Safety Evaluation Report Supporting the
Renewal of the Facility Operating License for the
Washington State University Modified TRIGA Nuclear Reactor
License No. R-76
Docket No. 50-027

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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 Washington State University (WSU, the licensee) for a 20-year renewal of Facility Operating License No. R-76 to continue to operate the Washington State University Modified TRIGA Nuclear Reactor. In its safety review, the NRC staff considered information submitted by the licensee, past operating history recorded in the licensee's annual reports to the NRC, inspection reports prepared by NRC personnel, and first-hand observations. On the basis of its review, the NRC staff concludes that WSU can continue to operate the facility for the term of the renewed facility license, in accordance with the license, without endangering public health and safety, facility personnel, or the environment.

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ABBREVIATIONS

$	dollar (of reactivity)
% Δk/k	excess reactivity in percent
10 CFR	Title 10 of the *Code of Federal Regulations*
AEA	Atomic Energy Act of 1954, as amended
ALARA	as low as low as is reasonably achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
Ar-41	argon-41
ARIES	auxiliary reactor emergency supply
BNC	boron neutron capture
BOL	beginning of life
C	Celsius
CCTV	closed-circuit television
cfm	cubic feet per minute
CHF	critical heat flux
Ci	curie
cm^3	cubic centimeter
CPI	Consumer Price Index
DAC	derived air concentration
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DOE	U.S. Department of Energy
DPR	Division of Policy and Rulemaking
EOL	end of life
ft	feet
FY	fiscal year
GA	General Atomics
HEU	highly enriched uranium
IFE	instrumented fuel element
IFR	instrumented fuel rod
in	inch
ISG	interim staff guidance
kW	kilowatt
kW(t)	kilowatt thermal
LEU	low-enriched uranium
LOCA	loss-of-coolant accident
LSSS	limiting safety system setting
μCi/mL	microcuries per milliliter
MDNBR	minimum departure from nucleate boiling ratio
MHA	maximum hypothetical accident
mg	milligram
mhos/cm	micromhos per centimeter
mm	millimeter
mrem	millirem
mrem/hr	millirem per hour
μs	microsecond
MW	megawatt

MWD	megawatt-days
MW(t)	megawatt thermal
N-16	nitrogen-16
NRC	U.S. Nuclear Regulatory Commission
NRCR	WSU Modified TRIGA Nuclear Reactor
NRR	Office of Nuclear Reactor Regulation
RAI	request for additional information
RSC	Reactor Safeguards Committee
SAR	safety analysis report
SER	safety evaluation report
SOI	statement of intent
SOP	standard operating procedure
SRM	staff requirements memorandum
TRIGA	Teaching Research Isotope General Atomics
TS	technical specification(s)
U^{235}	uranium-235
WSU	Washington State University

1. INTRODUCTION

1.1 Overview

By letter dated June 24, 2002, as supplemented by letters dated April 7, May 3, May 24, June 30, July 30, August 4, August 10, August 17, and September 22, 2010, and March 23, July 15, July 18, August 1, and August 26, 2011, Washington State University (WSU, the licensee) submitted to the U.S. Nuclear Regulatory Commission (NRC, the Commission) an application (Reference (Ref.) 1) for a 20-year renewal of Class 104c Facility Operating License No. R-76, NRC Docket No. 50-27, for the WSU Modified TRIGA Nuclear Reactor (NRCR).

The regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.51(a) (Ref. 2) state 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. The NRC reissued the WSU NRCR facility operating license on August 11, 1982, for a period of 20 years expiring on August 10, 2002. A renewal would authorize continued operation of the WSU NRCR facility for an additional 20 years. The WSU NRCR is located east of the main WSU campus in the Nuclear Radiation Center in Pullman, WA. Because the request for license renewal was filed in a timely manner, until the NRC staff completes action on the renewal request, the licensee is permitted to continue operation of the WSU NRCR under the terms and conditions of the existing license in accordance with 10 CFR 2.109, "Effect of Timely Renewal Application."

In 10 CFR 50.64, "Limitations on the Use of Highly Enriched Uranium (HEU) in Domestic Non-Power Reactors," the NRC requires licensees of research and test reactors to convert from the use of highly enriched uranium (HEU) fuel to low-enriched uranium (LEU) fuel, unless specifically exempted. In 2007, WSU proposed to convert the fuel in the WSU NRCR from HEU to LEU. In a letter dated August 15, 2007 (Ref. 3), WSU submitted its application for the conversion, requesting NRC approval of the fuel conversion and of changes to the technical specifications (TS). In its application, the licensee included a safety analysis report (SAR) for the conversion (hereafter called "the conversion SAR") (Ref. 3) in which the change from HEU to LEU and the TS changes were based. After an initial review, the NRC issued requests for additional information (RAI) to the licensee on November 20, 2007, and February 20, 2008. In letters dated December 14, 2007 (Ref. 4), January 15, 2008 (Ref. 5), June 13, 2008 (Refs. 6,7, 8, 9, 10, and 11), and August 4, 22, and 25, 2008 (Refs. 12, 13, and 14), the licensee provided an updated version of the conversion SAR, responses to the RAIs, and the TS that would be applicable after conversion.

The NRC issued an order for WSU to convert to LEU fuel on September 4, 2008. The order included a safety evaluation report (SER), as Enclosure 4 of the order to convert (hereafter called "the conversion SER"), which provided the results of the NRC staff's evaluation of the licensee's conversion request. The order also included changes in the TS that would be required for operation of the facility with LEU fuel (Ref. 15). 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 on April 20, 2009 (hereafter called "the startup report") (Ref. 16).

In this SER for the current license renewal request (hereafter called “this SER”), the NRC staff has fully considered the information and conclusions in the conversion SER and the information provided in the startup report.

The NRC staff conducted its review, with respect to renewing the WSU NRCR facility operating license, on the basis of information contained in the license renewal application as well as in supporting supplements and licensee responses to RAIs. Specifically, the license renewal application included the June 2002 SAR (hereafter called “the license renewal SAR”) (Ref. 1), as supplemented by letters dated April 7, 2010 (Ref. 17), August 4, 2010 (Ref. 18), March 23, 2011 (Ref. 19), July 18, 2011 (Ref. 20), and August 1, (Ref. 51), and August 26, 2011 (Ref. 52). On June 30, 2010 (Ref. 21), July 30, 2010 (Ref. 22), August 10, 2010 (Ref. 23), August 17, 2010 (Ref. 24), September 22, 2010 (Ref. 25), and July 15, 2011 (Ref. 26), the licensee also submitted revised TS that the NRC staff considered during this review. The NRC staff also reviewed the conversion SAR (Ref. 3), the revised conversion SAR (Ref. 7), proposed TS (Refs. 3, 5, 7 and 21 through 26), the licensee’s responses to RAIs (Refs. 4, 5, 6, 8, 9, 10, 11, 12, 13, 14, 17, 18, and 19), the environmental report (Ref. 1), and the startup report (Ref. 16).

The licensee’s application and other material may be examined and copied for a fee at the NRC’s Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, MD. The NRC maintains an Agencywide Documents Access and Management System (ADAMS), which provides text and image files of the NRC’s public documents. Documents related to this license renewal review may be accessed through the online NRC Library at http://www.nrc.gov.

The dates and ADAMS accession numbers of the licensee’s renewal application and associated supplements are listed in Chapter 7 of this SER, “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,”10 CFR Part 30, “Rules of General Applicability to Domestic Licensing of Byproduct Material,” 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” 10 CFR Part 51, “Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions,” 10 CFR Part 70, “Domestic Licensing of Special Nuclear Material,” and 10 CFR Part 73, “Physical Protection of Plants and Materials”; the recommendations of applicable regulatory guides; and relevant accepted industry standards, such as the American National Standards Institute/American Nuclear Society (ANSI/ANS) 15 series. The NRC staff also referred to the recommendations contained in NUREG-1537, “Guidance for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,” issued February 1996 (Ref. 27). 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. 28), 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. 29). 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) (Ref. 30) 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 levels 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. 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 WSU license renewal application review using the final ISG (Ref. 30) and, since the licensed power level for the WSU NRCR is less than 2 MW(t), the NRC staff performed a focused review of 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 the 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.

In its application, the licensee states that the current existing "Physical Security Plan for the Washington State University TRIGA Facility," approved on September 12, 1984, meets the current standards and is still valid for the facility. Thus, a new plan need not be submitted because the plan remains in place and no changes were proposed as part of license renewal. 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, these changes did not decrease the effectiveness of the plan and the NRC-approved plan remains in place.

In its application, the licensee states that the emergency plan of September 1963, as amended, is still valid for the facility. Because the plan remains in place and no changes have been proposed as part of license renewal, a new plan need not be submitted. By letter dated June 11, 1984, the NRC staff informed the licensee that the Emergency Plans for WSU submitted on September 6, 1983, was acceptable. 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.

In its application, the licensee states that the current existing "Operator Requalification Program for the Washington State University TRIGA Facility," dated March 22, 1989, meets the current requirements. Because the existing plan remains in place and no changes have been proposed as part of the license renewal, a new program need not be submitted. However, by letter dated August 26, 2009, the licensee submitted an updated requalification plan, dated January 16, 2008, as supplemented by letter dated May 4, 2011. In accordance with the guidance provided in the ISG, the requalification plan was not reviewed as part of this license renewal. The requalification plan for WSU has been reviewed under the current licensing process and was approved by letter dated May 19, 2011.

The purpose of this SER is to summarize the findings from the safety review of the WSU NRCR and to delineate the technical details considered in evaluating the radiological safety aspects of continued operation. This SER provides the basis for renewing the license for operation of the WSU NRCR at steady-state thermal power levels up to and including 1.0 MW(t) and short-duration power pulses with reactivity insertions up to approximately $2.00. 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 A. Francis DiMeglio, Project Manager from the NRC's Office of Nuclear Reactor Regulation (NRR), Division of Policy and Rulemaking (DPR), Research and Test Reactors Projects Branch, and 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 <u>Summary and Conclusions on Principal Safety Considerations</u>

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 WSU NRCR, 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 license renewal SAR (Ref. 1), as supplemented, in accordance with the TS are safe, and safe operation can reasonably be expected to continue.

- The expected consequences of a broad spectrum of postulated credible accidents and a maximum hypothetical accident (MHA) have been considered, emphasizing those that could lead to a loss of integrity of fuel rod cladding and a release of fission products. The licensee performed conservative 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 10 CFR Part 20 doses 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 that provide 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 is 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 license renewal SAR (Ref. 1), as supplemented, 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's procedures for training its reactor operators and the operator requalification plan give reasonable assurance that the licensee will continue to have qualified personnel who can safely operate the reactor.

- 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).

- The licensee maintains an emergency plan in compliance with 10 CFR 50.54(q) and Appendix E to 10 CFR Part 50, which provides reasonable assurance that the licensee will continue to be prepared to assess and respond to emergency events. All changes to the Emergency Plan have been made in accordance with 10 CFR 50.54(q).

On the basis of these findings, the NRC staff concludes that WSU can continue to operate the WSU NRCR in accordance with the Atomic Energy Act of 1954, as amended (AEA), NRC regulations, and the renewed facility operating license without endangering public health and safety.

1.3 General Description of the Facility

In March 1961, the U.S. Atomic Energy Commission issued a facility operating license to WSU for operation of a research reactor on its campus located near Pullman, WA. Facility Operating License No. R-76 authorized the WSU reactor to operate at steady-state power levels up to 100 kilowatts (kW). In 1967, the reactor was converted to the use of General Atomics training reactor and isotopes production (TRIGA) fuel, and the facility operating license was amended to allow operation up to its current power level of 1,000 kW steady state and pulsed to an average maximum power level of 1,200 MW(t). In September 2008, the reactor license was amended to permit conversion of the reactor fuel from a combination of HEU and LEU to all LEU. The WSU NRCR returned to steady-state operation in October 2008 and pulsing operation in March 2009.

Currently, the WSU NRCR operates with two types of TRIGA LEU fuel: standard and 30/20. The standard TRIGA fuel is 8.5 weight percent uranium, enriched to 19.75 percent of uranium-235 (U^{235}). The 30/20 TRIGA fuel is 30 weight percent uranium, enriched to 19.75 percent of uranium-235 (U^{235}). Water is used as the coolant and moderator. The primary coolant system consists of a 26-foot-deep concrete pool in which the reactor core is submerged. Heat generated from the reactor core is directly transferred to the pool water by natural convection. Reactor pool water is kept at 50 degrees C or less by a closed-loop cooling system with a design flow rate of 450 gallons per minute. A pump takes water from a pipe connected to the reactor pool, passes it through the tube side of a stainless steel heat exchanger, and returns it through a pipe to the reactor pool. This system provides the heat removal capability for the water in the reactor pool. Heat is removed from the shell side of the heat exchanger by an evaporative cooling tower. Water from the cooling tower basin is pumped through the heat exchanger at a nominal flow rate of 900 gallons per minute and returned to the tower. Makeup to the tower basin necessitated by evaporation losses is provided by the domestic water supply and is controlled by level instrumentation located in the basin. Design features of this system allow transfer of reactor heat from the primary system under all operating conditions, but this transfer 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 rods (blades) using boron carbide as the neutron absorber are moved in and out of the reactor core by individual mechanical drives. Three blades can be disengaged to drop by gravity into the core for safety purposes. An additional transient control rod with a combination of pneumatic-electromechanical drive may be used either as a control or transient rod generating a neutron pulse in pulsing mode operation.

1.4 <u>Shared Facilities and Equipment</u>

The WSU NRCR is contained in a separate room, with minimal penetrations, within the Nuclear Radiation Center building. Shared facilities include demineralized water and the drain system for radioactive liquid waste. Potentially contaminated drains from laboratories, the reactor equipment rooms, and the reactor pool room flow into a common collection system. The collection system discharges into a retention tank that is sampled before release to the campus sanitary system. Offices for the reactor program personnel and some laboratories are located in the adjoining rooms in the building. The reactor pool room does not rely on other building services such as electricity or ventilation. Power to the reactor facility is provided by a separate circuit, with circuit breakers independent from the other electrical power to the Nuclear Radiation Center building. The main power transformer serving the Nuclear Radiation Center building and its main breaker are the only electrical common points. Air from the reactor facility is exhausted through an independent ventilation system that is monitored continuously and can be routed through an absolute filter during an emergency. The exhaust air can also be diluted with outside air and exhausted through a stack extending from the roof of the building. This 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 (Ref. 31). The WSU NRCR is very similar in design to TRIGA reactor facilities at Texas A&M University and the University of Wisconsin. Instruments and controls used in the WSU NRCR facility are similar to most nonpower reactors licensed by the NRC. The pool size and experimental facility configuration differ among the three reactors, but basic reactor behavior and accident analyses are similar.

The reactor at Texas A&M University converted its fuel to the same LEU fuel elements used in the WSU NRCR. The TRIGA Mark F reactor at GA in San Diego was operating with a core partially made up of high-density LEU fuel when it was permanently shut down. There have been no performance issues with respect to the use of this fuel in these reactors (Ref. 32).

1.6 Summary of Operations

The WSU NRCR is used for nuclear research and a range of irradiation services such as isotope production, neutron activation analysis, and boron neutron capture (BNC) therapy experiments. Use of the reactor facility over the past 20 years has been high. Since the reactor was converted to the use of LEU in 2008, it has accumulated 859.5 MW-hours of operation as of June 30, 2010, and has been pulsed 39 times. The licensee expects this use of the reactor for the upcoming license renewal period to continue (Ref. 1).

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 a facility operating license for a research or test reactor, that the applicant shall 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 DOE has determined that universities and other government agencies operating non-power reactors have entered into contracts with DOE providing that DOE retains title to the fuel and is obligated to take the spent fuel and high-level waste for storage or reprocessing (Ref. 33). An e-mail sent from James Wade of DOE to Paul Doyle of the NRC, dated May 3, 2010 (Ref. 34), reconfirms this obligation with respect to the fuel at the WSU NRCR (DOE Contract No. 74511), valid from June 1, 2008, to December 31, 2012. By entering into such a contract with DOE, WSU has satisfied the requirements of the Nuclear Waste Policy Act of 1982.

1.8 Facility Modifications and History

Review of the modifications made during the last 20 years indicates that most modifications were technological upgrades to instrumentation or minor changes to the existing design that either enhanced capabilities or improved reactor operations. All of these modifications were subject to evaluation under 10 CFR 50.59, “Changes, Tests and Experiments,” to ensure there was no impact on the safety of the WSU NRCR. Review of the license amendments showed that they were mostly administrative in nature.

On September 4, 2008, the NRC issued an order (Ref. 15) modifying the WSU NRCR facility operating license to allow conversion from HEU to LEU fuel. Following this order, the licensee replaced the HEU fuel, enriched to 70 percent, with TRIGA LEU new conversion fuel, enriched to less than 20 percent. The reactor returned to operation in April 2009 loaded exclusively with LEU fuel.

WSU did not request any substantive changes as part of this license renewal application.

1.9 Financial Considerations

1.9.1 Financial Ability To Operate the Facility

The regulation at 10 CFR 50.33(f) states the following:

> 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.

WSU does not qualify as an "electric utility," as defined in 10 CFR 50.2, "Definitions." Further, pursuant to 10 CFR 50.33(f)(2), applicants to renew or extend the term of any operating license for a non-power reactor shall include the financial information that is required in an application for an initial license. The NRC staff has determined that WSU must meet the financial qualifications requirements pursuant to 10 CFR 50.33(f) and is subject to a full financial qualifications review. WSU must demonstrate that it possesses or has reasonable assurance of obtaining the funds necessary to cover estimated operating costs for the period of the license. Therefore, WSU must submit estimates of the total annual operating costs for each of the first 5 years of facility operations from the expected license renewal date and indicate the sources of funds to cover those costs.

In a supplement to the application dated April 7, 2010 (Ref. 35), WSU submitted its projected operating costs for the WSU NRCR for each of the fiscal years (FYs) 2012 through 2016. The projected operating costs for the WSU NRCR are estimated to range from $441,495 in FY 2012 to $516,486 in FY 2016. Funds to cover operating costs will be provided by State funding, additional sources of income, such as work for others and grants, and extramural support (e.g., contracts, grants). The NRC staff reviewed WSU's estimated operating costs and projected sources of funds to cover those costs and finds them to be reasonable.

The NRC staff finds that WSU 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 WSU has met the financial qualifications requirements in 10 CFR 50.33(f) and is financially qualified to engage in the proposed WSU NRCR activities.

1.9.2 Financial Ability To Decommission the Facility

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

The licensee's application for license renewal, dated June 24, 2002 (Ref. 1), referenced a decommissioning cost estimate of $4,994,615 in 2000 dollars based on a detailed technical and cost proposal for the decommissioning of the WSU NRCR that was provided by the Nuclear Division of Westinghouse Electric Corporation in 1989. In supplements to the application dated April 7, 2010 (Ref. 35), and May 3, 2010 (Ref. 36), WSU updated the decommissioning cost estimate to $7,093,119 in 2011 dollars using the U.S. Bureau of Labor Statistics Consumer Price Index (CPI) for all urban consumers. According to WSU, upon further examination of current waste disposal costs and comparing the annually updated WSU cost estimate based on annual CPI incremental increases, WSU reevaluated the decommissioning cost estimate to determine whether or not a more comprehensive determination was warranted. As a result, WSU estimated the decommissioning cost to be between $12,000,000 and $14,600,000 in 2011 dollars. The cost estimate summarized costs by labor, radioactive wastes disposal, energy, and a 25-percent contingency factor. (WSU combined labor, tools, equipment, and supplies into a single category of "labor" and combined energy, fees, insurance, and travel into a single category of "energy.") According to WSU, the decommissioning cost estimate is based on information provided in NUREG-1307, "Report on Waste Burial Charges: Changes in Decommissioning Waste Disposal Costs at Low-Level Waste Burial Facilities" (Ref. 37), the U.S. Social Security Administration's Average Wage Index, the 2010 cost schedule provided by U.S. Ecology Washington for the Richland, WA facility, and the CPI. According to WSU, the decommissioning cost estimate will be adjusted annually, with the values for labor and energy adjusted annually based on the CPI or Average Wage Index values, and the waste disposal costs will be determined according to the standard published rates for waste disposal. The NRC staff has reviewed the material submitted by WSU (Chapter 17 of the license renewal SAR) (Ref. 1 and responses to the RAIs) concerning decommissioning of the reactor facility and the cost estimates provided in the material and also recognizes concerns surrounding current limited disposal options. The NRC staff concludes that the decommissioning approach and cost estimates submitted by WSU are reasonable.

WSU 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.

WSU provided an SOI, dated May 11, 2010 (Ref. 38), stating that the signator "intends to seek and obtain such funds sufficiently in advance of decommissioning to prevent delay of required activities." The decommissioning cost estimate is between $12,000,000 and $14,600,000 for the DECON option.

To support the SOI and WSU's qualifications to use an SOI, the application stated that WSU is a land grant educational institution and a part of the Government of the State of Washington and included documentation that corroborates this statement. The application also provided information supporting WSU's representation that the decommissioning funding obligations of WSU are backed by the full faith and credit of the State of Washington. WSU also provided documentation verifying that Elson S. Floyd, President of WSU, the signator of the SOI, is authorized to execute contracts on behalf of WSU.

The NRC staff reviewed WSU's information on decommissioning funding assurance and finds that WSU is a State of Washington government licensee under 10 CFR 50.75(e)(1)(iv); the SOI is acceptable; the decommissioning cost estimate as well as the costs for the DECON option are reasonable; and WSU'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 resulting from the availability of disposal facilities, and that WSU has an obligation under 10 CFR 50.9, "Completeness and Accuracy of Information," to update any changes in the projected cost, including changes in costs resulting from increased disposal options.

1.9.3 Foreign Ownership, Control, or Domination

Section 104d of the AEA prohibits the NRC from issuing a license under Section 104 of the AEA to "any corporation or other entity if the Commission knows or has reason to believe it is owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government." The regulation in 10 CFR 50.38, "Ineligibility of Certain Applicants," contains language to implement this prohibition. According to the application, WSU is a State of Washington government licensee and is not owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government. The NRC staff does not know or have reason to believe otherwise.

