Wide Documents Access and Management System
ALARA	as low as low as reasonably achievable
ALI	annual limit on intake
ANS	American Nuclear Society
ANSI	American National Standards Institute
Ar	argon
BOL	beginning-of-life
C	Celsius
cc	cubic centimeter(s)
CFR	*Code of Federal Regulations*
cfm	cubic foot (feet)/minute
CHF	critical heat flux
Ci	curie(s)
Ci/m^3	curie(s) per cubic meter
Ci/s	curie(s) per second
cm	centimeter(s)
cps	count(s) per second
DAC	derived air concentrations
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DOE	U.S. Department of Energy
EOL	end-of-life
F	Fahrenheit
FLIP	fuel lifetime improvement program
ft	feet
GA	General Atomics
gpm	gallon(s) per minute
H	hydrogen
HEPA	high-efficiency particulate air
HEU	high-enriched uranium
HVAC	heating, ventilating and air conditioning
I&C	instrumentation and control
IFE	instrumented fuel element
in.	inch(es)
ISG	interim staff guidance
K	kelvin
kV	kilovolt(s)
kVA	kilovoltampere(s)
kW	kilowatt(s)
LCO	limiting condition for operation
LEU	low-enriched uranium
LOCA	loss-of-coolant accident
LSSS	limiting safety system setting
μCi/mL	microcurie(s) per milliliter
μS/cm	microsiemen(s) per centimeter
MCNP	REBUS-MCNP, a computer code

MDNBR	minimum departure from nucleate boiling ratio
mg	milligram(s)
MHA	maximum hypothetical accident
MΩ-cm	megaohm(s)-centimeter
MJ	megajoule(s)
mm	millimeter(s)
MOL	middle-of-life
mrem/yr	millirem per year
m/s	meter(s) per second
MTR	materials test reactor
MW	megawatt
MWd	megawatt-day(s)
MW(t)	megawatt thermal
NRC	U.S. Nuclear Regulatory Commission
RAI	request for additional information
RPS	reactor protection system
RSC	Reactor Safety Committee
RSO	Radiation Safety Office
RTR	research test reactor
s	second
SAR	safety analysis report
SER	safety evaluation report
sfcm	standard cubic feet per minute
SRM	staff requirements memorandum
TEDE	total effective dose equivalent
TLD	thermoluminescent dosimeter
TRIGA	training, research, isotopes, General Atomics
TS	technical specification(s)
U	uranium
UPS	uninterruptable power supply
URSC	University Radiation Safety Committee
UW	University of Wisconsin
UWPD	University of Wisconsin Police Department
UWNR	University of Wisconsin Nuclear Reactor
V	volt(s)
wt-%	weight-percent
Zr	zirconium

1 INTRODUCTION

1.1 Overview

By letter dated May 9, 2000, as supplemented by letters dated September 7, 2004, October 17, 2008; June 16 (two letters), July 8, August 11, November 22, and December 8, 2010; and January 28, and February 8, 2011, the University of Wisconsin (UW, the licensee) submitted to the U.S. Nuclear Regulatory Commission (NRC) an application for a 20-year renewal of the Class 104c Facility License No. R-74, Docket No. 50-156 for the UW Nuclear Reactor (UWNR). In its October 17, 2008, letter, UWNR provided an updated to its application with a new revision to the safety analysis report, "Safety Analysis Report for the University of Wisconsin Nuclear Reactor," Revision 2, (2008 SAR (Ref 9)).

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. UW is the holder of Facility License No. R-74 (the license), issued on November 23, 1960, for a period of 40 years. A renewal would authorize continued operation of the UWNR for an additional 20 years. Because UW filed the request for license renewal in a timely manner, until the NRC staff completes action on the renewal request, the licensee is permitted to continue operation of the UWNR under the terms and conditions of the existing license, in accordance with 10 CFR 2.109, "Effect of Timely Renewal Application."

The UWNR was licensed in 1960 at a maximum power level of 10 kilowatts (kW) as a teaching and research facility. On December 7, 1964, after a license amendment approval, the UWNR increased its power level to 250 kW, using the original flat-plate aluminum clad fuel. Operations with the original core ended October 1, 1967, after 2,268.5 critical hours and 105.65 megawatt (MW) hours of core exposure. The licensee installed a cooling system and converted the UWNR to a 1,000 kW TRIGA (training, research, isotopes, General Atomics) reactor with pulsing capability in 1967. The licensee received Amendment No. 8 to its facility license for the conversion on November 13, 1967.

UWNR currently operates at 1.0 MW steady-state power with pulsing capability using reactivity insertions up to 1.4% $\Delta k/k$. A renewal would authorize continued operation of the TRIGA-type research reactor facility, located on the campus of the university in a building located at 1513 University Avenue, Madison, Wisconsin.

The regulations in 10 CFR 50.64, "Conditions of Licenses," require licensees of research and test reactors to convert from the use of high-enriched uranium (HEU) fuel to low-enriched uranium (LEU) fuel, unless specifically exempted. In 2008, the licensee proposed to convert the fuel in the UWNR from HEU to LEU. In a letter dated August 25, 2008 (Ref 10), the licensee requested approval of the fuel conversion and of changes in the technical specifications (TS). To support this action, the licensee submitted the "Safety Analysis Report for the Conversion of the University of Wisconsin-Madison TRIGA Reactor from HEU to LEU Fuel," dated August 25, 2008 (LEU conversion SAR (Ref 10)), on which it based the HEU to LEU conversion and the TS changes. The licensee supplemented the LEU Conversion SAR on October 17, 2008 (Ref 9), April 10, 2009 (Ref 11), May 1, 2009 (Ref 12), and June 8, 2009 (Ref 13).

The NRC issued the order for UWNR to convert on June 11, 2009 (Ref 5). The order included a conversion safety evaluation report (Conversion SER (Ref 5)), providing the results of the NRC staff's evaluation of the licensee's conversion proposal. The order also included changes to the TS that the NRC would require for operation of the facility with LEU. The order also required the licensee to submit a startup report to the NRC within 6 months of the completion of the conversion. The licensee submitted the startup report by letter dated June 21, 2010 (Ref 14).

The NRC staff's review, with respect to renewing the UW facility license, was conducted on the basis of information contained in the renewal application, as well as supporting supplements and license responses to request for additional information (RAIs). Specifically, the renewal application included the SAR (as supplemented), an environmental report, and the TS (as supplemented). Minor typographical changes were made by the NRC staff to the TS (Ref 29) following discussion and agreement by the licensee via telephone conversation on March 10, 2011. Several site visits were conducted at the facility to observe facility conditions. With the exception of the security plan and the emergency plan, material pertaining to this review may be examined or copied, for a fee, at the NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike, Rockville, MD. The NRC maintains an Agencywide Documents Access and Management System (ADAMS), which provides text and image files of NRC's public documents. Documents related to this license renewal may be accessed through the NRC's Public Electronic Reading Room 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 Public Document Room staff by telephone at 1-800-397-4209 or 301-415-4737, or send an e-mail to 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 emergency plan is withheld from public disclosure, as it is considered Security-Related Information. Parts of the SAR and RAI responses from the licensee contain Security-Related Information and are protected from public disclosure.

The dates and associated ADAMS accession numbers of the licensee's renewal application and associated supplements are listed in Chapter 12, "References."

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," Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," Part 50, "Domestic Licensing of Production and Utilization Facilities," Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," and Part 70, "Domestic Licensing of Special Nuclear Material;" the recommendations of applicable regulatory guides; and relevant accepted industry standards, such as the American National Standards Institute/American Nuclear Society (ANSI/ANS) 15 series. The NRC staff also referred to the recommendations contained in NUREG-1537, "Guidance for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," (Ref 2). 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 22), the NRC staff provided the Commission with information regarding plans to revise the review of license renewal applications for research and test reactors. The Commission issued its staff requirements memorandum (SRM) for

SECY-08-0161, dated March 26, 2009 (Ref 23). The SRM directed the NRC staff to streamline the renewal process for such reactors, using some combination of the options presented in SECY-08-0161. The SRM also directs the NRC staff to implement a graded approach whose scope is commensurate with the risk posed by each facility. The graded approach incorporates elements of the alternative safety review approach discussed in Enclosure 1 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 interim staff guidance (ISG) to assist in the review of license renewal applications. The streamlined review process is a graded approach based on licensed power level. Under the streamlined review process, the facilities are divided into two tiers. Facilities with licensed power level of 2 megawatts thermal (MW(t)) and greater would undergo a full review using NUREG-1537. Facilities with a licensed power level less than 2 MW(t) would 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. Specifically, for reactors with licensed power levels less than 2 MW(t), the review will focus on the sections of the safety analysis report that are most significant to safety under a focused license renewal approach. The NRC 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 conducted the UWNR license renewal application review using the final ISG (Ref 6) and since the licensed power level for the UWNR is less than 2 MW(t), the NRC staff performed a focused review on the licensee's application for license renewal. Specifically, the NRC focused on reactor design and operation, accident analysis, TS, radiation protection, waste management programs, financial requirements, environmental assessment, and changes to facility after submitting application.

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

The UWNR facility physical security plan was approved by the NRC in June 1991. The licensee maintains a program for providing for the physical protection of the facility and its special nuclear material 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 licensee stated, in response to RAI No. 3 in a letter dated June 16, 2010 (Ref 20), that the UWNR security plan required no further changes as a result of the license renewal.

The UWNR Emergency Plan was approved by the NRC in July 1984. The licensee maintains an emergency plan 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 licensee created the current revision of the emergency plan during the UWNR conversion to LEU fuel, and the NRC staff finds it to be acceptable.

The licensee identified a standard procedure (UWNR 004, "University of Wisconsin Nuclear Reactor Operator Proficiency Maintenance Program") for its requalification program as part of its application for license renewal. Because the licensee did not make changes to the approved requalification plan, and in accordance with the guidance provided in the ISG, the NRC staff has determined that the requalification plan will not be reviewed as part of this license renewal.

The purpose of this SER is to summarize the findings of the safety review of the UWNR and to delineate the technical details considered in evaluating the radiological safety aspects for continued operation. This SER provides the basis for renewing the license for operation of the UWNR at steady-state thermal power levels up to and including 1.0 MW and short duration power pulses with reactivity insertions up to 1.4% $\Delta k/k$. The pulses were calculated not to raise the fuel temperature at the hottest core location above 830 degrees Celsius (C).

This SER was prepared by Christian B. Cowdrey, Project Manager from the NRC's Office of Nuclear Reactor Regulation (NRR), Division of Policy and Rulemaking (DPR), Research and Test Reactors (RTR) Projects Branch, Geoffrey A. Wertz, Project Manager from the NRC's NRR/DPR, RTR Licensing Branch, Jo Ann Simpson, Financial Analyst from the NRC's NRR/DPR, Financial Analyst Branch. Brookhaven National Laboratory, the NRC's contractor, provided substantial input to this SER.

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 UWNR, the NRC staff made the following findings:

- The design and use of the reactor structures, systems, and components important to safety during normal operation discussed in Chapter 4 of the 2008 SAR (Ref 9), 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 a maximum hypothetical accident (MHA), emphasizing those that could lead to a loss of integrity of fuel element cladding and a release of fission products. The licensee performed analyses of the most serious credible accidents and the MHA and determined that the calculated potential radiation doses outside the reactor room would not exceed doses in 10 CFR Part 20 for unrestricted areas.

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

- The systems provided for the control of radiological effluents, when operated in accordance with the TS, are adequate to ensure that releases of radioactive materials from the facility are within the limits of the Commission's regulations and are as low as 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 2008 SAR (Ref 9) and the LEU conversion SAR (Ref 10), and the TS will continue to 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 special nuclear material 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) and therefore, according to the licensee, they will not decrease the effectiveness of the plan.

- 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. The licensee created the current revision of the emergency plan during the UWNR conversion to LEU fuel, and the NRC staff found it to be acceptable.

On the basis of these findings, the NRC staff concludes that UW can continue to operate the UWNR in accordance with the Atomic Energy Act of 1954, as amended (AEA), NRC regulations 10 CFR, Chapter 1, and Renewed Facility License No. R-74 without endangering public health and safety.

1.3 General Description of the Facility

The UWNR is located within the city of Madison on the University of Wisconsin, Madison Campus, in the Mechanical Engineering Building. The building also contains classrooms, laboratories, shops, and staff offices for the departments of mechanical engineering, industrial engineering, and engineering physics. This license, R-74, authorizes UWNR to operate at steady-state power levels up to 1.0 MW with pulsing capability using reactivity insertions up to 1.4% $\Delta k/k$. On June 11, 2009, the NRC amended the reactor license to permit conversion of the reactor from HEU to LEU fuel.

The reactor is a heterogeneous pool-type nuclear reactor currently fueled with TRIGA 30/20 fuel in four-element fuel assemblies. The coolant is light water, which circulates through the core by natural convection. The core is reflected by water and graphite. The maximum steady-state power level is 1.0 MW. The 30/20 fuel is 30-percent uranium (U), enriched to 19.75 percent. Water is used as the coolant and moderator. The primary coolant system consists of a 27.5-foot deep, aluminum-lined concrete pool in which the reactor core is submerged under a nominal 19-feet (ft) column of water. Heat generated from the reactor core is directly transferred to the pool water by natural convection. Reactor pool water is kept below 54.4 degrees C (130 degrees Fahrenheit (F)) by a closed-loop cooling system with a flow rate of approximately 500 gallons per minute (gpm). A pump takes water from a pipe connected to the reactor pool, passes it through a primary heat exchanger, and returns it through a pipe to the reactor pool. The primary heat exchanger provides the heat removal capability for the water in the reactor pool. Heat is removed from the primary heat exchanger by a second, intermediate, closed-loop heat exchanger operating at a higher pressure than the primary heat exchanger. An intermediate pump circulates water in this closed loop. Heat is transferred to water from the campus chilled water system, which is pumped through the intermediate heat exchanger. Both heat exchangers have a flat-plate design. Design features of this system allow the transfer of reactor heat from the primary system under all operating conditions, but it is only required for core heat removal during power operations. These systems are controlled remotely from the control room.

The reactor's experimental facilities include space adjacent to the reactor core, a pneumatic transfer system, beam tubes, and a thermal column. Four control blades, three safety and one regulating, using boron carbide (B_4C) as the neutron absorber, are moved in and out of the reactor core by individual mechanical drives or can be disengaged to drop by gravity into the core for safety purposes. An additional transient control rod with a combination pneumatic-electromechanical drive may be used either as a normal control rod or as a transient rod used to generate a neutron pulse in pulsing mode operation.

1.4 Shared Facilities and Equipment

The UWNR is a separate room within the Mechanical Engineering Building that contains minimal penetrations. Shared facilities include adjacent laboratory spaces, chilled water, instrument air, emergency alternating current (ac) power, and the water waste system. Potentially contaminated drains from laboratories, the reactor equipment rooms, and the reactor pool room flow into a common collection system. This system discharges into a holdup tank that is continuously monitored and sampled before release into the campus sanitary system. Adjoining rooms in the building contain offices for reactor program personnel and some laboratories.

A separate circuit provides power to the reactor facility, with circuit breakers independent from the electrical power supplying the rest of the Mechanical Engineering Building. The main power transformer serving the Mechanical Engineering Building and its main breaker are the only electrical common points. There is an emergency diesel generator capable of supplying power to the entire Mechanical Engineering Building, located in the basement. It starts automatically on undervoltage and transfers ac power to the building's various designated emergency buses, some of which supply the UNWR. The emergency generator and a dedicated uninterruptable power supply (UPS) can provide power to the control console; heating, ventilating, and air conditioning (HVAC) system; radiation monitors; and other reactor-related systems, if normal campus-supplied power is lost.

The campus chilled water system provides the ultimate heat sink for the reactor coolant system, which is only needed when the reactor is at power. The campus chilled water system serves no safety function to UWNR. There is also a common building air compressor that supplies 90 pounds per square inch of instrument air. This air powers air-operated valves and solenoids on UWNR systems.

Air from the reactor facility is exhausted through an independent ventilation system that is monitored continuously and routed through a high-efficiency particulate air (HEPA) filter during normal operation or an emergency. It is also diluted with outside air to a stack extending from the roof of the building. The raised elevation of the stack provides additional atmospheric dilution.

1.5 Comparison with Similar Facilities

The TRIGA-type nuclear reactor built by General Atomics (GA) is one of the most widely used research and training reactors in the United States. TRIGA reactors exist in a variety of configurations and capabilities. The UWNR is very similar in design to TRIGA reactor facilities at Texas A&M University and Washington State University. Instruments and controls used in the UWNR facility are similar to most non-power reactors licensed by the NRC. The pool size and experimental facility configuration differ on the three reactors, but basic reactor behavior and accident analyses are similar.

The reactors at Texas A&M University and Washington State University converted their fuel to the same LEU fuel elements used in the UWNR reactor. The TRIGA Mark III reactor at GA in San Diego also operated with a core partially made up of high-density LEU fuel before it was permanently shut down. There have been no performance issues with the use of this fuel in these reactors.

1.6 Summary of Operations

The UWNR facility is used for nuclear research and teaching, as well as a range of irradiation services. Use of the reactor facility over the past 20-year period has been high. Since 1967, it has accumulated 20,000 MW hours of power operation without a fuel element failure. Expectations for the upcoming license renewal period are to at least maintain or improve the high usage rate achieved.

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 email sent from James Wade of DOE to Paul Doyle (NRC) (Ref 25) reconfirms this obligation with respect to the fuel at UWNR (DOE Contract No. 9722, valid from November 1, 2001 – November 1, 2011. By entering into such a contract with DOE, UW has satisfied the applicable requirements of the Nuclear Waste Policy Act of 1982.

1.8 Facility Modifications and History

Review of the modifications made up to the year 2000 involved minor technological changes, inclusion of security, training, and emergency plans, and changes to the amount and type of special nuclear material used in connection with the existing operating license. Review of the license amendments showed that they were mostly administrative in nature and did not have any effect on the operating characteristics of the reactor. All of these modifications were subject to an evaluation in 10 CFR 50.59, “Changes, tests, and experiments,” to ensure there was no impact on the safety of the UWNR facility.

On May 9, 2000 (Ref 24), UW submitted a new SAR as part of the license renewal application. The SAR contained two changes that the UWNR Reactor Safety Committee (RSC) approved were based on 10 CFR 50.59 evaluations. The changes eliminated two reactor trips that were not applicable, because they were not TS-required and did not provide protection for a pulsing TRIGA reactor.

By letter dated September 7, 2004 (Ref 30), the licensee submitted Revision 1 to the UWNR SAR to reflect installation of the new reactor coolant system in 2003. Revision 1 was not reviewed because, from 2004 through 2007, the Mechanical Engineering Building was completely rebuilt and renovated, and other reactor system modifications were planned during this period (Ref 9). Revision 2 of the SAR, which UW submitted via letter on October 17, 2008 (Ref 9), included all facility modifications made during this period. Although all modifications received a 10 CFR 50.59 review by the licensee's RSC before implementation, the NRC staff reviewed these modifications and the evaluation is included in this SER. Specifically, review of the reactor coolant system modification is documented in Section 3.1, the new pneumatic tube transfer system is documented in Section 2.1.3, the instrumentation upgrades to the reactor control console is document in Sections 4.1 through 4.7, the emergency electrical power system is documented in Section 5.2, and the new ventilation system is documented in Section 6.1. One additional modification has been implemented since the 2008 SAR (Ref 9). The regulating blade was replaced with a similar blade, but containing a welded stainless steel connecting rod in-lieu of the original riveted aluminum rod. The licensee performed a 10 CFR 50.59 review as required.

The UWNR is a teaching and research facility which was initially licensed in 1960 at a maximum power level of 10 kilowatts (kW). After a license amendment, dated October 2, 1964, the UWNR increased its power level to 250 kW on December 7, 1964, using the original flat-plate aluminum clad fuel. Operations with the original core ended October 1, 1967, after 2,268.5 critical hours and 105.65 MW hours of core exposure. UW installed a cooling system and converted the reactor to a 1,000 kW TRIGA reactor with pulsing capability in 1967. UW received Amendment No. 8 to its facility operating license for the conversion on November 13, 1967.

On June 11, 2009 (Ref 5), the NRC issued an order modifying the UWNR operating license to allow conversion from HEU to LEU fuel (Ref 5). The reactor is currently operating with the TRIGA LEU conversion fuel, enriched to less than 20 percent.

2 REACTOR DESCRIPTION

2.1 Summary Description

2.1.1 Introduction

The UWNR TRIGA reactor was originally designed using materials test reactor (MTR) plate-type fuel assemblies. GA developed a fuel system with fuel assemblies that can hold up to four TRIGA fuel elements each to replace the MTR plate-type fuel assemblies.

The UWNR reactor normally operates in steady-state mode at a maximum power level of 1.0 MW and is cooled by natural convection. It can also operate in the pulsed mode with a reactivity insertion not to exceed 1.4% Δk/k initiated from a steady-state power level not to exceed 1 kW. UW recently replaced the highly-enriched fuel, (70 percent U-235) TRIGA fuel lifetime improvement program (FLIP), with an LEU fuel that has a nominal enrichment of 19.75 U-235 in a fuel element with the same geometry and cladding as the FLIP fuel. The HEU-to-LEU fuel conversion only affected the fuel type, the core configuration, and the TS for the core operating limits. This conversion to LEU fuel did not result in any other changes to the UWNR facility.

During the conversion, UW inserted four additional graphite reflectors in the core to reduce the number of fuel bundles to satisfy the TS on core reactivity limitations and to increase the expected core lifetime to 1,800 MW-days (MWd). The NRC staff previously reviewed the impact of this modification during the conversion and found it acceptable (Ref 5, Section 2.4).

In addition, to reduce the radiation dose to employees, the licensee replaced the fuel-assembly hardware and transient rod guide tube during the LEU conversion. The NRC staff reviewed the impact of these changes on the operation and safety of the LEU core and determined it to be acceptable (Ref 5, Section 2.1).

2.1.2 Summary of Reactor Data

Table 2-1 below contains a summary of pertinent thermal-hydraulic and neutronic design data for the UWNR LEU core.

Table 2-1 UWNR LEU Core Thermal-Hydraulic Design Data

Reactor Parameters for the LEU Core	
Licensed Reactor Power (MW)	1.0
TS Safety Limit on Maximum Power (MW)	1.5
Number of Fuel Elements in Core	83
Number of Control Elements in Core	5
Maximum Fuel Temperature at 1.0 MW (ºC)	491
Maximum Fuel Temperature at 1.3 MW (ºC)	596
Fuel Temperature Coefficient (%Δk/k / K x 10^3)	-5.55×10^{-3}
Prompt Negative Temperature Coefficient of Reactivity (Δk/k/K x 10^3), 350 K–1,000 K	3.05 to 13.34
Coolant Void Coefficient (Δk/k/% void x 10^3)	-1.25 to -1.49
Coolant Temperature Coefficient (Δk/k / K x 10^4)	0.82 to 1.01

Maximum Rod Power at 1 MW (kW)	19.36
Power Peaking Factors (minimum, maximum)	0.46, 1.61
Average Rod Power at 1 MW (kW)	12.0
Maximum Rod Power at 1.3 MW (kW)	27.2
Average Rod Power at 1.3 MW (kW)	10.9
Maximum Rod Power at DNB (kW)	51.7
DNBR at Licensed Operating Power	1.52
Prompt Neutron Lifetime (µs)	27.1µs to 29.4µs
Effective Delayed Neutron Fraction	0.0076 to 0.0078
Safety Parameters for the LEU Core	
IFE LSSS (ºC), D4-SW and E3-NE	500 and 400
Reactor Power LSSS (MW)	1.25
Minimum DNBR (at 1.0 MW)	1.52
Minimum DNBR (at 1.3 MW)	1.33
LCO on Maximum Pulsed Reactivity Insertion (%Δk/k)	1.4
Calculated Pulsed Reactivity Insertion to Reach T = 830ºC (%Δk/k)	1.5 to 1.6
Peak Pulsed Fuel Temperature (ºC)	830

2.1.3 Experimental Facilities

Experimental facilities at the UWNR consist of four radial beam ports, a thermal column, a vertical hydraulic irradiation tube (whale system), and a horizontal pneumatic tube (rabbit system). The licensee replaced the pneumatic tube system in 2004 through 2007, when it extensively modified the Mechanical Engineering Building.

The new 2.25-inch (in.)-diameter steel pneumatic tube system transports polyethylene cylinders (rabbits), samples 1.25 in. in diameter and 5.5 in. long, not to exceed 12 ounces in weight, in and out of the reactor pool adjacent to the core for short-duration irradiations. Both ends of the tube terminate in a fume hood with a HEPA filter located in Room B1135C with a dedicated exhaust fan (EF-13). Room B1135B contains system controls and the blower system. A duplicate set of controls and indications reside in the reactor control room. The system can only be started from the control room. This configuration ensures the reactor operator is always aware of the system operating status. Additionally, an interlock prevents the pneumatic system from being placed in operation without starting the fume hood exhaust fan. This interlock ensures that any residual argon (Ar)-41 in the tube will be discharged up the stack rather than leaking out of the fume hood, creating an airborne radiation hazard in the room. Additionally, having the tube terminate in a fume hood prevents the spread of contamination, should a sample become damaged.

Pressurized carbon dioxide (CO_2) gas provides the motive force for rabbit movement through the tube. Using CO_2 instead of compressed air reduces the production and discharge of Ar-41. Air-operated ball valves installed adjacent to the biological shield at each tube penetration are closed when the system is not in operation. The air-operated ball valves prevent air from leaking into the tube, which would result in greater Ar-41 production. Since the pneumatic tube penetrates the reactor tank below the water surface, a leak in the tubing could drain the reactor pool. These valves also prevent the total loss of pool water, if a tube leak inside the pool were to occur. On loss of air or power, these valves fail closed. An auxiliary compressor powered from an emergency bus can be started to supply air to operate the pneumatic system valves, if retrieval of a rabbit capsule becomes necessary.

In the unlikely case of a tube break occurring outside the biological shield concurrent with a tube leak inside the pool and one of the air-operated ball valves failing open, the system is designed to prevent draining more than 8 ft of water from the pool. The elevation of the tubes, combined with the check-valve-protected solenoid valve siphon breakers on the system's high points, limit the amount of water that could be drained from the pool.

The pneumatic tube system, as described in the 2008 SAR (Ref 9) is designed to minimize airborne Ar-41 hazards in the facility, contains instrumentation adequate to operate the system in a safe manner, and provides for proper diagnosis of potential abnormal conditions. The ability to start the system from the control room helps ensure the system will not degrade the safety systems at the UWNR. The NRC staff finds the new design acceptable.

2.2 Reactor Core

TS 1.3 defines reactor core elements as follows:

TS 1.3 Definitions

FUEL ELEMENT:
A fuel element is a single TRIGA fuel rod of LEU 30/20 type.

FUEL BUNDLE:
A fuel bundle is a cluster of three or four fuel elements secured in a square array by a top handle and a bottom grid plate adaptor.

CORE LATTICE POSITION:
The core lattice position is that region in the core (approximately 3" x 3") over a grid hole. It may be occupied by a fuel bundle, an experiment or experimental facility, or a reflector element.

INSTRUMENTED ELEMENT:
An instrumented element is a special fuel element in which thermocouples are embedded for the purpose of measuring fuel temperatures during reactor operation.

LEU 30/20 CORE:
A LEU 30/20 core is an arrangement of TRIGA LEU 30/20 fuel in the reactor grid plate.

OPERATIONAL CORE:
An operational core is an LEU 30/20 core for which the core parameters of shutdown margin, fuel temperature, power calibration, and maximum allowable pulse reactivity insertion have been determined to satisfy the requirements of the Technical Specifications.

These are standard definitions used in research reactor TS and are therefore acceptable to the NRC staff.

The LEU core consists of 21 fuel assemblies arranged in the same pattern as the previous HEU core. One of the central assemblies of the core contains a transient control rod in replacement of one fuel element for pulsed operation. This results in a total of 83 fuel elements within the cores, two of which have internal thermocouples and are referred to as instrumented fuel elements (IFEs). Table 2-2 provides a list of the major components of the UWNR LEU core. 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 Core Components for the LEU Core

Core Configuration	LEU 30/20
Non-instrumented Fuel Elements	81
Instrumented Fuel Elements	2
Transient Rod (water followed)	1
Aluminum Clad Reflector Elements	14
Shim-Safety Blades	3
Stainless Steel Regulating Blade	1

An operational reactor core is defined in TS 1.3 in Section 2.1of this SER. TS 5.4 presents the design feature requirements for the UWNR core as follows:

<u>TS 5.4 Reactor Core</u>

1. The core shall be an arrangement of TRIGA LEU 30/20 uranium-zirconium hydride fuel-moderator bundles positioned in the reactor grid plate.

2. The reactor shall not be operated with a core lattice position vacant except for positions on the periphery of the core assembly.

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

TS 5.4 (1) defines the reactor core as consist of standard TRIGA LEU fuel elements.

TS 5.4 (2) ensures that internal core lattice positions are occupied by standard fuel elements to controlling power peaking in fuel elements and reducing the probability of an accidental reactivity insertion.

TS 5.4 (3) places a limitation on reflector elements to prevent reactivity excursions.

TRIGA cores have been in use for years, and their characteristics are well documented. GA and Texas A&M University have also operated reactors with LEU cores including 30/20 fuel. The UW analysis indicates that the LEU 30/20 core safely satisfies all operational requirements. Vacant core lattice positions will contain experiments or an experimental facility to prevent accidental fuel additions to the reactor core. They will be permitted only on the periphery of the core to prevent power perturbations in regions of high-power density.

The UWNR uses solid fuel elements in which the zirconium hydride moderator is homogenously combined with low-enrichment fuel (U-ZrH_x fuel). The fuel rods are in a three- or four-rod cluster configuration. The fuel clusters are contained within an assembly or bundle, which resembles the outline of the original MTR fuel element and allows the three- or four-rod clusters to be plugged into a grid plate. The assembly is supported in a grid box suspended from a movable bridge by four flooding corner posts that form a suspension frame. The grid plate in the grid box provides a 7x9 array of square holes for fuel bundles. The grid plate also accepts reflector elements, experiment devices, and a neutron source holder. Two shrouds in which three standard control elements (shim-safety blades) and one regulating control element (servo blade) move in and out of the core are permanently attached in the grid box.

Reactor core components are positioned on the grid plate in the grid box. The UWNR uses five control elements: three Boral shim-safety blades, one stainless steel regulating blade, and one aluminum-clad boron-carbide transient pulsing rod. The transient rod moves in the fourth position of a three-rod cluster fuel assembly. A number of instrument channels monitor the reactor neutron flux and power level with neutron sensors located in the corner posts of the suspension frame.

The primary reactor water is de-ionized and routinely monitored both for quality and to identify any significant radioactivity increase. The reactor core is cooled by natural convection of this water, which also serves as reflector, moderator, and shield. A diffuser system reduces the radiation exposure level from nitrogen (N)-16 on the bridge. Using a closed-loop system, water is taken from the pool with a pump and discharged down directly toward the top of the reactor core. The net effect is an increase in the time that it takes the N-16 to reach the pool surface, allowing decay time for the 7.4 second half-life of N-16.

The licensee has proposed changes to the bases to reflect the design features of the LEU fuel (Ref 10). Because they reflect the conversion to LEU fuel, the NRC staff finds these proposed changes to TS 5.4 acceptable for the current LEU core, based on the LEU conversion SER (Ref 5, Section 2.0).

TS 1.3 defines reactor core parameters as follows:

TS 1.3 Definitions

EXCESS REACTIVITY:

Excess reactivity is that amount of reactivity that would exist if all control elements were fully withdrawn from the core from the point the reactor is exactly critical (k_{eff}=1) in the reference core condition.

COLD CRITICAL:

The reactor is in the cold critical condition when it is critical in the reference core condition.

REFERENCE CORE CONDITION:

The reactor is in the reference core condition when the fuel and bulk water temperatures are at ambient conditions and the reactor is xenon free.

These are standard definitions used in research reactor TS, and the NRC staff therefore finds them acceptable.

TS 3.1 presents the limiting conditions for operation (LCO) and surveillance requirements for the control of the configuration and operation of the reactor core parameters, reactor control and safety systems, and in-core experiments. They apply to all modes of operation. The objective is to ensure that the reactor can be shut down at all times and to ensure that the fuel temperature safety limit will not be exceeded.

TS 3.1 Reactor Core Parameters

TS 3.1.1 Excess Reactivity

The excess reactivity shall not exceed 5.6% Δk/k.

TS 3.1.1 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 design.

TS 3.1.2 Shutdown Margin

The reactor shall not be operated unless the shutdown margin provided by control rods shall be greater than 0.2% Δk/k with:

1. the highest worth non-secured experiment in its most reactive state.

2. the highest worth control element and the regulating blade (if not scrammable) fully withdrawn, and

3. the reactor in the reference core condition.

TS 3.1.2(1)–(3) define the shutdown margin, ensuring that the reactor can be shut down by an acceptable margin.

TS 3.1.2(1) places constraints on the core condition by considering the highest worth non-secured experiment to be in its most reactive state, to ensure that the reactor remains subcritical, should a nonsecured experiment move to its most reactive position.

TS 3.1.2(2) ensures that the reactor can be shut down even if one of the control rods becomes stuck out of the reactor core.

TS 3.1.2(3) establishes the core reference conditions for deriving the shutdown margin. 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 shutdown margin.

TS 3.1.3 Pulse Limits

1. The reactivity to be inserted for pulse operation shall be determined and mechanically limited such that the reactivity insertion will not exceed 1.4% $\Delta k/k$.

2. Pulses shall not be initiated at power levels exceeding 1 kilowatt.

TS 3.1.3(1) establishes the maximum reactivity addition for pulsing, ensuring that the reactor can be safely pulsed without fuel damage. Pulse reactivity is limited to ensure that the fuel temperature stays below the pulse safety limit of 830 degrees C.

TS 3.1.3(2) places a limit on the initiation of a pulse to prevent a fuel temperature increase above the pulse temperature limit of 830 degrees C. (Even though a pulse at a lower power will produce a higher power peak and release more energy than a pulse at higher powers, the initial fuel temperature is lower and the resulting peak fuel temperature is below the 830 degrees C pulse temperature limit during the pulse transient.)

TS 3.1.4 Core Configuration

1. The core shall be an arrangement of TRIGA LEU 30/20 uranium-zirconium hydride fuel-moderator bundles positioned in the reactor grid plate.

2. The reactor shall not be operated with a core lattice position vacant except for positions on the periphery of the core assembly.

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

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

5. Control elements shall not be manually removed from the core unless the core has been shown to be subcritical with all control elements in the full out position.

TS 3.1.4(1) controls a design criterion by requiring the use of only TRIGA LEU 30/20 fuel elements.

TS 3.1.4(2) ensures that all core lattice positions except on the periphery are filled with fuel controlling water holes in the reactor core. The purpose of controlling water holes is to control power peaking in fuel elements.

TS 3.1.4(3) specifies the reflector locations as filled with water or a combination of water and graphite, except experiments or experimental facilities.

TS 3.1.4(4) ensures that the reactor core remains subcritical during fuel movement, preventing inadvertent reactivity accident scenarios caused by fuel assembly movements (see Section 8).

TS 3.1.4(5) ensures that the reactor core remains subcritical during the removal of control rods, preventing inadvertent reactivity accident scenarios (see Section 8).

TS 4.1 presents the surveillance specifications for the reactor core parameters. These TS apply to the reactivity condition of the reactor and the reactivity worths of control rods. They apply to all modes of operation. The objective is to ensure that the reactor can be shut down at all times and to ensure that the fuel temperature safety limit will not be exceeded.

<u>TS 4.1 Reactor Core Parameters</u>

1. Excess reactivity

   Excess reactivity shall be determined at least annually and after changes in either the core, in-core experiments, or control elements for which the predicted change in reactivity exceeds the absolute value of the specified shutdown margin.

2. Shutdown margin

   The shutdown margin shall be determined at least annually and after changes in either the core, in-core experiments, or control elements.

3. Pulse limits

   The reactor shall be pulsed semiannually to compare fuel temperature measurements and peak power levels with those of previous pulses of the same reactivity value.

4. Core configuration

   Each planned change in core configuration shall be determined to meet the requirements of Sections 3.1(4) and 5.3 of these specifications before the core is loaded.

TS 4.1(1)–(4) are consistent with the definitions referenced in NUREG-1537, Chapter 14, “Technical Specifications,” and ANSI/ANS-15.1-2007, “The Development of Technical Specifications for Research Reactors.” Operational experience has shown that these surveillance intervals are acceptable to detect changes in core behavior. The NRC staff concludes that these surveillances will help ensure that these core parameters are within their TS limits.

Before the conversion to LEU, the NRC staff evaluated the fuel and core design for the mixed LEU core, including physical and chemical composition, the nuclear design, calculation methodology, core parameters, criticality, kinetic parameters, reactivity coefficients, neutronic behavior, thermal behavior, effects of burnup and temperature, power peaking, and pulsing. The NRC staff finds that the changes in nuclear design of the core, represented in part by the TS described above, are acceptable for the current LEU core, based on the conversion analyses (Ref 5, Section 2.3).

The TS related to the normal operating conditions of the reactor core include limits on excess reactivity, minimum shutdown margin, allowable core configurations, and surveillance requirements for the fuel parameters and reactivity worth of the control rods. These TS are consistent with NUREG-1537 and ANSI/ANS-15.1-2007. The NRC staff finds that the analyses presented in the 2008 SAR (Ref 9) and the LEU conversion SAR (Ref 10) adequately justify these TS and show that normal reactor operation will not lead to the release of fission products from the fuel. Based on these considerations, the NRC staff concludes that the licensee has adequately analyzed the expected normal reactor operation during the period of the renewed facility operating license. The NRC staff further concludes that the TS provide reasonable assurance that normal operation of the UWNR core will not pose a significant risk to public health and safety or the environment.

2.2.1 Reactor Fuel

These design specifications apply to the fuel elements used in the reactor core and specify the design requirements for the fuel. The objectives are to ensure 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. The reactor is fueled with TRIGA LEU 30/20 fuel rods, as indicated in Table 2-3 below.

Table 2-3 Description of TRIGA LEU Fuel Elements

TRIGA 30/20 LEU Fuel	
Uranium content	30 wt-%
Enrichment, nominal U-235	19.75 %
Erbium content	0.9 wt-%
Diameter of fuel meat	34.823 mm
Length of fuel meat	381 mm
Cladding	304 stainless steel
Cladding thickness, nominal	0.508 mm

The design features of the fuel are given in TS 5.3 as follows:

<u>TS 5.3 Reactor Fuel</u>: <u>TRIGA LEU 30/20 Fuel</u>

The individual unirradiated TRIGA LEU 30/20 fuel elements shall have the following characteristics:

1. Uranium content: maximum of 30 Wt-% enriched to maximum of 19.95 Wt-% with nominal enrichment of 19.75 Wt-% Uranium 235.

2. Hydrogen-to-zirconium atom ratio (in the ZrH_x): nominal 1.6 H atoms to 1.0 Zr atoms with a maximum H to Zr ratio of 1.65.

3. Natural erbium content (homogeneously distributed): nominal 0.9 Wt-%.

4. Cladding: 304 stainless steel, nominal 0.020 inch thick.

TS 5.3(1) specifies the maximum uranium enrichment of 19.95 percent for the TRIGA LEU 30/20 fuel. Although this is greater than the design value of 19.75 percent enrichment by about 1 percent, this increase in the uranium enrichment should only increase the power density by about 1 percent.

TS 5.3(2) specifies the maximum Hydrogen to Zirconium (H-to-Zr) ratio as 1.65. 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 1.65, along with the conservative safety limit of 1,150 degrees C, the pressure is at least a factor of 5 lower than would be necessary for clad failure. Because the maximum value of the H-to-Zr ratio is adequate to account for uncertainties in clad strength and manufacturing tolerances, the NRC staff finds TS 5.3.2 acceptable.

TS 5.3(3) specifies a variation of erbium content for a single fuel element that could increase the fuel element power density by about 1-2 percent. The licensee stated that such an increase in local power density would only reduce the safety margin by less than 2 percent. The NRC staff finds that because the bases reflect the design features of the LEU fuel, TS 5.3(3) is acceptable.

TS 5.3(1)–(4) control important aspects of the design of the LEU fuel. The licensee rewrote TS 5.3(1)–(4) as a result of the LEU conversion (Ref 10) and the NRC staff previously approved these TS (Ref 5). On this basis, the NRC staff finds TS 5.3 acceptable.

TS 3.1.6 Fuel Parameters

The reactor shall not be operated with damaged fuel except for purposes of identifying the damaged fuel. A fuel element shall be considered damaged and must be removed from the core if:

1. In measuring the transverse bend, its sagitta exceeds 0.125 inch over the length of the cladding;

2. In measuring the elongation, the length of the cladding exceeds its original length by 0.125 inch;

3. A clad defect exists as indicated by detection of release of fission products.

4. The fuel has not been visually inspected within the previous 15 months.

5. The burnup of uranium-235 in the U-ZrH fuel matrix exceeds 50 percent of the initial concentration.

TS 3.1.6(1)–(5) establish inspection requirements for the fuel, to detect gross failure or visual deterioration. Attributes inspected include the fuel element transverse bend and length, and a visual inspection is conducted for bulges or other cladding defects. The limits on transverse bend and length are based on values from GA, the reactor designer.

TS 4.1(5) describes the surveillance requirements for the fuel as follows:

TS 4.1 Reactor Core Parameters

5. Fuel Parameters

   a. All fuel elements shall be inspected visually for damage or deterioration annually.

   b. Uninstrumented fuel elements which have been resident in the core during the previous year shall be measured for length and sagitta annually. Fuel elements shall not be added to a core unless a measurement of length and sagitta has been completed within the previous fifteen months.

   c. Fuel elements in the hottest assumed location, as well as representative elements in each of the rows, shall be measured for possible damage in the event there is indication that the Limiting Safety System Setting may have been exceeded.

TS 4.1 (5)(a)–(c) specify the surveillance requirements for the UWNR LEU fuel. The surveillance frequencies are consistent with NUREG-1537 and ANSI/ANS-15.1-2007 and other TRIGA facilities. The intervals have proven to be sufficient to ensure that fuel element integrity is maintained and any deterioration in cladding integrity can be detected.

Based upon a review of the information provided by the licensee, the NRC staff concludes that the licensee has adequately described the fuel elements used in the UWNR, including design limits, and the technological and safety-related bases for these limits. The licensee has discussed the constituents, materials, and components for the fuel elements in the LEU conversion SAR (Ref 10). Compliance with the TS limits will ensure uniform characteristics and compliance with design bases and safety-related requirements. There is reasonable assurance that the fuel will function safely in the UWNR core for the renewal period without adversely affecting public health and safety. The reactor fuel and its associated TS are, therefore, acceptable to the NRC staff.

2.2.2 Control Elements

Reactivity at the UWNR is controlled and altered by the positioning of three shim safety blades, a regulating blade, and a transient rod. TS 1.3 defines these control elements as follows:

TS 1.3 Definitions

SHIM-SAFETY BLADE:
A shim-safety blade is a control blade having an electric motor drive and scram capabilities. Its position may be varied manually or by the servo-controller.

TRANSIENT ROD:
The transient rod is a control rod with scram capabilities that can be rapidly ejected from the reactor core to produce a pulse. Its position may be varied

manually or by the servo-controller. It may have a voided or solid aluminum follower.

REGULATING BLADE:
The regulating blade is a low worth control blade that need not have scram capability. Its position may be varied manually or by the servo-controller.

The NRC staff reviewed these definitions and finds them to be acceptable and consistent with NUREG-1537 and ANSI/ANS-15.1-2007.

TS 5.5 describes the TS design requirements for the control rods as follows:

TS 5.5 Control Elements

1. The safety blades shall be constructed of boral plate and shall have scram capability.

2. The regulating blade shall be constructed of stainless steel.

3. The transient rod shall contain borated graphite or boron and its compounds in a solid form as a poison in an aluminum or stainless steel clad. The transient control rod shall have scram capability and may incorporate an aluminum or air follower.

These design specifications give the requirements for the control blades and transient control rod. The objective is to ensure that control elements are fabricated to reliably perform their intended control and safety function. TS 5.5(1)–(3) control the important aspects of the design of the control and transient rods to ensure they will perform their safety function and are, therefore, acceptable to the NRC staff.

The shim-safety elements are blade-type elements, which move in and out of the core in shrouds attached to the grid box. The poison section of the blade is Boral sheet 40.5 in. long, 10.5 in. wide, and 0.375 in. thick. It is clad with 0.125 in. aluminum. A reversible electric motor with an integral worm-gear drive mechanism is used to move the shim-safety blades in and out of the core at the appropriate speed. The shim-safety blades are coupled to the associated rod drive mechanism through an out-of-water electromagnet attached to the rod drive and a steel armature disk attached at the upper end of a long shaft attached to the shim-safety blade. Deenergizing the electromagnet allows the shim-safety blade to fall into the core by gravity. An underwater dashpot is positioned at the upper end of the shaft to act as a shock absorber to decelerate the last 5 in. of fall. Continuous indication of blade position is provided.

The regulating element is a blade-type element, which moves in and out of the core in a shroud attached to the grid box. The poison section of the blade is a stainless steel sheet about 11 in. wide and 40 in. long. The regulating blade is nonscrammable and is therefore directly attached to its drive mechanism. The regulating blade, or servo control element drive, is similar to the shim-safety drive, except that there is no scram capability and a servo motor and tachometer generator are used as the blade drive mechanism. The regulating blade may be moved manually by the operator to control core reactivity or automatically by the servo system. Continuous indication of blade position is provided.

The transient control element is a solid borated graphite cylinder contained in a 1.25-in. diameter stainless steel or aluminum tube. The poison section of the transient rod is 15 in. long. The transient rod moves in and out of the core in a guide tube inserted in the fourth position of a three-rod fuel cluster. A hold-down tube extends from the top of the guide tube up to the bottom of the transient rod drive. The transient rod drive employs a combination pneumatic-electromechanical drive assembly, which allows the rod to be used as a control rod or transient rod. The pneumatic portion is a single acting pneumatic cylinder mounted on the reactor bridge. A piston within the cylinder is attached to the poison end of the transient rod by means of a connecting rod. Compressed air from an accumulator mounted on the bridge is admitted at the lower end of the cylinder to drive the piston, and the transient rod, upward. A shock absorber decelerates the piston at a controlled rate during its final in. of travel. The resulting reactivity insertion is a function of the initial position of the cylinder. The electromechanical portion of the transient rod drive acts through a reversible electric motor and consists of a ball-nut drive assembly and the externally threaded air cylinder, which act as the screw in the ball-nut drive assembly. A system of limit switches indicates the position of piston and the transient rod.

The NRC staff reviewed the design and performance of the control elements and has determined the Boral safety blades and stainless steel regulating blade used in the UWNR reactor have demonstrated that they provide adequate reactivity worth, structural rigidity, and reliability, to ensure reliable operation and long life under all operating conditions. The scrammable control elements have further demonstrated the ability to scram without challenging the integrity of other reactor systems. The transient control rod materials and fabrication techniques 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 UWNR safety basis requirements, as stated in the 2008 SAR (Ref 9), and is, therefore, acceptable to the NRC staff.

TS 3.2 contains the following requirements for control rod operability:

TS 3.2 Reactor Control and Safety Systems

<u>TS 3.2.1 Operable Control Rods</u>

The reactor shall not be operated unless at least three control elements are operable and scrammable in accordance with TS 3.2.2.

<u>TS 3.2.2 Reactivity Insertion Rates (Scram time)</u>

The scram time measured from the instant a simulated signal reaches the value of the LSSS to the instant that the slowest scrammable control element reaches its fully inserted position shall not exceed 2 seconds.

TS 2.2 of the UWNR TS defines Limiting Safety System Settings (LSSS) as the scram setting which prevents the safety limit from being reached. The NRC reviewed TS 3.2 and determined that TS 3.2.1 ensures that the control rods will promptly shut down the reactor upon a scram signal. The UWNR will not be operated if three scrammable control elements are not operable and scrammable. The licensee also demonstrated that pin power and peaking factors remained bounded by the results of the LEU Conversion SAR (Ref 10), Section 4.7.7 in its response to the NRC RAI dated November 22, 2010. The licensee proposed to modify TS 3.2.1 by adding a reference to the TS 3.2.2 scram time, allowing it to define the operability requirement for a control element using scram times. TS 3.2.1 is acceptable to the NRC staff, because it supports the basic design requirements for control rod operability and the proposed modification is consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007.

TS 3.2.2 ensures that, during the normal operation of the UWNR, 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. Experience and analysis have 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 safety analysis assumed a 2 second value during the LEU conversion, which is common to other TRIGA reactors. Based on the discussion above, the NRC staff finds TS 3.2.2 acceptable as TS supports the basic design requirements to prevent reactor fuel damage.

TS 4.2 contains the following surveillance requirements for the control rods:

TS 4.2 Reactor Control and Safety Systems

1. Reactivity worth of control elements
   The reactivity worth of control elements shall be determined annually and following significant changes in core composition or arrangement that increase reactivity by a value greater than or equal to the shutdown margin.

2. Transient Rod and Associated Mechanism
   The transient rod drive cylinder and associated air supply system shall be inspected, cleaned, and lubricated as necessary annually.

3. Scram times of control and safety elements
   The scram time for all scrammable control elements shall be measured annually and following maintenance to the control elements or their drives.

TS 4.2(1)–(3) specify surveillance intervals to ensure the operability of the control rods. NRC staff finds these requirements are consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007 and therefore, is acceptable to the NRC staff.

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 reactor 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. A review of the design and functional description of the transient rod system offers reasonable assurance that pulses will be reproducible and limited to values that maintain fuel integrity. The control rod design for the UWNR includes reactivity worths that can control the excess reactivity planned for the UWNR, 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 UWNR control elements are acceptable.

2.2.3 Neutron Moderator and Reflector

The pool water serves as the moderator for the core and as the reflector above, below, and on the sides of the core not provided with graphite reflectors or other reflector elements. Individual fuel elements contain an internal 3.5-in. long graphite end reflector above and below the active fuel. This provides neutron reflection above and below the core by a mixture of graphite and water. The reflector elements remained unchanged during the HEU to LEU conversion; however, four reflectors were added in positions B5 and F5, in place of fuel bundles, and in positions D3 and D7 in place of irradiation baskets.

The four additional reflectors were inserted to increase core lifetime and reduce the number of fuel bundles to meet excess reactivity requirements. The NRC staff evaluated the calculational methodology and impacts on excess reactivity, control element worth, and shutdown margin as a result of the additional reflector modification. Based on the analysis performed for the LEU conversion (Ref 5, Section 2.4), the NRC staff finds the impact of these reflectors to be acceptable.

2.2.4 Neutron Startup Source

The neutron startup source is a 100-milligram (mg) radium-beryllium source, which provides a neutron flux greater than 10^7 neutrons/second. This is the same neutron startup source that was used in the HEU core. It is encapsulated in a 0.515-in.-diameter by 3.10-in.-long stainless steel welded cylindrical capsule, which in turn contains two 1.25-in.-long welded stainless steel capsules. The neutron source fits into a source holder, which, in turn, fits into a grid location adjacent to the core; it is usually left in the core during full-power operation.

One of the primary functions of a neutron source is to provide sufficient counts so that instrumentation will function properly during startup. TS 3.2.5 Interlocks, Table 3.2.5, contains operational interlocks, one of which is an interlock on control rod withdrawal if there is a count rate less than 2 counts per second (cps).

The UWNR uses a neutron source that is similar to the one used in other TRIGA reactors. Based on its review of the information provided by the licensee in the 2008 SAR (Ref 9), the NRC staff concludes that the source has sufficient strength to allow a controlled reactor startup and is therefore acceptable.

2.2.5 Core Support Structure

The conversion of the UWNR core from HEU to LEU fuel did not require any modifications to the core support structure (Ref 10). The core is suspended from an all-aluminum frame, which extends from the grid box to a height of about 1 ft above the pool surface. The reactor bridge (mounted over the pool) supports the core suspension frame. The 2008 SAR (Ref 9) describes in detail the system for grid position designation, the location of experimental facilities, and the radiation detector orientation. It also includes the letter and number codes used in later descriptions to identify fuel and reflector reactivity worths. An opening in the side of the header, 24 in. wide by 12 in. high, allows cooling water flow for natural convection. A diffuser pump and jet above the core deflect the cooling water streaming from the core to reduce N-16 activity above the core.

On the basis of its review of the 2008 SAR (Ref 9), the NRC staff concludes that the core support assembly accurately positions and aligns the fuel elements for all anticipated operating conditions. The arrangement ensures a stable and reproducible reactivity. The core support provides sufficient coolant flow. The NRC staff concludes that reasonable assurance exists that the core support structure will continue to operate as designed throughout the renewal period.

2.3 Reactor Pool

The reactor pool is a reinforced concrete, aluminum-lined, above-ground pool with a water volume of approximately 18,000 gallons. A thermal column and four beam tubes penetrate the pool. The pool water level is normally 20 ft above the top of the core. An alarm sounds if the water level drops or increases more than 2 in. from normal.

The TS define the reactor pool inventory requirements as follows:

TS 3.3 Reactor Pool Water Systems

1. A pool level alarm shall indicate loss of coolant if the pool level drops one foot or more below normal level.

TS 3.3(1) ensures that the reactor pool water level is maintained at 20 ft above the core to guarantee sufficient water for effective fuel cooling and to ensure that radiation levels at the top of the reactor are within acceptable levels. In response to RAI No. 14 in a letter dated June 16, 2010 (Ref 20), the licensee proposed to modify TS 3.3(1) to indicate that the pool level alarm will initiate when the water level drops “one foot or more below normal level” instead of “one foot or less.” The modification is acceptable to the NRC staff, since an alarm will be initiated at 2 in. below normal water level.

The reactor pool water level is continuously monitored with a water level monitoring device that provides a high-water-level alarm when the level is 2 in. above normal and a low-water alarm if the water level drops 2 in. below normal. This alarm is monitored in the reactor control room and, in off-hours, at the campus police station. If the water level drops more than 1 ft below normal, a scram is initiated.

Since the pool water level is continuously monitored and the frequency and quantity of makeup water is logged, pool water leakage will be detected. In the event of a catastrophic leak, the water would be collected in the drainage sump and holdup tank, where it would be analyzed before disposal. Based on the analysis above, the NRC staff finds TS 3.3(1) to be acceptable.

2. A pool water temperature alarm shall indicate if water temperature reaches 130°F.

TS 3.3 (2) ensures that the bulk water temperature is maintained below 54.4 degrees C (130 degrees F) to minimize evaporation of reactor coolant and thermal expansion of the aluminum liner of the shield tank wall. This temperature limit also ensures that the pool water temperature is consistent with assumptions used in the 2008 SAR (Ref 9) thermal-hydraulic analyses (Ref 9). During power operation, if the pool water temperature exceeds 54.4 degrees C (130 degrees F), a reactor scram is initiated. This alarm setpoint ensures the pool cooling system is adequately removing the reactor heat and also keeps the rate of pool water evaporation within acceptable limits. Based on the discussion above, the NRC staff finds this specification acceptable.

3. The reactor shall not be operated if the conductivity of the pool water exceeds 5 μS/cm (<0.2 MΩ-cm) when averaged over a period of one week.

TS 3.3(3) ensures that the conductivity of the tank water is maintained at less than 5 μS/cm (micro-Siemens per centimeter - one Siemens is equal to one mho) to control corrosion. 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 the heat transfer between the cladding and coolant will not degrade because of oxide buildup on the cladding. A limit of 5 μS/cm is consistent with the conductivity limits at other TRIGA reactors. Based on the discussion above, the NRC staff finds this specification to be acceptable.

4. The reactor shall not be operated if the radioactivity of pool water exceeds the limits of 10 CFR Part 20, Appendix B, Table 3 for radioisotopes with half-lives >24 hours.

TS 3.3(4) limits the radiation levels in the primary coolant system. This protects personnel working in close proximity to the reactor, by minimizing exposure from these radiation hazards, and is consistent with the ALARA principle. Since this requirement promotes compliance with 10 CFR Part 20 requirements for occupational radiation exposure, the NRC staff finds this specification acceptable.

5. The reactor shall not be operated if the pH of the pool water is greater than 7.5 or less than 5.5.

TS 3.3 (5) ensures that the pH of the tank water is maintained between 5.5 and 7.5. Similar to TS 3.3 (3) above, this specification helps to control corrosion of reactor components and minimize the activation of impurities in the pool water. A pH limit between 5.5 and 7.5 is consistent with other TRIGA reactors and the guidance provided in NUREG-1537. Based on the discussion above, the NRC staff finds this specification to be acceptable.

Based upon the information provided in the 2008 SAR (Ref 9) and the response to RAI No. 1 in a letter dated June 16, 2010 (Ref 20), the NRC staff concludes that TS 3.3 ensures that the water level exceeds 15 ft above the core during all modes of reactor operation and that the reactor pool water quality will be adequate for reactor operation. The NRC staff also concludes that the probability of a significant release to the environment resulting from pool leakage is extremely low.

The surveillance requirements associated with the reactor pool are as follows:

<u>TS 4.3 Coolant Systems</u>

1. The pool water conductivity and radioactivity shall be measured quarterly. This specification shall not be deferred during extended periods without operation.