1.9.4 Nuclear Indemnity

The NRC staff notes that WSU currently has an indemnity agreement with the Commission, which does not have a termination date. Therefore, WSU 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," WSU, as a nonprofit educational institution, is not required to provide nuclear liability insurance. The Commission will indemnify WSU 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 under 10 CFR 140.95, "Appendix E—Form of Indemnity Agreement with Nonprofit Educational Institutions," up to $500 million. Also, WSU is not required to purchase property insurance under 10 CFR 50.54(w).

1.9.5 Conclusions on Financial Considerations

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

2. REACTOR DESCRIPTION

2.1 Summary Description

The WSU NRCR is a natural convection, water-cooled, and shielded reactor that was converted to the use of TRIGA fuel. Aspects of its design are similar to many other research reactors operating in the United States and abroad. The WSU NRCR was originally designed for plate-type, material test reactor fuel assemblies but was converted to the use of TRIGA fuel rods contained in assemblies that can hold up to four rods each. The WSU NRCR, which operated for 27 years with a combination of HEU and LEU fuel, was converted to the use of LEU fuel and returned to operation using LEU fuel in April 2009. The TRIGA fuel is uranium-zirconium hydride ($U\text{-}Zr_x$). The reactor is presently fueled with two types of LEU fuel: 30/20 $U\text{-}Zr_x$ where x is approximately 1.6 and 8.5/20 $U\text{-}Zr_x$ where x is approximately 1.7.

The reactor core is contained within a grid box. The fuel in the core may be surrounded by graphite reflectors and irradiation facilities, including a pneumatic transfer tube, a rotating rack, irradiation tubes, a cadmium-lined experiment facility, beam tubes, and a thermal column.

The reactor core is located near the bottom of a water-filled concrete pool. The radial shielding for the reactor consists of water and ordinary concrete and, in some directions, lead and graphite. About 20 feet of water serve as shielding above the core. Four blades moving in shrouds fixed in the grid box serve as shim-safety control rods and a regulating rod. Additionally, a water-followed transient rod, located in the fourth position of a fuel assembly with a three-rod fuel cluster, is used for control and pulsing.

The reactor normally operates at a maximum thermal power level of 1 MW(t). The reactor can also be pulsed up to a peak power that corresponds to a peak fuel temperature of 830 degrees C. The reactor core is cooled through natural convection of the pool water.

The inherent safety of TRIGA reactors has been demonstrated by the extensive experience gained from similar designs used throughout the world. The TRIGA fuel is characterized by inherent safety, high fission product retention, and the ability to withstand water quenching at temperatures as high as 1,150 degrees C. The safety of the fuel arises from the strongly negative prompt temperature coefficient characteristic of $U\text{-}Zr_x$ fuel-moderator elements. As the fuel temperature rises, this coefficient immediately compensates for reactivity insertions. The WSU fuel temperature safety limit (TS 2.1) is specified not to exceed 1,000 degrees C for 8.5/20 $U\text{-}Zr_x$ fuel and 1,150 degrees C for 30/20 $U\text{-}Zr_x$ fuel under any conditions of operation. To ensure that this safety limit is not exceeded, the limiting safety system setting (LSSS), TS 2.2, is established for steady-state or pulse operation for fuel temperature to be less than or equal to 500 degrees C as measured in the instrumented fuel rod (IFR) located in specific locations of the core (see Section 2.5.3 of this SER for a discussion of the safety limit and LSSS).

On August 15, 2007, WSU submitted a supplemental conversion SAR (Ref. 3) requesting an amendment to its facility operating license that would allow conversion to the exclusive use of LEU fuel. WSU then submitted additional information, based on NRC RAIs, regarding the conversion (Refs. 4, 5, 6, 7, 8, 9, 10, 11, 12, 13, and 14). On the basis of its review, on September 4, 2008, the NRC issued an order for WSU to convert to LEU fuel. The order included the conversion SER that provided the results of the NRC staff's evaluation of the licensee's conversion request. The order also included changes in the TS that would be required for operation of the facility with LEU fuel (Ref. 15). The reactor returned to steady-state operation in October 2008 and pulsing operation in March 2009.

2.2 Reactor Core

The WSU NRCR uses solid fuel rods in which the zirconium hydride moderator is homogenously combined with low-enrichment fuel. The fuel rods are in a three- or four-rod cluster configuration. The fuel clusters are contained within an assembly or bundle that resembles the outline of the original material test reactor 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 consisting of a cast aluminum grid plate suspended from a movable bridge by four flooding corner posts that form a suspension frame. The grid plate in the grid box provides a seven-by-nine array of square holes for fuel assemblies. The grid plate also accepts reflector elements, experiment devices, a neutron source holder, and other elements. Two shrouds, in which three standard control rods (shim-safety blades) and one regulating rod (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 WSU NRCR uses five control rods: three scrammable, motor driven, shim-safety blades; one nonscrammable, motor driven, regulating (servo) blade; and one scrammable, water-followed, pneumatic, electromechanical transient rod. The transient rod moves in the fourth position of a three-rod cluster fuel assembly. The control rods are discussed in Section 2.2.2 of this SER.

The primary reactor water is deionized and routinely monitored 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 is used to reduce the radiation exposure level on the bridge from nitrogen-16 (N-16). 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 7.13-second half-life of N-16.

Four instrument channels monitor reactor neutron flux and power level, with neutron sensors located in the corner posts of the suspension frame.

TS 1.0 defines the reactor core components as follows:

TS 1.0 DEFINITIONS

30/20 Fuel: 30/20 fuel is TRIGA fuel that contains a nominal 30 weight percent of uranium with a ^{235}U enrichment of less than 20% and erbium, a burnable poison.

(….)

Control Rod: A control rod is a device fabricated from neutron-absorbing material that is used to initiate neutron flux changes and to compensate for reactivity changes. A control rod can be coupled to its drive unit allowing it to perform a safety function when the coupling is disengaged. All such reactor control devices for the WSU reactor are referred to as control rods, irrespective of the specific geometry of the devices. The following types of control rods are in use:

(1) Regulating Rod: The regulating rod is a low-worth (low reactivity) control rod fabricated from stainless steel, and is used primarily to maintain an intended power level and does not have scram capability. Its position is varied by means of an electric motor-operated positioning system. The electric motor-operated positioning system moves the control rod into or out of the reactor core in response to a signal initiated by the reactor operator when the console mode selector switch is set in the manual or auto position or in response to a signal generated within the control console when the console mode selector switch is set in the auto position.

(2) Transient Rod: The transient rod is a control rod that has a scram capability and is capable of providing rapid reactivity insertion to produce a pulse. The transient rod is positioned by controlled movement of a pneumatic cylinder that moves the cylinder and control rod together when the console mode selector switch is in the manual or auto mode and air pressure is applied, or the control rod can be rapidly moved by application of air pressure to move the control rod drive within the pneumatic cylinder when the console mode selector switch is in the pulse mode.

(3) Standard Control Rod: Standard control rod shall mean any control rod which has a scram capability, which is utilized to vary the reactivity of the core, and which is positioned by means of an electric motor-operated positioning system. The electric motor-operated positioning system moves the control rod into or out of the reactor core in response to a signal initiated by the reactor operator when the console mode selector switch is set to the manual or auto position.

Core Configuration: The core configuration includes the number, type, or arrangement of fuel rods, reflector elements and regulating, transient, or standard control rods occupying the core grid.

Core Lattice Position: Core lattice position refers to specific locations in the WSU reactor core. The core lattice positions are denoted by a letter-number sequence with the letters A through G and the numbers one through nine, where the letters denote rows and the numbers denote columns. Each letter-number sequence may be followed by a directional indicator, NE, SE, SW or NW, which are compass directional indicators denoting a particular quadrant in a core lattice position.

(….)

Fuel Assembly: A fuel assembly is a cluster of three or four fuel rods fastened together in a square array by a top handle and bottom grid plate adapter. A fuel assembly is also sometimes referred to as a fuel bundle.

Fuel Rod: A fuel rod is a single TRIGA-type fuel rod of either Standard TRIGA or 30/20 TRIGA fuel.

(…)

Instrumented Fuel Rod: An instrumented fuel rod is a fuel rod in which thermocouples have been embedded for the purpose of measuring the fuel temperature during reactor operation. The Instrumented Fuel Rod or Instrumented Fuel Element is sometimes referred to by the acronyms "IFR" or "IFE."

(….)

Mixed Core: A mixed core is a core arrangement containing Standard and 30/20 TRIGA fuels.

(….)

Operational Core: An operational core is any arrangement of TRIGA fuel that is capable of operating within the maximum licensed power level and that satisfies all the requirements of the Technical Specifications.

(….)

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

(….)

Standard Fuel: Standard fuel is TRIGA fuel that contains a nominal 8.5 weight percent of uranium with a ^{235}U enrichment of less than 20%.

(….)

Vacant Core Position: A vacant core position is a core grid position which does not have a fuel assembly, reflector, or experimental apparatus installed in the grid position.

These are standard definitions used in research reactor TS and are therefore, acceptable to the NRC staff. These definitions are also consistent with the guidance provided in ANSI/ANS-15.1-2007, "The Development of Technical Specifications for Research Reactors" (Ref. 39).

During the startup testing of the conversion core, an operational core configuration was established with 13 30/20 LEU fuel assemblies containing 51 fuel rods, 1 of which is an IFR, and 17 8.5/20 LEU fuel assemblies. The 30/20 LEU fuel assemblies were located in a contiguous block in the central region of the core. There were no contiguous water holes in the 30/20 LEU region of the core. The IFR in the core was located in the 30/20 LEU fuel region. Adjustments were made in the core until the cold, xenon-free critical excess reactivity of the core was $7.44 (5.58 percent excess reactivity (% Δk/k)). After calibration of the control rods, the shutdown margin was determined to $1.23 and is acceptable. This core was designated as operational core 35A. Based on this core configuration, calculations were performed to determine the locations and setting for the instrumented fuel element (IFE) that would prevent the safety limit from being exceeded. See further discussion in Section 2.5.3 of this SER.

TS 5.3 provides the design requirements for the WSU NRCR core as follows:

TS 5.3 Reactor Core

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

(2) The TRIGA core may be composed of 30/20 fuel or a combination of standard and 30/20 fuel (mixed cores) provided that the 30/20 fuel region contains at least 51 30/20 fuel rods located in a contiguous block in the central region of the core.

(3) A reactor core fueled with a mixture of fuel types shall not be operated with a vacant core lattice position in the 30/20 fuel region. Water holes in the 30/20 fuel region shall be limited to single-rod holes. Lattice positions in the fueled region of the core that are not occupied by fuel assemblies, reflectors or experiments shall be occupied by fixtures that will prevent the installation of a fuel assembly into a position not occupied by a fuel assembly, reflector or experiment.

(4) The reflector, excluding experiments and experimental facilities, shall be water or a combination of graphite, aluminum and water.

TS 5.3(1) controls a design criterion requiring the use of TRIGA fuel rods. TRIGA cores have been in use for years, and their characteristics are well documented.

TS 5.3(2) helps ensure that reactor lattice positions contain either standard or 30/20 fuel rods, with the central locations occupied by only 30/20 fuel rods to control power peaking.

TS 5.3(3) helps ensure that internal core lattice positions are occupied with fuel rods, with water holes limited to one single-rod hole, thereby controlling power peaking. Vacant lattice positions are filled with mechanical devices that reduce the probability of an accidental reactivity insertion.

TS 5.3(4) places a limitation on reflector elements requiring the use of only water or a combination of water with graphite and aluminum.

TS 5.3 establishes the design criteria for the reactor core. The NRC staff finds TS 5.3 acceptable for continued operation of the WSU NRCR based on the analysis in the LEU conversion SER (Ref. 15, Enclosure 4, Section 2).

TS 3.1.5 specifies the limitation on core configuration as follows:

TS 3.1.5 Core Configuration Limitation

(1) The 30/20 LEU fueled region in a mixed core shall contain at least 51 30/20 fuel rods in a contiguous block of fuel in the central region of the

reactor core. Water holes in the 30/20 LEU region shall be limited to nonadjacent single fuel rod holes.

(2) Instrumented fuel elements shall be placed in the core grid positions specified in Section 2.2, Limiting Safety System Settings.

TS 3.1.5(1) helps ensure that the central core lattice positions are filled with 30/20 fuel rods and controls the location of the water holes in the reactor core. The purpose of controlling the location and minimum number of 30/20 LEU fuel rods and water holes is to control power peaking in fuel rods.

TS 3.1.5(2) requires that the IFE be placed in specific locations in the core. The allowable core locations where an IFE may be placed were determined during the conversion of the core to use LEU fuel (Ref. 12) and were reviewed by NRC staff. On the basis of its review, the NRC staff found the specific core locations selected for IFE placements acceptable (Ref. 15). The licensee's analysis has shown that, for both the hottest and the coldest thermocouples in the IFE, an IFE located in the allowable core positions stated in TS 2.2 would protect the fuel temperature safety limit of 1,150 degrees C for 30/20 LEU fuel at reactor power levels that are less than 1.7 MW(t) and limit the maximum steady-state temperature in the 30/20 fuel region to less than 800 degrees C (see further discussion in Section 2.5.3 of this SER).

The surveillance requirement for the core configuration limitation is presented in TS 4.1.5 as follows:

TS 4.1.5 Core Configuration Limitation

(1) Proposed changes in core configuration shall be analyzed to determine whether amendments to the reactor license or Technical Specifications are required.

(2) Changes in fuel configuration shall be documented by a Safety Analysis Report and recorded in a fuel inventory log.

(3) Each change to the core configuration shall be evaluated to determine the allowed locations for the Instrumented Fuel Element.

TS 4.1.5(1), (2), and (3) require that a proposed change in core configuration be analyzed to determine if a license or TS amendment is required and to determine power peaking and heat transfer characteristics and to determine acceptable positions for the IFE.

Because TS 3.1.5 and TS 4.1.5 establish core configuration limitations based on the analysis performed for the conversion to LEU, the NRC staff finds TS 3.1.5 and TS 4.1.5 acceptable for continued operation of the WSU NRCR based on the LEU conversion SER (Ref. 15).

The reactivity limitations for the core are defined by TS 3.1.3 and TS 3.1.4 as follows:

TS 3.1.3 Shutdown Margin

The reactor shall not be operated unless the shutdown margin provided by control rods is $0.25 or greater with:

(1) all experiments with positive reactivity in the most reactive state;

(2) the value of all experiments with negative reactivity not used in the shutdown margin determination;

(3) the highest worth scrammable control rod and the non-scrammable control rod fully withdrawn;

(4) the reactor in the reference core condition.

TS 3.1.3(1) through (4) define the shutdown margin, ensuring that the reactor can be shut down by an acceptable margin.

TS 3.1.3(1) and (2) place constraints on the core condition by considering that all experiments be in their most reactive state to ensure that the reactor is not subcritical because of an experiment that could be removed from the core.

TS 3.2(3) helps ensure that the reactor can be shut down even if the highest worth control rod and nonscrammable control rods become stuck out of the reactor core.

TS 3.2(4) establishes the reference core 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 3.1.4 Maximum Excess Reactivity

The maximum excess reactivity based on the reference core condition shall not exceed 5.6% $\Delta k/k$.

TS 3.1.4 establishes a limit on excess reactivity, allowing operational flexibility while limiting the reactivity available for reactivity addition accidents. The maximum excess reactivity helps establish a basis for ensuring that an adequate shutdown margin is available by control rod insertion.

The excess reactivity is equivalent to $7.46. During the initial startup testing of core 35A, the maximum reactivity in excess of cold, clean was measured at $7.44. After operation to build in fission products, the excess reactivity was measured at $7.35. After installation of four experimental tubes and a cadmium-lined experiment tube, the final core excess reactivity of core 35A was measured at $7.12 (5.34 % $\Delta k/k$).

The calculated core lifetime for core 35A with no fuel changes is approximately 1,000 megawatt-days (MWD) with full equilibrium xenon. At the end of life (EOL), 1,000 MWD, the reactor core still has about $0.40 excess reactivity for experiments.

The corresponding surveillance requirements for the shutdown margin and the excess reactivity are presented in TS 4.1.3 and TS 4.1.4 as follows:

TS 4.1.3 Shutdown Margin

The surveillance requirements in this section may be postponed during periods of reactor shutdown. If the surveillance requirement occurs during a period of reactor shutdown the surveillance shall be completed upon resumption of reactor operation.

(1) The reactivity worth of each control rod and the shutdown margin shall be determined annually.

(2) The reactivity worth of each control rod and the shutdown margin shall be determined after a change to the type or location of fuel, reflector, or control rods in the reference core, or after any change to the reference core that results in or could result in a change of reactivity of $0.25 or more.

(3) The reactivity worth of an experiment shall be estimated or measured, as appropriate, before reactor operation with the experiment.

TS 4.1.3(1) and (2) require that the shutdown margin be determined annually and after changes are made in the core, which could result in a $0.25 or larger change of reactivity.

TS 4.1.3(3) requires that the reactivity worth of an experiment be determined before reactor operation with the experiment.

TS 4.1.4 Maximum Excess Reactivity

The core excess reactivity shall be determined annually or following a change to the core that causes a change in reactivity greater than $0.25. This surveillance requirement may be postponed during periods of reactor shutdown. If this surveillance requirement occurs during a period of reactor shutdown the surveillance shall be completed upon resumption of reactor operation.

TS 4.1.4 requires that the excess reactivity be determined annually or following core changes that cause a change in reactivity greater than $0.25.

The NRC staff finds that TS 4.1(1) through (4) are consistent with NUREG-1537 and ANSI/ANS-15.1-2007. On this basis, the NRC staff concludes that this surveillance will help ensure that the shutdown margin and core excess reactivity 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, the effects of burnup and temperature, power peaking, and pulsing. The NRC staff also reviewed an LEU startup plan designed to experimentally determine some of these parameters. The startup plan required a startup report to be prepared by the licensee and submitted to the NRC. The NRC staff also reviewed the startup report (Ref. 16). On the basis of its review, the NRC staff finds that the changes in nuclear design of the core resulting from the fuel conversion, represented in part by the TS described above, were acceptable (Ref. 15).

TS 5.3, 3.1.5, 3.1.3, and 3.1.4, related to the normal operating conditions of the reactor core, include limits on the allowable core configurations, shutdown margin, and excess reactivity. TS 4.1.5, 4.1.3, and 4.1.4 relate to the corresponding surveillance requirements and reactivity worths of the control rods. The NRC staff finds these TS consistent with NUREG-1537 and ANSI/ANS-15.1-2007. The NRC staff finds that the analysis presented in the 2002 SAR (Ref. 1) and the LEU revised conversion SAR (Ref. 7) justify these TS and show that normal 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 operation during the period of the renewed facility operating license. The staff further concludes that the TS provide reasonable assurance that normal operation of the WSU NRCR core will not pose a significant risk to the public health and safety or the environment.

2.2.1 Reactor Fuel

The WSU NRCR fuel components are defined in TS 1.0 and described in Section 2.2 of this SER. TRIGA fuels have safety features that include a large prompt negative temperature coefficient of reactivity, high fission product retention, chemical stability when quenched from high temperatures in water, and dimensional stability over a wide range of temperatures. Over 25,000 pulses have been performed domestically and abroad with TRIGA fuel rods, with temperatures reaching peaks of about 1,150 degrees C.

The WSU NRCR is fueled by two types of LEU TRIGA fuel rods, as described in Table 2-1.

Table 2-1 Description of TRIGA Low-Enriched-Uranium Fuel Rods

	TRIGA 30/20 LEU	**TRIGA 8.5/20 LEU Standard**
Uranium content, wt-%	30	8.5
Enrichment, nominal U^{235}, %	19.75	19.75
U^{235} content, nominal	—	—
Erbium content, wt-%	0.9	0
Diameter of fuel meat, mm	34.823	34.823
Length of fuel meat, mm	381	381
Cladding	304 stainless steel	304 stainless steel
Cladding thickness, nominal, mm	---	---

A hole is drilled through the center of the active fuel section to facilitate hydriding, and a zirconium rod is inserted in this hole once hydriding is complete. Two 3.5-inch sections of graphite are placed above and below the fuel to serve as top and bottom reflectors for the core. Stainless steel end fittings used for handling and positioning are welded to both ends. Three or four of these fuel rods are installed in an assembly, creating a fuel bundle. This assembly is required for placement of the fuel on the grid plate.

The WSU NRCR uses one IFE, which is identical to the 30/20 LEU fuel rod with the exception of three thermocouples embedded in the fuel. The sensing tips are located halfway between the outer radius and the vertical centerline on the fuel section and 1 inch above and below the horizontal center. The IFE allows the licensee to directly measure the temperature of the fuel. Although the IFE has been located in position C4NW in core 35A, it may be placed in other locations as specified in TS 2.2, discussed in Section 2.5.3 of this SER.

The NRC has approved the behavior of LEU TRIGA fuels with the above uranium content generically in NUREG-1282, "Safety Evaluation Report on High-Uranium Content, Low-Enriched Uranium-Zirconium Hydride Fuels for TRIGA Reactors," issued August 1987 (Ref. 40), and specifically for the WSU NRCR in the NRC's September 4, 2008, order to convert from HEU to LEU fuel, which included the conversion SER and the TS (Ref. 15).

The requirements of the WSU NRCR fuel are defined in TS 5.2, TS 3.1.6, and TS 4.1.6 as follows:

TS 5.2 Reactor Fuel

(1) The unirradiated 30/20 fuel rods shall have the following characteristics:

(a) the uranium content shall be a maximum of 30% by weight uranium, enriched to less than 20% ^{235}U;

(b) the hydrogen to zirconium ratio (in the ZrH_x) shall be a nominal 1.6 H atoms to 1.0 Zr atoms with a maximum H to Zr ratio of 1.65;

(c) the erbium content shall be homogeneously distributed with a nominal 0.90% by weight;

(d) the cladding shall be 304 stainless steel with a nominal thickness of 0.020 inches.