TS 4.3 specifies the surveillance requirement for measuring pool water conductivity and radioactivity content. Pool water conductivity is continuously monitored but would be manually monitored on a quarterly basis if the instruments failed. Radioactivity is indirectly monitored by an area radiation monitor near the demineralizer bed, so gross activity increases in the pool water would be detected. Experience with TRIGA reactors indicates the earliest detection of fuel-clad leaks is usually from airborne activity, rather than pool water activity. However, analysis of the quarterly pool water measurement can identify specific radionuclides. The staff reviewed the surveillance frequency for TS 4.3 and finds that it is consistent with guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007, and, therefore, the NRC staff concludes that TS 4.3 is acceptable.

2. Pool water pH shall be measured quarterly during periods of time when the conductivity is greater than 0.1μS/cm (resistivity < 10MΩ-cm). This specification shall not be deferred during extended periods without operation.

The licensee proposed, in response to RAI No. 1 in a letter dated December 8, 2010 (Ref. 28), to substitute conductivity measurements for pH measurements of the reactor pool water when the conductivity is less than 5 μS/cm (resistivity > 0.2 MΩ-cm). However, the licensee stated that at this conductivity limit, the pH range would be from 5.58 to 8.42. The staff noted that this range exceed the guidance from NUREG-1537, which provided a pH range from 5.5 to 7.5. In a response provided in a letter dated January 28, 2011 (Ref. 29), the license proposed reducing the conductivity value specified in TS 4.3 item 2, to 0.1 μS/cm (resistivity < 10MΩ-cm). With a conductivity value less than 0.1 μS/cm, the staff concluded that the pH range would be between 6.5 and 7.5, based on the information provided by the licensee in the December 8, 2010 RAI response. The NRC staff reviewed the licensee's response, as well as other information relating conductivity and pH, and finds that this method and the proposed TS 4.3, item 2, acceptable.

The UWNR reactor pool has a history of minor pool leakage to the compacted fill beneath the facility. The only confirmed leak path has been through cracks in the corner welds around the pool penetration associated with the thermal column. The licensee has attributed the cracked welds to thermal cycling of the pool liner caused by overcooling or overheating of the reactor coolant. The licensee first identified the pool leak in 1986. In 2003, the licensee installed a new cooling system capable of maintaining a constant pool temperature in any operating condition, to prevent further thermal cycling of the pool liner. The leak reinitiated on two occasions after the new cooling system was installed. The licensee attributed the leaks to an overheating event in October 2004 and an overcooling event in May 2008. The historical leak trend is depicted in Figure 2.1 below. To prevent further overcooling and overheating events, the licensee installed a high/low temperature alarm on the reactor console. With the measures taken to control the pool temperature, UWNR has seen no leakage from the pool since October 2008.

The licensee has committed to continuing a pool leak monitoring program. The program monitors the pool makeup rate and compensates for evaporation and other expected losses. The licensee reports any volume of lost coolant that exceeds normal losses as an environmental release. The licensee monitors the radioactive content of the pool water on a regular basis and reports the monthly average pool leakage and activity in its annual report to the NRC. The licensee has not detected any pool water activity with a half-life greater than 24 hours, with the exception of tritium. Typical concentrations of tritium in the pool water equate to approximately 10 percent of the effluent release limit in Table 2 of Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage," to 10 CFR Part 20. Any significant increase in the radioactive content of the reactor pool would be indicated by high levels on the air radiation monitor at the surface of the pool and the demineralizer area radiation monitor.

In the event that the pool leak reinitiates and an increased trend in leakage is identified, the licensee has established limits for further operation of the facility. Through the licensee response to RAI No. 1 in a letter dated June 16, 2010 (Ref 20), the licensee stated that it will adhere to the mandates of the UWNR RSC, which, in 2003, mandated a reactor shutdown and an immediate repair if either of the following conditions occurs:

(1) pool water makeup greater than 2,200 gallons per month, or

(2) pool water activity approaching 80 percent of 10 CFR Part 20, Appendix B, Table 2, water effluent concentration limits for isotopes with half-lives greater than 24 hours.

As the RSC limitations would prevent the licensee from exceeding 10 CFR Part 20, Appendix B, Table 2, limits, the NRC staff finds the above commitment acceptable.

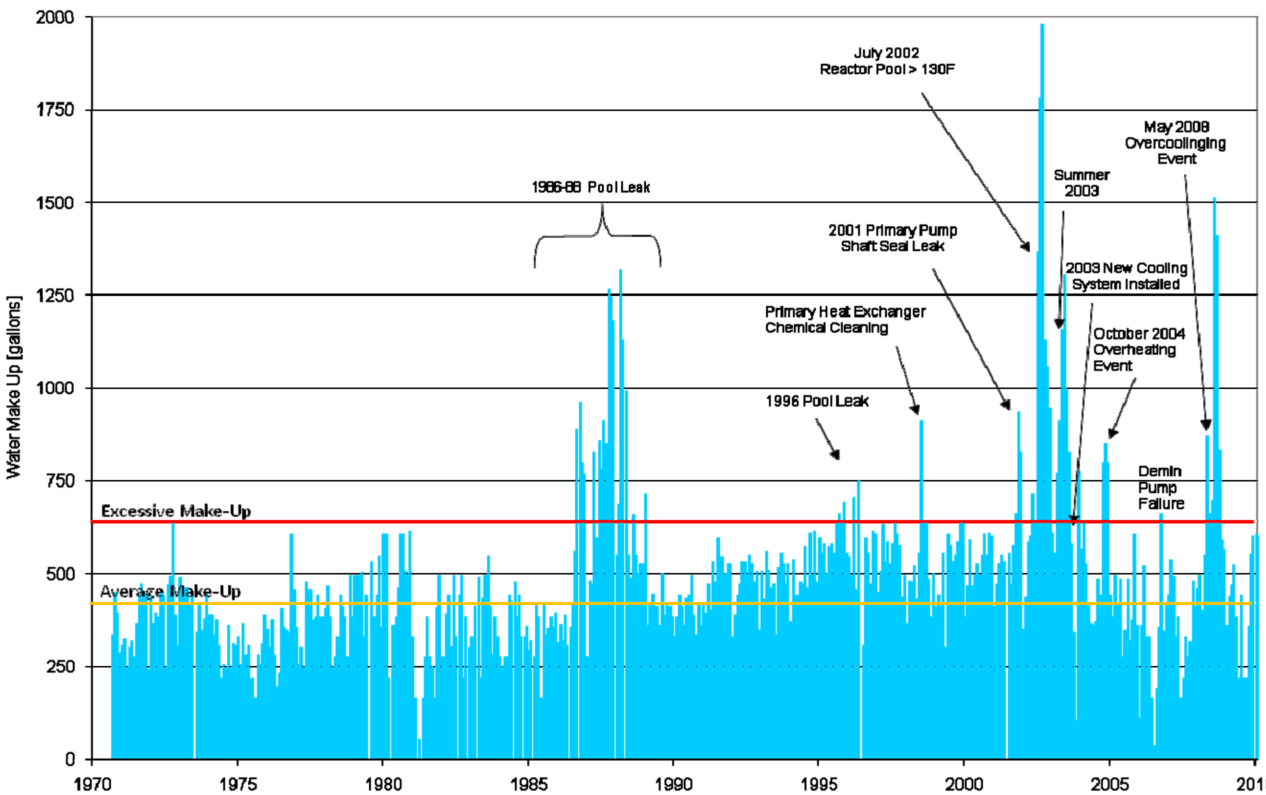

Figure 2.1 Historical Pool Water Makeup Trend

In the event of any further releases of pool water to the soil beneath the pool, the impact on public safety and the environment would not be significant, as the amount of water would be small and the radioactivity level would be low. As described in Chapter 11 of the 2008 SAR (Ref 9), the typical radioactivity concentrations in the primary coolant are very low. The equilibrium concentrations of predominant radionuclides in the primary coolant are within the limits in Appendix B to 10 CFR Part 20. The NRC staff concludes that potential leakage of primary coolant will not have a significant impact on public health and safety or the environment.

After reviewing the technical adequacy of the RAI response and the commitments made by the licensee, the NRC staff finds the licensee's policy and procedures on detectable pool leakage to be acceptable. The NRC staff concludes that the reactor pool will continue to perform acceptably during the license renewal period and that the licensee has the ability to detect any further leakage from the pool.

2.4 Biological Shield

The reactor biological shield consists of the concrete pool structure and the pool water. The NRC inspection program routinely reviews the licensee's radiation protection program. Based on a review of the information provided by the licensee, operational experience, and results from the NRC inspection program, the NRC staff concludes that there is reasonable assurance that during the renewal period, the biological shield will limit exposures from the reactor and reactor-related sources of radiation, so that the UWNR will not exceed the limits of 10 CFR Part 20.

2.5 Nuclear Design

The information discussed in this section establishes the design bases for the content of other chapters, specifically the safety analysis and portions of the TS. The modified TRIGA-fueled reactor nominally operates in the steady-state mode at a steady-state thermal power level of 1.0 MW. The reactor may also be operated in the pulsed mode from an initial power of 1 kW. TS 1.3 defines the reactor components and the operational state of the reactor. Several of these definitions are reproduced below for reference, and Section 2.5.2 below discusses the safety limits and limiting safety system settings (LSSSs). Section 2.2 previously discussed the limiting conditions for operations (LCOs) for the reactor core and the reactor control and safety systems.

TS 1.3 Reactor Components

REACTOR SHUTDOWN:

The reactor is shut down when the reactor is subcritical by least 0.7% $\Delta k/k$ of reactivity.

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 present in the reactor to attain criticality upon optimum available conditions of moderation and reflection, or

2. The following conditions exist:

a. All control elements are fully inserted, with the exception of the regulating blade in the event of an emergency,

b. The reactor is shut down,

c. The console key switch is in the "off" position and the key is removed from the console and under the control of a licensed operator or stored in a locked storage area, and

d. No work is in progress involving core fuel, core structure, control elements, or control element drives unless the work on the drive cannot move the control element, and

e. No experiments are being moved or serviced that have, on movement, a positive reactivity worth exceeding 0.7%Δk/k.

REACTOR OPERATION:
Reactor operation is any condition wherein the reactor is not secured.

These definitions describe the operational state of the reactor. The NRC staff reviewed these definitions and finds them to be acceptable and consistent with NUREG-1537, and ANSI/ANS-15.1-2007.

Steady-State Operation

The UWNR reactor is licensed to operate up to 1.0 MW. The safety limits, as provided in the 2008 SAR (Ref 9), require that the maximum temperature in the TRIGA 30/20 fuel not exceed 1,150 degrees C, and that the reactor power level not exceed 1.5 MW steady state under any condition of operation. To comply with these safety limits, LSSSs, as provided in the 2008 SAR (Ref 9), are established for the fuel temperature, as measured by two instrumented fuel elements: 400 degrees C for an instrumented fuel element (IFE) in a core location with a power peaking factor in the range of 0.87 to 1.16 and 500 degrees C for an IFE in a core location with a power peaking factor greater than or equal to 1.16. The LSSS for the reactor power level is 1.25 MW. The licensee stated that analyses performed by the vendor indicate that TRIGA 30/20 fuel may be safely operated up to 1.5 MW under conditions of natural convection cooling. Therefore, steady-state operation of the UWNR at 1.0 MW with an LSSS setting of 1.25 MW allows for a sufficient safety margin. The NRC staff reviewed the safety limits provided by the licensee and finds these limits acceptable. The NRC staff also reviewed the LSSS and finds that these limits are acceptable. The NRC's staff evaluation is provided in Section 2.5.2 of this SER.

Pulse Mode Operation

The licensee stated that UWNR can be operated in a pulsed mode from a low-power level to a high-power level by the rapid insertion of reactivity by ejecting a transient control rod from the central fuel cluster in the core. In this mode of operation, the maximum reactivity insertion is limited to 1.4% Δk/k, and the reactivity pulse may not be initiated from a core power in excess of 1 kW. Pulsing from a power level greater than 1 kW is prevented by required interlocks, which prevent the application of air to fire the transient rod. Assuming the most limiting core configuration (LEU core at MOL) and a reactor scram at 5 seconds following the pulse, the licensee calculated that a prompt reactivity insertion of 1.4 %Δk/k will result in a prompt peak fuel temperature of 790 degrees C (1,454 degrees F). Even if the pulse continued for the maximum pulse timer setting of 15 seconds, fuel temperature will not exceed 830 degrees C (1,526 degrees F). Calculations provided by the licensee show the maximum fuel temperature following a prompt reactivity insertion of 1.4 %Δk/k is 826 degrees C (1,519 degrees F) (Ref 10). GA recommended the 830 degrees C fuel temperature limit during pulsing, based on the fuel damage experience at another similar reactor. On the basis of its review, the NRC staff finds that the peak fuel temperature limit and the associated maximum reactivity insertion limit for pulsing ensure that the reactor can be safely pulsed without any concern regarding fuel damage, and therefore is acceptable.

The following sections of the SER will discuss additional details of the nuclear design, including both steady-state and pulsed operation.

2.5.1 Normal Operating Conditions and Reactor Core Physics Parameters

Calculational Methodology

To perform the UWNR neutronic analyses, the licensee used REBUS-MCNP (MCNP) (a computer code developed by Argonne National Laboratory to analyze reactor fuel cycles) for the burnup and inventory analyses. It modeled two different cores for the analyses: the HEU core, consisting of 91 HEU FLIP fuel elements, and the LEU core, containing 83 30/20 LEU fuel elements. The licensee used these calculations to benchmark the analytical methods and to confirm the nuclear design performance for the LEU core (Ref 10).

The licensee benchmarked the MCNP core model by comparing the calculated and measured values of the three control blades, regulating blade, and transient rod. It used two methods to calculate the transient rod worth for the HEU core: the rod drop and the rising period rod bump methodology of measuring reactivity. Both values agreed well with the measured value for the HEU core of 1.374% Δk/k. The licensee calculated the integral worth of the transient rod for the LEU core to be 1.369% Δk/k.

The NRC staff reviewed the licensee’s use of codes for the LEU conversion analysis (Ref 10) and finds the licensee to be knowledgeable in applying its use of code applications. Because the licensee used documented codes that are well accepted and have been validated against data for other TRIGA reactors, including UWNR, the NRC staff concludes that the licensee’s calculational methodology is acceptable (Ref 5, Section 2.4).

Excess Reactivity:

The licensee provided calculated and measured values for excess reactivity. If the licensee directly replaced the former HEU-fueled core with LEU fuel, the excess reactivity of the LEU core would be high enough that the core would be supercritical under the shutdown margin conditions defined in TS 3.1. The solution to this problem was to replace eight fuel elements in the LEU core with graphite reflectors. Even though the LEU core excess reactivity values are considerably lower than for the HEU core, they are sufficient to permit all modes of operation. Because the conversion from HEU to LEU fuel does not affect the ability of the licensee to meet TS 3.1, the NRC staff finds the values for excess reactivity acceptable for the current LEU core, based on the conversion analyses (Ref 5, Section 2.4).

Control Element Worth:

Using the MCNP code to model the control element worth, the licensee calculated the control element worths by using either the rod drop or the rising period method of measuring reactivity (Ref 10). For the previous HEU core, calculations of integral control elements compared favorably with measured values for the control elements. Calculated integral worths of the control elements for the LEU core are shown in Table 2-4, along with the measured values of the control element integral worths from the “Reactor Startup Report,” dated June 21, 2010. As the calculated values are in agreement with the measured values, the NRC staff finds the values for control element worth acceptable.

Table 2-4 Calculated and Measured Values for the Control Elements

Control Element	LEU 30/20	
	Calculated Integral Worth (% Δk/k)	Measured Integral Worth (% Δk/k)
Control Blade 1	2.13	2.22
Control Blade 2	2.09	1.99
Control Blade 3	2.60	2.36
Regulating Blade	0.38	0.37
Transient Rod	1.37	1.36

Shutdown Margin:

The licensee calculated the shutdown margin for both the HEU and LEU cores using the MCNP-5 model. The calculated shutdown margin at beginning of life (BOL), with no experiment installed, was 0.903% Δk/k for the HEU core and 0.994% Δk/k for the LEU core (Ref 10). If the licensee installed the maximum allowable experiment of 0.7% Δk/k, both would continue to meet TS 3.1, which states that this margin should be at least 0.2% Δk/k. Because the conversion from HEU to LEU fuel does not affect the ability of the licensee to meet TS 3.1, the NRC staff concludes that these values are acceptable for the current LEU core, based on the conversion analyses (Ref 5, Section 2.4).

Effective Delayed Neutron Fraction (β_{eff}):

The licensee calculated the effective delayed neutron fraction, β_{eff}, by using the MCNP code for both the HEU and LEU cores (Ref 10). It calculated β_{eff} for the HEU core to be 0.0075 at BOL, 0.0076 at the middle of life (MOL), and 0.0073 at the end of life (EOL). It calculated β_{eff} for the LEU core to be 0.0078 at BOL, 0.0077 at MOL, and 0.0073 at EOL. The calculated values of β_{eff} are consistent with values determined for other research reactors.

The licensee conducted pulse mode operation tests during startup testing, starting with a small pulse (1.00% Δk/k), followed by larger pulses, each increased by an increment of 0.05% Δk/k up to a maximum of 1.363% Δk/k. The licensee measured the peak power, peak fuel temperature, and integrated pulse power during each pulse and used the measurements to validate calculated values of neutronic parameters, including β_{eff}. The results showed good agreement; the calculated β_{eff} was 0.0078 and the measured β_{eff} was 0.0073.

Based on the above discussion, the conversion analyses, and the agreement between the calculated and the measured β_{eff} (Ref 5, Section 2.4), the NRC staff concludes that the values in β_{eff} are acceptable for the current LEU core.

Fuel Temperature Coefficient of Reactivity (α_{Fuel}):

One of the inherent safety features of the TRIGA fuel design is the large negative fuel temperature coefficient of reactivity, α_{Fuel}. With conversion to LEU fuel, the amount of uranium was increased from approximately 8.5 wt% (HEU) to 30 wt% (LEU), with no change in the physical size of the fuel. Therefore, the amount of zirconium hydride is reduced from approximately 91.5 wt% (HEU) to 70 wt% (LEU). It is expected that this, along with the increase in the amount of U-238 in the fuel, will affect the fuel temperature coefficient. Table 2-5 shows the licensee's calculated values of α_{Fuel} for the HEU and LEU cores at BOL, MOL, and EOL, as a function of temperature (Ref 10).

Table 2-5 Calculated Fuel Temperature Coefficients of Reactivity for HEU and LEU Cores

Temperature (K)	Negative Prompt Fuel Temperature Coefficient (% Δk/k / Kx10^3)					
	BOL		MOL		EOL	
	HEU	LEU	HEU	LEU	HEU	LEU
350	4.57	4.08	3.75	3.26	3.64	3.05
500	7.17	6.57	6.00	5.65	5.64	5.33
700	10.38	8.20	7.63	6.75	6.41	5.86
1000	19.18	13.34	15.44	11.00	13.29	9.54

Measurements during the startup of the conversion LEU core also indicate good agreement with the calculated values, as follows (Ref 14):

Table 2-6 Predicted vs. Measured Temperature Coefficients

	LEU Predicted	LEU Measured
Fuel temperature coefficient [% Δk/k/°C]	-4.08×10^{-3} (300-400K) -6.58×10^{-3} (400-600K)	-5.55×10^{-3}

The decrease in magnitude from the HEU core to the LEU core is expected, because of the added U-238 in the LEU fuel. However, when pulsing, it is the zirconium hydride that dominates the value of α_{Fuel}. Although the value of the fuel temperature coefficient decreases in magnitude from the HEU core to the LEU core during pulsing, the limitations on pulsing will continue to protect the integrity of the fuel, and this is acceptable for the current LEU core, based on the conversion analyses (Ref 5, Section 2.4).

As part of the conversion, the NRC staff reviewed the values for α_{Fuel} provided by the licensee for various times during the core life. Because the values for α_{Fuel} continue to provide for the inherent safety of the reactor with increasing fuel temperature, the NRC staff concludes that the changes in α_{Fuel} are acceptable (Ref 5, Section 2.4).

Void Coefficient of Reactivity:

The licensee calculated the void coefficient of reactivity for both the HEU and the LEU cores, using the MCNP model, by uniformly reducing coolant density 2.5 percent to introduce a void in the system. The calculated void coefficients, as shown in Table 2-7, are more negative for the LEU core than for the HEU core (Ref 10).

Table 2-7 Calculated Void Coefficients of Reactivity for HEU and LEU Cores

Fuel Condition		HEU (Δk/k / %void x 10^3)	LEU (Δk/k / %void x 10^3)
BOL	Cold (27ºC)	-1.13	-1.49
	Hot (327ºC)	-1.06	-1.35
MOL	Cold (27ºC)	-0.971	-1.35
	Hot (327ºC)	-0.962	-1.28
EOL	Cold (27ºC)	-0.673	-1.36
	Hot (327ºC)	-0.633	-1.25

A comparison of calculated (conversion SAR) and measured (startup report) void coefficient values show good agreement, as follows:

Table 2-8 Predicted vs. Measured Void Coefficient

	LEU Predicted	LEU Measured
Void coefficient [% Δk/k/%void]	-1.49	-1.40

The NRC staff reviewed the values for the void coefficient of reactivity for various times during core life, and it concludes that the changes in the void coefficient of reactivity are acceptable for the current LEU core, based on the conversion analyses (Ref 5, Section 2.4) and their agreement with the measured value from the startup report.

Coolant Temperature Coefficient of Reactivity:

The licensee calculated the coolant temperature coefficient of reactivity, as shown in Table 2-7, using the MCNP model to uniformly increase the coolant temperature by 100ºkelvin (K) (Ref 10). Although heating the coolant has a small positive temperature coefficient, raising the coolant temperature would also raise the fuel temperature. The magnitude of the coolant temperature coefficient is small when compared to the magnitude of the negative fuel temperature coefficient. The overall reactivity effect is negative when considering both the coolant and fuel temperature coefficients. The NRC staff concludes that the changes in the coolant temperature coefficient of reactivity are acceptable for the current LEU core, based on the conversion analyses (Ref 5, Section 2.4).

Table 2-9 Calculated Coolant Temperature Coefficients of Reactivity for HEU and LEU Cores

Fuel Condition		HEU (Δk/k / ºK x 10^4)	LEU (Δk/k / ºKx10^4)
BOL	Cold (27ºC)	0.931	0.816
	Hot (327ºC)	0.350	0.905
MOL	Cold (27ºC)	1.01	0.848
	Hot (327ºC)	1.09	0.923
EOL	Cold (27ºC)	1.21	0.910
	Hot (327ºC)	1.39	1.01

2.5.2 Operating Limits

The regulations in 10 CFR 50.36(d)(1) require reactors to specify safety limits and LSSSs. These regulations define safety limits as limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. LSSSs for nuclear reactors are defined as settings for automatic protective devices related to those variables having significant safety functions. Where an LSSS is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective actions will correct the abnormal situation before a safety limit is exceeded.

The safety limits for the UWNR are as follows:

TS 2.1 Safety Limits

1. The temperature in a TRIGA LEU 30/20 fuel element shall not exceed 1150°C under any conditions of operation.

2. The reactor steady-state power level shall not exceed 1500 kW under any conditions of operation

TS 2.1(1) specifies the fuel rod temperature, which is an important parameter for a TRIGA reactor. A loss of integrity of the fuel element cladding could arise from a buildup of excessive pressure between the fuel moderator and the cladding, if the fuel temperature exceeds the safety limit. The pressure is caused by air, fission product gases, and hydrogen from dissociation of the fuel moderator. The fuel moderator temperature and the ratio of H to Zr in the alloy determine the magnitude of this pressure.

The safety limit for the 30/20 LEU fuel is based on data that indicate the stress in the cladding because of the hydrogen pressure from the disassociation of zirconium hydride will remain below the ultimate stress, provided the temperature of the fuel does not exceed 1,150 degrees C and the fuel cladding is water cooled. The NRC evaluated the properties and performance of the TRIGA higher uranium wt-% LEU fuel, including the 30/20 LEU fuel, in NUREG-1282, “Safety Evaluation Report on High-Uranium Content, Low-Enriched Uranium-Zirconium Hydride Fuels for TRIGA Reactors,” issued August 1987 and approved for use with the provision that a case-by-case analysis would discuss individual reactor operating conditions when using the fuel. The UWNR safety limit is set at the temperature established in NUREG-1282.

The NRC staff concludes that TS 2.1(1) establishes a fuel temperature safety limit for the UWNR that is consistent with the safety limits used for other TRIGA reactors and is approved by the NRC. The fuel temperature safety limit is therefore acceptable to the NRC staff.

TS 2.1(2) specifies a steady-state power limit, ensuring that adequate cooling is provided to the fuel rods under natural circulation conditions. The LEU conversion SAR (Ref 10) shows, by analysis, that a power level of 1.5 MW corresponds to a peak fuel temperature of 665 degrees C, providing adequate cooling relative to the maximum heat flux preventing the departure from nucleate boiling (DNB) and the resulting rapid increase in clad temperature that would lead to cladding failure. In addition, experience has shown that operating TRIGA reactors of similar design at a power level of 1.5 MW will not result in damage to the fuel. Several reactors of this type have operated successfully for several years at power levels up to 1.5 MW.

Based on the analysis from the LEU conversion SAR and the history of safe operation at 1.5 MW by other TRIGA reactors, the NRC staff concludes that the safety limit of 1.5 MW, as established in TS 2.1(2), is acceptable.

The LSSSs for the UWNR are as follows:

TS 2.2 Limiting Safety System Setting

1. The limiting safety system setting for fuel temperature shall be 400°C as measured in an instrumented fuel element with a pin power peaking factor between 0.87 and 1.16, or 500°C as measured in an instrumented fuel element with a pin power peaking factor of at least 1.16.

2. The limiting safety system setting for reactor power level shall be 1.25 MW.

TS 2.2(1) specifies the LSSS limit for fuel temperature as measured in the IFEs. In the LEU conversion SAR, the licensee conducted analyses in support of locating an IFE connected to the fuel temperature safety channel in the periphery and central regions of the core. According to the analysis, if the IFE is located where the IFE power peaking factor is between 0.87 and 1.16, then a scram at 400 degrees C (752 degrees F) ensures the maximum fuel temperature will not exceed 678 degrees C (1,252 degrees F). Additionally, if an IFE were located where the local power peaking factor is at least 1.16, then a scram signal generated at 500 degrees C (932 degrees F) ensures that the maximum fuel temperature will not exceed 678 degrees C (1,252 degrees F). The LSSS values provide a margin of 472 degrees C (882 degrees F) to the safety limit of 1,150 degrees C (2,100 degrees F). If the IFE is located in either the periphery or central region of the core, both would continue to meet TS 2.1, which states that the temperature in a TRIGA LEU 30/20 fuel element shall not exceed 1,150 degrees C (2,100 degrees F) under any conditions of operation.

The NRC staff concludes that the analysis performed by the licensee is acceptable and establishes LSSS values that are sufficient to prevent reaching the safety limit. Therefore, TS 2.2(1) is acceptable to the NRC staff.

TS 2.2(2) specifies an LSSS for reactor power that is below the safety limit with sufficient margin for uncertainty. The NRC staff concludes that the LSSS of 1.25 MW is sufficient to prevent exceeding the reactor power safety limit, and, therefore, TS 2.2(2) is acceptable.

In the pulse mode of operation, the same LSSS will apply. However, the temperature channel will have no effect on limiting the peak powers generated because of its relatively long time constant as compared with the width of the pulse. In this mode, however, the temperature trip will act to reduce the amount of energy generated in the entire pulse transient, in the event the pulse rod remains stuck in the fully withdrawn position.

The NRC staff concludes that the LSSS and the accompanying conditions are sufficient to protect the safety limit and are acceptable. The NRC staff concludes that the safety limit and LSSS for the reactor are based on acceptable analytical and experimental investigations, are consistent with those approved by the NRC and used at other TRIGA-type reactors, and are acceptable for the current LEU core, based on the conversion analyses (Ref 5, Section 2.5).

2.6 <u>Thermal-Hydraulic Design</u>

2.6.1 Calculational Methodology

The licensee presented the thermal-hydraulic design of the UWNR in the LEU conversion SAR (Ref 10). For steady-state operation, it presented the thermal-hydraulic design in two parts. The first part presented the methodology and results for the HEU core, using operational data to validate the analytical results. It then used the same computational technique in the second part to analyze the expected performance of the LEU core.

The following sections of this SER will discuss additional details of the thermal-hydraulic design, including both steady-state and pulsed operation.

2.6.2 Steady-State Operations

The LEU conversion SAR (Ref 10) presents the thermal-hydraulic analysis for the HEU and LEU cores and considers steady-state and pulsing operation with natural circulation cooling at various times in core life (Ref 10).

<u>Calculational Methodology</u>

To carry out the steady-state thermal-hydraulic analysis and the core pulse analysis, the licensee performed calculations for the HEU core using RELAP5/MOD3.3 and validated them against operational data from the HEU core.

Inputs to the RELAP5 analysis included the following: inlet coolant temperature, system pressure at the top of the core, radial and axial heat source distribution, spacing of heat source nodes, inlet and exit pressure loss coefficients, and geometric parameters for the coolant channels and the fuel rods. The output parameters included: channel flow rate, axial fuel centerline temperature distribution, fuel radial temperature distribution, axial clad temperature distribution, axial bulk coolant temperature distribution, and axial DNB ratio (DNBR).

The NRC staff reviewed the licensee's use of codes for the thermal-hydraulic analysis. Because the licensee used documented codes that are well accepted and have been validated against data for other TRIGA reactors, including UWNR, the NRC staff concludes that the calculational methodology used by the licensee is acceptable.

Results of Thermal-Hydraulic Analysis

Using the established methodology, the licensee carried out the steady-state analyses for the HEU and LEU cores at three different power levels: 1.5 MW, 1.3 MW (1.25 MW trip set point plus uncertainty), and 1.0 MW (the licensed operating power) at a water inlet temperature of 54 degrees C (normal operating temperature is between 25 and 30 degrees C at 1.0 MW). The steady-state analysis also assumed a pool water level of 5.8 meters above the top of core. This assumption is consistent with TS 3.3(1), "Reactor Pool Water Systems." Results of the steady-state analysis included fuel and coolant temperatures and the minimum DNBR (MDNBR).

The steady-state analysis considered the maximum-powered fuel element and the coolant subchannel associated with this single fuel rod. The column of water surrounding the core supplied the driving force for the core flow. A natural circulation flow rate is established to balance the driving head against the core entrance and exit pressure losses and frictional, acceleration, and hydrostatic head losses in the core flow channel. The licensee calculated the inlet and outlet pressure loss coefficients by considering a series of abrupt contractions and expansions caused by flow channel geometry changes. An examination of the pressure differential between the hot channel and an adjacent channel indicated that cross flow would occur from the cold to the hot channel. The net increase in flow in the hot channel tends to increase the margin to critical heat flux (CHF). The thermal-hydraulic calculations conservatively ignored cross flow between adjacent channels in the LEU conversion SAR (Ref 10).

The licensee performed the steady-state thermal-hydraulic analysis for the maximum-powered channel. For both the HEU and the LEU core, the maximum-powered fuel element is next to the transient rod. In the single-channel, steady-state model, the coolant subchannel was conservatively assumed to be bordered on all sides by a fuel rod having the same characteristics as the hot rod. Because of channel geometry, Table 4.7.3 of the LEU conversion SAR demonstrates that the channel with the highest thermal power (e.g., hot rod) is not the channel with the minimum flow rate (e.g., rod next to the control blade shroud). The licensee established through analysis that the channel associated with the hot rod had the lowest MDNBR and is therefore the limiting channel.

Table 2-10 summarizes the results of the steady-state, thermal analysis for both the HEU and LEU core. Compared to the HEU core (91 fuel elements), the LEU core (83 fuel elements) was expected to have higher hot rod power and higher maximum fuel temperatures. Results of the analysis confirmed this expectation, and the differences in MDNBR and the maximum fuel temperatures between the HEU and LEU cores are consistent with the predicted hot rod powers. As the HEU analysis was performed solely to validate the model, only the BOL case was analyzed.

Table 2-10 Summary of Steady-State Analysis at 1.5 MW

Core Type	Hot Channel Peak Factor	Hot Rod Thermal Power* (kW)	Maximum Fuel Temperature (ºC)	MDNBR	Maximum Exit Bulk Coolant Temperature (ºC)
HEU-BOL	1.60	26.40	642.0	1.29	100.4
LEU-BOL**	1.61	29.04	673.9	1.23	101.0
LEU-MOL	1.60	28.89	665.1	1.22	101.2
LEU-EOL	1.57	28.33	641.9	1.23	100.7

*Hot rod thermal power corresponds to a core power of 1.5 MW.
**The LEU-BOL results are the revised values from the UWNR supplemental information.

During the LEU conversion startup, the licensee completed a fuel temperature mapping, allowing a comparison with the LEU core analysis calculations (Ref 14). It placed two IFEs, bundles 62 and 72, in different locations to measure the fuel temperature, as follows:

Table 2-11 Predicted vs. Measured IFE Temperatures

	LEU Predicted	LEU Measured	
		Bundle 62	Bundle 72
D4 SW Temp (C)	391-480	342.3	405.3
E3 NE Temp (°C)	264-317	269.1	318.6

The differences between the two bundles are most likely caused by differences in fuel swelling and physical contacts between the fuel and the thermocouples. The NRC staff concludes the results are within predicted uncertainty and show reasonable agreement with the analysis.

The core conversion study for the UWNR considered two different correlations for calculating the CHF, the Groeneveld 2006 Lookup Tables and the Bernath correlation. Between the two correlations, the Bernath correlation consistently produced lower DNBR values. At the nominal operating core power of 1.0 MW, the MDNBRs calculated by the Bernath correlation were 1.60 and 1.53 for the HEU and LEU cores, respectively, at BOL. This represents a 5-percent reduction in the MDNBR from the HEU to LEU core for a corresponding 10-percent increase in hot rod power.

In applying the Groeneveld Lookup Tables to the UWNR, a set of correction factors is required to adapt the geometry used in the lookup tables to the UWNR geometry. The LEU conversion SAR identified three correction factors that are applicable—K1 (hydraulic diameter), K2 (rod bundle effects), and K4 (axial heated position). The other three correction factors—K3 (grid spacer effect), K5 (saturated boiling effect), and K6 (down flow effect), were evaluated and found not applicable to the UWNR and were assumed to be equal to 1.0. In evaluating the DNBR using both the Groeneveld Lookup Tables and the Bernath correlation, the outer diameter of the fuel rod represented both the heated and wetted diameter.

Table 2-12 summarizes the results of the critical rod powers for both the HEU and LEU cores. In all cases, the Bernath correlation predicted lower DNBR values than the Groeneveld 2006 Lookup Tables.

Table 2-12 Critical Rod Powers

DNBR Correlation	Power (MW)	HEU-BOL (kW/rod)	LEU-BOL* (kW/rod)	HEU-MOL (kW/rod)	HEU-EOL (kW/rod)
Groeneveld 2006	1.5	52.4	53.5	52.8	54.6
	1.3	49.6	52.75	49.6	54.7
	1.0	47.6	51.9	51.6	51.3
Bernath	1.5	34.0	36.3	35.8	35.5
	1.3	31.9	33.4	33.5	33.2
	1.0	28.2	29.6	29.4	29.1

*The LEU-BOL results are the revised values from the UWNR supplemental information.

2.6.3 Pulsed Operation

For the pulse analysis, the licensee calculated the transient power by the point kinetics model in RELAP5 (Ref 10). It modeled the reactor with two separate hydraulic channels: the hot rod channel (the same as for the steady-state analysis) and an average channel representing the rest of the fuel elements. The only reactivity feedback incorporated in the pulse analysis was the prompt negative fuel temperature coefficient.

Analysis has shown that the time lag in heat transfer from the fuel to the moderator makes the moderator feedback inconsequential. The licensee performed the pulsing analysis with the assumption that the reactor core maintains the same power shape as the steady state. To show that the assumptions are valid, the licensee performed a sensitivity analysis with a more severe pin power peaking (12 percent higher, LEU-MOL with the transient rod full out and control blades at the cold critical bank height) and the same 1.4% Δk/k pulse at the most limiting state-point of the LEU core. The resulting peak pulse temperature was increased from 727 degrees C to 790 degrees C but is still below GA's recommended fuel temperature pulsing limit of 830 degrees C. The licensee analyzed all pulse transients with an initial reactor power of 1 kW. The analysis also assumed that the transient rod remained out of the core for 15 seconds and then all scrammable control elements were fully inserted into the core. This assumption is consistent with the requirement in TS 3.3.3(g), "Reactor Safety System—Preset Timer," that a scram must occur in 15 seconds or less after the initiation of a pulse.

The licensee conducted an analysis at various stages in core life on the maximum pulse of 1.4% Δk/k allowable by TS 3.2. It performed an additional analysis to determine the pulse sizes needed to reach GA's recommended fuel temperature pulsing limit of 830 degrees C and the fuel temperature safety limit of 1,150 degrees C. Tables 2-13 and 2-14 summarize the results of the pulse analyses.

Table 2-13 Summary of Results for a 1.4% Δk/k Pulse

Core Type	Maximum Fuel Temperature (ºC)	Peak Pulse Power (GW)	Total Pulse Energy (MJ)
HEU-BOL	661	2.29	28.1
LEU-BOL	663	2.64	26.9
LEU-MOL	727	2.52	31.9
LEU-EOL	723	3.06	35.2

Table 2-14 Pulse Sizes to Exceed the Fuel Temperature Limits

Core Type	Pulse to Exceed Fuel Temperature Operational Limit (% Δk/k)	Pulse to Exceed Fuel Temperature Safety Limit (% Δk/k)
HEU-BOL	1.6	2.2
LEU-BOL	1.6	2.1
LEU-MOL	1.5	2.0
LEU-EOL	1.5	2.0

The licensee performed pulsing measurements during the LEU conversion core startup with pulses up to 1.363% Δk/k. The measured energy release of 9.59 megajoules (MJ) at the highest tested pulse of 1.363% Δk/k is substantially below the similar 28.1 MJ value at 1.4% Δk/k that the analytical calculations predicted.

The pulse results presented in Table 2-3 above indicate that the highest predicted maximum fuel temperature occurs in the LEU core at MOL. The maximum fuel temperatures shown in Table 2-13 are prompt peaks occurring near the outside edge of the fuel. As heat is redistributed over time, the maximum fuel temperature occurs at the fuel centerline. It is noted from Table 2-14 that, at no time during the reactor life of the LEU core, will a 1.4% Δk/k pulse ever exceed the fuel temperature operational limit of 830 degrees C.

The NRC staff concludes that the thermal-hydraulic analysis for the UWNR adequately demonstrates that the reactor can operate at its licensed power level with sufficient safety margins, with regard to thermal-hydraulic conditions, and conduct pulsing operations, as limited by TS, without exceeding operational or safety limits on the fuel temperature. The licensee's analyses used qualified calculational methods and acceptable assumptions.

2.7 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 UWNR core without undue risk to public health and safety or the environment. The NRC staff's 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 UWNR LEU conversion SAR demonstrates that the LEU core results in acceptable safety margins with regard to thermal-hydraulic conditions. While UWNR is administratively limited to the licensed power level of 1.0 MW, the results of the thermal-hydraulic analysis indicate that the reactor could be safely operated to at least 1.3 MW with adequate critical heat flux margins.

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 mixed HEU core. The NRC staff reviewed the analysis of the pulsed operation of the UWNR reactor and has finds that, with pulse sizes up to the administrative limit of 1.4% $\Delta k/k$, the maximum core fuel temperature will remain below the limit set by the known mechanical and thermal properties of the fuel. The NRC staff concludes that the UWNR 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. On the basis of its review, the NRC staff concludes that there is reasonable assurance that the UWNR is capable of safe operation, as limited by the TS, for the period of the requested license renewal.

3 Reactor Coolant Systems

3.1 Summary Description

In 2003, the licensee substantially modified the reactor coolant system. All changes were made in accordance with the 10 CFR 50.59 process. The previous design consisted of a single-loop cooling system with a primary coolant pump that circulated 500 gpm of reactor coolant through a flat-plate heat exchanger that transferred heat to a dedicated closed-loop cooling tower system. A second pump circulated 1,000 gpm of cooling water between the heat exchanger and cooling tower to keep the primary coolant temperature below 54.4 degrees C (130 degrees F). The new system consists of a three-loop cooling system that uses the campus chilled water system to ultimately remove heat from the primary coolant system when the reactor is at power. Each subsystem is described below. The reactor coolant system at the UWNR has no safety function, as it is only needed as a heat removal mechanism when the reactor is at power.

3.2 Primary Coolant System

This system continuously circulates reactor pool water through a primary heat exchanger using a dedicated pump. Both the inlet and outlet pipes in the reactor pool have siphon break holes to preclude draining more than 1 ft of water from the pool. The inlet pipe discharges coolant approximately 5 ft below the core, while the suction pipe takes coolant from the pool approximately 3 ft below the surface. A level switch ensures the coolant level in the pool is maintained at 20 ft above the active core. The siphon breakers ensure a minimum of 19 ft of coolant above the core, in the event of a primary coolant piping rupture.

3.3 Intermediate Coolant System

This closed-loop system continuously circulates demineralized water though the secondary side of the primary heat exchanger and the primary side of the intermediate heat exchanger, using a dedicated pump. It is also equipped with an expansion tank, with manual makeup capability through this tank. The expansion tank accommodates volumetric changes and is used to maintain the intermediate coolant pressure above the primary coolant pressure at all times when the reactor is at power. Therefore, any leak developed in the primary heat exchanger would result in intermediate coolant leaking into the primary system. This prevents any radioactivity in the primary coolant from contaminating other systems in the facility and loss of pool water caused by a heat exchanger leak. Heat exchanger tube leakage would be detected by a high pool level alarm. In the event of a significant heat exchanger leak or piping rupture, a low intermediate pressure annunciator on the console would alarm. There is also an interlock on the reactor control console that prevents the primary coolant pump from starting unless the intermediate coolant pump is running.

3.4 Campus Chilled Water System

A dedicated pump circulates chilled water from the campus system through a filter, then through the intermediate flat-plate heat exchanger, cooling the intermediate loop coolant and returning it to the campus system. Since the campus chilled water system operates at a higher pressure than the intermediate loop, it precludes any inadvertent loss of pool water caused by leaks developing in the coolant systems or the spread of contamination.

3.5 Primary Coolant Cleanup System

The primary cleanup system extracts 18 gpm of primary coolant using a 2.0 horsepower pump and passes it through a mixed bed demineralizer to keep the reactor pool water chemistry within operational limits. Both the intake and discharge lines to the reactor pool are check-valve protected and have siphon breakers installed to minimize pool inventory loss in the event of a pipe break. This system is operated manually and is controlled by written operating procedures. This system is designed to provide reasonable assurance that the required water quality and inventory for all modes of operation are maintained. The primary cleanup system is designed such that no malfunction or leakage will lead to an uncontrolled release of reactor coolant. The primary cleanup system also ensures that oxide buildup on the fuel cladding will not exceed acceptable limits for corrosion and thus will not affect the rate of heat transfer from the fuel. The suction pipe is connected to the primary coolant pump discharge piping above the reactor pool, well above the core elevation.

3.6 Primary Coolant Makeup System

The primary coolant makeup system replaces water that has been lost by evaporation from the reactor pool surface. During normal operation, makeup water supplied by either an internal still supply or a filtered city potable water source is added to the reactor pool by the recirculating pump approximately 3 ft below the pool surface. This system is operated manually and is controlled by written operating procedures. This system is designed to provide reasonable assurance that the required water inventory for all modes of operation is maintained and designed such that no malfunction or leakage will lead to an uncontrolled release of reactor coolant.

3.7 Nitrogen-16 Control System

There were no changes to the Nitrogen-16 control system associated with the new cooling system design.

3.8 Conclusions

Review of the reactor coolant system design changes showed it has the capacity to adequately cool the reactor core under all modes of power operation. Heat exchanger tube leakage is accounted for by designing an intermediate cooling loop that operates at a higher pressure than the primary coolant loop. The use of campus chilled water as the heat sink enables the facility to maintain a constant pool temperature during any normal operating conditions, thus reducing the thermal fatigue stresses on the pool liner. The new design is therefore acceptable to the NRC staff.

4 Instrumentation and Control Systems

4.1 Summary Description

From 2003 to 2007, the licensee made substantial changes to the instrumentation and control (I&C) systems. All changes were made in accordance with the 10 CFR 50.59 process. It made most of them to facilitate modifications to the UWNR support systems, such as installation of a UPS for the instrumentation, a pneumatic tube replacement, and a modification to the dedicated HVAC system. Others were to take advantage of technological advances made in instrumentation design, such as the use of digital displays.

4.2 Design of Instrumentation and Control Systems

A signal generated from the firing of the transient rod actuates a new power pulse channel that computes and displays the peak and integrated power of the pulse automatically. Since the fuel temperature is also displayed after each pulse, the operator can more closely monitor it during pulsed mode operations.

4.3 Reactor Control System

The transient control rod logic incorporated a timer, operable in pulse mode only, that drops the rod to the full in-core position within 15 seconds after it has been initiated. This limits the pulse duration that, in turn, limits the temperature rise in the fuel for a given power pulse.

4.4 Reactor Protection System (Scram Circuits)

The licensee has completed several minor modifications to the reactor protection system (RPS) in accordance with the 10 CFR 50.59 process. Installation of the log N modification ensures that, in pulse and square wave modes of operation, the transient control rod cannot be removed if the reactor power is above 1 kW. The addition of a 76-minute battery power UPS capability on the reactor control console and instrumentation would preclude the automatic scram on loss of power. Therefore, the licensee installed, in the trip logic, a failsafe scram and alarm to a power source with no backup power supply to ensure the control blades will drop on loss of normal power. This scram is necessary, even with the emergency backup power capability to the reactor support systems, because these support systems would stop functioning for the duration of the power transfer. This precludes placing the reactor in a configuration that is outside its licensed safety basis and the potential for damage to reactor components during this power transfer period. Since the conversion to TRIGA fuel, prompt critical operation has become a normal mode of operation for the reactor. This modification made the short period trip and the 60 millisecond SCRAM delay inconsequential, as they were legacy systems in place to prevent prompt critical conditions. Therefore, the licensee removed the electronic SCRAM and period SCRAM from the SCRAM circuitry.

4.5 Engineered Safety Features Actuation Systems

There are no engineered safety features actuation systems at UWNR.

4.6 Control Console and Display Instruments

Modifications to the pneumatic tube system, dedicated HVAC system, and reactor pool coolant system required replacement of the control panels and the instrumentation for each of these systems. The NRC staff reviewed the reactor console changes during a site visit on March 3, 2010, and determined these panels showed an adequate display of parameters and indicators on critical components. The NRC staff further concludes the licensee has incorporated the proper interlocks in the controls to preclude any system interactions that could create a hazardous condition or degrade the reactor safety envelope.

The licensee also replaced various display monitors and indicators as a result of the reactor console upgrade. These changes were technological upgrades, and the functionality remained unchanged, with the exception of the compensated ion chambers. They now require a negative compensating voltage. Therefore, the licensee modified the high-voltage distribution system and the high-voltage monitor to accommodate the new instruments.

4.7 Radiation Monitoring Systems

As a result of the modification of the dedicated HVAC system, the licensee replaced the air stack monitor. This was a technological upgrade that increased the sensitivity of both the particulate and gaseous detection capability, which facilitates more accurate monitoring of lower concentrations in the exhaust stream.

4.8 Conclusions

The licensee made changes to the I&C systems at UWNR in accordance with the requirements of 10 CFR 50.59; this did not reduce the ability of the systems to perform safety-related functions. The technological upgrades will enhance the operator’s ability to ascertain the status of system parameters and do not affect the overall functionality of the I&C systems at UWNR. The NRC staff finds the designs acceptable.

5 Electrical Power Systems

5.1 Normal Electrical Power Systems

Normal power to the UWNR comes from an existing 14.8 kilovolt (kV), three-phase, 60-hertz electrical feeder off the campus distribution system that terminates in the basement of the Mechanical Engineering Building. It feeds two circuits. One circuit feeds a 3,000 kilovolt-ampere (kVA) transformer that supplies 480 volt (V) power through a 3,000 ampere (A) breaker to three-breaker-protected (225A, 200A, and 1600A) circuits that supply all UWNR electrical power loads. The second circuit feeds a 1,000 kVA transformer that supplies 208/120V power to panels B1215 and B1215 for support spaces and laboratories within the UWNR. A circuit in B1215 powers a 6,000 VA UPS with 76-minute battery backup capability, which in turn powers panel ELP-2. ELP-2 powers both the stack air monitor and the continuous air monitor. It also powers the fire protection systems for the building.

The 225A breaker feeds panel 4RI, which distributes power to electrical loads throughout the UWNR. Two circuits from panel 4RI power transformers T1 and T2. T1 supplies 240V power to panel R that powers the motor on the thermal column shield door, small motors, and 240V outlets throughout the facility. T2 supplies 208/120V power to Panel 2, which powers small motors and electrical outlets throughout the facility. Panel 2 also supplies power for reactor control and instrumentation, which also has a dedicated UPS that can be powered by batteries for up to 76 minutes that conditions this power.

The 200A breaker supplies 277V power to panel B1135 for the lighting throughout the UWNR and its support spaces. The 1600A breaker supplies 480V power to the dedicated HVAC system, a backup air compressor, and a 75 kVA transformer that supplies 208/120V loads. If normal power is lost or interrupted, both the 200A and 1600A circuits can be powered by the emergency electrical power described in Section 5.2 below.

5.2 Emergency Electrical Power Systems

A loss of normal ac power will initiate an automatic scram of the reactor by causing the control rods to drop, by gravity, into the core, placing the reactor in a safe-shutdown condition. However, electrical power monitors the orderly shutdown of the reactor in an emergency. This emergency power is supplied to the UWNR by a diesel-driven 800 kW 480/277V generator located in Room B1002A of the Mechanical Engineering Building. The generator starts automatically on undervoltage and supplies power through an automatic transfer switch to the 200A and 1600A breakered electrical loads. The licensee installed this shared emergency diesel capability when it renovated the Mechanical Engineering Building between 2004 and 2007. All changes were made in accordance with the 10 CFR 50.59 process.

A review of the modifications to the electrical power systems demonstrated that they have increased system reliability without degrading the capability to provide power to safety-related services. The design change will not interfere with the safe shutdown of the reactor. The modifications are therefore, acceptable to the NRC staff.

5.3 Conclusions

On the basis of its review, the NRC staff concludes that the normal electrical power system at the UWNR facility provides reasonable assurance of adequate protection. In addition, the NRC staff concludes that the loss of normal electrical power will lead to a safe shutdown of the facility, and emergency power is not required to maintain the reactor in a safe shutdown condition.

6 Auxiliary Systems

6.1 Heating, Ventilation, and Air Conditioning Systems

The licensee installed the new UWNR dedicated HVAC system in 2003. All changes were made in accordance with the 10 CFR 50.59 process. Its design consists of a 9,150 standard cubic feet per minute (scfm) air-handling unit that supplies 70 degrees F air at a relative humidity of 30 percent, two exhaust fans with a 9,600 scfm flow rate capability each, and a filter bank consisting of 16 nuclear grade HEPA filters with an efficiency of 99.97 percent for particles 0.1 micron and larger. Normally, only one exhaust fan is operational to maintain a 1.5-in. water gauge pressure, which is sufficient to establish a confinement on the UWNR facility. If the operating fan fails, the standby fan will automatically start on loss of motor function on the other fan. If both exhaust fans were to fail, an interlock automatically shuts down the air-handling unit to prevent a positive pressure from occurring in the reactor room.

For emergency planning and rapid air change purposes, there is an emergency venting mode capability that involves running both exhaust fans simultaneously. Two switches, one in the control room and the other in Room 1200J on the first floor automatically place the HVAC system in this configuration. Since this would place an extreme negative pressure on the facility, a solenoid-operated exhaust makeup damper opens to minimize the negative pressure when one of the two switches is depressed.

A second ventilation system for the experimental facilities is designed to sweep residual Ar-41 into the main ventilation system exhaust duct before the main filter bank. It consists of a 2 horsepower booster fan and adjustable makeup damper that can maintain 960 cfm with all experimental facilities opened simultaneously. The vent on the waste holdup tank discharges into the exhaust duct of this system before the filter, which ensures all potentially contaminated air is monitored before release to the environment through the stack.

The third system is on the pneumatic tube system fume hood. It includes a booster fan and HEPA filter bank and discharges to the main ventilation exhaust duct before the main filter bank. Instrumentation includes a pressure drop across the filter, indication in the control room that the booster fan is running, and an interlock that automatically starts the booster fan when the pneumatic tube system is started. The NRC staff concludes that this design ensures any residual Ar-41 in the tube will be discharged up the stack, rather than leaking out of the fume hood, which could create an airborne radiation hazard in the room.

A review of the HVAC systems showed they are designed to minimize airborne Ar-41 hazards in the facility. The licensee establishes adequate confinement by maintaining the UWNR at a negative pressure relative to the outside environment and other occupied areas of the Mechanical Engineering Building. The new systems contain instrumentation adequate to operate the systems as intended. The new design ensures that radiation doses during normal operations and under potential accident conditions are within 10 CFR Part 20 requirements. These new designs are, therefore, acceptable to the NRC staff.

6.2 Conclusions

On the basis of its review, the NRC staff concludes that the UWNR HVAC systems is adequate to provide for the controlled release of radioactive effluents during normal operation and in the event of abnormal or accident conditions. The UWNR staff, students and members of the public will be adequately protected from airborne radioactive hazards associated with the operation of the reactor

7 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

7.1 Radiation Protection

Activities involving radiation at the UWNR are controlled under the Radiation Protection Program, which must meet the requirements of 10 CFR 20.1101, "Radiation Protection Programs," and minimize radiation exposure. The regulations in 10 CFR 20.1101 specify, in part, that each licensee shall develop, document, and implement a radiation protection program and shall use, to the extent practical, procedures and engineering controls based on sound radiation protection principles to achieve occupational doses and doses to members of the public that are ALARA. The basic aspects of the radiation protection program include occupational and general public exposure limits, surveys and monitoring, and personnel dosimetry. The NRC inspection program routinely reviews radiation protection and radioactive waste management at UWNR.

7.1.1 Radiation Sources

The NRC staff reviewed the descriptions of potential radiation sources, including the inventories of each physical form and their locations. This review of the radiation sources included the identification of potential radiation hazards, as presented in Chapters 11 and 13 of the 2008 SAR (Ref 9).

7.1.1.1 Airborne Radiation Sources

During normal operations, the primary airborne source of radiation is Ar-41. Ar-41 results from irradiation of the air in experimental facilities and the dissolved air in the reactor pool water. The primary means of Ar-41 production is the reactor pool. Other production sources may include Ar-41 that is induced in the air that flows through the reactor thermal column, beam ports, and pneumatic tube system, which are near the reactor core, as well as Ar-41 evolving from the pool water surface. Since the core is cooled by natural convection of the pool water, that heated water rises to the surface of the pool, along with the air dissolved in it. Some of that air containing Ar-41 escapes into the surrounding atmosphere. Since the reactor area has a ventilation system that facilitates removal of Ar-41 through the building exhaust system, the Ar-41 dose to workers is limited. The licensee combines the air from the reactor room exhaust, beam ports, and thermal column, filters it, and vents it to the atmosphere after monitoring it for Ar-41 content. The calculated dose from a maximum release rate of Ar-41 from the stack with Ar-41 production resulting from the reactor operating continuously at 1.0 MW and all four beam ports and the thermal column open showed that a maximally exposed public receptor would receive 0.6 millirem per year (mrem/yr). Over the years, the concentration of Ar-41 above the pool surface during full-power operation has been about one-third of the derived air concentration (DAC) for occupational workers. The Ar-41 activity in the surrounding laboratory areas has been about a factor of 6 below the DAC for occupational workers. TS 3.7.2 specifies the Ar-41 discharge limit for the UWNR as follows:

TS 3.7.2 Effluent (Argon-41) Discharge Limit

> The concentration of Ar-41 in the effluent gas from the facility, as diluted by atmospheric air in the lee of the facility as a result of the turbulent wake effect, shall not exceed 1×10^{-8} μCi/ml averaged over one year.

The licensee calculated the dose from normal operations to a person in the unrestricted area caused by the Ar-41 release from the stack. The licensee assumed a 3.54 meter per second (m/s) wind speed and continuous discharge of all Ar-41 produced. The calculated release rate was 3.3×10^{-9} microcuries per milliliter (μCi/mL), which is well below the 10 CFR Part 20, Appendix B, limit of 1×10^{-8} μCi/ml. The licensee stated, in response to RAI No. 2 in a letter dated June 16, 2010 (Ref 20), regarding the dose calculations, that the maximum concentration to which the public would be exposed, 3.31E-9 curies per cubic meter (Ci/m^3) assumes the ventilation system is inoperable. If the ventilation system is assumed to be operable, the comparable dose rate, as calculated by the EPA COMPLY computer code, would be 0.6 mrem/yr at the nearest residence, about 133 m to the west of the reactor site (approximately 80 m from the west wall of the facility). The analysis used averaged wind rose data ranging from 4.01 to 5.30 m/s specifically compiled for Madison and obtained from the National Weather Service for the period of record from 1948 to 1995.

The licensee also performed an analysis using an updated methodology in the LEU conversion SAR (Ref 10). The maximum ground-level concentration with the ventilation system in operation is 4.78E-10 Ci/m^3 with a resulting dose rate of 3.6 mrem/yr (Ref 20).

The licensee repeated the EPA COMPLY code calculation with the receptor at the site boundary, rather than the nearest residence. The calculated dose was 67 mrem/yr. However, this assumes a maximum hypothetical Ar-41 release rate of 13.3 μCi/s with continuous year-round operation, resulting in a hypothetical annual release of 420 Ci. The highest recorded annual release is only 3.04 Ci, which would result in an EPA COMPLY code calculated dose of only 0.5 mrem/yr.