(2) The unirradiated standard fuel rods shall have the following characteristics:

(a) the uranium content shall be a maximum of 9.0% by weight enriched to less than 20% ^{235}U;

(b) the hydrogen to zirconium atom ratio (in the ZrH_x) shall be between 1.5 and 1.8;

(c) the cladding shall be 304 stainless steel with a nominal thickness of 0.020 inches.

TS 5.2.1(a) through (d) specify the uranium content, enrichment hydrogen to zirconium ratio limits, the burnable poison distribution and quantity, and the clad material and thickness for the 30/20 LEU fuel.

TS 5.2.1(a) specifies a uranium enrichment of less than 20 percent, which is the normal description of 30/20 LEU TRIGA fuel. The licensee states that, although this enrichment may be higher by about 1 percent than the design value of 19.75 percent used in the analysis for the conversion to LEU, this increase in the uranium enrichment should only increase the power density by about 1 percent.

TS 5.2.1(b) specifies limits on the hydrogen to zirconium ratio with a maximum of 1.65. The licensee states that the hydrogen to zirconium ratio influences the fuel rod internal pressure during operation. The internal pressure of the fuel rod influences the stress in the clad. As shown in the conversion SER, at the maximum upper limit of the ratio, 1.65, and 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 design value of the hydrogen to zirconium ration is 1.6, even at a value of 1.65, a factor of 5 is adequate to account for uncertainties in clad strength and manufacturing tolerances.

TS 5.2.1(c) specifies that the erbium content be homogeneously distributed in the fuel rod with a nominal 0.90 percent by weight. The licensee states that this could lead to a variation of erbium content for a single fuel rod of about 1 to 2 percent over the content used in the analysis for the conversion to LEU. However, such an increase in local power density would only reduce the safety margin by less than 2 percent.

TS 5.2.(2)(a) through (c) specify the uranium content, enrichment, hydrogen to zirconium ratio limits, and the clad material and thickness for standard 8.5/20 LEU fuel rods. The licensee states that these specifications are the normal description for 8.5/20 LEU fuel rods. In the conversion SAR, the licensee analyzed the effect of a maximum uranium content of 9 percent by weight U^{235} for the standard TRIGA elements and found it to be about 6 percent greater than the design value of 8.5 percent by weight U^{235}. Such an increase in loading would result in an increase in the power density of 6 percent and reduce the safety margin by 10 percent at most. The maximum hydrogen to zirconium ratio of 1.8 could result in the maximum stress under accident conditions in the fuel rod clad being about a factor of 2 greater than the value resulting from a hydrogen to zirconium ratio of 1.60. However, this increase in the clad stress during an accident would not exceed the rupture strength of the clad (Ref. 32).

TS 5.2(1) and (2) control important aspects of the design of the 30/20 LEU and standard TRIGA fuels. The LEU fuel design was reviewed by NRC staff during the conversion of the core to LEU fuel (Ref. 12) and the characteristics of the 30/20 and standard 8.5/20 LEU fuels were found acceptable (Ref. 15). On the basis of the LEU conversion, the NRC staff finds TS 5.2(1) and (2) acceptable for the current mixed-LEU core (Refs. 7 and 15).

Fuel growth and deformation can occur during normal operations, as described in NUREG-1537 (Ref. 28) and General Atomics Report E-117-833, “The U-ZrH_x Alloy: Its Properties and Use in TRIGA Fuel,” issued February 1980 (Ref. 32). Damage mechanisms include fission recoils and fission gases, both of which are strongly influenced by thermal gradients. Swelling of the fuel is dependent on the amount of time the fuel spends over a temperature threshold of about 750 degrees C. At 1 MW(t), the current steady-state power level, a computed IFE operating temperature of about 430 degrees C corresponds to a maximum calculated fuel temperature of about 500 degrees C (Ref. 6, Table 30), and swelling would be minimal, if present at all. Although fuel temperatures could go above 750 degrees C during pulsing, the time at temperature is short enough that pulsing should not cause fuel swelling by these mechanisms (Ref. 6). The NRC staff reviewed the data provided by the licensee and concludes that there is reasonable assurance that fuel swelling by the above mechanism is precluded.

The licensee inspects the fuel cladding to detect gross failure or visually observed deterioration. Attributes inspected include the fuel rod transverse bend and length, and a visual inspection is performed for bulges and other cladding defects. TS 3.1.6 defines the following requirements for maintaining the integrity of the reactor fuel:

TS 3.1.6 Fuel Parameters

The reactor shall not be operated with damaged fuel rods, except for the purpose of identifying damaged fuel rods. A fuel rod shall be considered damaged if any of the following occur:

(1) the sagitta of transverse bend exceeds 0.125 in. over the length of the cladding;

(2) the length exceeds the original length by 0.125 in.;

(3) a cladding defect exists as indicated by release of fission products;

(4) visual inspection reveals bulges, gross pitting, or corrosion.

TS 3.1.6(1) through (4) 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 performed for bulges or other cladding defects. The limit of transverse bend has been shown to result in no difficulty disassembling the core. The elongation limit has been specified to ensure that the cladding material will not be subjected to stresses that could cause a loss of the integrity of the clad containing the fuel and to ensure adequate coolant flow.

The limits on transverse bend and length are based on values from GA, the reactor designer. Although the reactor may not operate with damaged fuel, there have been instances of fuel-cladding defects in which fission products are only detected during reactor operation. Under these circumstances, the reactor needs to be operated to locate the fuel rod with the cladding defect. The NRC staff finds that licensee has used the standard definition of damaged fuel for TRIGA reactors. On this basis, the NRC staff concludes that TS 3.1.6 is acceptable.

TS 4.1.6 describes the surveillance requirements for the fuel as follows:

TS 4.1.6 Fuel Parameters

(1) At least 20% of the fuel rods comprising the core shall be visually inspected annually for damage or deterioration and annually measured for bowing or elongation such that each fuel rod in the core is inspected at least once over a five year period.

(2) The failure of a single fuel element to pass inspection shall trigger a required inspection of all fuel elements in the reactor core.

TS 4.1.6 specifies the surveillance requirements for the WSU NRCR fuel.

TS 4.1.6(1) has been revised by the licensee to provide for an inspection frequency based on time rather than the total worth of all reactor pulses, as was previously the case for the WSU research reactor. The increased inspection frequency will provide information on the condition of the research reactor fuel rods. The surveillance frequencies are consistent with NUREG-1537, 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.

TS 4.1.6(2) provides for a trigger event for inspection of all fuel rods if a single rod fails to pass inspection because of the need to provide a high degree of confidence that the remaining fuel can be safely used.

An important parameter in ensuring fuel rod integrity is the fuel rod temperature. TS 3.2.2, "Reactor Measuring Channels," and TS 3.2.3, "Reactor Safety System," require a fuel rod temperature measuring channel and a fuel rod temperature safety channel. To help ensure that the fuel rod temperatures are properly monitored, the surveillance requirements of the fuel rod temperature measuring channel and fuel rod temperature safety channel are defined in TS 4.2.2 and TS 4.2.3 as follows:

TS 4.2.2 Reactor Measuring Channels

(1) A channel test of each of the required operable measuring channels listed in Table 3.1 for the intended mode of operation shall be performed before each day's operation or before each operation extending more than one day.

(2) A channel check of the fuel rod temperature measuring channel shall be made each time the reactor is operated in the steady state mode by comparing the indicated instrumented fuel rod temperature with previous indicated temperature values for the same core configuration and power level.

TS 4.2.2(1) requires a channel test of the fuel rod temperature measuring channel, which is listed in Table 3.1 of the TS.

TS 4.2.2(2) provides for a check of the fuel rod temperature measuring channel to ensure proper operation.

TS 4.2.3 Reactor Safety System

The surveillance requirements in this section may be postponed during periods of reactor shutdown. If the surveillance requirement occurs during a period of reactor shutdown the surveillance shall be completed upon resumption of reactor operation.

(1) A channel test of each of the safety channels listed in Table 3.2, except for the bulk primary coolant temperature, for the intended mode of operation (steady-state or pulse) shall be performed before each day's operation or before each operation extending more than one day.

(2) (…)

(3) (…)

(4) A channel calibration of the fuel rod temperature measuring channel shall be performed semiannually by the substitution of a thermocouple simulator in place of the instrumented fuel rod thermocouple.

(5) (…)

TS 4.2.3(1) requires a fuel element temperature safety channel test before the operation of the reactor, which is listed in Table 3.2 and which provides a reactor scram for high temperature.

TS 4.2.3(4) requires a semiannual channel calibration of the fuel rod measuring channel.

The NRC staff finds that these surveillances of fuel rod parameters and fuel rod temperature measuring and safety channels and their intervals are consistent with the guidance provided in NUREG-1537, ANSI/ANS-15.1-2007, and the intervals used at similar research reactors. The NRC staff finds that these surveillance frequencies will ensure performance and operability of the fuel rods, including the fuel element temperature measuring systems or components.

The NRC staff reviewed the discussion regarding the constituents, materials, and components of the fuel rods provided in the renewal SAR, as supplemented. On the basis of its review, the NRC staff finds that the licensee has adequately described the fuel rods used in the WSU NRCR, including design limits, and the technological and safety-related bases for these limits. The NRC staff concludes that compliance with the TS limits will ensure uniform characteristics and compliance with design bases and safety-related requirements.

2.2.2 Control Rods

To vary the reactivity of the core, the WSU NRCR uses four motor-driven control rods—three scrammable standard control rods (shim-safety blades) and one nonscrammable regulating rod (regulating blade)—and one scrammable, pneumatic electromechanical transient rod.

The definition of control rods has been presented in Section 2.2 of this SER.

TS 5.4 defines the design requirements for the control rods as follows:

TS 5.4 Control Rods

(1) Standard control rods shall have scram capability and contain borated graphite, B_4C powder, boron or boron compounds in solid form within aluminum or stainless steel cladding.

(2) The regulating control rod does not have scram capability and shall be stainless steel.

(3) The transient control rod shall have scram capability and contain borated graphite or boron compounds in a solid form within aluminum or stainless steel. The transient rod shall have an adjustable upper limit to allow variation of reactivity insertions. The transient control rod does not incorporate a fueled follower.

TS 5.4 has been modified by the licensee to more accurately reflect the control rods as they currently exist and as they were for the conversion of the WSU NRCR to LEU. These design specifications give the requirements for the standard, regulating, and transient rods. The objective is to ensure that control rods are fabricated to reliably perform their intended control and safety function. TS 5.4(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 standard control rods (shim-safety blades) are blade-type rods that 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 inches long, 10.5 inches wide, and 0.375 inch thick. It is clad with 0.125-inch 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. De-energizing 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 inches of fall. Continuous indication of blade position is provided.

The regulating rod is a blade-type element that 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 inches wide and 40 inches long. The regulating rod (blade) is nonscrammable and is therefore directly attached to its drive mechanism. The regulating blade drive (or servo control rod drive), is similar to the shim-safety drive except that there is no scram capability, and a servo motor and tachometer generator is 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 rod is a solid borated graphite cylinder contained in a 1.25-inch diameter stainless steel or aluminum tube. The poison section of the transient rod is 15 inches 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 holddown 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 that 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 inch 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 that acts as the screw in the ball-nut drive assembly. A system of limit switches indicates the position of the piston and the transient rod.

The licensee projects that the lifetime of the standard and regulating control rods will extend beyond the period of renewal and has no plans to replace or store depleted control rods. The transient rod was replaced during the recent conversion to an all-LEU core. Its lifetime should extend beyond the period of the license renewal requested.

The worths of the control rods in core 35A were determined during the startup testing. While the permanently installed experimental devices were in place, there was no unsecured experiment during the determinations. The worths of the transient rod, regulating blade, and the three shim-safety blades for this core configuration are $3.20, $0.16, $1.44, $3.71, and $3.97, respectively. However, the worths of the control rods are dependent on the design of the core being used, and slight changes in core 35A may occur over time. Operational core 35A was determined to have an excess reactivity of $7.44. Using the rod worth data, the shutdown margin, based on TS 3.1.3, is $0.91. The NRC staff reviewed the above information and concludes that, because the excess reactivity and the shutdown margin of $0.91 are acceptable, the reactivity worths of the control rods are acceptable.

TS 3.2.1 defines requirements to ensure that the control rods will promptly shut down the reactor upon a scram signal as follows:

TS 3.2.1 Control Rods

(1) The reactor shall not be operated unless the control rods are operable. Control rods shall not be considered operable if damage is apparent to the rod or rod drive assemblies, or the scram time exceeds 2 seconds.

(2) The scram time from the time that a scram signal is initiated to the time that the slowest scrammable control rod reaches its fully inserted position shall not exceed 2 seconds.

TS 3.2.1 helps ensure that, during the normal operation of the WSU NRCR, the control rods are operable and the time required for the scrammable control rods to be fully inserted from the instant that a scram signal is initiated is rapid enough to prevent fuel damage. Adherence to this specification ensures that the reactor will be promptly shut down when a scram signal is initiated. For the range of transients anticipated for a TRIGA reactor, the specified scram time is adequate to ensure the safety of the reactor. The NRC staff finds that the requirements of TS 3.2.1 support the basic design requirements to prevent reactor fuel damage and, therefore, the NRC staff concludes that TS 3.2.1 is acceptable.

TS 4.2.1 defines the requirements for the measurement and verification of the worth, performance, and operability of the control rods as follows:

TS 4.2.1 Control Rods

The surveillance requirements in this section may be postponed during periods of reactor shutdown. If the surveillance requirement occurs during a period of reactor shutdown the surveillance must be completed upon resumption of reactor operation.

(1) The control rods shall be visually inspected at biennial intervals.

(2) The scram time shall be measured annually.

(3) The transient rod drive cylinder and associated air supply system shall be inspected, cleaned, and lubricated semiannually.

TS 4.2.1(1) through (3) specify surveillance intervals to ensure the operability of the control rods. TS 4.1.3, discussed in Section 2.2 of this SER, specifies the intervals for determination of the reactivity worths of the control rods. The NRC staff finds that these intervals for control rod inspection, scram time determination, rod worth determinations, and transient rod maintenance are sufficient to help ensure operability. The NRC staff finds TS 4.2.1, in conjunction with TS 4.1.3, to be acceptable to ensure the performance of the control rods.

TS 5.4, TS 3.2.1, and TS 4.2.1, along with TS 4.1.3, help to ensure that the control rods will promptly shut down the reactor upon a scram signal. The WSU NRCR shall not be operated if any damage is found on the control rods or drives. The scram time for the control rods is specified to be less than 2 seconds and is measured annually by the licensee. The 2-second value for scram time is a margin far enough above normal scram times to allow for some variation in performance while bounding acceptable performance. The 2-second value is assumed in the safety analysis.

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 WSU NRCR includes reactivity worths that can control the excess reactivity planned for the WSU NRCR, including the assurance of an acceptable shutdown reactivity and margin. The licensee has justified appropriate design limits, limiting conditions for operation, and surveillance requirements for the control rods. Based on the above discussion, the NRC staff concludes that the requirements related to the WSU NRCR control rods are acceptable.

2.2.3 Neutron Moderator and Reflector

The predominant moderator of the WSU NRCR reactor core is the zirconium hydride incorporated into the LEU fuel rods. The pool water between the fuel rods in the fuel bundles also serves as a moderator.

The top and bottom 3.5 inches of a fuel rod contain graphite that serves as a reflector. In addition, individual 3-inch by 3-inch by 31-inch aluminum-enclosed graphite reflectors may be plugged into the grid plate to surround the core. These graphite reflectors are fabricated with nuclear-grade graphite. The aluminum-clad enclosure is evacuated during fabrication and sealed to collapse the aluminum onto the graphite for improved thermal conductivity. Absent a graphite reflector piece, a device for experiments, a neutron source holder, or pool water serves as a reflector.

Core 35A contains 20 graphite reflectors on two faces of the core. The remainder of the core is reflected by water.

With sufficient neutron irradiation, graphite is known to grow. Such growth may distort the graphite and cause the aluminum clad to rupture. The licensee recently replaced all graphite moderator blocks and indicated that visual inspections are periodically performed on the blocks to detect significant structural changes. Graphite reflectors are known to maintain their structural integrity and that visual inspections during reactor core changes are sufficient to ensure that no major degradations have occurred. Base on its review of the license renewal SAR, the NRC staff concludes that there is reasonable assurance that the reflectors will function safely in the WSU NRCR core for the renewal period without adversely affecting public health and safety.

2.2.4 Neutron Startup Source

The startup source for the WSU NRCR is an aluminum-clad antimony-beryllium neutron source, located in a neutron source holder in a graphite reflector element that may be located in any position of the grid box. A typical position is a grid location diagonally opposite the neutron detector for the startup channel. Because the antimony is activated by core neutrons, the source normally remains in the reactor during operation. A neutron-source clad failure would be detected during the routine analysis of pool water as required by TS 4.3 (discussed in Section 2.3 of this SER) to periodically measure the radioactivity content of the reactor pool. The NRC staff finds that the surveillance requirements specified in TS 4.3 are acceptable for limiting the radioactivity content of the pool water, reducing personnel exposure, and detecting potential damage to the source cladding.

The primary function of the neutron source is to provide sufficient counts on a neutron monitoring channel during startup of the reactor to show that the instrumentation is functioning properly. TS 3.2.3, discussed in Chapter 5 of this SER, requires that the control and safety system have an interlock that forbids rod withdrawal when the neutron count is less than 2 counts per second.

The NRC staff reviewed the information provided in the license renewal SAR (Ref. 1), as supplemented. Based on the information provided in the licenses renewal SAR (Ref. 1, Section 4.2.4) as supplemented by the analysis performed for the LEU conversion (Ref. 3, Section 4.2.4), the NRC staff concludes that the neutron startup source is adequate to allow controlled reactor startup and is therefore acceptable.

2.2.5 Core Support Structure

The WSU NRCR core is suspended in the reactor pool from a movable bridge that is mounted on rails. The bridge and entire structure may be moved laterally by hand so that the reactor core may be positioned anywhere along the centerline of the pool. Most portions of the bridge are constructed of steel. The core suspension framework, which is suspended from the bridge, is constructed of aluminum with some stainless steel fasteners. This framework supports the grid box in which there is a grid plate. The fuel bundles and other core components are plugged into this grid plate. The hollow-free flooding corner posts of the suspension framework serve as guide tubes for the nuclear instrumentation detectors. The control rod drives are connected to and supported by the bridge structure.

Deck plates mounted on the top side of the bridge structure form a floor area on the bridge around the control drives. The floor area provides a workspace for maintaining the reactor and associated facilities. A railing system is connected to the bridge structure to prevent personnel from falling off the bridge. The licensee performs a visual inspection of the bridge structure and would observe significant structural degradations. The licensee indicates that the bridge structure maintains its structural integrity and that visual inspections during reactor core changes are sufficient to recognize significant degradations. On the basis of its review, the NRC staff concludes that there is reasonable assurance that the reactor bridge will function safely for the renewal period without adversely affecting public health and safety.

2.3 Reactor Tank or Pool

The reactor pool is a reinforced, aboveground, concrete, two-section pool with a volume of 247,000 liters (65,250 gallons). The pool is penetrated by a thermal column and has provisions for up to 12 beam tubes. The pool water level is normally approximately 20 feet above the top of the core. An alarm sounds if the pool water level drops by 8 inches.

The TS define the following reactor pool inventory requirements:

TS 5.8 Reactor Pool Water System

(1) The reactor core shall be cooled by natural convection water flow.

(2) All piping extending more than 5 ft below the surface of the pool shall have adequate provisions to prevent inadvertent siphoning of the pool.

(3) A pool level alarm shall be provided to indicate in the reactor control room and at a remote location of a loss of coolant if the pool level drops more than 8 inches below the normal level.

(4) The reactor primary coolant pool shall provide for at least 16 feet of water above the top of the core.

TS 5.8(1) through (4) state important design aspects of the reactor pool water system to provide cooling for the reactor core, to prevent loss of core cooling because of siphoning of the pool, to detect pool water leakage, and to provide the coolant pressure assumed in the thermal-hydraulic calculations. In addition, the TS require an alarm to indicate water loss in the event of a leakage path developing that could potentially drain the reactor pool.

TS 5.8(1) through (4) specify important design aspects of the reactor pool water system. The NRC staff finds that these design aspects are acceptable and therefore concludes that these TS are acceptable.

The normal water level in the reactor pool is 20 feet above the top of the reactor core. The reactor pool water level is continuously monitored by a water-level monitoring device that provides signals at three levels: normal, a high-water alarm signal when the level is 3.5 inches above normal, and a low-water alarm signal when the level drops 4.5 inches below normal. The TS for pool water loss has been established at 8 inches below normal to allow for operational flexibility. An additional water level switch, installed at the normal water level, activates the makeup system when the level drops below normal and deactivates the makeup supply when the water level is at or above the normal water level. Because the pool water level is continuously monitored and the frequency and quantity of makeup water is logged, pool water leakage will be detected. The requirements for pool makeup water to replace water lost to evaporation are well known to the staff and off-normal changes would be investigated by the licensee. A small leak will start the makeup water system that will be able to compensate for small leakages. Any further drop in water level will activate a low-water alarm. Most leakage pathways through pipes, fittings, beam ports, and the pool wall would result in leakage into observable areas and would likely be discovered in a 24–72-hour time period. The water would collect in the drainage sump and holdup tank and be analyzed before disposal; it would not result in an unmonitored and uncontrolled release. In the event that leakage is not observable and water leaks to the environment, the periodic monitoring of the pool water ensures that the radioactivity level of the released water would be very low, well within the 10 CFR Part 20 limits for effluent releases (Ref. 18).