The analysis demonstrates that the airborne sources released from the UWNR during normal operations do not present a significant exposure hazard. Compliance with the TS 3.7.2 discharge limit ensures that Ar-41 emissions are within the requirements of the regulations. After reviewing the licensee's response, the NRC staff finds TS 3.7.2 acceptable.

Routine effluent releases from the UWNR are within the regulatory limits for a radiation dose to members of the public. The occupational exposure from Ar-41 is also within regulatory limits. The "2008–2009 Annual Operating Report for the University of Wisconsin Nuclear Reactor Laboratory," dated August 10, 2009 (Annual Operating Report), indicates that the average activity discharged through the facility stack was 1.92 Ci, with a concentration at the stack release point of 0.2×10^{-6} μCi/mL, corresponding to a maximum downwind air concentration of about a factor of 40 lower than the 10 CFR Part 20, Appendix B, limit of 1×10^{-8} μCi/mL.

Based on the analysis above, demonstrating that the UWNR routine effluent releases are well within the limitations of 10 CFR Part 20 criteria, the NRC staff concludes that the production and control of Ar-41 are acceptable. The NRC staff further concludes that the TS provides reasonable assurance that, during the continued normal operation of the UWNR, airborne radioactive releases will not pose a significant risk to public health and safety or the environment.

7.1.1.2 Liquid Radioactive Sources

The dose rate from liquid radiation sources at the top of the reactor pool is, for the most part, from N-16, produced when the neutrons from the reactor activate oxygen in the pool water. N-16 has a half-life of 7.4 seconds. Occupational exposure from the N-16 is minimal, because access to the bridge above the core is minimal during reactor operation. The short half-life of N-16 makes exposure to the public negligible. Piping that carries pool water that has been circulated through the reactor core and radioactive corrosion products produced during normal operation make the reactor coolant system another source capable of producing a whole body exposure to personnel.

Measurements of the dose rate at the pool surface, with the reactor operating at full power, is 80–220 mrem/hr with the diffuser system off and 2–4 mrem/hr with the diffuser operating. The diffuser directs a small flow of water downwards and across the top of the core area, which significantly slows down the upward flow of heated water containing the N-16. The dose rate from N-16 decay is thereby reduced in the area near the surface of the pool.

Another potential low-level radiation source would be small quantities of liquid waste that are accumulated from operations and stored before disposal. Radiation exposures from these liquid radiation sources at the UWNR are small and, as a result, do not present a significant hazard to either operating personnel or the public.

The NRC staff concludes that liquid radioactive sources from the continued normal operation of the UWNR are small and access to them is controlled. Therefore, the NRC staff concludes that these sources do not present a significant hazard to the public or operating personnel.

7.1.1.3 Solid Radioactive Sources

Radioactive waste is the only solid radiation source at the UWNR. Solid radioactive waste production at the UWNR reactor facility is minimal. The reactor fuel constitutes the bulk of the solid waste. Nonfuel sources include activated reactor components, resins from the primary water demineralizer, and irradiated samples. The licensee estimates the final radioactivity before it conducts experimental irradiations, so both shielding and storage duration will be known. Solid radioactive waste handling has not resulted in any significant personnel exposure at the UWNR since it began operations.

The NRC staff concludes that solid radioactive sources from the continued normal operation of the UWNR are controlled and have resulted in no significant personnel exposure at UWNR. Therefore, the NRC staff concludes that the control of solid radioactive sources at UWNR is acceptable.

7.1.2 Radiation Protection Program

The regulations in 10 CFR 20.1101(a) require that each licensee develop, document, and implement a radiation protection program. The radiation protection program at UWNR is governed by UW-Madison radiation safety regulations as submitted for the University Broad License, License Number 25-1323-01. The NRC inspection program routinely reviews the radiation protection program at the UWNR. Review of the annual operating reports and NRC inspection reports for UWNR demonstrated that adequate measures are in place to minimize radiation exposure to personnel and provide adequate protection against operational releases of radioactivity to the environment. TS 6.3 covers the radiation protection program, as discussed in Section 9.6.3 of this SER. Based on the above discussion, the NRC staff finds the radiation protection program at UWNR to be acceptable.

7.1.3 ALARA Program

To comply with the regulations in 10 CFR 20.1101, UW has established and implemented a policy that all operations are to be planned and conducted in a manner to keep all exposures ALARA. The program to implement this policy is based on the guidelines of ANSI/ANS-15.11. The program is applied through written procedures and guidelines. The licensee reviews all proposed UWNR experiments and operational procedures for ways to minimize potential exposure to personnel. The UW health physics staff participates in experiment planning to minimize both personnel exposure and the generation of radioactive waste. Additionally, unanticipated or unusual reactor-related exposures are investigated to develop methods to prevent recurrence. The review of controls for limiting access and personnel exposure in the UWNR provides reasonable assurance that radiation doses to the environment, the public, and facility personnel will be ALARA. The UWNR management adequately supports the ALARA program. TS 6.3 covers the ALARA program, as discussed in Section 9.6.3 of this SER.

The NRC staff concludes that the UWNR ALARA program complies with 10 CFR 20.1101, is acceptable, and provides reasonable assurance that radiation exposure will be maintained ALARA for all facility activities.

7.1.4 Radiation Monitoring and Surveying

The regulations at 10 CFR 20.1501(a) state that each licensee shall make or cause to be made, surveys that have the following characteristics:

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

(2) Are reasonable under the circumstances to evaluate–
(i) The magnitude and extent of radiation levels; and
(ii) Concentrations or quantities of radioactive material; and
(iii) The potential radiological hazards.

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.

The licensee has a comprehensive set of portable radiation survey instrumentation that covers, with sufficient ranges, the various types of radiation that may be encountered at the UWNR; this includes radiation monitors, as specified in TS 3.7.1.

TS 3.7.1 Monitoring System

The reactor shall not be operated unless the radiation monitoring channels listed in Table 3.7.1 are operable.

Table 3.7.1 Radiation Monitoring Systems

	Radiation Monitoring Channels*	Function	Number
1	Area Radiation Monitor	Monitor radiation levels within the reactor room	3
2	Exhaust Gas Radiation Monitor	Monitor radiation levels in the exhaust air stack	1
3	Exhaust Particulate Radiation Monitor	Monitor radiation levels in the exhaust air stack	1
4	Environmental Radiation Monitors	TLD dosimeters in area surrounding the stack	4

* For periods of time, not to exceed 1 week, for maintenance to the radiation monitoring channels, the intent of this specification will be satisfied if they are replaced with portable gamma sensitive instruments having their own alarms or which shall be kept under visual observation.

In response to RAI No. 17 in a letter dated June 16, 2010 (Ref 20), the licensee proposed to limit to 1 week the time during which it may replace the radiation monitoring channels with portable instruments. By adding a time limit of 1 week, where no time limit previously existed, the proposed modification ensures that the radiation monitoring channels are restored to an operable condition in a timely fashion. This time period is consistent with guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007, Section 3.7.1. On the basis of its review, the NRC finds the licensee proposal acceptable.

The NRC staff notes that all radiation monitors except the environmental radiation monitors scram the reactor and activate the building evacuation system in the event that radiation levels exceed a preset value. The radiation monitors are set to alarm at radiation levels just above levels experienced during normal reactor operation, and can be adjusted by agreement between the reactor operations supervisor and the UW senior health physicist, based on anticipated changes from normal radiation levels. This method provides the necessary operational flexibility to perform experiments while providing assurance that occupational doses are maintained in accordance with ALARA principles. The NRC staff concludes that TS 3.7.1 requires sufficient monitors to evaluate potential radiation hazards. Routine effluent releases are within regulatory limits and the discussion in Section 8 of this SER shows that the consequences of accidents are acceptable.

TS 4.7.1 contains the following surveillance requirements for the radiation monitoring systems:

TS 4.7.1 Radiation Monitoring Systems

1. The radiation monitoring and stack monitoring systems shall be calibrated annually and shall be verified to be operable by monthly source checks or channel tests.

2. The environmental dosimeters shall be evaluated on a quarterly basis.

Monthly operability checks and channel tests, combined with annual surveillances, constitute a schedule sufficient to identify any changes to the operating characteristics of the monitoring systems. The NRC staff reviewed the information provided in 2008 SAR (Ref 9), Chapter 11 and finds that finds that the surveillance frequency is consistent with guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of its review, the NRC staff concludes that the surveillance requirements in TS 4.7.1 for the radiation monitoring channels are acceptable.

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

7.1.5 Radiation Exposure Control and Dosimetry

Thermoluminescent dosimeters (TLDs) monitor personnel exposure and are assigned to individuals who have the potential to be exposed to radiation. The licensee uses portable equipment to perform radiation surveys and issues personnel protective equipment as needed. It has facilities and equipment to decontaminate persons, if required. Operating procedures govern the use of this equipment.

The licensee uses survey meters to measure dose rates from radiation fields, and those measured rates are posted where required. These provisions provide assurance that external and internal radiation monitoring of all individuals needing to be monitored meets the requirements of 10 CFR Part 20 and the goals of the facility's ALARA program. A review of the annual operating reports revealed that the highest annual whole body exposure received by a facility employee was 26 mrem. The highest annual extremity exposure for that period was 117 mrem. The highest dose received by a member of the public visiting the reactor lab was 0.1 mrem. All UWNR staff and visiting members of the public received significantly lower radiation doses than the limits in 10 CFR 20.1201, "Occupational Dose Limits for Adults," and 10 CFR 20.1301, "Dose Limits for Individual Members of the Public."

The licensee maintains personnel exposure records and effluent and environmental monitoring readings for the life of the UWNR. The review verified that engineered radiation exposure controls were located in appropriate areas of the facility and were adequate to alert workers to radiation hazards within the facility.

The NRC staff reviewed the licensee's exposure control and dosimetry program. Based on historically low doses and a review of facility programs, the NRC staff concludes that personnel exposures at UWNR are controlled through satisfactory radiation protection and ALARA programs. Therefore, the NRC staff concludes that the licensee's control of its exposures and dosimetry program are acceptable.

7.1.6 Contamination Control

Contamination surveys are performed on a daily, weekly, or, at a minimum, monthly basis, depending on the frequency that radioactive material is used or handled. Written procedures control the handling of any radioactive material within the UWNR. Employees are trained to work with radioactive materials, including learning ways to limit its spread while entering and exiting an area containing radioactive material. The facility surveys have routinely shown no detectable contamination in non-radiological areas. This review showed that adequate controls exist to prevent the spread of radiological contamination within the facility. Based on a review of UWNR Radiation Protection Program, Contamination Control described in the 2008 SAR, Section 11.1.6 (Ref 9), and a history of satisfactory contamination control, the NRC staff concludes that adequate controls exist to prevent the spread of contamination within the facility.

7.1.7 Environmental Monitoring

Environmental monitoring at UWNR is part of the Radiation Monitoring Program and is described in the 2008 SAR (Ref 9), Chapter 11.1.7. The environmental monitoring consists of taking various samples at 25 different locations, which include unrestricted areas adjacent to the UWNR and outside the Mechanical Engineering Building. TLDs measure direct gamma radiation, and the licensee evaluates these measurements on a quarterly basis. The RSC audits the environmental monitoring program annually to ensure that it contains an adequate number of samples and locations and that the frequency of collection is sufficient to provide an early indication of any environmental impact caused by reactor operations. To maintain awareness of this impact, UW monitors effluent concentrations at the point of release and routinely samples pool water to ensure any pool leakage is within effluent limits. Table 7.1 of the 2008-2009 Annual Operating Report indicates the dose a person would have received if continuously present in the indicated area for the entire 2008 calendar year (Ref 15) . All locations recorded dose rates well below the limits in 10 CFR Part 20.

Table 7-1 Annual Environmental Monitoring Dose Data

Location	Annual Dose (mrem)
Dose Inside Reactor Laboratory Stack	16.0
Highest Dose in Occupied* Nonrestricted Area	46.0
Average Dose in all Nonrestricted Areas	21.9

Based on its review, the NRC staff concludes that the environmental monitoring program is sufficient to assess the radiological impact of the UWNR on the environment.

7.2 Radioactive Waste Management

The Radioactive Waste Management Program is described in the 2008 SAR (Ref 9), Chapter 11.2. The purpose of radioactive waste management is to minimize the generation of radioactive waste and ensure that it receives proper handling, storage, and disposal. The heath physics staff oversees, and procedures control, all operations for handling radioactive waste at UWNR.

7.2.1 Radioactive Waste Management Program

The UWNR Radioactive Waste Management Program is a campus program referenced in the 2008 SAR (Ref 9) as the Radiation Safety Regulations, University of Wisconsin-Madison, Revision 2, January 1997. UWNR operating procedures adequately described the movement process and practices for release of radioactive waste from controlled to uncontrolled areas. The NRC staff reviewed the UWNR radioactive waste release practices during a site visit on March 3, 2010 as well as UWNR annual operating reports and determined that the UWNR demonstrated reasonable assurance that radiological releases from the facility will not exceed applicable regulations nor pose unacceptable radiation risk to the environment and the public. This review also verified that adequate controls are in place to prevent uncontrolled personnel exposures from radioactive waste operations and that they provide the necessary accountability to prevent any potential unauthorized release of radioactive waste.

7.2.2 Radioactive Waste Control

The UWNR Radioactive Waste Control program is described in the 2008 SAR (Ref 9), Chapter 11.2.2. Low-level solid waste from laboratory experiments or disposable protective clothing items are accumulated and stored in authorized containers. Activated equipment and irradiation samples are properly stored in controlled areas to be reused or to decay to low-level activity limits. When filled, the low-level waste containers are sealed and stored until a licensed carrier ships them off the campus to a licensed facility for disposal. Adequate procedures are in place to: monitor the radiation exposure from waste storage areas within the facility; perform required handling operations, such as packaging and transfer; and prepare the proper documentation associated with such shipments.

7.2.3 Release of Radioactive Waste

The UWNR Release of Radioactive Waste program is described in the 2008 SAR (Ref 9), Chapter 11.2.3. Normal operation of the UWNR does not produce significant liquid radioactive waste. However, small quantities are periodically generated by minor leakage and sampling of the reactor pool and the primary coolant cooling equipment. The licensee collects and stores these liquid wastes in a 2000-gallon holdup tank until it determines final disposition. It periodically releases nonalpha-emitting liquid radioactive waste from the retention tank to the sanitary system, in accordance with an approved discharge permit issued by the State of Wisconsin. The licensee monitors all radioactive liquids before discharging them to the environment (sanitary sewer) and controls their release by operating procedures. For liquids being directly discharged to the sewer system, the maximum activity before dilution is 2×10^{-8} μCi/mL. For higher activities, further analysis identifies isotopic content. Discharge limits for the isotopes identified are in accordance with the limits for liquid effluents in 10 CFR Part 20, Appendix B, Table 3. Fresh water can be added to the holdup tank if dilution or ph adjustment is necessary before discharge.

TS 4.7.2 specifies the requirements that apply to all gaseous and liquid discharges from the UWNR.

TS 4.7.2 Effluents

1. Liquid radioactive waste discharged to the sewer system shall be sampled for radioactivity to assure levels are below applicable limits before discharge. Results of the measurements shall be recorded and reported in the Annual Report.

2. The total annual release of gaseous radioactivity to the environment shall be recorded and reported in the Annual Report.

In response to RAI No. 24 in a letter dated June 16, 2010 (Ref 20), the licensee stated that it based TS 4.7.2 on the standard ANSI/ANS-15.1-1990 approved at the time it submitted its license renewal application, which did not specifically address offsite monitoring. However, the

UWNR environmental monitoring program includes the following offsite monitoring: the licensee monitors all liquid releases to the sewer and air effluents to the stack and verifies that they are below effluent limits. The licensee routinely analyzes pool water for radioactivity, according to TS 4.3; monitors any water makeup beyond normal evaporative losses; and verifies that it is below effluent limits. The licensee monitors environmental TLD badges located at various positions offsite to monitor exposure. The NRC staff reviewed the licensee's RAI response and finds that offsite monitoring performed by the licensee is adequate. On the basis of its review, the NRC staff concludes that TS 4.7.2 (1) is acceptable.

Sampling requirements ensure that all discharges are consistent with the UWNR ALARA program and are within 10 CFR Part 20 limits. The specific annual reporting requirement related to radioactive effluent releases is consistent the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. Therefore, TS 4.7.2 (2) is acceptable to the NRC staff.

The NRC staff concludes that liquid radioactive sources from the continued normal operation of the UWNR and the controls currently in place for releases to the sewer system provide reasonable assurance they will not pose a significant risk to public health and safety or the environment.

7.3 Conclusions

On the basis of its evaluation of the information presented in the licensee's 2008 SAR, observations of the licensee's operations, review of annual operating reports, and results of the NRC inspection program, the NRC staff concludes as follows:

(1) The UWNR radiation protection program complies with the requirements in 10 CFR 20.1101(a), is acceptably implemented, and provides reasonable assurance that the NRC staff, the environment, and the public are protected from unacceptable radiation exposures. The radiation protection program is acceptably staffed and equipped. The radiation protection staff has adequate lines of authority and communication to carry out the program.

(2) The UWNR ALARA program complies with the requirements of 10 CFR 20.1101(b). A review of controls for radioactive material in the UWNR provides reasonable assurance that radiation doses to the environment, the public, and facility personnel will be ALARA.

(3) The results of radiation surveys carried out at the UWNR, doses to the persons issued dosimetry, and results of the environmental monitoring program help verify that the radiation protection and ALARA programs are effective.

(4) The licensee adequately identifies and describes potential radiation sources and sufficiently controls them.

(5) Facility design and procedures limit the production of Ar-41 and N-16 and control the potential for facility staff exposures. Conservative calculations of the quantities of these gases released into restricted and unrestricted areas give reasonable assurance that doses to UWNR staff and public will be below applicable 10 CFR Part 20 limits.

(6) 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.

8 ACCIDENT ANALYSES

8.1 Accident Analysis Initiating Events and Determination of Consequences

The LEU conversion SAR analyzed three hypothetical accidents: the MHA, the insertion of excess reactivity, and a loss-of-coolant accident (LOCA). The LEU conversion SAR (Ref 10) also considered other accidents that are identified in NUREG-1537.

8.1.1 Maximum Hypothetical Accident

To support the conversion from HEU to LEU fuel, the licensee performed a complete reanalysis of the MHA. The UWNR MHA is defined as an assumed cladding failure of one highly irradiated fuel element in air after continuous operation at 1.3 MW, followed by the instantaneous release of the noble gases and halogen fission products. Since 1.0 MW is the licensed power and the scram set point of the UWNR is 1.25 MW, 1.3 MW is representative of the maximum power accounting for power level uncertainty. The MHA is based on a single fuel rod cladding failure in air in the reactor room. Boundary conditions and assumptions for the most conservative cases analyzed for both the HEU and LEU cores included the following:

- The reactor core operates continuously at 1.3 MW for 50 MWd insuring that the radioactive halogen and noble gas fission products are at saturation.

The licensee used the ORIGEN2 computer code Version 2.1 with the PWRUS cross section library to calculate the core fission product inventories. This code is well known and used throughout the nuclear industry to calculate core fission product inventories. The results presented in the LEU conversion SAR showed that the fission product inventories for the LEU core are generally higher than the revised HEU core for most isotopes. This difference is caused by the increased power density as a result of fewer fuel bundles in the core. Table 13.1.1 of the LEU conversion SAR contains the fission product inventories, as calculated (Ref 10).

- All of the gap activity is released into the reactor room and is instantly mixed uniformly in the reactor room air space with a volume of 2000 m^3. All of the fuel pin air gap noble gases and halogen inventory become instantaneously released and airborne (except for the pool water activity calculations, which assume all of the gap activity released is dissolved in the pool water).

There will be a time delay between the time of fuel rod failure and fission product release to the building environment. This assumption ignores radioactive decay during the finite mixing time.

- The fuel element has a maximum temperature of less than 600°C. After averaging over the fuel volume and temperature a release fraction is applied to arrive at the total gap activity. The temperature dependent fission product release fraction is based on measurements performed by GA (Ref 18).

The fission product release fraction from the fuel into the fuel pin air gap is a function of the fuel temperature. Since the previous HEU SAR assumed a maximum fuel temperature of 600 degrees C, the calculation in Section 4.7.4 of the LEU conversion SAR (Ref 10) determined the maximum fuel temperature for the revised HEU core and LEU core parametric analysis. The review of the analysis showed a maximum fuel temperature of 576 degrees C for the HEU core and 596 degrees C for the LEU MOL core. The licensee calculated the release fractions on this basis and included the released fission product inventories in Table 13.1.2 of the LEU conversion SAR (Ref 10). In response to RAI No. 43 in a letter dated April 10, 2009, (Ref 11), the licensee revised the fission product release fraction based on a more realistic, integrated volumetric and temperature average through the fuel rod. A complete update of the MHA doses was provided in UWNR letter dated February 8, 2011 (Ref 21).

- Building ventilation system is unavailable. Fission products leak to the outside environment as ground release,

The availability of the ventilation system would reduce the radiation dose to the occupational staff and the public. A ground release is a conservative assumption as opposed to an elevated release through the stack increasing the occupational and public dose. This assumption does not account for the removal of any radioisotope due to radioactive decay and/or from plating out of the air. It also does not allow for any sealing of the reactor room.

- If the ventilation system is available, then all of the noble gases and halogens from the gap are released to the environment through the facility stack.

The availability of the ventilation system would reduce the dose to occupational staff and the public. The radiation dose values are bounded by the scenario assuming that the system is unavailable. The gaseous material that is discharged from the ventilation system is assumed to be diluted by atmospheric air and dispersed by the prevailing wind in the direction of the receptor. Ground level concentrations and external dose levels were calculated using a Gaussian plume dispersion model.

- It will take 5 minutes to evacuate persons from the reactor room and additional 5 minutes to evacuate the building. Building occupants outside the reactor room would leave the building in 10 minutes.

The use of a five/ten-minute occupational and building occupant exposure is considered reasonable since evacuation drills conducted by the facility have demonstrated the ability to evacuate personnel from the reactor room within the five minute and the building within the ten minute timeframe.

The NRC staff reviewed the analytical modeling, input data, and the methodology used in deriving occupational and public radiation dose values for both the ground and stack releases. The NRC staff concludes that the methodology is consistent with the guidance provided in NUREG-1537 and adequate to calculate occupational and public radiation doses. Assumptions and boundary conditions were consistent with accepted nuclear industry practices.

Using these inventories, the licensee calculated doses to three receptors: occupational doses, building occupant doses, and public doses in the unrestricted area outside the reactor building.

1) Occupational dose - The occupational dose for a member of the operating staff was calculated by the licensee in the reactor room assuming an evacuation time of 5 minutes from the reactor room. Boundary conditions for these calculations included assuming failure of the hottest fuel element, incorporating the calculated release fractions, and assuming the

reactor room has a volume of 2,000 m^3 (an added conservatism, since this is at least 20 percent smaller than the actual volume). With these conservative assumptions, the maximum whole-body (which occurs for the MOL condition) and thyroid doses are below the occupational limit set in 10 CFR 20.1201. Table 8-1 gives the occupational doses.

Table 8-1 MHA Occupational Dose*

	External Dose (mrem)	**Thyroid Dose (mrem)**	**TEDE (mrem)**
Previous HEU SAR	10.0	N/A	N/A
Revised HEU Analysis	20.3	2,110	83.7
LEU BOL Analysis	36.1	3,730	148
LEU MOL Analysis	34.6	3,670	145
LEU EOL Analysis	27.4	2,960	116

From Ref 21.

2) Non-occupational, public dose - the licensee analyzed the following situations:

   a) Person located inside the Mechanical Engineering Building but outside the reactor confinement,

The UWNR is located in a reactor confinement room inside the Mechanical Engineering Building, which has several floors with non-restricted access that are occupied by the public not associated with the UWNR. Therefore, the other building occupants are considered members of the general public. For calculating the dose to a member of the public in the Mechanical Engineering Building, the licensee made the following additional assumption:

- Uniform and immediate dispersal of fission products throughout the reactor confinement plus the auxiliary support space; and, after the confinement space is filled, uniform dispersal of fission products throughout the first four floors of the Mechanical Engineering Building.

The first four floors are connected to an open-air atrium in the central wing of the building, which allows for ready mixing of the building air. The licensee's calculations excluded the volume of the fifth floor of the building, because there is no ready flow path for mixing the building air. This is conservative and provides a higher concentration of fission products when calculating the dose rates to all of the building's occupants. Actual dose rates to building occupants would be lower if the volume of the fifth floor were included in the calculations. Table 8-2 gives the dose to a member of the public located inside the Mechanical Engineering Building outside the reactor room:

Table 8-2 MHA Building Occupant Dose without Ventilation*

	External Dose (mrem)	Thyroid Dose (mrem)	TEDE (mrem)
Previous HEU SAR	N/A	N/A	N/A
Revised HEU Analysis	1.26	131	5.18
LEU BOL Analysis	2.24	231	9.16
LEU MOL Analysis	2.14	227	8.95
LEU EOL Analysis	1.70	183	7.20

from Ref 21.

b) Ground release - a person standing by the wall just outside the Mechanical Engineering Building. This category also includes the location of nearest residence due to its close proximity to the Building

The fission products released inside the reactor room may leak out into the outside atmosphere as a ground release. The ground release was modeled with a conservative atmospheric dispersion factor of $\chi/Q=0.01$ with the release lasting 1 hour allowing the leakage of all fission products from the reactor room and building to the outside. Table 8-3 gives the dose to a member of the public located just outside of the Mechanical Engineering Building including at the location of the nearest residence:

Table 8-3 MHA Public Dose Outside Building without Ventilation*

	External Dose (mrem)	Thyroid Dose (mrem)	TEDE (mrem)
Previous HEU SAR	0.153	1,019	30.7
Revised HEU Analysis	1.35	141	5.58
LEU BOL Analysis	2.41	249	9.86
LEU MOL Analysis	2.31	245	9.64
LEU EOL Analysis	1.83	198	7.75

* from Ref 21.

c) Stack release - a person located at the maximum ground level concentration due to a stack release, which is approximately 150m from the Mechanical Engineering Building.

In this scenario the ventilation system is expected to be in operation and the fission products would be transported to the outside atmosphere from the reactor room through the building stack. For conservatism the licensee assumed that the occupational dose to the workers would remain the same as presented in Table 8-1. The dose to occupants of the Mechanical Engineering Building would be eliminated as the fission products exit the building through the stack. The dose to a person outside the building would be reduced because the ventilation would reduce the concentration of radioactivity released in the air through dilution. The MHA case with the ventilation system operating results in an elevated stack release to the environment. Downwind dispersion is calculated using a Gaussian plume model, assuming an urban environment. Maximum dose values were calculated under different Pasquill stability classes using an average wind speed of 3.4 m/sec. The most limiting concentration value, Pasquill Stability Class C at 150 m from the facility provided the basis for the public dose values. Table 8-4 shows the results for the MHA case with ventilation:

Table 8-4 MHA Public Dose Outside Building with Ventilation*

	External Dose (mrem)	Thyroid Dose (mrem)	TEDE (mrem)
Previous HEU SAR	N/A	N/A	N/A
Revised HEU Analysis	0.0049	0.506	0.0200
LEU BOL Analysis	0.0087	0.893	0.0354
LEU MOL Analysis	0.0083	0.879	0.0347
LEU EOL Analysis	0.0066	0.710	0.0279

* from Ref 21.

The dose to an occupant of the Mechanical Engineering Building shown in Table 8-2, and to an individual in the unrestricted zone outside the building, shown in Table 8-3 without and Table 8-4 with the ventilation system operational, would be below the 100 mrem yearly dose limit for an individual member of the public established in 10 CFR 20.1301.

The licensee provided other calculations, for what it termed near-MHA conditions, in which it assumed that the reactor pool was not drained. In these cases, the bromine and iodine isotopes were assumed to be scrubbed. The scrubbing resulted in lowering the whole-body occupational dose and dose to the members of the public outside the building by slightly more than a factor of 2 and the thyroid doses by almost a factor of 10.

The NRC staff reviewed the MHA for the LEU core. Because the calculated occupational and public radiation exposures for the MHA are within the limits of the current 10 CFR Part 20, the staff finds the results of the MHA to be acceptable.

8.1.2 Insertion of Excess Reactivity

The licensee hypothesized that a pulsing accident from full power would be initiated by the ejection of the transient rod or the ejection of an experiment up to the value of the transient rod (1.4% $\Delta k/k$, or $1.79). The analysis assumed the reactor was operating at 1.3 MW and the water inlet temperature was 54 degrees C. It assumed that the blades did not move for 2 seconds and then instantaneously were set at the fully inserted position, including the transient rod, which was not in the pulse mode. The licensee calculated the temperature of the highest power fuel pin. For the LEU fuel, the maximum temperature was calculated to occur at EOL and reached 997 degrees C within 2 seconds. The peak fuel temperature for this insertion of excess reactivity accident is 167 degrees C above the 830 degrees C pulse limit recommended by GA but 153 degrees C below the maximum fuel safety limit of 1,150 degrees C.