The pool water level alarm requirement is presented in TS 3.2.4 and the surveillance requirement for the alarm is presented in TS 4.2.4 as follows:

TS 3.2.4 Pool Level Alarm

The pool level sensor shall initiate an alarm signal if the reactor pool level falls 8 inches or more below the normal level. The pool alarm sensor shall initiate a signal at the reactor control console and at a monitored remote location.

TS 4.2.4 Pool Level Alarm

The reactor pool level alarm shall be monthly tested for operability.

TS 3.2.4 requires that the control system provide an alarm if the pool water level is 8 inches or more below normal. The alarms are monitored in the reactor control room and at a monitored remote station, and an automatic recorded message is sent to the on-call reactor operator or senior reactor operator. Procedures are in place for responding to the alarms for both the remote monitors and the reactor on-call staff.

TS 4.2.4 provides for monthly testing of the pool water level alarm.

Because the pool water level is continuously monitored and the frequency and quantity of makeup water is logged, pool water leakage will be detected. The requirements for pool makeup water to replace water lost to evaporation are well known to the staff, and off-normal changes would be investigated by the licensee. Based on the analysis presented above, the NRC staff concludes that TS 3.2.4 and 4.2.4 are acceptable.

A rapid loss of pool water would result in a loss-of-coolant accident (LOCA), discussed in Section 4.1.3 of this SER.

The WSU NRCR pool has two compartments. In the event of a significant pool leak, the leak is not likely to occur in both sections at the same time. Accordingly, the reactor core can be moved to the nonleaking section, the (gate) dam inserted between the two sections, and appropriate repairs to the leaky section undertaken. The licensee did such repairs in 1999 by draining one section of the pool while the core remained underwater in the other section.

The reactor pool is filled with demineralized water. The conductivity and the pH of the pool water is controlled to minimize the corrosion of the fuel rod cladding, minimize the corrosion of reactor components, and minimize neutron activation of dissolved materials in the pool water.

The reactor pool water quality requirements are specified in TS 3.3, TS 4.2.3(2), and TS 4.3 as follows:

TS 3.3 Primary Coolant Conditions

(1) Conductivity of the primary coolant shall be no higher than 5×10^{-6} mhos/cm.

(2) The pH of the primary coolant shall be between 5.0 and 7.5.

(3) The bulk primary coolant temperature shall not exceed 50 °C.

(4) The radionuclide content of the primary coolant shall not exceed 10 CFR 20 effluent release limits.

(5) The reactor shall not be operated with less than 16 feet of water above the top of the core.

TS 3.3(1) and (2) ensure that the conductivity of the tank water is maintained at or less than 5 micromhos per centimeter (mhos/cm) and the pH level is kept between 5.0 and 7.5 to control corrosion. The licensee states that a small rate of corrosion continuously occurs in a water-metal system. Limiting this rate extends the longevity and integrity of the fuel cladding. It also ensures that the heat transfer between the cladding and coolant will not degrade because of oxide buildup on the cladding. A pH limit between 5.0 and 7.5 is consistent with other TRIGA reactors and the guidance provided in NUREG-1537. Based on the discussion above, the NRC staff concludes that this specification is acceptable.

TS 3.3(3) provides assurance that the pool water temperature is consistent with the assumptions used in the thermal-hydraulic calculations.

TS 3.3(4) helps ensure that the radioactive content of the primary coolant water will be low and known in the event of pool leakage.

TS 4.2.3 Reactor Safety System

The surveillance requirements in this section may be postponed during periods of reactor shutdown. If the surveillance requirement occurs during a period of reactor shutdown the surveillance shall be completed upon resumption of reactor operation.

(…)

(2) A channel check of the bulk primary coolant temperature shall be performed before each day's operation or before operation extending more than one day.

(…)

TS 4.2.3(2) requires that the bulk primary coolant temperature safety channels be tested for operability before each day's operation or before each operation extending more than one day.

TS 4.3 Primary Coolant Conditions

(1) The conductivity and pH of the primary coolant water shall be measured at least once every 2 weeks.

(2) The radionuclide content of the reactor pool water shall be monitored monthly. Steps shall be taken to isolate the source of the radioactivity and to mitigate the problem if the radionuclide content of the pool water in

the reactor pool exceeds one-third (1/3) of the 10 CFR 20 Appendix B, Table 3 value.

TS 4.3(1) provides for periodic monitoring of primary coolant water conductivity and pH to provide timely information of possible changes in primary coolant water chemistry.

TS 4.3(2) provides for monthly monitoring of the radionuclide content in the pool water to provide information as a means to detect, in a timely fashion, a leak of radioactive material by a sealed source. In addition, periodic monitoring of the pool water will provide 10 CFR Part 20 release limits under any circumstances or condition of operation.

The NRC staff reviewed the information provided in the WSU license renewal SAR, as supplemented, regarding pool water level and quality (Ref. 18). The NRC staff finds that the water-level instrumentation and the water quality program are adequate to ensure that the water level exceeds 16 feet at all times above the core, and that the water quality is maintained. In addition, pool water level is monitored, and leakage would be investigated by WSU reactor staff. The NRC staff concludes that significant release to the environment resulting from pool leakage is extremely low.

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 and performs independent measurements of radiation levels in the facility. Based on a review of the information provided by the licensee and results from the NRC inspection program, the NRC staff concludes that there is reasonable assurance that, during the renewal period, the WSU NRCR biological shield will limit exposures from the reactor and reactor-related sources of radiation so that the limits of 10 CFR Part 20 will not be exceeded.

2.5 Nuclear Design

The information discussed in this section establishes the design bases for the content of other chapters in this SER.

2.5.1 Normal Operating Conditions

The WSU NRCR nominally operates at a steady-state thermal power level of 1 MW(t). The following definitions delineate the operational state of the reactor and are used in part to determine the applicability of other TS to the reactor.

TS 1.0 provides the definitions for the operational states of the reactor as follows:

TS 1.0 Definitions

(….)

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

(…)

Reactor Secured: The reactor is secured when:

Either

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

Or

(2) The following conditions exist:

(1) The reactor is shutdown;

(2) All of the control rods are fully inserted;

(3) The console key switch is in the “off” position and the key is removed from the console lock;

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

(5) No experiments are being moved or serviced that have a reactivity worth equal to or greater than $1.00.

Reactor Shutdown: The reactor is shut down if it is subcritical by at least $1.00 in the reference core condition with the reactivity worth of all installed experiments included.

(….)

These definitions describe the operational states of the reactor. They are standard definitions used in research reactor TS and are consistent with the definitions in NUREG-1537 and ANSI/ANS 15.1-2007 and are therefore, acceptable to the NRC staff.

Steady-State Operation

The WSU NRCR is licensed to operate at a steady-state maximum power level of 1.0 MW(t). The revised conversion SAR (Ref. 7) shows by analysis that operation at a steady-state power level of 1.3 MW(t) corresponds to a peak fuel temperature of 540 degrees C with a departure from nucleate boiling ratio (DNBR) of 1.69. Several reactors of this type have operated successfully for many years at power levels up to 1.5 MW(t).

TS 3.1.1 specifies the steady-state operating power level as follows:

TS 3.1.1 Steady State Operation

The reactor power level shall not exceed 1.0 MW during steady-state operation.

TS 3.1.1 has been modified by the licensee to limit the maximum steady-state power level to the licensed power of 1 MW(t). Previously, the licensee had been permitted to operate the reactor at a steady-state power level of 1.3 MW(1) for the purpose of testing reactor scrams.

TS 3.1.1 specifies a steady-state power limit to help ensure that adequate cooling is provided for the fuel rods by natural convection of pool water. As will be discussed in the thermal-hydraulic analysis (Section 2.6 of this SER), operation of the WSU NRCR even at 1.3 MW(t) would allow for sufficient safety margins. Moreover, Table 3.2 of TS 3.2.3 indicates that a reactor scram would occur at a power level of 1.25 MW(t).

On the basis of the discussion above, the NRC staff finds that the requirement in TS 3.1.1 that the reactor power not exceed 1.0 MW(t) during steady-state operation provides an acceptable margin of safety for operation. Therefore, the NRC staff concludes that this specification is acceptable.

TS 4.1.1 requires that the reactor power level monitoring channels be calibrated as follows:

> **<u>4.1.1 Steady State Operation</u>**
>
> The surveillance requirements for the reactor safety systems that monitor reactor power level are described in Section 4.2.3.

TS 4.2.3 requires that the reactor power level monitoring channels be calibrated annually as follows:

> **<u>TS 4.2.3 Reactor Safety System</u>**
>
> The surveillance requirements in this section may be postponed during periods of reactor shutdown. If the surveillance requirement occurs during a period of reactor shutdown the surveillance shall be completed upon resumption of reactor operation.
>
> (….)
>
> (5) A channel calibration shall be made of the power level monitoring channels annually or after a core configuration change, by the calorimetric method.

TS 4.1.1 and 4.2.3(5) require that the reactor power measuring channels be calibrated annually using the calorimetric method, which is the standard method to perform the calibration of the reactor power measuring channels. Annual calibration provides assurance that the power level measuring channels are providing accurate power level indications. A recalibration is required after core configuration changes due to the possibility that a change in flux distribution could lead to inaccurate power level indications on the power level measuring channels.

TS 4.2.3 requires that the reactor safety system power measuring channels be checked and tested as follows:

> **<u>TS 4.2.3 Reactor Safety System</u>**
>
> The surveillance requirements in this section may be postponed during periods of

reactor shutdown. If the surveillance requirement occurs during a period of reactor shutdown the surveillance shall be completed upon resumption of reactor operation.

(1) A channel test of each of the safety channels listed in Table 3.2, except for the bulk primary coolant temperature, for the intended mode of operation (steady-state or pulse) shall be performed before each day's operation or before each operation extending more than one day.

(2) (…)

(3) A test of the interlocks in Table 3.3 for the intended mode of operation (steady-state or pulse) shall be performed before each day's operation or before each operation extending more than one day.

(4) (…)

Table 3.2 is presented in Section 5 of this SER. The TS presenting Table 3.2 requires that two power level safety channels be available to scram the reactor at 125 percent of maximum licensed power.

TS 4.2.3(1) requires that these power level safety channels be tested for operability before each day's operation or before each operation extending more than one day.

Table 3.3 is presented in Section 5 of this SER. It presents the interlocks required for steady-state and pulse operation.

TS 4.2.3(3) requires that these interlocks be tested before each day's operation or before each operation extending more than one day.

TS 4.2.3(5), already discussed above, requires annual calibration of the power level measuring channels.

The licensee states that experience has shown that a channel test of the two power level safety channels and a test of interlocks before reactor startup, when combined with the channel calibrations, provide assurance that the power level measuring channels are providing accurate power level indications that will assist in preventing the reactor power level from exceeding 1 MW(t).

On the basis of the discussion above, the NRC staff finds that the requirement in TS 4.1.1 and 4.2.3 that the reactor power will be accurately indicated is an aid in ensuring that the power level does not exceed 1 MW(t) during steady-state operation. Therefore, the NRC staff concludes that TS 4.1.1 and 4.2.3 are acceptable.

Pulse Mode Operation

The licensee states that the WSU NRCR is designed to be pulsed from a low to a high power level by the rapid insertion of reactivity. In this mode of operation, the maximum reactivity insertion is limited to that which will limit the peak fuel temperature to 830 degrees C, and the 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 a required interlock, which prevents the transient rod from firing. In addition, a timer is set to initiate a transient rod scram 15 seconds or less following the initiation of a pulse. TS 3.1.2 applies to the peak fuel temperatures in the reactor resulting from a rapid insertion of reactivity to ensure that fuel rod damage does not occur.

TS 3.1.2 specifies the limitation on pulse mode operation as follows:

> **TS 3.1.2 Pulse Mode Operation**
>
> The maximum reactivity inserted during pulse mode operation shall be such that the peak fuel temperature in any fuel rod in the core does not exceed 830ºC.

TS 3.1.2 establishes the criteria for determining the maximum reactivity addition for pulsing to ensure that the reactor may be safely pulsed without fuel damage.

The peak fuel temperature limitation and its associated maximum reactivity insertion for pulsing will ensure that the reactor can be safely pulsed without fuel damage. The large prompt negative temperature coefficient of reactivity of the U-Zr_x fuel moderator provides a basis for safe operation of the reactor in the nonpulsing mode and is the essential characteristic supporting the operational capability of the reactor in a pulse mode. Pulse capability is limited to ensure that the fuel temperature stays below the safety limit of 1,150 degrees C for the 30/20 fuel and 1,000 degrees C for standard fuel, as discussed in Section 2.5.3 of this SER, and the peak fuel temperature remains below 830 degrees C. The fuel temperature limit of 830 degrees C during pulsing is recommended by GA to ensure that no fuel damage occurs because of internal pressure caused by hydrogen migration (Ref. 41). The 830 degrees C limit was based on the fuel damage experience at the Texas A&M University TRIGA reactor.

During the startup testing of the LEU core, core 35A was established as the core that met the design goals. Calculations were performed to guide the experimental determination of the maximum pulse that would not exceed the temperature limitation. As shown in the startup report (Ref. 16), the maximum allowable pulse size corresponding to the insertion of $2.31 of reactivity would still result in a maximum fuel temperature below the 830 degrees C limit. The administrative reactivity insertion limit for core 35A was set at $2.00. This is discussed further in Section 2.6.2 of this SER.

TS 3.2.3 requires a preset timer to initiate a scram of the transient rod 15 seconds or less after the initiation of a pulse, and also an interlock to prevent pulsing when the reactor power level is 1 kW or above. TS 4.2.3 requires periodic testing of the timer and the interlock.

The licensee performed analyses of two unlikely events using the BLOOST code (Ref. 42), discussed in Section 13.5.3 of the revised conversion SAR (Ref. 7):

1. the rapid insertion of $2.00 reactivity into the reactor already operating at a power level of 1 MW(t) by the rapid removal (in 0.3 sec.) of the maximum worth secured experiment, or

2. the rapid insertion of $3.19 reactivity into the reactor by the ejection of the inserted transient rod.

The results of the analysis indicated that the fuel temperature in both of these accident sequences at beginning of life (BOL) and EOL would be less than the safety limit of 1,150 degrees C for 30/20 LEU rods and 1,000 degrees C for 8.5/20 LEU rods.

On the basis of its review, the NRC staff finds that the peak fuel temperature limit and the associated maximum reactivity addition limit for pulsing help ensure that the reactor can be safely pulsed without concern regarding fuel damage and therefore, is acceptable.

Core Changes

The LEU core 35A is expected to have a lifetime of about 1,000 MWD operating at 1 MW(t). Based on current operating schedules of less than or equal to 35 MW hours per week, the projected life of the present WSU NRCR core is about 13 years, long enough so that there is no anticipated reloading of the core.

The 30/20 LEU fuel contains erbium as a burnable poison. Because the erbium burns faster than the U^{235}, the excess reactivity of a core will increase with time until about midlife of the core. After that, the excess reactivity decreases with time as the U^{235} is depleted.

The licensee considered three scenarios for introducing new fuel rods into the core. The first scenario is the replacement of the partially burned 8.5/20 fuel rods when they become more fully burned. The second is replacing an IFE that has been operated in the core with a fresh IFE because of thermocouple failure. The third is the introduction of a fresh 30/20 LEU fuel rod into the core. In all cases, the licensee considered the effects of the changes on peaking factors during steady-state and pulsing operation. Calculations have been performed to determine the best core locations for fuel additions for each scenario. The licensee has demonstrated in the revised conversion SAR (Ref. 7), as supplemented, the LEU startup plan, and the startup report its ability to add fuel to the core while maintaining the requirements of TS 3.1.3, TS 3.1.4, and TS 3.1.5. The NRC staff reviewed the licensee’s approach to adding fuel to the core and concludes that it is acceptable.

Control Rod Worths

Sections 2.2 and 2.2.2 of this SER describe the reactor core and the control rods. For LEU core 35A, the reactivity worths of the scrammable control rods total $12.32, including the reactivity worth of the transient rod. The excess reactivity in the core may not exceed $7.46. The excess reactivity is reduced by the buildup in the core of long-lived fission product poisons, primarily samarium. Therefore, there is adequate margin to maintain the reactor shut down when all of the control rods are inserted and also to shut down the reactor while meeting the stuck-rod criteria and experiment reactivity criteria controlled by TS 3.1.3. Therefore, the NRC concludes that the values of the control rod worths are acceptable.

Excess Reactivity

The excess reactivity criteria are specified by TS 3.1.4, as discussed in Section 2.2 of this SER. The maximum reactivity in excess of cold, xenon-free critical is to be less than 5.6 % Δk/k, equivalent to $7.46. During the initial startup testing of core 35A, the maximum reactivity in excess of cold, clean was measured at $7.44. After operation to build in fission products, the excess reactivity was measured at $7.35. After installation of four experimental tubes and a cadmium-lined experiment tube, the final core excess reactivity of core 35A was measured at $7.12 (5.34 % Δk /k). The NRC staff finds the value for the excess reactivity acceptable based on the analysis performed for the conversion of the reactor to LEU and the startup testing (Ref. 16).

Shutdown Margin

Section 2.2 of this SER describes the criteria for shutdown margin and requires a shutdown margin of $0.25. The purpose of defining a shutdown margin is to ensure that the reactor can be shut down by an acceptable margin even if the most reactive control blade sticks in the out position. In addition to assuming that the most reactive control rod is not available to help shut down the reactor, the TS also places constraints on the core condition and experiments. The core conditions are those most limiting for determining shutdown margin. Nonsecured experiments are considered to be in their most reactive state to ensure that the reactor is not subcritical because of experiments that could be removed from the core. As demonstrated in the startup report (Ref. 16), the measurements performed by the licensee during the startup testing of the LEU core showed that the measured value of the shutdown margin met the above criteria.

The surveillance requirements to ensure compliance with shutdown margin and excess reactivity requirements are presented in TS 4.2.1 and TS 4.1.3, as discussed in Sections 2.2.2 and 2.2 of this SER. Core 35A has been established by the licensee as the operational core.

The NRC staff reviewed the licensee's analysis for the all-LEU core and finds that the established core 35A contains all of the components for an operable reactor core. The NRC staff finds that the licensee used input parameters justified by analysis presented in the revised conversion SAR (Ref. 7), as supplemented and justified by the results of startup testing as summarized in the startup report. The NRC staff finds that the licensee adequately analyzed the reactivity effects of individual core components and verified the analysis in the startup program as presented in the startup report. TS related to the normal operating conditions of the reactor core include limits on excess reactivity, the minimum shutdown margin, allowable core configurations, and surveillance requirements for the core reactivity parameters and reactivity worth of the control rods. These TS are consistent with the recommendations of ANSI/ANS-15.1-2007. The NRC staff finds that the analysis presented in the revised conversion SAR, as supplemented, and the information in the startup report 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 expected normal reactor operation during the period of the renewed license.

2.5.2 Reactor Core Physics Parameters

During the analysis for the conversion of the WSU NRCR to LEU fuel, the following core physics parameters were determined: the temperature coefficient of reactivity, neutron lifetime, effective delayed neutron fraction, power coefficient, and void coefficient

The licensee states that an important safety feature of a TRIGA reactor is the reactor core's inherent large, prompt, negative temperature coefficient of reactivity, resulting from an intrinsic molecular characteristic of the U-Zr_x matrix at elevated temperatures. The negative temperature coefficient results principally from the neutron hardening properties of the fuel matrix at elevated temperatures, which increases the leakage of neutrons from the fuel-bearing material into the water moderator material, where they are absorbed preferentially. This reactivity decrease is a prompt effect because the fuel and zirconium hydride are mixed homogeneously; thus, the zirconium hydride temperature rises essentially simultaneously with fuel temperature, which is directly related to reactor power. An additional contribution to the prompt, negative temperature coefficient is the Doppler broadening of uranium-238 resonances at high temperatures, which increases nonproductive neutron capture in these resonances.

Because of the large, prompt, negative temperature coefficient, a step insertion of reactivity resulting in an increasing fuel temperature will be compensated for by the fuel matrix rapidly and automatically. This can terminate the resulting power excursion without any dependence on the electronic or mechanical reactor safety systems or the actions of the reactor operator. Also, changes of reactivity resulting in a change in fuel temperature during steady-state operation can be rapidly compensated for by the fuel matrix, thus limiting the reactor steady-state power level (Ref. 31). This inherent characteristic of the U-ZrH_x fuel has been the basis for designing TRIGA reactors with a pulsing capability as a normal licensed mode of operation.

For the LEU core using 30/20 and 8.5/20 fuel, the prompt temperature coefficients of reactivity were calculated for the BOL and EOL over a range of temperatures from room temperature to about 800 degrees C. At the BOL, the coefficient is approximately -0.6×10^{-4} Δk/k per degree C at room temperature and -1.2×10^{-4} at 800 degrees C. At the EOL, these coefficients are -0.57×10^{-4} Δk/k per degree C and -1.1×10^{-4}, respectively. The calculation method for this reactor used the computer code DIF3D. The values quoted for this core are similar to values reported by GA and others for TRIGA reactors. Therefore, a large step insertion of reactivity will be compensated for by the fuel as the temperature increases, causing the transient to end. This large negative value of temperature coefficient, α_T, was designed into the fuel so that TRIGA reactors can be pulsed. In the pulsed mode of operation, TS 3.1.2 specifies a limit for the temperature of the fuel during a pulse. The licensee has calculated and determined experimentally that a step insertion of reactivity to \$2.31 is within the TS. GA performed many tests with step insertions up to \$5.00 before any fuel damage became apparent; therefore, the TS limit of \$2.31 is well within the safety envelope established by GA.

The thermal neutron flux for the WSU NRCR core 35A operating at 1 MW(t) is approximately 4×10^{12} neutrons per centimeter squared per second, which is consistent with other 1-MW(t) research reactors. The neutron lifetime has been calculated to be about 28 microseconds (μs). The best-estimate value for the effective delayed neutron fraction for the LEU core was found to be β_{eff} = .0075. These values are similar to those of other TRIGA reactors.

The power coefficients provided by the licensee show negative values, and all of the coefficients are similar to those of other TRIGA-type nuclear reactors. According to the licensee, the void coefficient has been calculated to be -0.135 % Δk/k water void. If an evacuated experimental tube in the core were to flood, the core reactivity would increase. The conversion SAR (Ref. 3) presents a calculation in which a 205-cubic centimeter (cm^3) dry experiment is placed in the core and becomes flooded, resulting in an insertion of $0.18. Prompt critical occurs when the reactivity insertion is $1.00 or greater, so a flooding will not pose a risk to the reactor. It is also true that reactivity insertion events are bounded by what is allowed to be inserted for pulsed operation.

Curves of excess reactivity as a function of burnup are given in the license renewal SAR (Ref. 1), as supplemented. After 1,000 MWD of full-power operation, and giving no credit for fuel shuffling to extend the life of the core, the mixed-LEU core will have $0.40 of excess reactivity. After 1,000 MWD, there are no significant changes to the calculated reactor parameters, β_{eff}, and neutron lifetime, ℓ. The value of β_{eff} is 0.0073, down from 0.0075, and the value of ℓ changes from 28.2 μs to 27.0 μs.

During operation, the reactivity of the core drops as a result of the strong negative fuel temperature coefficient of reactivity. For the mixed-LEU core, the drop at BOL is calculated to be $3.24 and at EOL is calculated to be lower because of the decrease in core average temperature with burnup. The licensee states that these values are typical.

The NRC staff reviewed the licensee's analyses, as discussed above, and finds that the licensee considered appropriate core physics parameters. The NRC staff concludes that the methods used to determine values of the core physics parameters and the values of the core physics parameters are acceptable. These values are similar to those found acceptable at other TRIGA reactors.

2.5.3 Operating Limits

The regulations in 10 CFR 50.36(d)(1) require reactors to specify safety limits and LSSSs. Safety limits are defined in 10 CFR 50.36(d)(1) as limits on 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 action will correct the abnormal situation before a safety limit is exceeded.

The safety limit for the WSU NRCR is as follows:

TS 2.1 Safety Limit—Fuel Rod Temperature

(1) The maximum temperature in a Standard TRIGA fuel rod shall not exceed 1000°C under any condition of operation.

(2) The maximum temperature in a 30/20 TRIGA fuel rod shall not exceed 1150°C under any condition of operation.

TS 2.1(1) and (2) specify the maximum fuel rod temperatures for the 30/20 LEU and standard TRIGA fuels to prevent damage to the fuel.

The licensee states that an important parameter for a TRIGA reactor is the fuel rod temperature. This is well-suited to be a safety limit specification because it can be measured using an IFR. A loss in the integrity of the fuel rod cladding could arise from a buildup of excessive pressure between the moderator and the cladding if the fuel temperature exceeds the safety limit. The pressure is caused by the presence of air, fission product gases, and hydrogen from the disassociation of the hydrogen and zirconium in the moderator. The magnitude of this pressure is determined by the fuel-moderator temperature and the ratio of hydrogen to zirconium in the alloy.

The licensee also states that safety limit for the standard TRIGA fuel rod is based on data that include the large mass of experimental evidence obtained during high-performance reactor tests on this fuel. These data indicate that the stress in the cladding because of hydrogen pressure from the disassociation of zirconium hydride will remain below the ultimate stress, provided that the temperature of the fuel does not exceed 1,000 degrees C and the fuel clad is water cooled.

The safety limit for the 30/20 LEU fuel is based on data indicating that 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 below 500 degrees C (Ref. 32). The properties and performance of the TRIGA higher uranium-weight-percent LEU fuel, including the 30/20 LEU fuel, have been evaluated by the NRC in NUREG-1282 (Ref. 40) and approved for use with the provision that case-by-case analysis discusses individual reactor operating conditions when using the fuel. The WSU NRCR safety limit is set at the temperature established in NUREG-1282 (Ref. 40).

An additional consideration is the need to provide adequate cooling relative to the maximum heat flux to prevent departure from nucleate boiling (DNB) and the resulting rapid increase in clad temperature, which will lead to failure of the clad (see Section 2.6 of this SER). A power level limit is calculated that ensures that the fuel temperature safety limit will not be exceeded and that film boiling will not occur. The design-bases analysis has shown that operation at 1.3 MW(t), the power level at which a reactor scram occurs, across a broad range of core and coolant inlet temperatures with natural convection flow, will not lead to film boiling. This is discussed further in Section 2.6 of this SER.

The NRC staff concludes that TS 2.1(1) and (2) establish the maximum fuel rod temperature safety limit for the WSU NRCR standard TRIGA and 30/20 fuels that is consistent with the safety limits used for other TRIGA reactor fuel rods (supported by research conducted by GA) and is previously approved by the NRC. The fuel rod temperature safety limits are therefore, acceptable to the NRC staff.

The licensee states that the LSSS is the measured IFR temperature that, if exceeded, will initiate a scram to prevent the fuel rod temperature safety limit from being exceeded. For the NRCR, the LSSS is set equal to or less than 500 degrees C as measured in the IFR at specific locations in the core. Exceeding this limit causes a scram of the reactor and protects the fuel from exceeding the safety limit. The IFE is not measuring the hottest fuel location in the reactor core. The relationship between the measured temperature in the IFR and the actual temperature at the fuel hot spot in the core has been determined to show that the setting of 500 degrees C protects the safety limit at the hottest point in the core. The IFR contains three thermocouples that measure the fuel temperature.

The LSSS for the WSU NRCR is as follows:

TS 2.2 Limiting Safety System Settings

The limiting safety system setting shall be 500°C or less, as measured in an instrumented fuel rod located in the central region of the core. The instrumented fuel rod shall be located in one of the following core lattice positions in the region of the core containing the 30/20 fuel rods: D2NE, D2SE, C3 (except for C3NE), D3, E3 (except for E3SE), C4, E4NE, E4NW, C5 (except for C5SW), D5SE, E5NE, E5NW, C6NW, or D6.

TS 2.2 specifies the acceptable locations for the IFR and the temperature limit as measured by any one of the three thermocouples in the IFR. The locations are chosen based on calculations performed for the conversion of the WSU NRCR to LEU fuel in order to ensure that the hottest location is protected not only against the safety limit but also against DNB. The licensee has shown that, for both the hottest and the coldest thermocouples, an IFR located in the above core positions would protect the fuel temperature safety limit for reactor power levels that are less than 1.3 MW(t) and would limit the steady-state temperature in the 30/20 fuel region to less than 800 degrees C. The setting of 500 degrees C provides at least a 350 degrees C margin of safety for the 30/20 fuel and at least a 200 degrees C margin of safety for standard 8.5/20 fuel.

The licensee indicates in the license renewal SAR, as supplemented, that the LSSS is applicable not only in steady-state operation but also in pulse mode. However, the temperature channel will not limit the peak power generated during a pulse because of the response time of the temperature channel as compared with the width of a pulse. The temperature scram would limit the total amount of energy generated in a pulse by cutting off the tail of the energy transient in the event that the fuel temperature limit was exceeded. Thus, the fuel temperature scram provides an additional degree of safety in the pulse mode of operation.

Based on the discussion above, the NRC staff finds that the safety limit and LSSS for the WSU NRCR are based on acceptable analytical and experimental investigations and are consistent with those approved by the NRC and used at other TRIGA-type reactors. On this basis, the NRC staff concludes that the LSSS of 500 degrees C and the accompanying conditions are sufficient to protect the safety limit and are therefore, acceptable.

2.6 Thermal-Hydraulic Design

2.6.1 Steady-State Operations

The thermal-hydraulic design of the WSU NRCR was presented by the licensee in the revised conversion SAR (Ref. 7), as supplemented, prepared for the conversion of the reactor to LEU. The NRC staff analyzed this design. For steady-state operation, the thermal-hydraulics were presented in two parts. The first part presented the methodology and results for the mixed-HEU core (core 34A). Operational data were used to benchmark the analytical results. The second part used the same computational technique to analyze the mixed-LEU core (core 35A). Results of the thermal-hydraulic analyses include fuel and coolant temperatures and the minimum departure from nucleate boiling ratio (MDNBR) at power levels of 1.0 and 1.3 MW(t). Results for the mixed-LEU core, core 35A, are given in Table 2-2. The results for the mixed-HEU core are shown for comparison purposes only.

The WSU NRCR fuel rods are cooled by natural convection. A natural circulation flow rate is established, balancing the driving head against the core entrance and exit pressure losses and frictional, acceleration, and hydrostatic head losses in the core flow channels.

The RELAP5 (Version 3.2) code, a widely used and benchmarked system code for power and research reactors, was used for the thermal-hydraulic analysis. The steady-state analysis considered four different flow channels, each based on a four-rod cluster of fuel elements. The model examined a single fuel rod and a representative flow area associated with that single fuel rod. Each flow channel was assumed to experience the same driving head, but the model ignored crossflow from one channel to another. Analysis and experiments in packed geometries similar to the WSU NRCR indicate that crossflow tends to increase the heat flux at which critical heat flux (CHF) occurs in the limiting channel; therefore, neglecting crossflow is expected to result in conservative calculation of the CHF. The four channels represent the average channel (fuel rod with core average power), the maximum powered channel (maximum rod power), and two channels associated with two IFE rods.

For a set of given input parameters, including the inlet water temperature of 30 degrees C and 50 degrees C, reactor power of 1.3 MW(t), system pressure, local pressure-loss coefficients, and the axial power distribution for the bounding fuel rods, RELAP5 calculates the natural circulation flow rate and, along the axial length of the flow channel, the coolant temperature, wall heat flux, and clad temperature. Geometric parameters for the flow channel and the fuel rod are given in Tables 23 and 24 of the revised conversion SAR (Ref. 7). The maximum-powered rod (hottest fuel rod) was determined by the DIF3D diffusion code to be located in core position D4NE. The rod power peaking is represented by the rod power factor, defined as the power generation in a fuel rod relative to the core averaged rod power generation. Two other power peaking factors are defined for the NRCR; the axial power factor and the intrarod peaking factor. The axial power factor represents the axial peak-to-average power ratio within a fuel rod, and the intrarod peaking factor represents the peak-to-average power in a radial plane within a fuel rod. The power peaking factors for the LEU mixed core 35A are presented in Tables 17 and 18 of the revised conversion SAR (Ref. 7) for the BOL and EOL. The axial power factors for the maximum power rod, the average rod, and two IFEs are shown in Figure 32 of the revised conversion SAR (Ref. 7) for the mixed-LEU core 35A (at BOL).

Table 2-2 Washington State University LEU Core 35A Thermal-Hydraulic Design Data

	Mixed-HEU Core 34A	Mixed-LEU Core 35A
REACTOR PARAMETERS		
Reactor power		
Licensed power, MW(t)	1.0	1.0
Maximum analyzed power level, MW(t)[a]	1.3	1.3[a]
Max. fuel temperature at 1 MW(t), °C	435[b]	499[c]
Max. fuel temperature at 1.3 MW(t), °C	520[b]	540[c]
Cold clean excess reactivity, Δk/kβ, $	7.17	6.94
Prompt negative temp. coefficient of reactivity -Δk/k-°C 23–1,000 °C	$0.54–1.51\times10^{-4}$	$0.60–1.27\times10^{-4}$
Coolant void coefficient, Δk/k per 1% void	-0.080%	-0.135%
Maximum rod power at 1 MW(t), kW/rod	20.9	20.8
Average rod power at 1 MW(t), kW/rod	8.4	8.4
Maximum rod power at 1.3 MW(t), kW/rod	27.2	27.0
Average rod power at 1.3 MW(t), kW/rod	10.9	10.9
Maximum rod power at DNB = 1.0, kW/rod	51.7[b]	45.8[c]
DNB ratio at operating power	2.47[b]	2.20[c]
Prompt neutron lifetime, μs	30.7	28.2
Effective delayed neutron fraction	0.0076	0.0075
Shutdown margin, $, with most reactive rod and reg. rod stuck out	-0.70	-0.91
SAFETY PARAMETERS		
Limiting safety system setting, ˚C	500	500
Maximum analyzed power level, MW(t)	1.3	1.3[a]
Minimum DNB ratio at 1.0 MW(t)	2.47[b]	2.20[c]
Minimum DNB ratio at 1.3 MW(t)	1.90[b]	1.69[c]
Calculated maximum positive pulsed reactivity insertion to reach T = 830°C, ($)	2.02	2.04 BOL 2.20 EOL
Peak pulsed fuel temperature, °C	830	830

[a] Maximum permitted, steady-state power level = 1.0 MW(t).
[b] Pool water inlet temperature = 30 °C.
[c] Pool water inlet temperature = 50 °C (TS limit).

The RELAP5 code was also used to determine the MDNBR using the Bernath correlation, a CHF correlation historically used for TRIGA reactors. The conversion SAR noted that the default CHF correlation in RELAP5 (the Groeneveld 1986 correlation) gave a less conservative CHF than the Bernath correlation. By accounting for peaking in the cladding wall heat flux and using a maximum rod power factor, the predicted minimum DNBR for a reactor power of 1.3 MW(t) is 1.69 for the mixed-LEU core 35A.

The thermal-hydraulic analysis is supplemented by the use of a finite-difference code, TAC2D, to calculate the steady-state fuel temperature. The GA code (Ref. 43) has been benchmarked and used for many TRIGA reactors. The code calculates temperatures in two-dimensional problems, with radial and axial power distributions in the fuel given as input. The fuel rod model consists of the central zirconium rod, the fuel annulus, the fuel-to-clad gap, and the stainless steel clad. TAC2D requires a boundary condition given by an input from RELAP5, a clad surface temperature or a coolant temperature with the corresponding clad surface heat transfer coefficient. The gap between fuel and cladding was assumed to be filled with air. A cold gap of 0.2 mils was assumed throughout the core, and some gap closure was expected to occur as a result of relative expansion of the fuel and cladding at normal operating temperatures.

Results of the benchmarking steady-state thermal-hydraulic analysis for the mixed-HEU core 34A are summarized in Tables 25 and 26 of the revised conversion SAR. A summary of the calculated and measured fuel temperatures for the mixed-HEU core 34A are shown in Table 27 of the revised conversion SAR (Ref. 7).

The calculated fuel temperatures at the sensing point of the IFR thermocouples were about 6 to 7 percent higher than the measured fuel temperature at a 1.0 MW(t) power level. The calculated maximum fuel temperature and the average fuel temperature at 1.0 MW(t) are 435 degrees C and 221 degrees C, respectively. Section 4.5.7 of the license renewal SAR (Ref. 1, as supplemented) discusses that, based on the comparison of calculated versus measured reactivity loss (from cold critical to 1.0 MW(t)), the thermal calculation likely underpredicts the core average temperature. This discrepancy is attributed to the average core gap of the lower powered fuel rods being slightly larger than the assumed 0.2 mils. A larger average gap will result in a higher core average temperature, making the calculated and measured reactivity loss more consistent with each other. It is noted that, for the mixed-LEU core 35A, a larger gap of approximately 2 mils was assumed for the fuel rods.

A similar thermal-hydraulic analysis to that described above was conducted for the mixed-LEU core 35A. The steady-state results for a 1.0-MW(t) operating power are summarized in Tables 28 and 29 of the revised conversion SAR. The RELAP5 calculations assumed more limiting power factors at the BOL conditions, namely a hot rod factor of 2.474 and an axial peaking factor of 1.286.

Results of the steady-state thermal-hydraulic analysis for the mixed-LEU core 35A are summarized in Tables 28 and 29 of the revised conversion SAR. Given that the maximum local heat flux occurs in the hot rod, the MDNBR of 1.92 is predicted for reactor power level 1.3 MW(t). At the maximum licensed power level of the WSU NRCR, there is significant margin to DNB.

A summary of the calculated fuel temperatures by TAC2D for the mixed-LEU core 35A are shown in Table 30 of the revised conversion SAR. The calculated fuel temperatures at the thermocouples of the two IFEs at 1.0-MW(t) power level are 427 degrees C and 440 degrees C. These calculated temperatures are for the bottom thermocouple (there are three thermocouples in each IFR), which tends to have the highest reading because of power tilting toward the lower half of the core from control blade insertion. The calculated maximum fuel temperature and the average fuel temperature at 1.0 MW(t) are 500 degrees C and 304 degrees C, respectively.

The thermal-hydraulic analysis was repeated for the mixed-LEU core 35A with the pool water temperature at the administrative limit of 50 degrees C and a reactor power level of 1.3 MW(t). At the combination of higher pool water inlet temperature and higher power, the peak fuel temperature was increased by 40 degrees C (from 500 degrees C to 540 degrees C), while the peak clad temperature was increased by 22.4 degrees C (from 142.6 degrees C to 165 degrees C). The corresponding MDNBR (calculated by the Bernath correlation) was reduced from 2.5 to 1.69. Even at the bounding conditions of 50 degrees C pool inlet temperature and a reactor power of 1.3 MW(t), the thermal analysis shows that the facility can operate safely.

The thermal-hydraulic analysis for the mixed-LEU core 35A contains discussions of two effects that have an impact on the MDNBR and the peak fuel temperature. Owing to its location adjacent to the transient rod, the maximum-powered fuel rod (located in core position D4NE) is shown to have a flow area about 3.5 percent smaller than at other core positions. Based on an analysis performed for a similar flow channel for the Texas A&M University HEU-to-LEU conversion (Ref. 44), a 3.3 percent reduction in the calculated MDNBR is expected from the reduced flow area.

The effect of radial gap on peak fuel temperature is shown in Figure 36 of the revised conversion SAR, indicating a higher temperature for a wider manufactured gap because the gap acts to insulate the fuel meat. According to the conversion SAR, the manufactured radial gap between the fuel and cladding is limited to a maximum of 2 mils, and the averaged radial gap is limited to less than 1.75 mils. From past operating experience, offset swelling of the fuel tends to close the gap once the core is put in operation. This implies a lower measured peak fuel temperature as the new core undergoes burnup. For the mixed-LEU core 35A fuel temperature calculations, a gap of 1.75 mils was assumed for the average fuel rod, and a gap of 2 mils was assumed for the hot fuel rod and the IFR.

On the basis of the discussion above, the NRC staff finds that the analysis performed by the licensee used qualified calculation methods and conservative or justifiable assumptions. The applicability of the analytical methodology used for the mixed-LEU core was demonstrated by comparing analytical results for the mixed-LEU core with measurements obtained from the mixed-HEU core. On this basis, the NRC staff concludes that the thermal-hydraulic analysis in the WSU NRCR conversion SAR demonstrates that the LEU core 35A results in acceptable safety margins in regard to thermal-hydraulic conditions.

2.6.2 Pulsed Operation

The limiting condition for pulsed operation of the WSU NRCR is the peak fuel temperature. The fuel temperature is limited to a maximum of 830 degrees C on the basis of early experience with TRIGA fuel, which demonstrated that fuel damage could occur as a result of hydrogen gas accumulation and redistribution in the hydride fuel if the reactor is pulsed after an extended period of operation at 1 MW(t) (Ref. 41). As a result of this physical feature of the fuel, the pulsed maximum reactivity insertion is limited to $2.31, and the allowable initial power level at the instant of pulse initiation is set below or at 1 kW, as discussed below.

The performance of a TRIGA reactor core in a step input of reactivity has been generally computed using the GA BLOOST computer code (Ref. 42). The code performs combined reactor kinetics-heat transfer calculations using a point kinetics model that analyzes reactor core power and fuel temperature reactivity transients with a variable-temperature fuel heat capacity model. The fuel rod is reasonably considered adiabatic for the short duration of the neutronic pulse. The BLOOST model can predict either the core average fuel temperature response or it can use power peaking factors to model the highest power density locations in the reactor core. The revised conversion SAR (Ref. 7), as supplemented, presented the following steady-state and pulsing calculations for core 34A (mixture of HEU/LEU fuel) and core 35A (LEU fuel):

1. Core 34A analysis - comparison of BLOOST predictions to available experimental data demonstrated that the code over predicts the energy generation and peak power, and also over predicts the fuel temperatures with reactivity insertions larger than $1.5. The code also calculated the maximum reactivity insertion, $2.02 that corresponds to the limiting fuel temperature of 830°C. Based on comparison with data, specifically the measured energy release and actual fuel temperatures, it is concluded that the BLOOST calculations are conservative.

2. Core 35A analysis - at beginning of life (BOL) the BLOOST analysis predicts that a reactivity insertion of $2.04 would lead to the limiting fuel temperature of 830°C. The analysis also predicts that at a $2.00 reactivity insertion the core average fuel temperature is 276°C, the peak temperature at the hottest fuel location is 800°C, and the temperature at the IFE location is 331°C. At end-of-life (EOL) the analysis also indicates that the energy release and corresponding maximum fuel temperature would be significantly lower than at BOL. Based on the conservative BLOOST analysis WSU established an administrative reactivity insertion limit of $2.00 for Core 35A, which is identical to the limit used for Core 34A. Any pulse larger than $2.0 requires the approval of the Reactor Supervisor and Facility Director.

The maximum allowable reactivity insertion was determined by the licensee and presented in the startup report (Ref. 16) for core 35A using an analytical procedure combined with measured data of pulse energy release versus reactivity insertion. The dependence of the pulse energy release as a function of reactivity insertion is measured using a high-speed pulse power channel that is based on the output of an uncompensated ionization chamber during a pulse. Using the temperature-dependent heat-capacity data of the LEU fuel, together with the assumed maximum fuel temperature of 795 degrees C (reduced from 830 degrees C to account for measurement uncertainties) and GA-calculated power peaking factors, WSU derived a limiting value for the maximum allowable energy release. The measured data for energy release versus pulse size were then used together with the computed maximum allowable energy release to determine the limiting maximum reactivity insertion of $2.31 (Ref. 16).

WSU established an administrative procedure and physical design features to prevent the initiation of a pulse from power levels exceeding 1 kW. The accidental pulsing of the transient rod requires the failure of the 1-kW interlock that prevents air from being applied to the transient rod piston for reactor power level above 1 kW and the failure of the operator to follow written procedures. BLOOST code analysis in the revised conversion SAR (Ref. 7) demonstrated that the maximum fuel temperature at the peak power location for a pulse initiated at a 1-kW power level would be 812 degrees C, below the GA-recommended pulsing limit of 830 degrees C.