The NRC staff reviewed the insertion of excess reactivity accident for the LEU core during the conversion analysis (Ref 5, Section 2.6). This accident scenario is extremely unlikely and requires the failure of an operator to follow written instructions, together with the failure of an interlock. Even in these unlikely scenarios with extremely conservative assumptions, the results of the BLOOST code indicate that the TS safety limit of 1,150 degrees C would not be exceeded.

8.1.3 Loss of Coolant

Instrumentation and the reactor staff will detect tank failures caused by corrosion or other failures that lead to a slow loss of water. Adding water to the pool would compensate for slow leaks. In the event of a massive loss of water from the reactor tank, the reactor would scram upon receipt of the high- or low-water level alarm. The licensee has estimated that it would take 836 seconds to lose water cooling the fuel in case of a sheared and open beam port. The reactor pool water level is continuously monitored; alarms would sound if the level varied by 2 in. from normal. These alarms are monitored in the control room. Receipt of this alarm would initiate the emergency response.

The licensee calculated the consequences of the assumption that all water was removed from the pool. The licensee states that Standard 8.5/20 LEU fuel will maintain its integrity and not release fission products at temperatures up to 900 degrees C; for the 30/20 LEU fuel, the corresponding temperature is stated as 940 degrees C. On the basis of its review, the NRC staff finds that although the cladding temperature may be greater than 500 degrees C, as could be expected during a LOCA with air cooling, no fuel damage is expected, if the fuel temperature does not exceed 900 degrees C. Therefore, the risk to the public would be from direct gamma radiation, not from fission product release.

The LOCA analysis calculated what would happen under the limiting assumption that water was removed from the pool. In calculating the fuel temperature in a LOCA, the transient was divided into two portions: a water-cooled portion and an air-cooled portion. The water-cooled portion spans the period from the start of the LOCA to the time when the pool water reached the top of the fuel. As soon as the pool water reached the top of the fuel, the remaining water in the vessel was assumed to disappear immediately. This initiated the start of the air-cooled portion of the LOCA. The licensee's acceptance criterion for the maximum air-cooled fuel temperature is given as 950 degrees C. This value is long-standing, and the NRC staff has previously found it to be acceptable (NUREG-1282).

In a complete LOCA, the water in the reactor tank is assumed to drain completely. For this scenario, the maximum fuel temperature was calculated by using RELAP5-3D, assuming air cooling of the fuel rods by natural convection. In a partial LOCA, the water initially drained down to the bottom of the beam port. Assuming the reactor operated continuously at 1.02 MW for 7 days a week, maximum fuel temperatures for the hot rod (19.7 kW) were calculated to be 585 degrees C for the complete LOCA. The licensee stated that analyses have shown that a complete LOCA is the more limiting and bounds a partial LOCA scenario and that the maximum fuel temperature remains below the maximum air-cooled temperature limit of 950 degrees C.

These conclusions are supported by studies performed for TRIGA fuels. As long as the operating power is less than 1.5 MW(t), the fuel cladding should not breach during a LOCA, per GA-6596, "Simulated Loss-of-Coolant-Accident for TRIGA Reactors," August 1965.

Based on the review of the above analysis and experimental results, the NRC staff concludes that neither full nor partial LOCAs would damage the reactor fuel.

The main consequence from a catastrophic loss of coolant is the gamma ray dose from the exposed core. The licensee calculated direct gamma dose rates at the top of the pool, at the console, and above the reactor in an adjacent classroom. The direct gamma dose rates from the unshielded core are high; however, evacuation of the reactor room and the building, and exclusion of the public from the vicinity of the facility boundary, would ensure that the 10 CFR Part 20 dose guidelines to the workers, building occupants, and the public are satisfied.

The licensee described the general evacuation procedure, UWNR 150, “Reactor Accident, Fission Product Release, or Major Spill of Radioactive Materials,” identifying the areas to be evacuated, and the evacuation alarm system, which alerts members of the public to evacuate. The members of the public are instructed by evacuation notices, which are posted throughout the building. The licensee performs an evacuation drill annually to verify operation of the evacuation alarm system and to provide training for both reactor staff and building occupants. Historically, evacuation drills have demonstrated that building occupants are evacuated within 5 minutes.

In the event of an evacuation alarm, the operating staff verifies that the site boundary is evacuated, with assistance from the UW Police Department (UWPD) as necessary. The operating staff and UWPD control access to the evacuated site boundary.

The NRC staff reviewed the LOCA analysis for the LEU core as its evaluation is documented in the NRC SER for the LEU conversion (Ref 5, Section 2.6). The NRC staff finds that because the maximum fuel temperature is below the air-cooled limit of 950 degrees C and doses to the public at the site boundary are within the requirements in 10 CFR Part 20, the results for the LOCA analysis are acceptable.

8.1.4 Loss-of-Coolant Flow

The loss-of-coolant flow accident is not applicable to the UWNR. Natural convection cools its core.

8.1.5 Mishandling or Malfunction of Fuel

This accident is included as the MHA, when combined with failure of the ventilation system and loss of pool water. The effect of a fuel clad failure with a normal pool water level and with the ventilation system operating would be bounded by the MHA and have no significant effect on the building occupants or the public.

8.1.6 Experiment Malfunction

The licensee controls and limits experiments to prevent a step change in reactivity greater than 1.4% $\Delta k/k$ for fixed experiments and 0.7% $\Delta k/k$ for movable experiments. It evaluates experiments that contain chemical and explosive hazards during the review process and limits experiments containing fissionable material to ensure releases will be lower than for the MHA. Therefore, experiment malfunction will not result in consequences more severe than those listed in other parts of this chapter.

8.1.7 Loss of Normal Electrical Power

Loss of normal electrical power will cause the reactor to shut down and will not result in any release of radioactive material. In the event of a loss of electrical power, all control rods would insert into the core automatically by gravity caused by a loss of power to the electromagnets and to the transient rod, the three-way solenoid valve that holds the control rods in position.

Upon loss of electrical power, the primary and intermediate coolant pumps would stop. Reactor decay heat would be dissipated through natural circulation in the primary coolant. There is sufficient coolant in the reactor pool to absorb the decay heat from the reactor without the need for the primary or intermediate cooling system. The UWNR has a number of backup electrical power systems that power the reactor console and control instrumentation, allowing the operator to monitor the shutdown of the reactor even if electrical power is lost. Additionally, licensee staff can verify that the control elements are fully inserted by visual inspection at the reactor bridge. The NRC staff concludes that loss of normal electric power poses little risk to the health and safety of the public or the UWNR staff.

8.1.8 External Events

The licensee stated that floods, hurricanes, tornados, and earthquakes are not considered to pose a threat to the reactor. Aircraft collisions are considered unlikely and not able to breach the concrete shield at core level. The threat of malevolent acts are mitigated by the security plan and implementing procedures. The NRC staff concludes that the potential consequence of any external events would be bounded by the MHA.

8.1.9 Mishandling or Malfunction of Equipment

The licensee considered the possibility that water from the reactor or from the radioactive liquid waste storage tanks could affect the city water supply. The floor drains in the reactor room empty into a holding tank. If the entire contents of the reactor pool were released into the room, some water could get into the sewer system through the drain thimble. There are four ways water may leave the reactor room: (1) by flow pumped from the radioactive waste storage tanks through the elevated drain thimble provided for emptying the tanks, (2) directly into the drain thimble, should the pool be completely ruptured, (3) by loss through the floor and into the ground, (4) by rupture of the liquid radioactive waste storage tank directly into the soil under the laboratory floor. The licensee considered these four failure modes in the 2008 SAR (Ref 9) and indicated that none present a credible hazard to the municipal water supply. The NRC staff reviewed the analysis performed by the licensee and finds it acceptable.

8.2 Conclusions

The NRC staff reviewed the radiation source term and MHA calculation for the HEU-to-LEU fuel conversion at the UWNR. The review of the calculation, including assumptions, demonstrated that the inventory of radioactivity assumed and other boundary conditions used in the analysis are acceptable. While the radiological consequences to the public and occupational workers at the UWNR from a postulated MHA for the LEU-fueled reactor are expected to be slightly higher than the radiological consequences calculated for the HEU-fueled reactor, these doses are in conformance with the requirements in 10 CFR Part 20. As a result of this review, the NRC staff concludes that, from a radiological standpoint, continued operation of the reactor poses no undue risk to the UWNR staff or the public from the MHA.

The licensee did not identify any reactivity addition accidents that are not bounded by those analyzed for the LEU-fueled reactor (Ref 10). The design features and administrative restrictions that prevent accidental pulsing from occurring at full power and that preclude damage if such pulsing occurs continue to exist for the LEU core. Therefore, risk to the health and safety of the reactor staff and to the public does not increase above that previously found acceptable for the HEU core from reactivity addition accidents.

The review of the calculations, including assumptions, demonstrated that a LOCA would not result in unacceptable fuel element temperatures. The radiological consequences to the public and occupational workers at the UWNR from a postulated LOCA for the proposed LEU-fueled reactor are expected to be similar to that of the HEU-fueled reactor, which were in conformance with 10 CFR Part 20 requirements. As a result of this review, the NRC staff concludes that air cooling the reactor after a LOCA is sufficient to prevent cladding failure and that continued operation of the reactor during the license renewed period poses no undue risk to the UWNR staff or to the public.

9 TECHNICAL SPECIFICATIONS

As part of the 2008 SAR (Ref 9), the licensee submitted a new version of the proposed TS for the license renewal. After the 2008 SAR was submitted, the facility converted to LEU fuel and implemented Amendment 17, which included changes to the original version of the UWNR TS. In response to RAI No. 5 in a letter dated June 16, 2010 (Ref 20), the licensee incorporated the TS changes from Amendment 17 into the proposed TS from the 2008 SAR and included a complete version of the proposed TS as Attachment 1 to the RAI response letter dated June 16, 2010 (Ref 20). In response to the NRC's RAI, UWNR provided responses in letters dated November 22, 2010 (Ref 27) and December 8, 2010 (Ref 28), respectively. An updated version of the UWNR TS was submitted to the NRC by UWNR staff on January 28, 2011 (Ref 29). The following chapter documents the formal review of the proposed TS for UWNR.

9.1 TS 1.3 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:

COLD CRITICAL:
The reactor is in the cold critical condition when it is critical in the reference core condition.

CONFINEMENT:
Confinement is an enclosure of the facility that is designed to limit the release of effluents between the enclosure and its external environment through controlled pathways. This is room 1215 of the Mechanical Engineering Building.

EXCESS REACTIVITY:
Excess reactivity is that amount of reactivity that would exist if all control elements were fully withdrawn from the core from the point where the reactor is exactly critical (k_{eff}=1) in the cold critical condition.

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

REACTOR SECURED:
The reactor is secured when:

(a) Either there is insufficient moderator available in the reactor to attain criticality or there is insufficient fissile material present in the reactor to attain criticality upon optimum available conditions of moderation and reflection or,

(b) The following conditions exist:

a. All control elements are fully inserted, with the exception of the regulating blade in the event of an emergency,
b. The reactor is shut down,

c. The consol key switch is in the “off” position and the key is removed from the console and under the control of a licensed operator or stored in a locked storage area, and

d. No work is in progress involving core fuel, core structure, control elements, or control element drives unless the work on the drive can not move the control element, and,

e. No experiments are being moved or serviced that have, on movement, a positive reactivity worth exceeding 0.7% $\Delta k/k$.

REFERENCE CORE CONDITIONS:

The reactor is in the reference core condition when the fuel and bulk water temperatures are at ambient conditions and the reactor is xenon free.

REPORTABLE OCCURANCE:

A reportable occurrence is any of the following:

1. Operation with any safety system setting less conservative than specified in the technical specifications;

2. Operation in violation of a Limiting Condition for Operation listed in Section 3;

3. 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;

4. Any unanticipated or uncontrolled change in reactivity greater than 7% $\Delta k/k$, excluding reactor trips from a known cause;

5. 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 a condition which could result in a violation of technical specifications, or applicable regulations;

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

SCRAM TIME:

The time from the initiation of a scram signal to the time that the slowest scrammable control element reaches its fully inserted position.

SHALL, SHOULD, and MAY:

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

These are standard definitions used in research reactor TS, consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. The licensee’s proposed additions and modifications of these TS definitions are therefore, acceptable to the NRC staff. Other TS 1.3 definitions are consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007 and are therefore, acceptable to the NRC staff.

9.2 Safety Limits and Limiting Safety System Settings

9.2.1 TS 2.1 Safety Limits

See Section 2.5.2 of this SER.

9.2.2 TS 2.2 Limiting Safety System Settings

See Section 2.5.2 of this SER.

9.3 Limiting Conditions of Operation

9.3.1 TS 3.1 Reactor Core Parameters

See Section 2.2 of this SER.

9.3.2 TS 3.2 Reactor Control and Safety Systems

These LCO specifications apply to a number of reactor control and safety system functions that must be operable in order to operate the reactor. The objectives of these requirements are to insure that the reactor may be safely operated, adequately controlled, and shut down from any condition of operation.

9.3.2.1 TS 3.2.1 Operable Control Rods

See Section 2.2.2 of this SER.

9.3.2.2 TS 3.2.2 Reactivity Insertion Rates (Scram Time)

See Section 2.2.2 of this SER.

9.3.2.3 TS 3.2.3 Other Pulsed Operation Limitations

Limitations other than those on core configuration and pulsed reactivity insertion limits are not required for UWNR.

TS 3.1.3 includes limits on pulse reactivity, and the NRC staff finds them to be acceptable (see Section 2.1.3 of this SER.)

9.3.2.4 TS 3.2.4 Reactor Safety System

TS 3.2.4 specification is as follows:

> The reactor shall not be operated unless the safety channels described in Table 3.2.4 are operable.

Table 3.2.4 Reactor Safety System Channels

	Safety Channel	Setpoint and Function	Number Operable in Specified Mode		
			SS	SW	PU
1	Fuel Temperature	Scram if fuel temperature exceeds 400°C in the fuel temperature safety channel for an instrumented fuel element pin power peaking factor of 0.87-1.16, or 500°C for an instrumented fuel element pin power peaking factor greater than 1.16.	1	1	1
2	Linear Power Level	Scram if power > 125% full power.	2	2	-
3	Manual Scram	Manually initiated scram.	1	1	1
4	Preset Timer	Transient rod scram 15 seconds or less after pulse.	-	-	1
5	Reactor Water Level	Scram if < 19 feet above top of core.	1	1	1
6	High Voltage Monitor	Scram on loss of high voltage to neutron and gamma ray power level instrument detectors	1	1	1
7	Reactor water temperature	Scram if > 130°F	1	1	1

TS 3.2.4 confirms that, during the normal operation of the UWNR, in the specified mode of operation (i.e., steady state, square wave, or pulse), the required number of safety channels are operable to ensure safe shutdown of the reactor before it exceeds a safety limit. The requirement for two linear power channels and one fuel temperature channel provides the necessary diversity and redundancy to prevent the reactor from exceeding the fuel temperature safety limit. The two linear power channels prevent the reactor from exceeding the 1.5 MW power safety limit with setpoints at 1.25 MW.

The manual scram allows the operator to shut down the system if an unsafe or abnormal condition is present. The preset timer ensures that a reactivity pulse does not exceed 15 seconds in duration and that the power level rise caused by a reactivity pulse is reduced to low levels. The reactor pool water level scram ensures that the reactor shuts down if the water level drops below 19 ft. The high-voltage monitor prevents operation of the reactor when the detector high-voltage supplies are unavailable. The pool water temperature scram ensures that the reactor is not operated in an unanalyzed condition.

The reactor safety system requirements are consistent with other TRIGA reactors and, because the number of operable channels ensures the necessary redundancy and diversity to prevent the UWNR from exceeding a safety limit, the NRC staff finds TS 3.2.4 to be acceptable.

9.3.2.5 TS 3.2.5 Interlocks

TS 3.2.5 specification is as follows:

> The reactor shall not be operated in the indicated modes unless the interlocks in Table 3.2.5 are operable.

Table 3.2.5 Interlocks

	Channel	Setpoint and Function	Number Operable in Specified Mode		
			SS	SW	PU
1	Log Count Rate	Prevents control element withdrawal when neutron count rate < 2 per second	1	1	1
2	Transient Rod Control	Prevents application of air to fire transient rod unless drive is at IN limit.	1	0	0
3	Log N Power Level	Prevents application of air to fire transient rod when power level is above 1 kW and transient rod is not full in.	1	1	1
4	Pulse Mode Control	Prevents withdrawal of control blades while in pulse mode.	0	0	1

TS 3.2.5 ensures that, during the normal operation of the UWNR, in the specified mode of operation (i.e., steady state or pulse), the minimum number of reactor safety system channels required for safe operation of the reactor are operable. The minimum number of operable reactor safety interlocks shown in Table 3.2.5 of the TS will provide the following safety functions:

(1) An interlock to prevent startup of the reactor with less than 2 cps ensures that sufficient neutrons are available for a controlled reactor startup.

(2) An interlock prevents the application of air to the transient rod air cylinder unless the control switch is placed in the IN position.

(3) An interlock to prevent the initiation of a pulse above 1 kW is to ensure that the magnitude of the pulse will not cause the fuel element temperature safety limits to be exceeded.

(4) An interlock to prevent withdrawal of the standard or regulating control elements in the pulse mode is to prevent the reactor from being pulsed while on a positive period.

The staff noted during its review that UWNR TS 3.2.5 did not contain an interlock to prevent multiple control rod withdraw and proposed an RAI via letter dated November 18, 2010. UWNR provided a response, in a letter dated December 8, 2010, which demonstrated that the peak fuel temperature and energy released during an event involving the simultaneous withdraw of all control elements was less than a routine pulsing event at the TS limit of 1.4 % $\Delta k/k$.

The interlocks in TS 3.2.5 ensure that the reactor control system interlocks, as described in the 2008 SAR, are fully operable in the required modes of operation to prevent an insertion of positive reactivity that would challenge the reactor safety limits. TS 3.2.5 is acceptable to the NRC staff, because it means that the reactor will not be operated unless the required minimum number of reactor safety channel interlocks are operable to ensure the safe operation of the reactor.

9.3.2.6 TS 3.2.6 Backup Shutdown Mechanisms

TS 3.2.6 specifies:

> Backup shutdown mechanisms are not required for this reactor.

The NRC staff reviewed the 2008 SAR (Ref 9) and finds no scenarios in Chapter 13, Accident Analysis, which required the implementation of backup shutdown mechanisms. In accordance with the guidance provided in NUREG-1537, which indicates that most non-power reactors are required to only use control and safety rods for shutdown, unless the SAR indicates otherwise. On this basis, the NRC staff concludes that TS 3.2.6 is acceptable.

9.3.2.7 TS 3.2.7 Bypassing Channels

TS 3.2.7 specifications are as follows:

> The log count rate interlock in Table 3.2.5 may be bypassed under either of the following two conditions:
>
> (1) during fuel loading, to allow control element withdrawal necessary for the fuel loading procedure
>
> (2) when the log power level and linear power level channels are on-scale

TS 3.2.7 specifies the conditions when the log count rate interlock may be bypassed. During the early stages of fuel loading, the count rate may be below the setpoint of 2 cps. The interlock is bypassed in this condition to allow control element movements that may be required during fuel loading. During normal operation, once the other power indications are available, the log count rate channel is not required and the interlock is not necessary. The bypasses associated with the log count rate interlock do not affect the ability of the reactor safety system to prevent safety limits from being exceeded; therefore, the bypasses are acceptable to the NRC staff.

9.3.2.8 TS 3.2.8 Control Systems and Instrumentation Required for Operation

TS 3.2.8 specification is as follows:

The reactor shall not be operated unless measuring channels listed in Table 3.2.8 are operable.

Table 3.2.8 Instrumentation and Controls Required for Operation

	Channel	Function	Number Operable in Specified Mode		
			SS	SW	PU
1.	Fuel Temperature	Input for fuel temperature scram	1	1	1
2.	Linear Power Level	Input for safety system power level scram	2	2	0
3.	Log Power Level	Wide-range power indication, permissive for initiation of pulse mode	1	1	0
4.	Startup Log Count Rate	Wide-range power indication, permissive for control element withdrawal	1*	1*	0
5.	Pulsing Power Level	Pulse power level indication	0	0	1

* Required during startup only until the log power level and linear power level channels are on-scale. See TS 3.2.7.

TS 3.2.8 ensures that, during the normal operation of the reactor, in the specified mode of operation (i.e., steady state, square wave, or pulse), the operator has sufficient information to ensure the safe operation of the reactor. The minimum number of I&Cs shown in Table 3.2.8 of the TS will provide the operator with a fuel temperature indication and power level monitors. These requirements for instrumentation ensure that there are measuring channels available for the reactor safety systems and operator monitoring of parameters associated with the reactor safety limits.

The licensee, in response to RAI No. 11 in a letter dated June 16, 2010 (Ref 20), proposed to remove an exemption in TS 3.2.8 related to the availability of replacement IFEs that allowed continued operation of an operational core in the absence of an operable IFE, with the linear power level scram setpoints reduced to 110 percent of full power. The condition was caused by concerns of the potential burnout of the IFE thermocouples in the past. The licensee does not anticipate that all available IFE thermocouples would burn out in the expected operational life of the reactor core. Therefore, it proposed to remove the exemption and not allow the reactor core to operate without an IFE under all modes of operation. The NRC staff reviewed the proposed change and concluded that the requirement for at least one IFE to be in place, in accordance with TS 2.2, provides redundancy and defense-in-depth and, therefore, is acceptable.

The licensee noted, in response to RAI No. 13 in a letter dated June 16, 2010 (Ref 20), that TS 3.2.8 addresses ANSI/ANS-15.1-2007, Section 3.2(5), specifying the number of channels required for reactor operation, while TS 3.2.7 addresses ANSI/ANS-15.1-2007, Section 3.2(8), “permitted bypassing of channels.” The NRC staff reviewed the channels required to be operable and the permitted bypassing of channels and finds that the licensee addressed the requirements in ANSI/ANS-15.1-2007; therefore, TS 3.2.7 and TS 3.2.8 are acceptable.

TS 3.2.8 is acceptable to the NRC staff because it serves to prevent the reactor from being operated unless the required minimum number of I&Cs are operable to ensure the safe operation of the reactor.

9.3.3 TS 3.3 Reactor Pool Water Systems

See Section 2.3 of this SER.

9.3.4 TS 3.4 Confinement

These specifications provide restrictions on the release of airborne radiation to the environment.

9.3.4.1 TS 3.4.1 Operations that Require Confinement

TS 3.4.1 specification is as follows:

> Confinement is required for reactor operation or any movement of irradiated fuel or fueled experiments with significant fission product inventory outside containers, systems, or storage areas.

TS 3.4.1 defines the conditions requiring the establishment of confinement. Confinement, which is defined in TS 3.4.2, ensures that any potential radioactivity released as a result of operations associated with the reactor passes through a filter train in a manner that can be quantified by the plant ventilation stack air monitor and at an elevation that limits the consequences to the public. The NRC staff finds that TS 3.4.1 ensures confinement is set when a potential for radiological release is present and is consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. On this basis, the NRC staff concludes that TS 3.4.1 is acceptable

9.3.4.2 TS 3.4.2 Equipment to Achieve Confinement

TS 3.4.2 specification is as follows:

> To achieve confinement, the ventilation system must be operating in accordance with TS 3.5.

In response to RAI No. 15 in a letter dated June 16, 2010 (Ref 20), the licensee proposed revising TS 3.4 to conform to ANSI/ANS-15.1-2007. The licensee removed the previous specifications for minimum free volume and minimum exhaust height because it addressed them in TS 5.1. The NRC staff finds the revision acceptable, as it specifies the conditions requiring confinement and is in accordance with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007.

TS 3.4.2 is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007 and ensures that the ventilation system is able to establish confinement. Therefore, TS 3.4.2 is acceptable to the NRC staff.

9.3.5 TS 3.5 Ventilation System

This specification ensures that the ventilation system is operating to mitigate the consequences of the possible release of airborne radioactive materials resulting from reactor operations. TS 3.4.2 specification is as follows:

> The reactor shall not be operated unless the ventilation system is operating. The ventilation system is considered operating if:
>
> 1. One stack exhaust fan is operating
> 2. Exhaust flow-rate is at least 9600 scfm.
> 3. Exhaust filter total pressure drop is less than 2.5 inches of water column.

TS 3.5 states that one of the two stack exhaust fans must be operable to establish a minimum exhaust flow rate of 9,600 scfm with a water pressure drop of less than a 2.5 in. across the exhaust filter. The NRC staff finds that these conditions ensure the ventilation system is operable, as described in the 2008 SAR (Ref 9), and thus is capable of limiting the exposure to the public in the event of a radiological release. On this basis, the NRC staff concludes that TS 3.5.1 is acceptable.

In response to RAI No. 16 in a letter dated June 16, 2010 (Ref 20), the licensee clarified the operations of the ventilation system and proposed to modify TS 3.5 to conform to ANSI/ANS-15.1-2007, including the elimination of an exemption allowing 2 days of reactor operation with the ventilation system inoperable. The NRC staff finds that the proposed modifications ensure that the operation of the ventilation system establishes adequate confinement during reactor operations. On this basis, the NRC staff concludes that TS 3.5 is acceptable.

9.3.6 TS 3.6 Emergency Power

TS 3.6 specifies:

> Emergency power is not required for this facility

The 2008 SAR (Ref 9) indicated that no electrical power supplies were necessary for the safe shutdown of the UWNR. 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 information provided in the 2008 SAR (Ref 9) and concludes that emergency power systems are not required for the UWNR.

9.3.7 TS 3.7 Radiation Monitoring System and Effluents

9.3.7.1 TS 3.7.1 Monitoring Systems

See Section 7.1.4 of this SER.

9.3.7.2 TS 3.7.2 Effluent (Argon-41) Discharge Limit

See Section 7.1.1.1 of this SER.

9.3.8 TS 3.8 Experiments

The TS 3.8 specifications limit the effects that all experiments can have on reactivity, describe the type of materials that may be contained in experiments, and state what corrective actions are necessary if an experimental capsule fails.

9.3.8.1 TS 3.8.1 Reactivity Limits

TS 3.8.1 specification is as follows:

> The reactor shall not be operated unless the following conditions governing experiments exist:
>
> 1. The sum of the absolute values of the reactivity worths of all non-secured experiments does not exceed 0.7% $\Delta k/k$.
>
> 2. The reactivity worth of any single secured experiment does not exceed 1.4% $\Delta k/k$.
>
> 3. The sum of the absolute values of the reactivity worths of all experiments, both secured and non-secured, does not exceed the maximum excess reactivity specified in TS 3.1.1

In TS 3.8.1(1), the reactivity limit of less than 0.7% $\Delta k/k$ for a single non-secured experiment is designed to prevent prompt criticality from occurring and is substantially below the analyzed maximum allowable pulse size.

TS 3.8.1.(2) establishes the experimental reactivity worth limit of 1.4% $\Delta k/k$ for single experiments classified as "secured." Since the experiment is held stationary in the reactor, the likelihood that it would fall away from the core to produce an undesirable step increase in reactivity is minimized. In TS 3.8.1(2), the reactivity limit of 1.4% $\Delta k/k$ for single secured experiments is designed to be below the analyzed maximum allowable pulse size. The energy release, after removing 1.4% $\Delta k/k$ reactivity worth of an experiment from a cold critical reactor will not result in fuel temperatures exceeding the safety limit. Removal of an experiment at full power would result in a relatively slow power increase, allowing the reactor protective system to scram the reactor before it reaches the fuel temperature safety limits.

In response to RAI No. 18 in a letter dated June 16, 2010 (Ref 20), the licensee proposed to add TS 3.8.1(3), which states that the sum of all experimental absolute values of the reactivity worths, both secured and non-secured, will not exceed the maximum excess reactivity specified in TS 3.1.1. This specification is designed to ensure that the reactor remains subcritical in the unlikely event of simultaneous removal of all experiments. The requirement to use the absolute values of the reactivity worths further ensures that the shutdown margin is met in any situation where an experiment is removed and consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007, Section 3.8.1(2), and is, therefore, acceptable to the NRC staff.

TS 3.8.1 limits the reactivity of experiments to levels within bounds shown to prevent exceeding reactivity requirements or violating the shutdown margin. Therefore, TS 3.8.1 is acceptable to the NRC staff.

9.3.8.2 TS 3.8.2 Materials

TS 3.8.2 limits the amount of hazardous material that can be placed in the reactor as part of an experiment. The specifications are as follows:

1. Explosive materials, such as gunpowder, TNT, nitroglycerin, or PETN, in quantities greater than 25 milligrams shall not be irradiated in the reactor or experimental facilities. Explosive materials in quantities less than 25 milligrams 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.