The administrative limit of $2.00 reactivity pulse provides adequate protection for maintaining core 35A fuel temperature below 830 degrees C as required by TS 3.1.2. The maximum allowable reactivity insertion limit of $2.31 is based on measured data incorporating conservatisms to account for uncertainties in the measurements and the analytical procedure and is acceptable to the NRC staff. The NRC staff concludes that the administrative limit of $2.00, together with the requirements for review and approval when using reactivity insertions above $2.00, provide acceptable safety margins for limiting the maximum fuel temperature to below 830 degrees C. The analyses were done with 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 indicating that the analytical results are conservative. Because the BLOOST code has been shown to conservatively predict the pulse power and energy deposition in the fuel, its use for these calculations is acceptable.

The reactor can be safely pulsed to $2.31 without exceeding the 830 degrees C temperature limitation, but an administrative limit is conservatively set at $2.00.

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

2.7 Conclusions

Based on the above considerations, the NRC staff concludes that the licensee has presented adequate information and analyses to demonstrate its technical ability to configure and operate the WSU NRCR core without undue risk to the health and safety of the public or the environment. The NRC staff's review of the facility has 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 WSU NRCR conversion SAR demonstrates that the LEU core 35A results in acceptable safety margins with regard to thermal-hydraulic conditions. Although, the WSU NRCR is operated at 1.0-MW(t) power, the calculations show that the reactor can be safely operated to at least 1.3 MW(t) with adequate CHF 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 WSU NRCR and finds that, with pulse sizes up to the administrative limit of $2.00, 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 WSU TS regarding the reactor design, reactor core components, reactivity limits, and related surveillance requirements provide reasonable assurance that the reactor will be operated safely in accordance with the TS. The design features of the reactor are similar to those typical of research reactors of the TRIGA type operating in many countries worldwide. On the basis of its review, the NRC staff concludes that there is reasonable assurance that the WSU NRCR is capable of safe operation, as limited by the TS, for the period of the requested license renewal.

3. RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

3.1 Radiation Protection

Activities involving radiation at the WSU NRCR are controlled under the radiation protection program that meets 10 CFR Part 20 requirements and the recommendations in ANSI/ANS-15.11, "Radiation Protection at Research Reactor Facilities." The licensee has used the guidance of ANSI/ANS 15.11-2004 specifically, (Ref. 45). The regulations in 10 CFR 20.1101, "Radiation Protection Programs," 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.

TS 6.4.4(9) requires that the licensee's Reactor Safeguards Committee (RSC) review the radiation protection program annually:

TS 6.4.4 Reviews

The responsibilities of the Reactor Safeguards Committee or designated subcommittee shall include the following:

(….)

(9) annual review of the radiation protection program;

(…)

The specific annual review requirement related to the radiation protection program is consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007 and, therefore, is acceptable to the NRC staff.

The NRC staff finds that the NRC inspection program routinely reviews radiation protection and radioactive waste management at the WSU NRCR. Review of the annual operating reports of the WSU reactor facility and the NRC inspection reports concerning the radiation protection program demonstrated that adequate measures are in place to minimize radiation exposure to personnel and to provide adequate protection against operational releases of radioactivity to the environment. Based on the above discussion, the NRC staff concludes that the radiation protection program at the WSU NRCR is acceptable.

3.1.1 Radiation Sources

The NRC staff reviewed the descriptions of potential radiation sources, including the inventories of each physical form and their locations. The review of radiation sources included identification of potential radiation hazards as presented in Chapters 11 and 13 of the license renewal SAR (Ref. 1) and verification that the hazards were accurately depicted and comprehensively identified.

3.1.1.1 Airborne Radiation Sources

During normal operations, the primary airborne sources of radiation are argon-41 (Ar-41) and N-16. Ar-41 results from irradiation of the air in experimental facilities and the dissolved air in the reactor pool water. The primary producers of Ar-41 are the reactor pool and the thermal column. Other production sources may include beam ports, the pneumatic irradiation system, and irradiation tubes. N-16 is produced when oxygen in the pool water is irradiated by the neutrons from the reactor. The N-16 radionuclide has a half-life of 7.13 seconds.

The dose rate at the top of the reactor tank is for the most part from N-16, with a small contribution from Ar-41. The dose rate with the reactor operating at full power is about 50 millirem per hour (mrem/hr) with the diffuser off and 2 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, thereby reducing the bridge dose rate. TS 3.5.1 requires that radiation monitoring equipment be in operation at specific locations, including the reactor bridge, and that the equipment provide information about radiation levels to the reactor operator (see Section 3.1.4 of this SER).

The licensee states that occupational exposure from N-16 is minimal because access to the bridge above the core is minimal during reactor operation. Because of the short half-life of N-16, exposure to the public is negligible.

Ar-41 is induced in the air that flows through the reactor thermal column and other experimental devices and is also evolved from the pool water surface. The core is cooled by natural convection of pool water that causes the heated water to rise to the surface of the pool along with the air dissolved in the water; some of the dissolved air containing Ar-41 escapes into the pool room.

The licensee has determined the occupational dose level in the reactor room from Ar-41 during normal operation. Using data from the reactor room radiation monitors, the licensee has determined that the occupational exposure of Ar-41 averages about 1.6×10^{-8} microcuries per milliliter (μCi/mL), which is 0. 5 percent of the derived air concentration (DAC) limiting value of 3×10^{-6} μCi/mL, established in Table 1 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. The data were collected during reactor operational hours over a 2-month period, with the largest single measurement of 1.6×10^{-7} μCi/mL. Even if the reactor operated continuously at this Ar-41 concentration level for 2,000 hours per year, the licensee concluded the occupational exposure would still be about 5 percent of the limiting DAC value.

The reactor area has a ventilation system facilitating the removal of Ar-41 through the building exhaust system and limiting the Ar-41 dose to workers. The pool room exhaust air and thermal column air are combined and vented to the atmosphere after being monitored for Ar-41 content. Only Ar-41 has been found in analysis of effluent samples. TS 3.5.2(1) and (4) specify the Ar-41 discharge limit as follows:

TS 3.5.2 Effluents

(1) The concentration of ^{41}Ar in the effluent gas discharged from the facility into the unrestricted area, after environmental dilution shall not exceed 1×10^{-8} µCi/mL averaged over one year.

(….)

(4) The total annual discharge of ^{41}Ar into the environment shall not exceed 20 Ci per 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 average measured release rate for Ar-41 over a 10-year period was 10 curies (Ci) per year. The licensee assumed a release rate that was a factor of 10 higher, 100 Ci per year, and calculated that the atmospheric concentration at the site boundary is well below the limit of $1x10^{-8}$ µCi/mL in 10 CFR Part 20, Appendix B, as well as below the ALARA criteria of 0.1 millisievert per year as per the 10 CFR 20.1101 limit. The analysis demonstrated that the airborne sources released from the WSU reactor facility during normal operations do not present a significant exposure hazard.

Based on the analysis above, demonstrating that the WSU NRCR routine gaseous 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 adherence to TS 3.5.2(1) and (4) provides reasonable assurance that, during the continued normal operation of the WSU NRCR, the airborne radioactive releases will not pose a significant risk to public health and safety or the environment.

TS 4.5.2(1), (4), and (5) present the surveillance requirements for monitoring Ar-41 release as follows:

TS 4.5.2 Effluents

(1) The level of ^{41}AR in the effluent gas shall be continuously monitored during operation of the reactor.

(….)

(4) The annual discharge of ^{41}Ar shall be calculated annually.

(5) The continuous air monitor shall be continuously monitored during operation of the reactor.

(…)

The NRC staff concludes that adherence to TS 4.5.2(1), (4), and (5) provides reasonable assurance that, during the continued normal operation of the WSU NRCR, the airborne radioactive releases of Ar-41 will be appropriately monitored and will not pose a significant risk to public health and safety or the environment.

3.1.1.2 Liquid Radiation Sources

The pool surface exposure rate is, for the most part, from N-16. This was discussed in Section 3.1.1.1 of this SER for gaseous radiation sources. The reactor coolant system is another source of exposure. Because the piping carries pool water that has been circulated through the reactor core, irradiated corrosion products produced during normal operation may be capable of producing a whole-body exposure to personnel. This occurs only at high power levels when reactor pool water flows through the core. Potential exposure to operating staff is minimized by the implementation of the ALARA policy (see Section 3.1.3 of this SER).

As required by TS 4.3 (see Section 2.3 of this SER), the licensee samples primary water for radioactive content on a monthly basis to help detect potential fission product leakage from the reactor fuel, leakage from sealed sources, or activation of materials in the coolant water.

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

TS 3.5.2(6) specifies the requirement for radioactive material content of discharged liquids as follows:

> **<u>TS 3.5.2 Effluents</u>**
>
> (….)
>
> (6) The quantity of radioactivity in liquid effluents released to the sewer system shall not exceed the limits stipulated in 10 CFR 20 Appendix B, Table 3.

TS 4.5.2(6) specifies the requirement for analysis of liquids before discharge as follows:

> **<u>TS 4.5.2 Effluents</u>**
>
> (….)
>
> (6) Before discharge, the facility liquid effluents collected in the holdup tanks shall be analyzed for their radioactivity content.

All radioactive liquids discharged to the environment (via sanitary sewer) are monitored before release. The maximum activity of the liquids for direct discharge to the sewer system before dilution shall be 2×10^{-8} μCi/mL. For liquids with higher activities, further analysis is made to identify isotopic content. Discharge limits for the isotopes identified will be in accordance with 10 CFR Part 20 criteria for liquid effluents.

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

3.1.1.3 Solid Radiation Sources

The licensee states that radioactive waste is the primary solid radiation source at the WSU NRCR. The fission products in the reactor fuel constitute the most significant solid radiation source. Nonfuel sources include sealed sources in the reactor pool, activated reactor components, resins from the primary water demineralizer, and irradiated samples. Because final radioactivity is estimated before experimental irradiations are performed, both shielding and storage duration requirements will be known. Solid radiation sources are controlled by the radiation protection program.

The reactor contains an aluminum-clad antimony-beryllium startup source and a Cobalt-60 sealed source irradiation facility located in the reactor pool. The startup source and its location in the grid box have been discussed in Section 2.2.4 of this SER.

In addition, the WSU NRCR stores and uses sealed sources in the reactor pool.

TS 3.7 specifies the conditions on sealed sources stored and used in the reactor pool as follows:

TS 3.7 Sealed Sources in the Reactor Pool

(1) Sealed sources shall not be stored or used closer than five (5) feet from an operating reactor core.

(2) The total radioactivity of all sealed sources stored in the reactor pool shall not exceed 100,000 curies.

(3) All sealed source configurations shall be designed so that a loss of pool water accident shall not cause a loss of sealed source encapsulation integrity and the sources shall be stored in an appropriate shield to prevent a significant radiation hazard in the event of a loss of reactor pool water.

(4) The storage and use of sealed sources shall be considered an experiment subject to all applicable provisions for experiments in the Technical Specifications.

(5) A written Standard Operating Procedure for the storage and use of sealed sources in the reactor pool shall be in effect.

TS 3.7(1) helps ensure that sources cannot be moved within 5 feet of the reactor to limit the source's effect on the operating reactor core and also ensure that the source is not activated by the reactor.

TS 3.7(2) and (3) limit the total activity stored in the pool and the method for storage to ensure that the dose rate above the unshielded core in the case of a total LOCA would increase by no more than 2 percent.

TS 3.7(4) requires that the RSC review and approve the placement and use of radioactive sources in the pool.

TS 3.7(5) requires that standard operating procedures (SOPs) be in effect for the storage and use of sealed sources in the pool.

The licensee has removed from TS 3.7 a requirement that the pool water be monitored for radioactive content for indication of a potential leaking source because it has been incorporated into TS 4.3, "Primary Coolant Conditions," discussed in Section 2.3 of this SER, which requires such monitoring.

TS 4.7 specifies the surveillance requirements for sealed sources stored and used in the reactor pool as follows:

<u>TS 4.7 Sealed Sources in the Reactor Pool</u>

(1) An inventory of sealed sources in the reactor pool shall be kept to demonstrate that the total radioactivity remains below 100,000 curies.

(2) The written Standard Operating Procedure for storage and use of sealed sources shall be reviewed and approved by the Reactor Safeguards Committee.

TS 4.7(1) and (2) provide for the maintenance of an inventory of sealed sources and for review of SOPs for the use and storage of sealed sources.

The NRC staff reviewed TS 3.7, 4.3, and 4.7. The NRC staff finds that TS 3.7 along with TS 4.3 and TS 4.7 contain the basic requirements for handling sources in the reactor pool and help ensure the detection of degradation of source encapsulation. Therefore, the NRC staff concludes that the TS are acceptable because they along the provision of the radiation protection program control the use and storage of sources in the reactor pool, providing reasonable assurance that the sources can be handled without endangering the safety of the reactor and the reactor staff.

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

3.1.2 Radiation Protection Program

The regulation in 10 CFR 20.1101(a) requires that each licensee shall develop, document, and implement a radiation protection program. Section 11.0 of the WSU NRCR license renewal SAR (Ref. 1) states that the facility shall have a formal radiation protection program that meets the requirements of ANSI/ANS 15.11-1993 and 10 CFR Part 20. The use of the 1993 revision of ANSI/ANS 15.11 was later modified to the use of ANSI/ANS 15.11-2004 as discussed in Section 5.6.5 of this SER. Section 11.1.2.1 of the license renewal SAR describes the management of the radiation protection program. The ultimate responsibility for all activities at WSU is vested in the President of the University. For nonreactor-related areas, the responsibility for radiation protection is delegated to the Radiation Safety Officer and the Radiation Safety Committee. For radiation protection at the reactor, the responsibility is delegated to the Director of the Nuclear Radiation Center, with the Radiation Safety Committee performing the review and function over the radiation protection activities. The day-to-day radiation protection activities at the facility are performed by the operating staff, with the supervisory function performed by the Reactor Supervisor. The organizational chart (see Section 5.6.1 of this SER) indicates that there is coordination and cooperation between the Radiation Safety Committee and the Nuclear Radiation Center Director’s Office.

The radiation protection program establishes the following:

- exposure limits
- procedures and record system for surveys and monitoring
- requirements and responsibilities for personnel dosimetry

The WSU NRCR has a structured radiation protection program with the following elements:

- management commitment and worker responsibility
- qualification of personnel and adequacy of resources
- adequacy of authority for responsible positions
- new staff training and continuing education for all personnel
- radiological design as integral aspect of facility and experiment design
- radiological planning as an integral aspect of operations planning
- performance reviews of designs and operations
- analysis of personnel exposure records
- periodic assessment and trend analysis of the radiological environment
- periodic assessment and audits of the protection program
- surveillance activities
- protective equipment (supply, quality assurance)
- calibration and quality assurance programs
- training

The radiation protection program is implemented using written SOPs. The NRC inspection program routinely reviews the radiation protection program. Review of the annual operating reports and NRC inspection reports for the WSU NRCR 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.

The health physics SOPs and document control is described in the license renewal SAR (Ref. 1) under Section 11.1.2.3. Radiation protection procedures include testing and calibration of the monitors and detection instrumentation; administrative guidelines for receiving, monitoring, handling, transporting, and testing radioactive materials; decontamination; investigation; training; ALARA measures; and personnel access. TS 6.1(2) and TS 6.5 cover the radiation protection program. TS 6.4.4(9) requires an annual review of the radiation protection program by the RSC (discussed in Section 3.1 of this SER).

All personnel entering the facility are issued the appropriate monitoring devices and any required protective clothing. Section 11.1.2.3(n) of the license renewal SAR (Ref. 1) states that “All reactor users are trained in the use of the reactor and must pass a written user certification exam, SOP # 32, for ‘Security and Emergency Plan Training for Nuclear Radiation Center, Radiation Safety Office and Campus Police Personnel.’”

General levels of training cited in the license renewal SAR (Ref. 1) include storage, transfer, and use of radiation and radioactive material in portions of the restricted area; radioactive waste management and disposal; health protection problems and health risks; precautions and procedures to minimize exposure (ALARA); purposes and functions of protective equipment; applicable regulations and license requirements for the protection of personnel from exposure to radiation and radioactive materials; responsibility of reporting potential regulatory and license violations or unnecessary exposure; appropriate response to warnings in events or unusual occurrences; and radiation exposure reports.

All personnel permitted unescorted access to the WSU NRCR vital area receive additional training to include access control rules, emergency procedures, dosimetry requirements, key checkout and return, safety in the reactor and control rooms, communication systems, security door requirements, general checkout procedures when exiting the reactor bay, and emergency equipment location and use. Experiments and reactor equipment areas are surveyed on a regular basis, and radiological conditions are posted for required areas within the facility.

All gaseous and liquid effluents are monitored before release according to TS 3.5.2 and TS 4.5.2 to comply with 10 CFR Part 20 limits. Health physics and radiation protection program procedures are audited on an annual basis by the RSC. This includes all procedures, personnel radiation doses, radioactive material shipments, radiation surveys, and radioactive effluents released to unrestricted areas. The Reactor Supervisor oversees the maintenance of radiation protection program records, including radiological survey data, personnel exposure reports, training records, inventories of radioactive materials, environmental monitoring results, and waste disposal records. Records are kept for the life of the facility.

The NRC staff concludes that the WSU NRCR radiation protection program, as described in Section 11.1 of the license renewal SAR, complies with 10 CFR 20.1101(a), is implemented in an acceptable manner, and provides reasonable assurance that, for all facility activities, the program will protect the operating staff, the environment, and the public from unacceptable radiation exposures.

3.1.3 ALARA Program

To comply with the regulations in 10 CFR 20.1101, the WSU NRCR 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 industry guidelines of ANSI/ANS-15.11-2004. The program is applied through written procedures and guidelines described in Section 11.1.3 of the license renewal SAR (Ref. 1). All proposed experiments and operational procedures at the WSU reactor facility are reviewed for ways to minimize potential exposure to personnel. The WSU health physics staff participates in experiment planning to minimize both personnel exposure and generation of radioactive waste. Additionally, unanticipated or unusual reactor-related exposures are investigated to develop methods of preventing recurrence. The review of controls for limiting access and personnel exposure in the WSU reactor facility provides reasonable assurance that radiation doses to the environment, the public, and facility personnel will be ALARA. The ALARA program is adequately supported by the WSU NRCR management.

Section 11.1.3 of the license renewal SAR describes the ALARA program, which is applied not only to operations within the facility but also, through TS 3.5.2 and 4.5.2, to radioactive effluent releases from the facility discussed in Sections 3.1.1.1 and 3.1.1.2 of this SER.

In addition to the TS requirements for gaseous and liquid effluent releases from the facility discussed in Sections 3.1.1.1 and 3.1.1.2 of this SER, TS 3.5.2 presents additional requirements for effluent releases as follows:

TS 3.5.2 Effluents

(…)

(2) An environmental radiation monitoring program shall be conducted to measure the integrated radiation exposure in and around the facility.

(3) The annual radiation exposure due to reactor operation, at the closest off-site point of extended occupancy, shall not, on an annual basis, exceed the average local off-site background radiation by more than 20%.

(…)

(5) The reactor shall be shut down if a fission product leak from a fuel rod or an airborne radioactive release from an irradiated sample is detected by the continuous air monitor, and the reactor shall remain shut down until the source of the leak is located and eliminated. However, the reactor may continue to be operated on a short-term basis as needed to assist with identification of the source of the leak provided that occupational values listed in Table 1 of 10 CFR 20 Appendix B are not exceeded and effluent concentrations listed in Table 2 of 10 CFR 20 are not exceeded.

(…)

TS 3.5.2(2) requires an environmental monitoring program in and around the facility.

TS 3.5(3) requires that the public exposure at the closest offsite occupied point shall not exceed the background level by more than 20 percent.

TS 3.5(5) requires the licensee to identify potential leaks from fuel or samples and to shut down the reactor in the event of a significant leak. Allowing the reactor to operate on a short-term basis to determine the source of the leak is acceptable because the operating time is used to identify the source of the leakage and such leaks often occur only when the reactor is operating.

In addition to the surveillance requirements for gaseous and liquid effluent releases from the facility discussed in Sections 3.1.1.1 and 3.1.1.2 of this SER, TS 4.5.2 presents additional requirements for effluent releases as follows:

TS 4.5.2 Effluents

(….)

(2) The environmental radiation monitoring program required by Section 3.5.2(2) shall measure the integrated radiation exposure in and around the Dodgen Research Facility on a quarterly basis.

(3) The radiation levels determined by the environmental monitoring program shall be tabulated and examined annually.

(….)

TS 4.5.2(2) sets a quarterly interval for the environmental surveillance requirement.

TS 4.5.2(3) requires that the results of the environmental monitoring be examined annually.

The licensee submits the results of the environmental monitoring program to the NRC in the annual report. An examination of these reports by the NRC staff shows that the impact on the environment from the gaseous and liquid releases of the WSU NRCR is acceptable.

In addition, the NRC inspection program routinely reviews the ALARA program and finds that the program as implemented meets the requirements of the regulations. TS 3.5 and 4.5 require the facility to operate in a manner to minimize radioactivity exposure to workers and to the public. The NRC staff further concludes that the WSU NRCR ALARA program, as described in Section 11.1.3 of the license renewal SAR (Ref. 1) and implemented through TS 3.5 and 4.5, complies with 10 CFR 20.1101, is acceptable, and provides reasonable assurance that radiation exposure will be maintained ALARA.

3.1.4 Radiation Monitoring and Surveying

The regulations in 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 in 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 has sufficient range to cover the various types of radiation that may be encountered at the WSU NRCR. Systems used at the WSU NRCR for monitoring radiation include radiation monitors specified in TS 3.5 as follows:

TS 3.5 Radiation Monitoring System and Effluents

TS 3.5.1 Radiation Monitoring Systems

The reactor shall not be operated unless the radiation monitoring channels listed in Table 3.4 are operable. Each channel shall have a readout in the reactor control room and be capable of sounding an audible alarm that can be heard in the reactor control room.

Table 3.4 Minimum Radiation Monitoring Channels

Channel*	Number
Reactor bridge radiation monitor	1
Beam room radiation monitor	1
Continuous air monitor	1
Exhaust gas monitor	1

*During maintenance to the radiation monitoring channels, the intent of this specification will be satisfied if they are replaced with portable gamma radiation sensitive instruments with alarms or that shall be kept under visual observation.

Section 5.6 of the WSU NRCR license renewal SAR (Ref. 1) states that the reactor bridge radiation monitor monitors the dose rate above the core and will alert the operator if the N-16 diffuser ceases to operate. In addition, the bridge radiation monitor scrams the reactor and activates the building evacuation system if the bridge radiation level exceeds a preset high radiation level. The beam room monitor will alert the operator for unusual dose rate in the beam room. The continuous air monitor will alert the operator for unusual radioactivity in the pool room air, as, for example, from a fuel rod fission product leak. In addition, the continuous air monitor will activate the emergency ventilation system. The exhaust gas monitor will alert the operator for unusual radioactivity in the ventilation system exhaust. The radiation monitors are set to alarm at radiation levels just above levels experienced during normal reactor operation and can be adjusted based on SOPs under the direction of the Reactor Supervisor 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.

TS 4.5 specifies the surveillance requirements for the radiation monitoring systems as follows:

TS 4.5 Radiation Monitoring System and Effluents

TS 4.5.1 Radiation Monitoring System

(1) All radiation monitoring systems shall be verified to be operable by a monthly channel test.

(2) The following surveillance activities shall be performed on an annual basis:

(a) The reactor bridge and beam room radiation monitoring system shall be calibrated using a certified radioactive source.

(b) A calibration shall be performed on the continuous air monitor in terms of counts per unit time per unit of radioactivity using calibrated beta-particle emitting sources.

(c) A calibration of the exhaust gas monitor system shall be done using at least two different calibrated gamma-ray sources.

Monthly channel tests combined with annual calibrations constitute a schedule sufficient to identify any changes to the operating characteristics of the monitoring systems. The NRC reviewed the information provided in the SAR, as supplemented, and finds that these surveillance requirements are consistent with guidance provided in NUREG-1537 and ANSI/ANS 15.1-2007 and are sufficient to detect any changes to the operating characteristics of the monitoring systems. On the basis of this review, the NRC staff concludes that the surveillance requirements in TS 4.5.1 for the radiation monitoring system are acceptable.

The NRC staff finds that the licensee has adequate instruments and equipment for quantitative radiation measurements and they are calibrated periodically and that TS 3.5 requires sufficient monitors to evaluate potential radiation hazards. Routine effluent releases are within regulatory limits, and the discussion in Section 4 of this SER shows that the consequences of accidents are acceptable. On this basis, the NRC staff concludes that TS 3.5 is 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 are appropriate to ensure compliance with 10 CFR 20.1501(a) and (b) and the facility ALARA program.

3.1.5 Radiation Exposure Control and Dosimetry

Reactor shielding is based on the combination of pool water and the concrete pool structure. The dose rate at the outside surface of the pool structure in the beam room is normally 0.2 mrem/hr when the reactor is at full power. The ventilation system maintains the reactor room at negative pressure with respect to outside areas and maintains Ar-41 and N-16 levels below the limits prescribed in 10 CFR 20.1201, "Occupational Dose Limits for Adults."

The licensee uses optically stimulated luminescent dosimeters to monitor personnel whole-body exposure, which are assigned to individuals who have the potential to be exposed to radiation. Thermoluminescent dosimeters are provided for extremity monitoring. Portable equipment is used to perform weekly radiation surveys by the Radiation Safety Office. Personnel protective equipment is used as needed. Facilities and equipment to decontaminate persons are available, if needed. Procedures exist that govern the use of this equipment.

The licensee uses survey meters to measure does rates from radiation fields, and these measured rates are posted where required. These provisions provide assurance that external and internal radiation monitoring of all individuals required to be monitored meet the requirements of 10 CFR Part 20 and the goals of the facility ALARA program.

The licensee maintains personnel exposure records and effluent and environmental monitoring readings for the life of the WSU NRCR. For this license renewal review, the NRC staff reviewed the annual operating and inspection reports for 2009. The review revealed that the highest annual whole-body exposure received by a single individual was 61 millirem (mrem). The highest annual extremity exposure for that period was 210 mrem shallow-dose equivalents, and the highest skin or other shallow dose was 77 mrem shallow-dose equivalents. All WSU NRCR staff received significantly lower radiation doses than the limits in 10 CFR 20.1201.

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 the WUS NRCR 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.

3.1.6 Contamination Control

The licensee performs contamination surveys on a daily, weekly, and monthly basis, depending on the frequency that radioactive material is used or handled. Handling of any radioactive material within the WSU reactor facility is controlled by written procedure. Workers are trained for working with radioactive material, including how to limit its spread during entering and exiting an area containing radioactive material. The facility surveys have routinely shown no detectable contamination in nonradiological areas of the facility. The NRC staff reviewed the WSU NRCR records and the NRC inspection reports. This review showed that adequate controls exist to prevent the spread of radiological contamination within the facility. Based on a review of the WSU NRCR Radiation Protection Program and on a history of satisfactory contamination control, the NRC staff concludes that adequate controls exist to prevent the spread of contamination within the facility.

3.1.7 Environmental Monitoring

TS 3.5.2 (see Section 3.1.3 of this SER) requires that an environmental radiation monitoring program shall be conducted to measure the integrated radiation exposure in and around the environs of the facility. TS 4.5.2 requires that the program be conducted on a quarterly basis.

The licensee's environmental monitoring consists of taking various samples at 20 different locations, which include unrestricted areas adjacent to the WSU NRCR facility and a number of offsite locations. The environmental monitoring program is audited by the Radiation Safety Committee on a semiannual basis to ensure that the environmental monitoring program contains an adequate number of samples, locations, and sufficient frequency of collection such that the analysis of the samples has sufficient sensitivity to ensure that the overall program complies with 10 CFR 20.1301, "Dose Limits for Individual Members of the Public," and will provide an early indication of any environmental impact caused by the reactor facility operation. Direct gamma radiation measurements are performed monthly. The licensee includes the results of the environmental monitoring program in its annual report.

The NRC staff reviewed the licensee's environmental monitoring program and the results of the program as reported in the licensee's annual reports. The NRC staff finds that the environmental monitoring program can properly assess the day-to-day operation of the facility to minimize the radiological impact on the environment. On this basis, the NRC staff concludes that the environmental monitoring program is sufficient to assess the radiological impact of the WSU NRCR on the environment.

3.2 Radioactive Waste Management

The purpose of the radioactive waste management program is to minimize radioactive waste and ensure that it is properly handled, stored, and disposed. All radioactive waste handling operations are controlled by procedure and overseen by the WSU NRCR health physics staff.

TS 6.10.4(6) through (9) specify that, annually, the facility report to the NRC as follows:

TS 6.10 Reports to the U.S. Nuclear Regulatory Commission

TS 6.10.4 Written Report to the U. S. NRC Within 60 days after June 30 of Each Year

(….)

(6) 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 before the point of such release or discharge:

(7) liquid waste (summarized on a monthly basis):

  (a) monthly radioactivity discharged;

  (b) total estimated quantity of radioactivity released (in curies);

  (c) an estimation of the specific quantity for each detectable radionuclide in the monthly release;

  (d) fraction of 10 CFR 20 Table 3, Appendix B limit for each detectable radionuclide taking into account the dilution factor from the total volume of sewage released by the licensee into the sewage system;

  (e) sum of the fractions for each radionuclide reported above;

  (f) total quantity of radioactive material released by the facility into the sewage system during the reporting period.

(8) gaseous waste (summarized on a monthly basis) radioactivity discharged during the reporting period, including:

  (a) total estimated quantity of radioactivity released (in curies) determined by an appropriate sampling and counting method;

  (b) total estimated quantity of ^{41}Ar released (in curies) during the reporting period based on data from an appropriate monitoring system;

  (c) estimated average atmospheric diluted concentration of ^{41}Ar released during the reporting period in terms of μCi/mL and fraction of the applicable DAC value;

(d) total estimated quantity of radioactivity in particulate form with half-lives greater than 8 days (in curies) released during the reporting period as determined by an appropriate particulate monitoring system;

(e) average concentration of radioactive particulates with half-lives greater than 8 days released in μCi/mL during the reporting period;

(f) an estimate of the average concentration of other significant radionuclides present in the gaseous waste discharge in terms of μCi/mL and fraction of the applicable DAC value for the reporting period if the estimated release is greater than 20% of the applicable DAC.

(9) solid waste (summarized on an annual basis):

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

(b) total radioactivity in solid waste in curies,

(c) the dates of shipment and disposal (if shipped off-site).

(….)

The specific annual reporting requirement related to radioactive effluent releases as required by TS 6.10.4 is consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. Based on its review of TS 6.10.4, the NRC staff concludes that the reporting requirement and the reporting in the WSU NRCR annual reports are acceptable to the NRC staff.

3.2.1 Radioactive Waste Management Program

The WSU reactor facility license renewal SAR described the movement process and release practices of radioactive waste from controlled to uncontrolled areas. The NRC staff reviewed the facility radioactive waste release practices as described in the license renewal SAR and finds that it 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. The NRC staff also finds that the licensee has adequate controls in place to prevent uncontrolled personnel exposures from radioactive waste operations and provide the necessary accountability to prevent any potential unauthorized release of radioactive waste.

Base on its review of the licensee's radioactive waste management program, the NRC staff concludes that the program is acceptable.

3.2.2 Radioactive Waste Packaging and Labeling

Low-level solid radioactive waste from laboratory experiments or disposable protective clothing items are accumulated and stored in authorized containers. Activated equipment and activated irradiation samples are stored in the reactor bay area for reuse or to decay to low-level activity limits. When filled, the low-level waste containers are sealed and transferred to the WSU

Radiation Safety Office until they are shipped off campus by a licensed carrier to a licensed facility for disposal. Procedures are in place to monitor the radiation exposure from waste storage areas within the facility and to perform required handling operations, such as packaging and transfer, and the preparation of proper documentation associated with shipment. During the 2009–2010 reporting year, the WSU NRCR disposed of 21 cubic feet of noncompacted solid waste containing 0.32 millicurie.

The NRC staff reviewed the licensee's radioactive waste packaging and labeling techniques and concludes that these techniques are acceptable.

3.2.3 Release of Radioactive Waste

Normal operation of the WSU reactor facility does not produce significant liquid radwaste. However, small quantities of liquid waste are periodically generated by minor leakages and sampling of the reactor pool and the primary coolant cooling equipment. These liquid wastes are collected and stored in a retention tank until determination of final disposition is made. Non-alpha-emitting liquid radwaste from the retention tank is periodically released to the sanitary system in accordance with approved discharge permit WAC 246-221-190 issued by the State of Washington Department of Health and Ecology. All releases are governed by written procedures to ensure that they are within the limits stated in 10 CFR Part 20, Appendix B, Table 3. Fresh water can be added to the retention tank if dilution is necessary before discharge.

The NRC staff reviewed the licensee's techniques for release of radioactive liquid waste to the sewer system and concludes that these techniques are acceptable.

3.3 Conclusions

On the basis of its review of the information presented in the license renewal SAR (Ref. 1), observations of the licensee's operations, and the results of the NRC inspection program, the NRC staff concludes the following:

- The WSU NRCR radiation protection program complies with the requirements in 10 CFR 20.1101(a), is acceptably implemented, and provides reasonable assurance that the facility 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.

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

- The results of radiation surveys carried out at the WSU NRCR, doses to the persons issued dosimetry, and the results of the environment monitoring program help verify that the radiation protection and ALARA programs are effective.

- Potential radiation sources have been adequately identified and described by the licensee. The licensee sufficiently controls radiation sources.

- 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 WSU NRCR staff and public will be below applicable 10 CFR Part 20 limits.

- The facility radioactive waste management program provides reasonable assurance that radioactive waste released from the facility will neither exceed applicable regulations nor pose unacceptable radiation risk to the environment and the public

4. ACCIDENT ANALYSES

4.1 Accident-Initiating Events and Scenarios

The accident analysis presented in the WSU NRCR license renewal SAR, as supplement, helped to establish safety limits and limiting conditions that are imposed on the WSU NRCR through the TS. The licensee analyzed potential reactor transients and other accidents. The analysis included the potential effects of natural hazards as well as potential accidents involving the operation of the reactor. In addition, the NRC staff obtained independent analysis of accidents with TRIGA-fueled reactors (Refs. 46, 47 and 48) and compared those results with accidents analyzed by the licensee. As will be demonstrated below, none of the potential accidents considered in the WSU NRCR license renewal SAR, as supplemented, would lead to significant occupational or public exposure.

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

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

4.1.1 Maximum Hypothetical Accident

The licensee states that the MHA for the WSU NRCR is an assumed cladding failure of one highly irradiated fuel rod with no radioactive decay followed by the instantaneous release of the noble gases and halogen fission products directly into the reactor pool room air. Boundary conditions and assumptions include using a conservative fuel rod power density during operation of 28 kilowatts thermal (kW(t)) in the rod with saturated inventories used for all released isotopes, and no credit for iodine absorption in the reactor pool water or dilution or filtration removal by the ventilation system. The occupational dose was calculated for an individual in the reactor pool room as well as the radiological exposure to the public, which included a person at 15 meters (50 feet) from the building and at 600 meters (2,000 feet). The nearest permanently occupied dwelling is approximately 626 meters to the southwest of the WSU NRCR facility. In addition, the dose for the closest individual in the nonreactor area of the Dodgen Nuclear Radiation Center was also calculated. A review of this methodology by the NRC staff showed that the WSU NRCR facility analysis was consistent with the TRIGA MHA recommended in NUREG-1537 and adequate to calculate occupational and public radiation doses.

Radioactive releases from WSU NRCR facility operations can only occur if the fuel cladding is breached. The NRC staff reviewed the licensee's analytically generated radionuclide inventory for the MHA, and the assumptions and boundary conditions used. The NRC staff finds they were representative of the LEU fuel used in the WSU NRCR. The NRC staff also finds that the use of a 5-minute exposure to occupationally exposed workers in the reactor pool room and a 5-minute exposure to individuals in the Dodgen Nuclear Research Center is considered reasonable because evacuation drills conducted by the WSU NRCR facility have demonstrated the ability to evacuate personnel from the reactor pool room and from the Dodgen Nuclear Research Center within the 5-minute timeframe.

Because the noble gases do not condense or combine chemically, it is assumed that any gases released from the cladding will diffuse in the air until their radioactive decay. On the other hand, the iodines are chemically active and are not volatile below about 180 degrees C. Therefore, some of the radioactive iodines will be trapped by the materials with which they come in contact, such as water and reactor building structures. Most of these iodines will not become nor remain airborne under many accident scenarios applicable to research reactors. However, to be certain that the calculations discussed below lead to the upper limit dose estimates for the MHA, the licensee assumed that all of the iodine released from the fuel remains airborne and could be released to the environment.

The MHA is based on a single fuel rod cladding failure in air in the reactor room. The licensee's analysis is based on the following generally conservative assumptions:

- The LEU rod has a power density of 28 kW based on reactor operation at 1 MW(t) for 200 days. Hence, the radioactive halogen and noble gas fission products (with the exception of krypton-85) are at saturation. The burnup calculation for krypton was extended to achieve saturation.

- The fuel element has a conservative maximum temperature of 535 degrees C. After averaging over the fuel volume and temperature, a release fraction of 2.6×10^{-5} is applied to arrive at the total gap activity (Ref. 47).

- All of the gap activity is released to the reactor room and is instantly mixed uniformly with the air.

- The most stable atmospheric class (Pasquill F) is assumed; wind speed of 1 meter/second is assumed; and deposition and meandering processes are neglected.

- All of the noble gases and halogens from the gap are available for release to the environment through the emergency ventilation system.

- The airborne release is assumed to be a ground release, which is a conservative assumption, as opposed to an elevated release through the stack.

- It will take 5 minutes to evacuate persons from the reactor room and the areas in the Dodgen Nuclear Research Center outside of the reactor room.

- The closest person to the building is at 15 meters (50 feet), within the 20-meter wake cavity formed by the building.

- The nearest resident is at 626 meters; the evaluation is performed for 600 meters (2,000 feet).

The calculation additionally assumes that the pool room ventilation system would be operating in the dilution mode and that any radiological species suspended in the air would be released to the environment through the reactor building stack, a short length of pipe approximately 8 feet above the level of the building. When operating in the dilution mode, the ventilation system removes reactor pool room air at not less than 300 cubic feet per minute (cfm) and air from an uncontaminated source and releases it to the environment through the reactor building stack. The gaseous material that is discharged from the ventilation system is assumed to be diluted by atmospheric air in the "lee of the building due to turbulent wake effects." A Gaussian plume dispersion model with ground-level release is used to compute the doses external to the building.

During normal operation, the pool room ventilation system is required to operate as per TS 3.4(1). For a high reading on the continuous air monitor, the ventilation system automatically turns to a dilution mode. On a reactor scram, the ventilation system automatically shuts down and isolates the reactor room. The operator may select to turn it on manually to the dilution mode. For these reasons, the licensee performed analyses for the MHA not only with the ventilation system in the dilute mode but also in the isolate mode. The calculations considered the following scenarios depending on the ventilation system status:

(1) <u>Scenario A</u>—pool room ventilation in isolation mode (shut down) without radioactive decay:

    (a) Occupational dose—due to inhalation and submersion of radioactive species inside the reactor room for the 5-minute evacuation period. The dose value is derived by increasing the Scenario A(a) radioactive source term by 20 percent to account for a scenario without ventilation and radioactive decay (Ref. 20).

    (b) Dose to a member of the public inside the Dodgen Nuclear Research Center—dose is due to gamma-ray shine through the reactor room door from the radioactive species suspended in the reactor room (Ref. 19). The licensee used a very conservative source term from the renewal license application (Ref. 1), which is larger than the source term used in all other scenarios based on the LEU revised conversion SAR (Ref. 7). The resulting dose due to gamma-ray shine is therefore conservative.

        In this analysis, the inhalation dose component was not considered, because leakage through the door is negligible due to the lack of differential pressure across the reactor door. All leakage from the reactor room is directed to the outside.

(c) Dose to nearest member of public—assumed a ground release from building leakage. The dose value is bounded by Scenario B(c) because air flow in dilution mode is much higher than the building leakage rate.

(d) Dose to an individual at nearest residence—a ground release with conservative weather conditions.

(2) Scenario B—pool room ventilation in dilution mode with radioactive decay:

(a) Occupational dose—individual in the reactor room who evacuates the room in 5 minutes; radioactive decay is considered (Ref. 7).

(b) Dose to member of public inside the Dodgen Nuclear Research Center—dose is due to gamma-ray shine only from the radioactive species suspended in the reactor room. Because the reactor room is under negative pressure that precludes any leakage into unrestricted areas, the calculations did not consider inhalation dose from inhaled radioactive particles (Ref. 19).

(c) Dose to nearest member of public outside the Dodgen Nuclear Research Center—assumed ground release, which is conservative since an elevated release would result in a lower dose (Ref. 7).

(d) Dose to an individual at nearest residence—conservative weather conditions are assumed to persist throughout the release (Ref. 7).

The analysis for the licensee for Scenario A was presented by the licensee in a response to a request for additional information (Ref. 20).

The analyses for Scenarios B(a), (c), and (d) were presented by the licensee in the SAR for the conversion of the WSU NRCR to LEU (Ref. 7).

The analysis for Scenario B(b) was presented by the licensee in a response to a request for additional information (Ref. 19).

Table 4-1 presents the licensee’s results for the whole-body dose for Scenarios A and B as follows:

Table 4-1 MHA Whole-Body Dose

		Dose, Scenario A mrem	Dose, Scenario B mrem
a	Occupational	26	22
b	Public, inside Dodgen Center building	2	2
c	Public, nearest outside location, perimeter	0.8	0.8
d	Public, nearest residence	0.52	0.04

Table 4-2 presents the licensee's results for the inhalation thyroid dose for Scenarios A and B as follows:

Table 4-2 MHA Thyroid Dose

		Dose, Scenario A mrem	Dose, Scenario B mrem
a	Occupational	3,324	2,770
b	Public, inside Dodgen Center building	n/a*	n/a*
c	Public, nearest outside location, perimeter	185	185
d	Public, nearest residence	0.72	9.8

* no inhalation dose, only gamma-ray shine

The dose limits presented in NUREG-1537 are as follows:

(a) occupational—5,000 mrem whole-body and 30,000 mrem thyroid

(b) public—500 mrem whole-body and 3,000 mrem thyroid

The licensee states that the calculated exposures are below the occupational and public exposure limits based on NUREG-1537 and are also within the limits given in the current versions of 10 CFR 20.1201 and 10 CFR 20.1301.

The NRC staff reviewed the analysis performed by the licensee regarding the consequences for the MHA. The NRC staff finds that the licensee's analyses used qualified methodologies with an acceptable radiation source term and incorporated conservative or justifiable assumptions on other boundary conditions. In conducting the MHA evaluation, the NRC staff used dose limits in 10 CFR Part 20 and finds that the calculated occupational and public radiation exposures for the MHA are within the limits 10 CFR Part 20. Therefore, the NRC staff concludes that the licensee's analysis of the MHA is acceptable.

4.1.2 Insertion of Excess Reactivity

In Section 13.5.3 of the revised conversion SAR (Ref. 7), the licensee analyzed two events that would create a sudden insertion of reactivity in the LEU core. These events are (1) the rapid insertion of a large amount of reactivity into the reactor operating at full power by the ejection of the inserted transient rod worth $3.19, and (2) the rapid insertion of a large amount of reactivity into the reactor operating at full power by the unplanned removal of a secured experiment whose reactivity worth is $2. The analyses used hand calculations and the BLOOST code (Ref. 42). The analyses were performed at BOL and EOL.