In response to RAI No. 19 in a letter dated June 16, 2010 (Ref 20), the licensee proposed that TS 3.8.2(1) be amended by the insertion of the word “half,” requiring the design pressure of the irradiation container to be twice the pressure produced upon the detonation of the explosive. This modification will make the TS consistent with the recommendations of Regulatory Guide 2.2, “Development of Technical Specifications for Experiments in Research Reactors,” issued November 1973. TS 3.8.2(1) now states that no explosive material in excess of 25 mg is allowed to be irradiated in the reactor. Explosive material up to 25 mg may be irradiated, provided the pressure produced upon detonation of the explosive has been calculated or experimentally demonstrated to be less than half the design pressure of the irradiation container. The 25 mg limit is a longstanding limit discussed in Regulatory Guide 2.2. The requirement for experiments containing explosive materials is within the guidance of Regulatory Guide 2.2 and is, therefore, acceptable to the NRC staff.

2. Experiment materials, except fuel materials, which could off-gas, sublime, volatilize, or produce aerosols under (1) normal operating conditions of the experiment or reactor, (2) credible accident conditions in the reactor, or (3) possible accident conditions in the experiment shall be limited in activity such that if 100% of the gaseous activity or radioactive aerosols produced escaped to the reactor room or the atmosphere, the airborne concentration of radioactivity averaged over a year would not exceed the limits of Appendix B of 10 CFR 20.

TS 3.8.2(2) limits the radioactive products in experimental materials that may release airborne radioactive particles. The purpose of the specification is to ensure that potential releases of radioactive material from experiments are bounded by the dose limits in 10 CFR Part 20 for UWNR staff and members of the public. This includes failures under normal reactor operations, credible reactor accident conditions, and accident conditions in the experiment. TS 3.8.2(2) is acceptable to the NRC staff, because it limits doses from potential experiment failure or malfunction to 10 CFR Part 20 limits.

3. In calculations pursuant to 2 above, the following assumptions shall be used:

    a. If the effluent from an experimental 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.

b. If the effluent from an experimental facility exhausts through a filter installation designed for greater than 99% efficiency for 0.3 micron particles, at least 10% of these vapors can escape.
c. For materials whose boiling point is above 130ºF and where vapors formed by boiling this material can escape only through an undisturbed column of water above the core, at least 10% of these vapors can escape.
d. An atmospheric dilution factor of 3.6×10^{-5} µCi/ml per Ci/s for gaseous discharges from the facility.

TS 3.8.2(3) contains standard research reactor TS assumptions and ensures that source term calculations are conservative. This is consistent with the guidance provided in NUREG-1537 and is, therefore, acceptable to the NRC staff.

4. Each fueled experiment shall be controlled such that the total inventory of iodine isotopes 131 through 135 in the experiment is no greater than 1.5 Ci.

TS 3.8.2(4) limits the iodine content of a fueled experiment to 1.5 Ci. TS 3.8.2(4) is acceptable to the NRC staff, because it limits doses from potential fueled experiment failure or malfunction to 10 CFR Part 20 limits.

5. Experiment materials that are corrosive to reactor components or highly reactive with coolants shall be double encapsulated.

TS 3.8.2 (5) prevents damage to reactor components resulting from failure of an experiment involving corrosive or highly reactive material.

9.2.8.3 TS 3.8.3 Experiment Failure and Malfunctions

TS 3.8.3 specifies the following actions to be taken in the event an experimental capsule fails:

> If a capsule fails and releases material, which could damage the reactor fuel or structure by corrosion or other means, removal and physical inspection of the capsule shall be performed to determine the consequences and need for corrective action. The results of the inspection and any corrective action taken shall be reviewed by the Reactor Director or designated alternate and determined to be satisfactory before operation of the reactor is resumed.

The NRC staff reviewed the licensee's limitations on experiments as specified in TS 3.8. The licensee's TS 3.8 covers the areas recommended in NUREG-1537 and ANSI/ANS-15.1-2007. The technical content of TS 3.8 is consistent with guidance and provides an envelope of performance against which proposed experiments can be evaluated. Therefore, the licensee's limitations on experiments are acceptable to the NRC staff.

9.4 <u>TS 4 Surveillance Requirements</u>

TS 4 specifies the surveillance testing requirements for certain systems to verify proper operation after modifications or maintenance.

> 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 approved by the Reactor Safety Committee. A system shall not be considered operable until after it has been successfully tested.

Scheduled surveillances, except those specifically required when the reactor is shut down, may be deferred during shutdown periods, but be completed prior to subsequent reactor startup unless operation is required for the performance of the surveillance. Scheduled surveillances which cannot be performed with the reactor operating may be deferred until a planned reactor shutdown. If the reactor is not operational in a particular mode, surveillances required specifically for that mode may be deferred until the reactor becomes operational in that mode.

TS 4 requires that changes to certain important systems be controlled to their original design and fabrication specifications or, if to new specifications, that those specifications have been reviewed. The TS also governs the scheduling of required surveillance testing to allow operational flexibility that does not affect safety. Since TS 4 maintains adequate control over the design change process, it is acceptable to the NRC staff.

9.4.1 TS 4.1 Reactor Core Parameters

See Section 2.2 of this SER.

9.4.2 TS 4.2 Reactor Control and Safety Systems

These surveillance specifications apply to a number of reactor control and safety system functions that must be operable to operate the reactor. The objectives of these requirements are to ensure that the reactor may be operated, controlled, and shut down from any condition of operation:

TS 4.2(1)–(3): See Section 2.2.2.

4. Scram and Power Measuring Channels
    a. A channel test of each Reactor Safety System measuring channel in Table 3.2.4 items (1) through (4) and the interlocks in Table 3.2.5 required for the intended modes of operation shall be performed within 24 hours before each day's operation or prior to each operation extending more than one day.

    b. A channel test of items (5), (6), and (7) in Table 3.2.4 shall be performed semi-annually.

    c. A channel calibration of items (1) and (2) in Table 3.2.4 shall be performed annually.

5. Operability Tests

    This concern is covered by the General Surveillance criterion at the beginning of this section.

6. Thermal Power Calibration-Natural Convection
   A Channel Calibration shall be made of the power level monitoring channels by the calorimetric method upon changes in core composition or arrangement that could change flux distributions in the region of the nuclear instrumentation and annually thereafter.

7. Control Element Inspection
   The control elements shall be visually inspected for deterioration biennially.

TS 4.2(4)–TS 4.2(7) specify the surveillance tests and their intervals for the scram and power measuring channels, power calibration requirements, and control element inspection, consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007 and the intervals used at similar research reactors. Experience has shown that these surveillance frequencies will ensure the performance and operability of the UWNR control and safety system. The specified intervals provide reasonable assurance that any component failure or degradation will be detected in a timely manner and the specified calibration frequencies are adequate to prevent significant drift in the detection ranges. TS 4.2(4)–(7) are therefore acceptable to the NRC staff.

In response to RAI No. 20 in a letter dated June 16, 2010 (Ref 20), the licensee discussed the interlock surveillance requirements. TS 4.2(4)(a) addresses interlock surveillances for the scram channels, while TS 3.2.5 (4) specifies that the pulse mode control interlock is required to be operable in pulse mode only. TS 4.1(3) requires semiannual pulsing of the reactor, which ensures that the pulse mode control interlock is verified to be operable at least semiannually. This response is acceptable to the NRC staff.

9.4.3 TS 4.3 Coolant Systems

See Section 2.3 of this SER.

9.4.4 TS 4.4 Confinement

TS 4.4 specification is as follows:

> The ventilation system shall be verified operable quarterly, in accordance with TS 4.5.

In response to RAI No. 22 in a letter dated June 16, 2010 (Ref 20), the licensee proposed modifying TS 4.4 to specify a surveillance requirement for the ventilation system in accordance with ANSI/ANS-15.1-2007. TS 4.4, Confinement, requires that the ventilation will be operable in accordance with TS 4.5, Ventilation Systems. TS 4.5 requires that the ventilation system will be verified quarterly or following maintenance. The NRC staff concluded that this surveillance requirement ensures a high degree of operability of the ventilation system to provide the confinement function. Therefore, TS 4.4 is acceptable to the NRC staff.

9.4.5 TS 4.5 Ventilation Systems

TS 4.5 specification is as follows:

> It shall be verified quarterly and following repair or maintenance that the ventilation system is operable in accordance with TS 3.5.

The quarterly and following repair and modification verifications of operability is adequate to confirm operability and is consistent with other research reactor TS. Therefore, TS 4.5 is acceptable to the NRC staff.

In response to RAI No. 23 in a letter dated June 16, 2010 (Ref 20), the licensee discussed the surveillance requirement of the ventilation system. TS 3.5 specifies the minimum requirements of one exhaust fan operating, a required exhaust flow-rate, and an acceptable exhaust filter pressure drop. TS 4.5, the corresponding surveillance, ensures operability of the ventilation system by quarterly testing, consistent with ANSI/ANS-15.1-2007. The NRC staff concludes that specifications in TS 3.5, together with TS 4.5, ensure that ventilation system operability will be maintained. Therefore, the response is acceptable to the NRC staff.

9.4.6 TS 4.6 Emergency Electrical Power Systems

TS 4.6 specifies:

> Not Applicable.

Emergency power is not required for the safe shutdown of the UWNR. See Section 9.3.6 of this SER.

9.4.7 TS 4.7 Radiation Monitoring Systems and Effluents

9.4.7.1 TS 4.7.1 Radiation Monitoring Systems Applicability

See Section 7.1.4 of this SER.

9.4.7.2 TS 4.7.2 Effluents

See Section 7.2.3 of this SER.

9.4.8 TS 4.8 Experiments

Any surveillance requirements placed on experiments are specified as part of the safety review performed by the RSC. There are no TS surveillances specific to experiments.

9.4.9 TS 4.9 Facility-Specific Surveillance

There are no facility-specific surveillance requirements for the UWNR.

9.5 Design Features

9.5.1 TS 5.1 Site and Facility Description

TS 5.1 gives the facility design requirements as follows:

1. The reactor shall be housed in a closed room designed to restrict leakage. The minimum free volume shall be 2000 cubic meters.

TS 5.1(1) states the minimum free volume maintained by the facility to provide a dilution capacity, which limits the dose from any airborne radioactivity from reactor operations released within areas of the UWNR occupied by personnel. Calculations in Chapter 13 of the 2008 SAR demonstrate that occupational exposures in the UWNR can be kept below 10 CFR Part 20 limits under accident conditions if the empty volume of the space housing the reactor is 2,000 m^3.

2. All air or other gas exhausted from the reactor room and the Beam Port and Thermal Column Ventilation System shall be released to the environment a minimum of 30.5 m above ground level.

TS 5.1(2) specifies the minimum height requirement for release of exhausted gases from the ventilation system ensuring adequate dispersion with outside air. This specification limits the concentration of any airborne radioactivity released to the environment from reactor operations. Based on a ground-level release of radioactive effluent, calculations show that concentrations of radioactive materials are within 10 CFR Part 20 limits in non-restricted areas for the accidents considered. Additional calculations based on release at the stack height of 30.5 m show a further reduction by a factor of 2.6 attributed to operation of the ventilation system. Since the confinement specifications keep radioactive releases below 10 CFR Part 20 limits, TS 5.1(1) and (2) are acceptable to the NRC staff.

3. The operations boundary shall be the Reactor Laboratory, room 1215 of the Mechanical Engineering Building. The operations boundary shall be a restricted area.

TS 5.1(3) defines the operations boundary as a restricted area and sets the boundary as Room 1215 of the Mechanical Engineering Building. This ensures that an area is defined where operating staff will have positive control over personnel access. The reactor director has direct authority over all activities within Room 1215 of the Mechanical Engineering Building.

4. The site boundary shall be that portion of the center and east wings of the Mechanical Engineering Building south of the north lobby, plus the portion of Engineering Drive south of the designated areas of the building. The site boundary may be a non-restricted area.

TS 5.1(4) defines the site boundary and establishes it as a non-restricted area. The reactor director has direct control over these areas when directing emergency activities.

In response to RAI No. 25 in a letter dated June 16, 2010 (Ref 20), the licensee proposed to revise TS 5.1 to conform to ANSI/ANS-15.1-2007 and added two new specifications defining the operations and site boundaries. The NRC staff concludes the proposed changes are acceptable, because they clearly define the operational and site boundaries at UWNR in terms of physical areas and set requirements for their accessibility.

TS 5.1(1)–(4) state important requirements of the facility physical design used to house the UWNR and define the boundaries of the facility that is being licensed. These specifications minimize occupational and public radiation doses and facilitate meeting 10 CFR Part 20 requirements. The added specifications define the operational and site area boundaries for the facilities, consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. Therefore, TS 5.1 is acceptable to the NRC staff.

9.5.2 TS 5.2 Reactor Coolant System

TS 5.2 gives the reactor coolant system design requirements as follows:

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

2. The pool water inlet pipe to the demineralizer shall not extend more than 15 feet into the top of the reactor pool when fuel is in the core. The outlet pipe from the demineralizer shall be equipped with a check valve and siphon breaker to prevent inadvertent draining of the pool.

3. Diffuser and other auxiliary systems pumps shall be located no more than 15 ft below the top of the reactor pool.

4. All other piping and pneumatic tube systems entering the pool shall have siphon breakers and valves or blind flanges which will prevent draining more than 15 ft of water from the pool.

5. A pool level alarm shall indicate loss of coolant if the pool level drops approximately one foot below normal level.

6. A pool water temperature alarm shall indicate if water temperature reaches 130ºF

TS 5.2(1)–(5) state the important UWNR design aspects used to prevent loss of core cooling by the reactor coolant system in the event that a leakage path develops that would potentially drain the reactor pool. The normal water level in the reactor pool is 20 ft above the top of the reactor core. The reactor pool water level is continuously monitored by a water-level monitoring device that provides alarm signals at two levels: a high-water alarm, when the level is 2 in. above normal, and a low-water alarm, when the level drops 2 in. below normal. The alarms are monitored in the reactor control room and at the campus police station. Procedures are in place for responding to the alarms. Since the pool water level is continuously monitored and the frequency and quantity of makeup water is logged, pool water leakage will be detected. A small leak will be detected by the low-water alarm; when the water level drops 2 in. below normal, the operator will manually start the makeup water system that will be able to compensate for small leakages. In the event of a catastrophic leak, the water would collect in the drainage sump and holdup tank.

TS 5.2(6) ensures that the reactor pool water is operating below 54.4 degrees C (130 degrees F) and, therefore, is not operating in an unanalyzed condition.

TS 5.2(1)–(6) specify important design aspects associated with the reactor pool to ensure that the pool design minimizes the chances of a leak developing that could drain the pool. These specifications further ensure that pool leakage will be detected and that the pool operates at a temperature that has been analyzed. Based on the discussion above, TS 5.2 is acceptable to the NRC staff.

9.5.3 TS 5.3 Reactor Fuel: TRIGA LEU 30/20 Fuel

See Section 2.2.1 of this SER.

9.5.4 TS 5.4 Reactor Core

See Section 2.2 of this SER.

9.5.5 TS 5.5 Control Elements

See Section 2.2.2 of this SER.

9.5.6 TS 5.6 Fissionable Material Storage

TS 5.6 describes the design features for fuel storage as follows:

1. All fuel elements shall be stored in a geometrical array where the value of k-effective is less than 0.8 for all conditions of moderation

2. Irradiated fuel elements and fueled devices shall be stored in an array, which will permit sufficient natural convection cooling by water or air, so that the fuel element or fueled device temperature will not exceed design values

TS 5.6(1) limits the k_{eff} value to 0.8, which is lower than the commonly used value of k_{eff} of 0.9 that is recommended in NUREG-1537 and ANSI/ANS-15.1-2007. The lower value of 0.8 is more conservative and thus acceptable to the NRC staff.

TS 5.6(2) is the basic design requirement that ensures adequate cooling by natural convection by either water or air. Stored fuel must have cooling sufficient to prevent the stored fuel elements from exceeding the fuel temperature safety limit.

These specifications control subcriticality and fuel temperature within acceptable limits to ensure the integrity of the stored fuel; therefore, TS 5.6 is acceptable to the NRC staff.

9.6 Administrative Control

Conduct of operations for the UWNR includes administrative controls over the facility's operation and its emergency and security plans. The administrative controls discussed in this section of the SER are the facility organization, training, operational review and audits, procedures, required actions, reports, and records. This section of the SER addresses Chapter 12 of the 2008 SAR (Ref 9) and Section 6 of the TS.

The primary guidance for the development of administrative control TS for research reactor operation is NUREG-1537 and ANSI/ANS-15.1-2007. The licensee's TSs are based on the information contained within these documents. The NRC staff used the 2007 version of ANSI/ANS-15.1-2007 in its review of the licensee's administrative controls. In some cases, the wording of the TS proposed by the licensee was not identical to that given in NUREG-1537 and ANSI/ANS-15.1-2007. However, the NRC staff reviewed these cases and determined that the licensee's proposed TS met the intent of the guidance and were acceptable.

9.6.1 TS 6.1 Organization

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

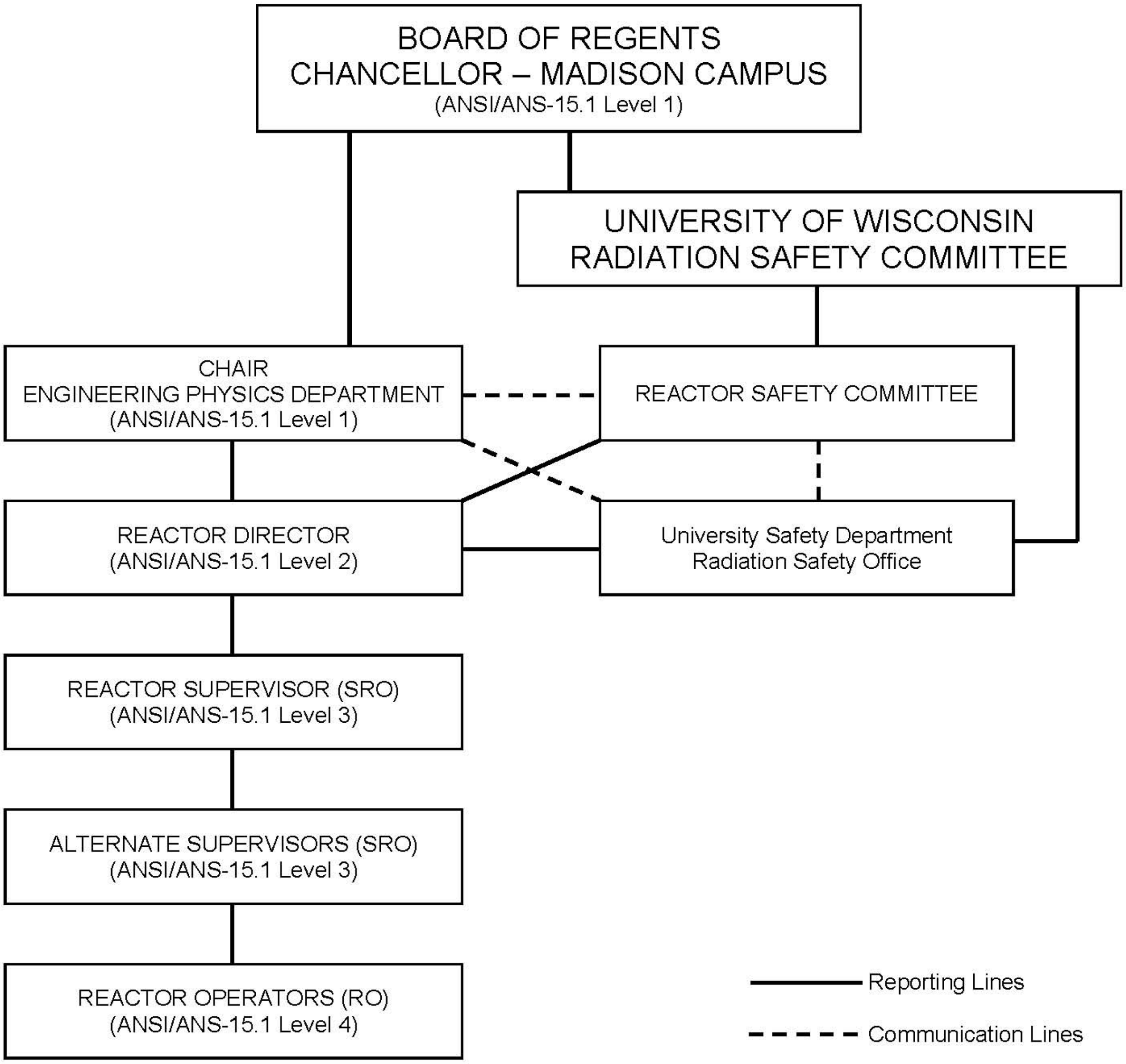

Figure -1 Organizational Chart

9.6.1.1 TS 6.1.1 Structure

TS 6.1.1 specifies the organizational structure of the UWNR facility.

> The reactor facility shall be an integral part of the Engineering Physics Department of the College of Engineering of the University of Wisconsin-Madison. The reactor shall be related to the University structure as shown in Figure -1.

> The Radiation Safety Office performs audit functions for both the Radiation Safety Committee and the Reactor Safety Committee and reports to both Committees, as well as to the Reactor Director.

In response to RAI No. 26 in a letter dated June 16, 2010 (Ref 20), the licensee clarified the structural responsibilities of the University Radiation Safety Committee (URSC) as follows: The URSC is the body that authorizes use of ionizing radiation on campus and is responsible for the oversight of all radioactive material on campus. This authority is delegated to the URSC by the Chancellor, who receives authority from the Board of Regents, the ultimate holder of the reactor license. The Radiation Safety Office is delegated by the URSC to implement, on a day-to-day basis, the authority of the URSC. The radiation safety officer (RSO) is in charge of the Radiation Safety Office and is a member of the URSC. The RSC is a standing subcommittee of the URSC, and the RSC chair is a member of the URSC. The RSO is also a member of the RSC. Therefore, the Radiation Safety Office operates under the authority of the URSC and reports to the URSC, department chair, RSC, and reactor director on review and audit functions at the facility. Each of these organizations has the authority to stop work at the reactor laboratory.

Certain Level 1 responsibilities of the Board of Regents of UW, the holder of the reactor license, are delegated to the Engineering Physics Department chair. The licensee revised the organizational chart (Figure -1) to reflect this clarification.

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

9.6.1.2 TS 6.1.2 Responsibility

TS 6.1.2 specifies the organizational responsibilities of the UWNR facility.

> The Reactor Director is responsible for all activities at the facility, including licensing, security, emergency preparedness, and maintaining radiation exposures as low as reasonably achievable.
>
> The reactor facility shall be under the direct control of a Reactor Supervisor designated by the Reactor Director. The Reactor Supervisor shall be responsible for assuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license, procedures, and the requirements of the Radiation Safety Committee and the Reactor Safety Committee.

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

9.6.1.3 TS 6.1.3 Staffing

TS 6.1.3 describes the minimum staffing necessary to operate the UWNR facility as follows:

> 1. The minimum staffing when the reactor is not secured shall be:

a. A licensed reactor operator in the control room (if senior operator licensed, may also be the person required in c).

b. A second designated person present at the facility capable of carrying out prescribed written instructions.

c. A designated senior reactor operator shall be readily available at the facility or on call. On call means the individual can be rapidly reached by phone or radio and is within 30 minutes or 15 miles of the reactor facility.

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.

3. A licensed senior reactor operator shall be present at the facility for:

a. Initial startup and approach to power.

b. All fuel handling or control-element relocations.

c. Relocation of any in-core experiment with a reactivity worth greater than 0.7% $\Delta k/k$.

d. Recovery from unplanned or unscheduled shutdown or significant power reduction.

TS 6.1.3(1) describes the minimum staffing necessary to safely operate the UWNR. 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.”

In response to RAI No. 27 in a letter dated June 16, 2010 (Ref 20), the licensee clarified when an operator is “on call,” defining it as being within 30 minutes or 15 miles of the reactor facility. This modification conforms to NUREG-1537 and ANSI/ANS-15.1-2007, Section 6.1.3(c), and is therefore acceptable to the NRC staff.

TS 6.1.3(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 provided in NUREG-1537 and ANSI/ANS-15.1-2007 and is therefore acceptable to the NRC staff.

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

The NRC staff reviewed and evaluated the requirements of TS 6.1.3 and finds that they are consistent with the guidance provided in NUREG -1537 and ANSI/ANS-15.1-2007. They further satisfy the requirements of 10 CFR 50.54(k) and 10 CFR 50.54(m). Therefore, TS 6.1.3 is acceptable to the NRC staff.

9.6.1.4 TS 6.1.4 Selection and Training Personnel

The licensee discussed training of personnel for key positions in Sections 12.1.4 and 12.10 of the 2008 SAR (Ref 9). TS 6.1.4 reads as follows:

> The selection, training, and re-qualification of operations personnel shall meet or exceed the requirements of ANSI/ANS-I5.4-2007, Sections 4-6.

The licensee uses ANSI/ANS 15.4-2007 as guidance for the selection and training of personnel. The NRC staff supports the use of ANSI/ANS 15.4-2007 by licensees for the selection and training of personnel. The 2008 SAR (Ref 9) describes the training requirements as two parts: a 4 credit hour training course and a 12 week candidacy program. The licensee indicated that the training program meets the requirements of 10 CFR Part 55 and is formalized in the licensee's procedure, UWNR-004. The NRC staff concludes that TS 6.1.4 is acceptable.

9.6.2 TS 6.2 Review and Audit

One of the responsibilities of the Reactor Safety Committee is to review and audit matters relating to the safe operation of the facility and the health and safety of the public and the environment. Overall responsibility, composition, and qualification of the RSC, as well as charter and rules, are described in Section 12.2 of the 2008 SAR (Ref 9) and UWNR TS Sections 6.2.1, 6.2.2, 6.2.3, and 6.2.4. NUREG-1537 and ANSI/ANS-15.1 specify that the purpose of the RSC is to provide independent oversight of reactor operations. TS 6.2 states the following:

> There shall be a Reactor Safety Committee which shall review and audit reactor operations to assure that the facility is operated in a manner consistent with public safety and within the conditions of the facility license.

The function of the RSC, as outlined in TS 6.2 and in the 2008 SAR (Ref 9), is consistent with the guidelines provided in NUREG-1537 and ANSI/ANS-15.1-2007 and is, therefore, acceptable to the NRC staff.

9.6.2.1 TS 6.2.1 Composition and Qualifications

TS 6.2.1 specifies the composition and qualifications for the RSC as follows:

> The Committee shall be composed of a least six members, one of whom shall be a Health, Physicist from the University of Wisconsin Safety Department Radiation Safety Office. The Committee shall collectively possess expertise in the following disciplines:
>
> 1. Reactor Physics;
>
> 2. Heat transfer and fluid mechanics;

3. Metallurgy

4. Instruments and Control Systems;

5. Chemistry and Radio-chemistry;

6. Radiation Safety.

Reactor staff shall not be members of the committee. This does not preclude reactor staff from participating on subcommittees

NUREG-1537 and ANSI/ANS-15.1-2007 specify that the purpose of the RSC is to provide independent oversight and that the operating staff should not constitute the majority of a quorum. The facility organization (Fig. 14-1) shows that the RSC reports to the University of Wisconsin – Madison Radiation Safety Committee, and to the Board of Regents/Chancellor – Madison Campus, an ANSI/ANS-15.1 Level 1 Position. The composition and qualifications of the RSC, as outlined in TS 6.2.1 and in the 2008 SAR (Ref 9), represent a broad spectrum of expertise in reactor technology. In response to a staff question concerning the independence of RSC membership, the licensee provided, in response to RAI No. 2 in a letter dated January 28, 2011 (Ref 29), an explanation how the RSC Charter precludes the Reactor Director (Level 2) from solely appointing RSC committee members. The staff finds the licensee's response acceptable. Therefore, TS 6.2.1 is acceptable to the NRC staff.

9.6.2.2 TS 6.2.2 Charter and Rules

TS 6.2.2 establishes the RSC and describes how it will operate.

The Committee shall meet at least annually.

The Committee shall formulate written standards regarding the activities of the full committee; minutes, quorum, telephone polls for approvals not requiring a formal meeting, and subcommittees.

A quorum shall be at least half the members.

The RSC is responsible for independent audits of the UWNR activities and conducts its review and audit functions in accordance with a written charter. Provisions include meeting frequency, voting rules, quorums, method of submission, use of subcommittees, and minutes. In response to RAI No. 28 in a letter dated June 16, 2010 (Ref 20), the licensee clarified that the RSC charter precludes reactor operating staff from being members of the committee. This clarification is consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007, Section 6.2.2. The NRC staff finds the licensee's response acceptable. The NRC staff concludes that TS 6.2.2 established requirements for meeting frequency and committee quorum and is consistent with NUREG-1537 and ANSI/ANS-15.1-2007.

9.6.2.3 TS 6.2.3 Review Functions

TS 6.2.3 establishes the review functions of the RSC.

The responsibilities of the Reactor Safety Committee shall include, but are not limited to, the following:

1. Review and approval of experiments utilizing the reactor facilities;

2. Review and approval of all proposed changes to the facility, procedures, license, and technical specifications;

3. Determinations that proposed changes in equipment, systems, tests, experiments or procedures are allowed in accordance with 10 CFR 50.59 without prior authorization by the NRC;

4. Review of abnormal performance of plant equipment and operating anomalies having safety significance;

5. Review of unusual or reportable occurrences and incidents, which are reportable under 10 CFR 20 and 10 CFR 50;

6. Review of audit reports; and

7. Review of violations of technical specifications, license, or procedures and orders having safety significance.

TS 6.2.3 specifies the review responsibilities of the RSC. The NRC staff reviewed TS 6.2.3 and 2008 SAR (Ref 9), Section 12.2.3. The NRC staff finds that the items that the committee will review and audit are consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. The NRC staff concludes that TS 6.2.3 is acceptable.

9.6.2.4 TS 6.2.4 Audit Functions

TS 6.2.4 establishes the audit functions for the UWNR facility.

> A Health Physicist from the University of Wisconsin Safety Department Radiation Safety Office shall represent the University Radiation Safety Committee and shall conduct an inspection of the facility at least monthly to assure compliance with the regulations of 10 CFR Part 20. The services and inspection function of the UW Health Physics Office shall also be available to the Reactor Safety Committee, and will extend the scope of the audit to cover license, technical specification, and procedure adherence.
>
> The committee shall annually audit operation and operational records of the facility, requalification program, security plan, and emergency plan and its implementing procedures. If the committee chooses to use the staff of the Health Physics organization for the audit function, the reports of audit results will be distributed to the committee and included as an agenda item for committee meetings.

In response to RAI No. 29 in a letter dated June 16, 2010 (Ref 20), the licensee proposed to modify TS 6.2.4 to clearly specify that the annual reviews of the requalification program, the security plan, and the emergency plan and its implementing procedures will not be performed by the reactor staff but by independent personnel. This modification ensures an independent review of the above areas and conforms to ANSI/ANS-15.1-2007, Section 6.2.4. The licensee's proposed TS is acceptable to the NRC staff.

The NRC staff reviewed TS 6.2.4 and 2008 SAR (Ref 9), Section 12.2.4. The NRC staff finds the licensee's audit functions acceptable and the items the committee will review and audit to be comprehensive and consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. On this basis, the staff concludes that TS 6.2.4 is acceptable.

9.6.3 TS 6.3 Radiation Safety

TS 6.3 states that UWNR shall meet the requirements of the university radiation safety regulations, as submitted for the University Broad License, License Number 25-1323-01, and is subject to the authority of the State license. It also specifies the following:

> The Reactor Laboratory shall meet the requirements of the University Radiation Safety Regulations as submitted for the University Broad License, License Number 25-1323-01 and is subject to the authority of the State license.
>
> The Reactor Director shall have responsibility for maintaining radiation exposures as low as reasonably achievable and for implementation of laboratory procedure for insuring compliance with 10 CFR Part 20 regulations.

According to 10 CFR 20.1101(a), each licensee must develop, document, and implement a radiation protection program, which this TS does. In response to RAI No. 30 in a letter dated June 16, 2010 (Ref 20), the licensee stated that TS 6.3 specifies that the reactor laboratory shall meet the requirements of the university radiation safety regulations and that the university radiation safety regulations meet the requirements of 10 CFR 20.1101(a) and the guidance of ANSI/ANS-15.11-1993 (R2004), as specified in ANSI/ANS-15.1-2007. Therefore, TS 6.3 is acceptable to the NRC staff, because it establishes a radiation safety program that commits to meeting the requirements of 10 CFR Part 20.

9.6.4 TS 6.4 Procedures

The licensee has specified, in Section 12.3 of the 2008 SAR and in TS 6.4, the type of written procedures that must be prepared and approved before use. TS 6.4 reads as follows:

> Written operating procedures shall be adequate to ensure 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 may have an effect on reactor safety;

5. Personnel radiation protection, consistent with the University of Wisconsin Radiation Protection Program, and applicable regulations;

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;

8. Use, receipt, and transfer of by-product material held under the reactor license.

Substantive changes to the above procedures shall be made only with the approval of the Reactor Safety Committee. Temporary changes to the procedures that do not change their original intent may be made by the Senior Operator in control or designated alternate. All such temporary changes shall be documented and subsequently reviewed by the reactor Safety Committee.

TS 6.4.(1)–(8) are consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. The NRC staff concludes that the procedural requirements in TS 6.4 provide reasonable assurance of the safe operation of the reactor and proper administration of the facility.

9.6.5 TS 6.5 Experiment Review and Approval

TS 6.5 describes the experimental review and approval requirements as follows:

1. Routine experiments may be performed at the discretion of the senior operator responsible for operation without the necessity of further review or approval.

2. Prior to performing any experiment which is not a routine experiment, the proposed experiment shall be evaluated by the senior operator responsible for operation. The senior operator shall consider the experiment in terms of its effect on reactor operation and the possibility and consequences of its failure, including where significant, consideration of chemical reactions, physical integrity, and design life, proper cooling, interaction with core components, reactivity effects, and interactions with reactor instrumentation. The experiment shall only be performed if the evaluation concludes that the requirements of 10 CFR 50.59(c) are met.

3. Modified routine experiments may be performed at the discretion of the senior operator responsible for operation without the necessity of further review or approval provided that the evaluation performed in accordance with Section 6.5(2) results in a determination that the hazards associated with the modified routine experiment are neither greater nor significantly different than those involved with the corresponding routine experiment which shall be referenced.

4. No special experiment shall be performed until the proposed experiment has been reviewed and approved by the Reactor Safety Committee.

5. Favorable evaluation of an experiment shall conclude that failure of the experiment will not lead directly to damage of reactor fuel or interference with movement of a control element.

TS 6.5(1)–(5) specify the different types of experiments that require review and approval before being performed at the UWNR. They further specify the extent of the analysis that should be submitted for review. TS 6.5 is acceptable to the NRC staff because the process and method for reviewing experiments at the UWNR ensures adequate management control over experiments and is consistent with the guidance provided in NUREG-1537.

9.6.6 TS 6.6 Required Actions

The licensee has defined the required actions for events in TS 6.7, including those actions to be taken in case of a safety limit violation and a reportable occurrence, as described below:

9.6.6.1 TS 6.6.1 Action To Be Taken in Case of Safety Limit Violation

In the event a safety limit is exceeded:

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

2. An immediate report of the occurrence shall be made to the Chairman, Reactor Safety Committee, and Reactor Director, and reports shall be made to the NRC in accordance with Section 6.7 of these specifications, and

3. A report shall be prepared which shall include an analysis of the causes and extent of possible resultant damage, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence. This report shall be submitted to the Reactor Safety Committee (RSC) for review and then submitted to the NRC when authorization is sought to resume operation of the reactor.

TS 6.6.1(1)–(3) require the facility to shut down in the event that a safety limit is exceeded. The facility may not resume operation without authorization from the NRC. The violation must also be reported to the 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 actions proposed by the licensee are consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007 and meet the requirements given in 10 CFR 50.36(c)(1) for actions to be taken if a safety limit is exceeded. Therefore, TS 6.6.1 is acceptable to the NRC staff.

9.6.6.2 TS 6.6.2 Action To Be Taken in the Event of an Occurrence of the Type Identified in 6.7.2(1) (b) and (c)

In the event of a reportable occurrence (see TS 1.3) the following actions shall be taken:

1. The reactor shall be shut down until operation is authorized by the Reactor Director.

2. The Director or designated alternate shall be notified and corrective action taken with respect to the operations involved,

3. The Director or designated alternate shall notify the Chairman of the Reactor Safety Committee,

4. A report shall be made to the Reactor Safety Committee, which shall include an analysis of the cause of the occurrence, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence, and

5. A report shall be made to the NRC in accordance with Section 6.7.2 of these specifications.

TS 6.6.2 requires 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 actions proposed by the licensee are consistent with the guidance provided in of NUREG-1537 and ANSI/ANS-15.1-2007. Therefore, TS 6.6.2 is acceptable to the NRC staff.

9.6.7 TS 6.7 Reports

TS 6.7 lists the type of required reports that must be prepared and submitted to ensure that the licensee adequately documents initial startup, routine operations, and special occurrences at UWNR.

9.6.7.1 TS 6.7.1 Operating Reports

TS 6.7.1 lists the operating reports that the licensee must prepare, as well as their schedules for completion and submittal:

1. An annual report covering the activities of the reactor facility during the previous calendar year shall be submitted (in writing to U.S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington, DC 20555) within six months following the end of each calendar year, providing the following information:

    a. A brief narrative summary of (1) operating experience (including experiments performed), (2) changes in facility design, performance characteristics, and operating procedures related to reactor safety and occurring during the reporting period, and (3) results of surveillance tests and inspections;
    b. Tabulation of the energy output (in megawatt days) of the reactor, hours reactor was critical, and the cumulative total energy output since initial criticality;

c. The number of emergency shutdowns and inadvertent scrams, including reasons therefore;

d. Discussion of the major maintenance operations performed during the period, including the effect, if any, on the safety of the operation of the reactor and the reasons for any corrective maintenance required;

e. A brief description, including a summary of the safety evaluations of changes in the facility or in the procedures and of tests and experiments carried pursuant to 50.59 of 10 CFR 50;

f. A summary of radiation exposures received by facility personnel and visitors, including dates and time of significant exposures and a summary of the results of radiation and contamination surveys performed within the facility; and

g. A description of any environmental surveys performed outside the facility.

h. 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;

(1) Liquid Effluents (summarized on a monthly basis)

Liquid radioactivity discharged during the reporting period tabulated as follows:

(a) Total estimated radioactivity released (in curies).

(b) The isotopic composition if greater than 1×10^{-7} microcuries/cc for fission and activation products.

(c) Total radioactivity (in curies), released by nuclide, during the reporting period based on representative isotopic analysis.

(d) Average concentration at point of release (in microcuries/cc) during the reporting period and the fraction of the applicable limit in 10 CFR 20.

(e) Total volume (in gallons) of effluent water (including diluent) during periods of release.

(2) Exhaust Effluents (summarized on a monthly basis)

Radioactivity discharged during the reporting period (in curies) for:

(f) Gases.

(g) Particulates with half lives greater than eight days.

(h) The estimated activity (in curies) discharged during the reporting period, by nuclide, for all gases and particulates based on representative isotopic analysis and the fraction of the applicable 10 CFR 20 limits for these values.

(3) Solid Waste

(a) The total amount of solid waste packaged (in cubic feet).

(b) The total activity involved (in curies).

(c) The dates of shipment and disposition (if shipped off site).

2. A report within 60 days after completion of startup testing of the reactor (in writing to the U.S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington, D.C. 20555) upon receipt of a new facility license or an amendment to the license authorizing an increase in reactor power level describing the measured values of the operating conditions or characteristics of the reactor under the new conditions including:

   a. An evaluation of facility performance to date in comparison with design predictions and specifications, and

   b. A reassessment of the safety analysis submitted with the license application in light of measured operating characteristics when such measurements indicate that there may be substantial variance from prior analysis.

Areas covered by these reports, including the schedules for submittal, are consistent with guidelines set forth in ANSI/ANS-15.1-2007. The contents of the reports are consistent with the guidance provided in NUREG-1537. Therefore, TS 6.7.1 is acceptable to the NRC staff.

9.6.7.2 TS 6.7.2 Special Reports

TS 6.7.2 lists the following special reports that the licensee must prepare, as well as their schedules for completion and submittal:

1. There shall be a report of any of the following not later than the following day by telephone or similar conveyance to the NRC Headquarters Operation Center, and followed by a written report describing the circumstances of the event and sent within 14 days to U.S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington, D.C. 20555:

   a. Any accidental release of radioactivity above permissible limits in unrestricted areas whether or not the release resulted in property damage, personal injury, or exposure;
   b. Any violation of a safety limit;
   c. Any reportable occurrences as defined in Section 1.3.1 of these specifications.

2. A written report within 30 days in writing to the U.S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington, D.C. 20555 of:

   a. Permanent changes in facility organization at Reactor Director or Department Chair level.
   b. Any significant change in the transient or accident analysis as described in the Safety Analysis Report.

Areas covered by these reports, including the schedules for submittal, are consistent with guidelines set forth in ANSI/ANS-15.1-2007. The contents of the reports are consistent with guidance provided in NUREG-1537. Therefore, TS 6.7.2 is acceptable to the NRC staff.

9.6.8 TS 6.8 Records

TS 6.8 lists the type of required records that must be retained and specifies the duration of that retention for various functions, as well as situations that may occur during UWNR operations. This ensures that important information is made available in a timely manner for proper management oversight, which is consistent with guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007.

9.6.8.1 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 lists the records required to be retained by the licensee for 5 years:

1. Normal reactor facility 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 where required by applicable regulations,

6. Experiments performed with the reactor,

7. Fuel inventories, receipts, and shipments,

8. Approved changes in operating procedures,

9. Records of meeting and audit reports of the review and audit group.

The NRC staff compared TS 6.8.1 with NUREG-1537 and ANSI/ANS-15.1-2007. The requirements in TS 6.8.1 were consistent with the guidance and, therefore, acceptable to the NRC staff.

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

TS 6.8.2 lists the records required to be retained by the licensee for one training cycle.

> Record of retraining and requalification of certified operations personnel shall be maintained at all times the individual is employed or until the certification is renewed. For the purposes of this technical specification, a certification is an NRC issued operator license.

In response to RAI No. 31 in a letter dated June 16, 2010 (Ref 20), the licensee proposed to modify TS 6.8.2 to conform to ANSI/ANS-15.1-2007 and 10 CFR 55.59(c)(5).

The NRC staff compared TS 6.8.2 with 10 CFR 55.59(c)(5) and ANSI/ANS-15.1-2007. The requirements in TS 6.8.2, including the proposed modifications, meet the regulations in 10 CFR 55.59(c)(5) and therefore, are acceptable to the NRC staff.

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

TS 6.8.3 lists the records required to be retained by the licensee for the lifetime of the UWNR.

> Annual reports that contain the information in Items 1 and 2 may be used as records for those items:
>
> 1. Gaseous and liquid radioactive effluents released to the environs,
> 2. Offsite environmental monitoring surveys required by technical specifications,
> 3. Radiation exposures for all personnel monitored,
> 4. Updated, corrected, and as-built drawings of the facility,
> 5. Notification that safety limit was exceeded.
> 6. Notification that automatic safety system did not function as required.
> 7. Notification of failure to meet limiting conditions for operation.

The NRC staff compared TS 6.8.3 with NUREG-1537 and ANSI/ANS-15.1-2007. The requirements in TS 6.8.3 were consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007 and, therefore, are acceptable to the NRC staff.

9.7 Conclusions

The NRC staff reviewed and evaluated the TS as part of its review of the application for renewal of Facility License No. R-74, Docket No. 50-156. The TS define certain features, characteristics, and conditions governing the operation of the UWNR. The TS are explicitly included in the renewed facility operating license as Appendix A. The NRC staff reviewed the format and content of the TS for consistency with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. The NRC staff specifically evaluated the content of the TS to determine if they meet the requirements in 10 CFR 50.36, "Technical Specifications." The NRC staff concludes that the UWNR TS do meet the requirements of the regulations. The NRC staff based this conclusion on the following findings:

- To satisfy the requirements of 10 CFR 50.36(a), the licensee provided proposed TS with the application for license renewal. As required by the regulations, the proposed TS included the appropriate summary bases. Those summary bases are part of the TS but not required by the regulations.

- The UWNR 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), the licensee provided TS derived from analyses in the 2008 SAR.

- To satisfy the requirements of 10 CFR 50.36(c)(1), the licensee provided TS specifying a safety limit on the fuel temperature and LSSSs for the reactor protection system to preclude reaching the safety limit.

- The TS contain LCOs on each item that meets one or more of the criteria specified in 10 CFR 50.36(c)(2)(ii).

- The TS contain surveillance requirements that satisfy the requirements of 10 CFR 50.36(c)(3).

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

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

The NRC staff finds the TS to be acceptable and concludes that normal operation of the UWNR within the limits of the TS will not result in radiation exposures in excess of the limits specified in 10 CFR Part 20 for members of the general public or occupational exposures. The NRC staff also finds that the TS provide reasonable assurance that the facility will be operated as analyzed in the 2008 SAR (Ref 9), and that adherence to the TS will limit the likelihood of malfunctions and the potential accident scenarios discussed in Chapter 8, "Accident Analysis," of this SER.

10 FINANCIAL QUALIFICATIONS

10.1 Financial Ability To Operate the Reactor

The regulation in 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.

UW 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 has determined that UW must meet the financial qualifications requirements of 10 CFR 50.33(f), and is subject to a full financial qualifications review. UW must demonstrate that it possesses or has reasonable assurance of obtaining the funds necessary to cover estimated operating costs for the period of the licensee. Therefore, UW 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.

According to the application dated May 9, 2000, the reactor laboratory is a part of the Engineering Physics Department of the College of Engineering at the University of Wisconsin-Madison and that the primary support for operation of the facility is from the state-funded university budget. In a supplement to the application dated June 16, 2010 (Ref 26), the licensee submitted its projected operating costs for the UWNR for each of the fiscal years (FYs) 2011 through 2015, which are estimated to range from $293,132 in FY 2011 to $309,158 in FY 2015. According to the University, the projected sources of funds to cover the operating costs will be provided by State funds, grants and gifts, which include existing state and federal grants and financial gifts made to the University in support of the facility, and auxiliary enterprise income that includes services provided by the facility to users for a fee (e.g., neutron activation analysis, isotope production, training, outreach and consultation). The licensee expects that these funding sources will continue for FYs 2011 through 2015. The NRC staff reviewed the licensee's projected operating costs and projected sources of funds to cover these costs and finds them to be reasonable.

The NRC staff finds that the UW 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 concludes that the University has met the financial qualification requirements in 10 CFR 50.33(f) and is financially qualified to engage in the proposed UWNR activities.

10.2 Financial Ability To Decommission the Facility

The NRC has determined that the requirements to provide reasonable assurance of decommissioning funding are necessary to ensure the adequate protection of public health and safety. 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).

In its application dated May 9, 2000, the licensee included a decommissioning cost estimate, based on the methodology of NUREG/CR-1756, "Technology, Safety and Costs of Decommissioning Reference Nuclear Research and Test Reactors," of $1,200,000 in 2000 dollars. In supplements to the application dated June 16 (Ref 26), July 8, and August 11, 2010, the licensee updated its decommissioning cost estimate basis from NUREG/CR-1756 to the new baseline, which was based on averaging the cost estimates of the Georgia Institute of Technology's research reactor, the Washington State University TRIGA Reactor, and the Oregon State University TRIGA Reactor. The licensee's new baseline decommissioning cost estimate was $9,208,564 in 2007 dollars, and escalated to $9,861,296 in 2010 dollars by using the formula provided in 10 CFR 50.75(c)(2) as the means of adjusting the decommissioning cost estimate for labor, energy, and burial costs. The licensee used factors for labor and energy taken from regional data from the U.S. Department of Labor Bureau of Labor Statistics and the factor for waste burial was taken from the most recent version of NUREG-1307, "Report on Waste Burial Charges," Revision 13, based on the vendor disposal option with the South Carolina disposal site. As of July 1, 2008, only members of the Atlantic Compact are permitted to dispose of waste at the South Carolina disposal site, but since the South Carolina site is closed, for estimating purposes, the use of factors for the South Carolina site are not unreasonable. The NRC staff reviewed the decommissioning cost estimate, and circumstances surrounding current limited disposal options, and concludes that the decommissioning cost estimate for the UWNR appears to be reasonable.

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

The licensee provided an updated SOI, dated June 16, 2010 (Ref 26), stating that the signator's intent is, "…to obtain funds necessary for decommissioning, when necessary, for the University of Wisconsin Nuclear Reactor…." The decommissioning cost estimate is estimated at $9,861,295 for the DECON option.

To support the SOI and the University's qualifications to use a SOI, the application stated that the University of Wisconsin is an agency of the State of Wisconsin, and a part of the state government of the State of Wisconsin. The licensee provided documentation which corroborates this statement. In its application, the licensee also provided information supporting the UW's representation that the decommissioning funding obligations are backed by the full faith and credit of the State of Wisconsin. The licensee also provided documentation verifying that Deborah A. Durcan, Vice President for Finance of the University of Wisconsin System, the signator of the SOI, is authorized to execute contracts on behalf of the University of Wisconsin.

The NRC staff reviewed the information provided by the licensee decommissioning funding assurance as described above and finds that the UW is a State of Wisconsin government licensee under 10 CFR 50.75(e)(1)(iv); the SOI is acceptable; the decommissioning cost estimate appears to be reasonable; and the licensee'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 the licensee 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.

10.3 Foreign Ownership, Control, or Domination

Section 104d of the Atomic Energy Act of 1954, as amended (AEA), prohibits the NRC from issuing a license under Section 104 of the AEA to "any corporation or other entity if the Commission knows or has reason to believe it is owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government." The NRC regulation 10 CFR 50.38, "Ineligibility of Certain Applicants," contains language to implement this prohibition. According to the application, the University of Wisconsin is a State of Wisconsin 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.

10.4 Nuclear Indemnity

The NRC staff notes that the University of Wisconsin currently has an indemnity agreement with the Commission, which does not have a termination date. Therefore, the licensee will continue to be a party to the present indemnity agreement following issuance of the renewed facility operating license. Under 10 CFR 140.71, "Scope," the University, as a non-profit educational institution, is not required to provide nuclear liability insurance. The Commission will indemnity the University of Wisconsin 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, the University of Wisconsin is not required to purchase property insurance under 10 CFR 50.54(w).

10.5 Conclusions

The NRC staff reviewed the financial status of the licensee and finds that the licensee has demonstrated reasonable assurance that the necessary funds will be available to support the continued safe operation of the UWNR and, when necessary, to shut down the facility and carry out decommissioning activities. In addition, the NRC staff does not know or have reason to believe that the licensee is owned, controlled or dominated by a foreign entity. The NRC staff concludes that there are no insurance issues that would preclude the issuance of a renewed facility operating license.

11 CONCLUSIONS

On the basis of its evaluation of the application, as supplemented, and as discussed in the previous chapters of this SER, the NRC staff concludes the following:

- The application for renewal of Facility License No. R-74 filed by the University of Wisconsin ("the licensee") dated May 9, 2000, as supplemented on September 7, 2004; October 17, 2008; twice on June 16, July 8, August 11, November 22, and December 8, 2010; and January 28, and February 8, 2011, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended, and the Commission's rules and regulations set forth in Title 10, Chapter 1, of the *Code of Federal Regulations*;

- The facility will operate in conformity with the application as amended, the provisions of the Act, and the rules and regulations of the Commission;

- There is reasonable assurance (i) that the activities authorized by this renewed license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

- The licensee is technically and financially qualified to engage in the activities authorized by this renewed license in accordance with the rules and regulations of the Commission;

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

12 REFERENCES

1. *Code of Federal Regulations*, Title 10, Chapter I, revised January 1, 2010, U. S. Government Printing Office.

2. U. S. Nuclear Regulatory Commission, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," NUREG-1537 Parts 1 and 2, February 1996.

3. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report on High-Uranium Content, Low-Enriched Uranium-Zirconium Hydride Fuels for TRIGA Reactors," NUREG-1282, August 1987.

4. U.S. Nuclear Regulatory Commission, "Development of Technical Specifications for Experiments in Research Reactors," Regulatory Guide 2.2, November 1973.

5. U. S. Nuclear Regulatory Commission, "University of Wisconsin Conversion Order and the U. S. NRC Safety Evaluation Report—Enclosure 5, June 11, 2009. (ADAMS Accession No. ML091390802).

6. U. S. Nuclear Regulatory Commission, "Interim Staff Guidance On The Streamlined Review Process For License Renewal For Research Reactors," October 15, 2009. (ADAMS Accession No. ML092240244).

7. American National Standards Institute/American Nuclear Society, ANSI/ANS-15.1-2007, "The Development of Technical Specifications for Research Reactors," La Grange Park, IL, 2007.

8. American National Standards Institute/American Nuclear Society, ANSI/ANS-15.11, "Radiation Protection at Research Reactor Facilities," La Grange Park, IL, 1993.

9. University of Wisconsin-Madison, "Safety Analysis Report for the University of Wisconsin Nuclear Reactor, Revision 2 (redacted version)," October 17, 2008, (ADAMS Accession No. ML100740573).

10. University of Wisconsin-Madison, "LEU Conversion Report for the the University of Wisconsin-Madison TRIGA Reactor (redacted version)," August 25, 2008, (ADAMS Accession No. ML090760776).

11. University of Wisconsin-Madison "Response to Request for Additional Information for the Conversion of the University of Wisconsin-Madison TRIGA Reactor from HEU to LEU Fuel (redacted version)," April 10, 2009, (ADAMS Accession No. ML091470391)

12. University of Wisconsin-Madison, "Response to Request for Clarification for the Conversion of the University of Wisconsin-Madison TRIGA Reactor from HEU to LEU Fuel (redacted version)," May 1, 2009, (ADAMS Accession No. ML091470390).

13. University of Wisconsin-Madison, “Response to Request for Clarification for the Conversion of the University of Wisconsin-Madison TRIGA Reactor from HEU to LEU Fuel,” June 8, 2009, (ADAMS Accession No. ML091610704).

14. The University of Wisconsin Nuclear Reactor Laboratory, “Reactor Start-Up Report,” June 21, 2010, (ADAMS Accession No. ML101740432).

15. University of Wisconsin-Madison, “2008–2009 Annual Operating Report for the University of Wisconsin Nuclear Reactor Laboratory,” August 10, 2009, (ADAMS Accession No. ML092240579).

16. Argonne National Laboratory, TRD 070.01006.05, Revision A, “Pulsing Temperature Limit for TRIGA LEU Fuel,” Prepared by General Atomics-ESI, April 2008.

17. J.R. Shoptaugh, Jr., GA-6596, “Simulated Loss-of-Coolant-Accident for TRIGA Reactors,” August 1965, transmitted by a letter dated September 2, 1970, (Docket No. 50-227).

18. M.T. Simnad, “The U-ZrH x Alloy: Its Properties and Use in TRIGA Fuel,” Nuclear Engineering and Design 64 (1981) pp. 403–422.

19. Letter from the Department of Energy (Morgan, R. L.) to the U.S. Nuclear Regulatory Commission (Denton, H.), May 3, 1983.

20. University of Wisconsin-Madison “Response to Request for Additional Information for License Renewal of Facility License No. R-74, University of Wisconsin Nuclear Reactor, TAC No. ME1585 (Technical RAI), dated June 16, 2010, (ADAMS Accession No. ML101690137).

21. University of Wisconsin-Madison, “Submission of Revised Dose Estimates for License Renewal to Facility License No. R-74, University Of Wisconsin Nuclear Reactor, TAC No. ME1585,” February 8, 2011. (ADAMS Accession No. ML110410534).

22. SECY-08-0161, “Review of Research and Test Reactor License Renewal Applications,” dated October 24, 2008, (ADAMS Accession No. ML082550140).

23. Staff Requirements memorandum (SRM) for SECY-08-0161, dated March 26, 2009, (ADAMS Accession No. ML090850159).

24. University of Wisconsin-Madison, “License Renewal of the University of Wisconsin Nuclear Reactor, License R-74, Docket 50-156,” dated May 9, 2000, (ADAMS Accession No. ML093570404).

25. Email from James Wade to Paul Doyle, Subject: Verification of DOE ownership of Fuel, dated May 3, 2010, (ADAMS Accession No. ML101250570).

26. University of Wisconsin-Madison “Response to Request for Additional Information for License Renewal of Facility License No. R-74, University of Wisconsin Nuclear Reactor, TAC No. ME1585 (Financial Qualification RAI), dated June 16, 2010, (ADAMS Accession No. ML101690083).

27. University of Wisconsin-Madison, “Response to Request for Additional Information for License Renewal to Facility License No. R-74 for the University of Wisconsin Nuclear Reactor,” November 22, 2010. (ADAMS Accession No. ML103330040).

28. University of Wisconsin-Madison, “Response to Request for Additional Information for License Renewal to Facility License No. R-74 for the University of Wisconsin Nuclear Reactor,” December 8, 2010. (ADAMS Accession No. ML103480028).

29. University of Wisconsin-Madison, “Submission of Revised Technical Specifications for the License Renewal to Facility License No. R-74 for the University of Wisconsin Nuclear Reactor,” January 28, 2011. (ADAMS Accession No. ML110340310).

30. University of Wisconsin-Madison, “License Renewal of the University of Wisconsin Nuclear Reactor, License R-74, Docket 50-156,” dated September 7, 2004. (ADAMS Accession No. ML093570441).