The accidental pulsing of the transient rod requires the failure of the 1-kW interlock that prevents air from being applied to the transient rod piston for reactor power level above 1 kW and the failure of the operator to follow written procedures.

Under this scenario, at the BOL, the licensee calculates that the maximum fuel temperature in the hottest fuel rod will be 997 degrees C and the TS safety limit would not be reached. At the EOL, the licensee calculates that the maximum fuel temperature in the hottest fuel rod will be 1,003 degrees C and the TS safety limit of 1,150 degrees C in the LEU 30/20 fuel rod would not be exceeded.

The accidental removal of a secured experiment requires violation of written procedures as the operator unfastens the experiment and pulls it rapidly out of the core (assumed to happen in 0.3 second).

The licensee demonstrates that this accident is less severe than the accidental pulsing accident described above, primarily because the removal of the experiment (and the introduction of the reactivity) occurs over a much longer time period than the accidental pulsing of the transient rod. The much slower withdrawal time will result in activating the 125-percent-power scrams at a lower portion of the reactivity insertion curve, resulting in lower peak power and lower peak fuel temperatures than those for the transient rod accidental pulsing.

Both of these scenarios are extremely unlikely because they require either the failure of an operator to follow written instructions together with the failure of an interlock, or deliberate violation of written procedures with rapid removal of an experiment. Even in these unlikely scenarios with extremely conservative assumptions, the BLOOST results indicate that the TS safety limit of 1,150 degrees C in the LEU 30/20 fuel rod would not be exceeded.

The NRC staff reviewed the information as discussed above and concludes that there are sufficient design features and administrative restrictions in place to make accidental pulsing or secured experiment removal unlikely and that the safety limit will not be exceeded if either event did happen.

4.1.3 Loss of Coolant

There are two scenarios that could result in a significant loss of coolant from the reactor pool: (1) pumping or draining of the water from the reactor pool from installed components, or (2) a pool penetration failure, such as failure of a beam tube. Pumping or draining all of the coolant water in the pool from installed components is not possible because of the location of the piping and siphon breaks in the piping. Pumping using a portable pump is administratively controlled.

Tank failures because of corrosion or other failures that lead to slow loss of water will be detected by instrumentation and the reactor staff. A drop in pool level will set off a low-level alarm as described in Section 2.3 of this SER. A loss-of-coolant alarm requires corrective action. During normal working hours, the alarm is observed in the reactor control room and at a monitored remote location, and the reactor staff would response to the alarm. During off hours, the alarm is monitored at the campus police station, which would contact the reactor staff. Slow leaks could be compensated for by adding water to the reactor pool.

The licensee calculated the consequences of the assumption that all water was removed from the pool. Because the TRIGA reactors have negative void coefficients, the reactor would scram even if the accident began at full power. The time for removal is assumed to be 900 seconds, based on the time it would take for the water to drain from a 10-inch beam tube after actuation of the low-level alarm for pool water. The licensee states that the 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. The NRC staff has previously accepted that no fuel damage is expected if the fuel temperature does not exceed 900 degrees C (Refs. 7 and 40).

Calculations of the maximum fuel temperature during a LOCA versus fuel rod power density during operation were performed by the licensee assuming that fuel uncovery occurs 15 minutes after a failure of a beam tube. The calculations show that the standard 8.5/20 LEU fuel can be operated up to 22.3 kW(t) per rod and the 30/20 LEU fuel up to 23.5 kW(t) per rod without exceeding the temperature limits of 900 degrees C for the 8.5/20 and 940 degrees C for the 30/20 LEU fuel. The maximum power density occurs in the 30/20 LEU fuel rod and is 20.8 kW(t) for the WSU NRCR operating at 1 MW(t). Allowing for a 4-percent level of uncertainty in reactor power, the maximum power density would be 21.6 kW(t) per rod occurring in the 30/20 LEU fuel. Therefore, the temperature limits for both standard 8.5/20 and 30/20 LEU fuels would not be exceeded during a LOCA.

These conclusions are supported by studies performed for TRIGA fuels, which show that, in general, as long as the operating power is less than 1.5 MW(t), the fuel cladding should not breach during a LOCA (Refs. 46 and 48).

A more likely scenario than complete draining of the reactor pool is that the core is only partially uncovered. GA performed experiments that concluded that the temperature rise for a partial loss of coolant is less severe than for a complete loss of coolant (Ref. 48).

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

The licensee states that another consequence of a LOCA is the gamma-ray dose from the exposed core. Dose rates at various locations in and around the facility have been calculated to determine the feasibility of recovery operations following a LOCA, such as pool refill and unloading of the core into storage racks. The dose rates were calculated at various positions for five time periods after shutdown with the assumption that the reactor has operated for a very long time at a power level of 1 MW(t). The positions used in the calculations were directly above the uncovered core on the reactor bridge, on the roof of the building, and at floor level in the pool room at the freight door at the east end of the room. The time periods were 10 seconds, 1 hour, 1 day, 1 week, and 1 month.

The calculations in the WSU license renewal SAR show (Ref. 1) that the reactor bridge and areas directly above the core will be inaccessible for a long period of time without some refill of the pool. However, the remainder of the building, where only scattered radiation occurs, will be accessible for recovery operations following an assessment of radiation levels.

In the unlikely event of a catastrophic leak from the reactor pool, the impact on public safety and the environment would not be significant. The water would drain into a sump and holdup tank that have a capacity of 4,000 gallons; ultimately, the water would collect in the sump and reactor bay floor. Any water would be collected and analyzed before disposal. The radioactivity level of the water is expected to be low and within the limits found in Appendix B to 10 CFR Part 20 for release to the sewer. The NRC staff concludes that potential total loss of primary coolant is an extremely unlikely event and, if it occurs, will not have significant impact on the health and safety of the public or environment.

The NRC staff reviewed the licensee’s LOCA analysis for the LEU core documented in the NRC SER for the LEU conversion (Ref. 15). The NRC staff finds that the LOCA does not result in damage to the reactor fuel because the maximum fuel temperature is below the air-cooled limit of 950 degrees C. In addition, as shown above, doses within the reactor facility are within the limits that would allow timely recovery operations to proceed. As a result, the NRC staff concludes that the results for the LOCA analysis are acceptable and the LOCA does not pose significant risk to the health and safety of the public or personnel.

4.1.4 Loss of Coolant Flow

The WSU reactor core is cooled by natural convection of pool water up between the fuel rods, and the pool water is cooled in a secondary cooling system by forced circulation to a cooling tower. In the event of a possible blockage of a coolant channel created by a foreign object lodged in the grid plate, the open fuel element lattice would ensure continuing cooling of all fuel rods as a result of coolant crossflow.

The NRC staff finds that loss of secondary cooling would be observed by operators using various indicators at the console, and the reactor would be shut down. The shut down reactor would continue to be safely cooled by natural circulation, while the reactor pool would slowly heat at decay heat level at a rate of 5.9 degrees C per hour per MW. On this basis, the NRC staff concludes that, even if the pool water should evaporate leaving the core uncovered, the fuel would be coolable, as described in Section 4.1.3 of this SER.

4.1.5 Mishandling or Malfunction of Fuel

The licensee has an established procedure for handling of fuel and has no recorded incidents of mishandling its fuel elements and no incidents of malfunctioning fuel. TS 3.1.6 and 4.1.6 require surveillance of fuel rods by visual inspection and measurement in order to verify the continuing integrity of the fuel rod cladding.

The NRC staff finds that in the event that a fuel rod is dropped while submerged, or that the fuel would suffer damage due to manufacturing defect or corrosion, the consequences of the incident would be bounded by the MHA since any fission product release would be significantly mitigated beyond the assumptions of the MHA analysis. This is due to the scrubbing effect of the pool water overlying the fuel rod. The NRC finds that the mishandling of a fuel rod would have no significant effect on the building occupants or members of the public.

The NRC staff concludes that any potential fission product release would be less than computed for the MHA analysis.

4.1.6 Experiment Malfunction

TS 3.6 and 4.6 place limits on experiments installed in the reactor and associated experimental facilities. The objectives of these limits are to prevent damage to the reactor and to limit any potential releases of radioactive materials and resulting exposures to personnel in the event of an experimental failure. Limits are placed on reactivity worths, mass of explosive materials, and other experiment materials to avoid accidental reactivity insertions, damage to reactor components, and release of radioactivity. TS 6.4.4 requires review and approval of all new experiments, during which the limits on experiments are analyzed and approved to ensure that releases will be lower than for the MHA analysis.

The NRC staff concludes that experiment malfunction will not result in consequences more severe than those computed for the MHA analysis.

4.1.7 Loss of Normal Electrical Power

The WSU NRCR has an auxiliary reactor emergency supply (ARIES). The licensee states that the ARIES system provides battery power for operation of the area monitoring system, building evacuation alarm system, pool level alarm, seismograph alarm, and the security system. However, the emergency electrical power system is not necessary to safely shut down the reactor and is not required to ensure public health and safety. In the event of a loss of electrical power without emergency power, all control rods would insert into the core automatically by gravity due to the loss of power to the electromagnets and, for the transient rod, the three-way solenoid valve that holds the control rods in position. Upon loss of electrical power, the primary and secondary coolant pumps would stop. Reactor decay heat would be dissipated through natural circulation in the primary coolant.

The NRC staff finds that there is sufficient coolant in the reactor pool to absorb the decay heat from the reactor without the need for the primary or secondary cooling system. The NRC staff concludes that loss of normal electric power poses little risk to the health and safety of the public or to WSU NRCR staff.

4.1.8 External Events

The likelihood of external events such as hurricanes, floods, and tornadoes is considered extremely low. Although the likelihood of an earthquake is also considered to be extremely low, the reactor control system contains a seismic switch, which would shut down the reactor in the event of a significant seismic event. Any such event, however unlikely, would lead to reactor shutdown. Calculations show that the fuel can be cooled in air as long as the reactor is shut down. The potential consequences of external events would be bounded by the MHA analysis.

The licensee has analyzed the potential crash of a small aircraft into the side of the reactor building. The analysis indicated that the potential consequences of such an event are extremely low and pose insignificant hazard to the members of the public. Aircraft collisions are considered unlikely and not able to breach the concrete shield at the core level. The NRC staff considers the likelihood of such an event extremely low and concludes that the members of the public are not subject to undue radiological risk following an aircraft crash.

The NRC staff concludes that external events pose insignificant risk to the health and safety of the public or to the WSU NRCR staff.

4.1.9 Mishandling or Malfunction of Equipment

The licensee states that the reactor is designed for "fail-safe operation" and that all conceivable failures at worst could only lead to reactor shutdown requiring repair. The licensee also stated that it cannot envision an equipment malfunction that would lead to a reactor accident type that has not already been analyzed.

The WSU NRCR is designed to shut itself down without damage in the event of a large reactivity insertion and would also shut itself down without damage at a power level of 1.3 MW(t). The worst conceivable accidents would lead to uncovering of the reactor core. The core uncovering would lead to negative reactor feedback and the reactor would shut itself down if it was not already in a shutdown state. Calculations performed by the licensee show that the fuel can be cooled in air for WSU NRCR decay heat loadings. The NRC staff finds that the physical limitations of the WSU NRCR design are such that mishandling or malfunction of equipment would lead to consequences that are bounded by the MHA. The NRC staff concludes, therefore, that the consequences of mishandling or malfunction of equipment pose negligible risk to the health and safety of the public or to the WSU NRCR staff.

4.2 Conclusions

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

- Under conservative assumptions, the MHA will not result in occupational radiation exposure of the facility staff or radiation exposure of the public in excess of the applicable NRC limits in 10 CFR Part 20.

- The administrative limit for pulse reactivity insertions is conservatively specified as $2.00. For accidents involving insertions of excess reactivity, the licensee has demonstrated that a pulse reactivity limit of $2.31 will result in peak fuel temperatures below the safety limits (830 degrees C).

- The LOCA does not result in damage to the reactor fuel. The reactor can be safely cooled with all fuel rods in an air environment. Doses within the reactor room and at the site boundary are below the limits of 10 CFR Part 20.

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

On the basis of its review, the NRC staff concludes that there is reasonable assurance that no credible accident would cause significant radiological risk to the facility staff, the environment, or the public.

5. TECHNICAL SPECIFICATIONS

5.1 Definitions

5.1.1 TS 1.0 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:

30/20 Fuel: [See Section 2.2 of this SER.]

Annual: Annual shall mean a time interval of 12 months, not to exceed 15 months.

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

Biennial: Biennial shall mean a time interval of 24 months, not to exceed 30 months.

BNC Design Modification: The term BNC Design Modification as applied to the BNC Facility beam refers:

(1) to a change that is shown to alter the dose vs. depth profile of the fast neutron, thermal neutron or gamma rays in the beam as sensed by the calibration check; and

(2) a change that has the potential to increase the amount of activation products in the BNC Facility.

BNC Experiment: BNC experiment shall be defined as a boron neutron capture experiment which utilizes the BNC facility, including the neutron irradiation of biological cells that have been enriched with boron.

BNC Facility: BNC facility shall mean the boron neutron capture facility that includes the BNC neutron beam, bridge moving system, beam monitoring equipment, beam shielding room, access door and experimental area viewing equipment. Experimental bench(s), experiment positioning equipment and other equipment used for neutron beam targets shall not be considered part of the BNC Facility for purposes of this definition except insofar as radiation safety (i.e., activation and/or contamination) is concerned.

BNC Neutron Beam Calibration: The term BNC neutron beam calibration shall be defined as the process of measuring the intensity and energy spectrum of a BNC neutron beam for the purpose of conducting a BNC Experiment.

BNC Radiation Fluence: The term “radiation fluence” means the total fluence of neutrons and gamma radiation that is emitted by the BNC facility beam. The determination of the ratio of gamma, fast neutron and thermal neutron fluences is part of the beam characterization, allowing the total radiation fluence to be monitored by the online detectors, which are neutron sensitive. Compliance with the limits specified for the radiation fluence is determined by reference to the fluence monitored by these detectors.

BNC Retracted Position: The retracted position is any position of the research reactor greater than 4 feet from the BNC beam port.

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

Channel Calibration: A channel calibration is an adjustment of a channel such that its output corresponds with acceptable accuracy to known values of the parameter that the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip setpoints, and shall include a channel test.

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

Channel Test: A channel test is the introduction of a signal into the channel to verify that it is operable.

Cold Critical: The reactor is in the cold critical condition when it is critical ($k_{eff} = 1$) with the fuel and pool water temperature both at ambient temperature.

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

Control Rod: [See Section 2.2 of this SER.]

Core Configuration: [See Section 2.2 of this SER.]

Core Lattice Position: [See Section 2.2 of this SER.]

Excess Reactivity: Excess reactivity is the amount of reactivity which would be added if all reactivity control devices were moved to the maximum reactive condition from the point at which the reactor is exactly critical ($k_{eff} = 1$) at reference core conditions or at a specified set of conditions.

Experiment: Any operation, hardware, or target (excluding devices such as detectors, foils, etc.) which is designed to investigate non-routine reactor characteristics or is intended for irradiation within the pool, on or in a beam port or irradiation facility. Hardware rigidly secured to a core or shield structure which is part of the design to carry out experiments is not normally considered to be an experiment. Experiments are classified as movable or secured as follows:

(1) Movable experiment: A movable experiment is one in which it is intended that all or part of the experiment can be moved in or near the core or into and out of the reactor while the reactor is operating.

(2) Secured Experiment: A secured experiment is any experiment, experimental apparatus, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means or by gravity and is not readily removable from the reactor. The restraining forces shall be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces which are normal to the operating environment of the experiment, or by forces that can arise as a result of credible malfunctions.

Fuel Assembly: [See Section 2.2 of this SER.]

Fuel Rod: [See Section 2.2 of this SER.]

Irradiation Facilities: Any in-pool experimental facility that is not a normal part of the core and which is used to irradiate devices and materials.

Instrumented Fuel Rod: [See Section 2.2 of this SER.]

License: The written authorization, by the responsible authority, for an individual or organization to carry out the duties and responsibilities associated with a personnel position, material or facility requiring licensing.

Licensed Area: The licensed area is the part of the Dodgen Research Facility building which is subject to the requirements of the WSU license R-76. This area includes:

(1) the reactor pool room (also known as Room 201) and adjacent rooms that allow direct unrestricted access to the pool room.

(2) the beam room (also known as Room 2) and adjacent rooms that allow direct unrestricted access to the beam room.

Licensee: An individual or organization holding a license.

Limiting Safety Systems Setting: Limiting safety systems settings are the settings for automatic protective devices related to those variables having significant safety functions.

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

Measuring Channel: A measuring channel is the combination of sensor, interconnecting cables or lines, amplifiers, and output devices that are connected for the purpose of measuring the value of a parameter.

Mixed Core: [See Section 2.2 of this SER.]

Monthly: Monthly shall mean a time interval of 30 days, not to exceed 45 days.

Off-site: Off-site shall mean any location which is outside the site boundary.

On-site: On-site shall mean any location which is within the site boundary.

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

Operability Test: An operability test is a test to determine whether a component or system is capable of performing its intended function.

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

Operational Core: [See Section 2.2 of this SER.]

Pool room ventilation system: The pool room ventilation system is the combination of fans, dampers, filters, ductwork and controls that provides controlled movement of air into and out of the reactor pool room.

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

Pulse Mode: Pulse mode operation shall mean operation of the reactor with the mode selector switch in the pulse position.

Quarterly: Quarterly shall mean a time interval of three months, not to exceed four months.

Reactor Bridge: The bridge is the structure that spans the research reactor pool to provide a support structure from which the research reactor is suspended.

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

Reactor Operating: [See Section 2.5.1 of this SER.]

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

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

Reactor Secured: [See Section 2.5.1 of this SER.]

Reactor Shutdown: [See Section 2.5.1 of this SER.]

Reference Core Condition: [See Section 2.2 of this SER.]

Reportable Occurrence: Any of the following events is a reportable occurrence:

(1) violation of a safety limit;

(2) release of fission products into the environment;

(3) release radioactivity from the site above limits established by 10 CFR 20 or specified within these Technical Specifications;

(4) operation with actual safety system settings for required systems less conservative than specified in the Technical Specifications;

(5) operation in violation of a Limiting Condition of Operation listed in Section 3;

(6) operation with a required reactor or experiment safety system component in an inoperative or failed condition which renders or could render the system incapable of performing its intended safety function;

(7) an unanticipated or uncontrolled change in reactivity greater than $1.00;

(8) abnormal and significant degradation in reactor fuel or cladding, or both, coolant boundary, or confinement boundary where appropriate;

(9) an observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy causes or could have caused the existence or development of an unsafe condition with regard to reactor operations.

Research Reactor: A research reactor is defined as a device designed to support a self-sustaining neutron chain reaction for research, development, education, training, or experimental purposes and that may have provisions for the production of radioisotopes.

Research Reactor Facility: The research reactor facility includes all areas within which the owner or operator directs authorized activities associated with the reactor.

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

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

Safety Limits: Safety limits are 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.

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

Semi-Annual: Semi-annual shall mean a time interval of six months, not to exceed 7.5 months.

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

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

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

SRO: The term "SRO" is an acronym for Senior Reactor Operator.

Standard Fuel: [See Section 2.2 of this SER.]

Steady-State Mode: Steady-state mode shall mean any operation of the reactor with the mode selector switch in the manual or auto position.

True value: The true value is the actual value of a parameter.

Unrestricted Area: Unrestricted area shall mean any location that is off-site.

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

Vacant Core Position: [See Section 2.2 of this SER.]

Weekly: Weekly shall mean a time interval of 7 days, not to exceed 10 days.

The licensee has relocated the definitions used in TS 3.8 and 4.8 to this definitions section of the TS. These definitions apply specifically to the BNC facility for BNC experiments. The definitions uniquely apply to the safe operation and use of the BNC facility for its intended purpose for BNC experiments in accordance with the guidance in Section 16.2 of NUREG-1537, Part I. The unique definitions, specifically applicable to the BNC facility, are acceptable to the NRC staff because they support the safe operation and use of the BNC facility for BNC experiments.

The remaining definitions 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.

5.2 Safety Limits and Limiting Safety System Settings

5.2.1 TS 2.1 Safety Limit—Fuel Rod Temperature

See Section 2.5.3 of this SER.

5.2.2 TS 2.2 Limiting Safety System Settings

See Section 2.5.3 of this SER.

5.3 Limiting Conditions of Operations

5.3.1 Reactor Core Parameter

5.3.1.1 TS 3.1.1 Steady-State Operation

See Section 2.5.1 of this SER.

5.3.1.2 TS 3.1.2 Pulse Mode Operation

See Section 2.5.1 of this SER.

5.3.1.3 TS 3.1.3 Shutdown Margin

See Section 2.2 of this SER.

5.3.1.4 TS 3.1.4 Maximum Excess Reactivity

See Section 2.2 of this SER.

5.3.1.5 TS 3.1.5 Core Configuration Limitation

See Section 2.2 of this SER.

5.3.1.6 TS 3.1.6 Fuel Parameters

See Section 2.2.1 of this SER

5.3.2 TS 3.2 Reactor Control and Safety System

TS 3.2.1, 3.2.2, and 3.2.3 specify the requirements for the reactor control and safety system. TS 3.2.1 helps ensure that the control rods will promptly shut down the reactor upon a scram signal. TS 3.2.2 and 3.2.3 specify the requirements for the safety measurement channels that must be available during reactor operation.

5.3.2.1 TS 3.2.1 Control Rods

See Section 2.2.2 of this SER.

5.3.2.2 TS 3.2.2 Reactor Measuring Channels

The TS 3.2.2 specification is as follows:

> The reactor shall not be operated in the specified mode of operation unless the measuring channels listed in Table 3.1 are operable.

Table 3.1 Required Operable Measuring Channels

Channel	Minimum Number Operable	Operating Mode	
		Steady State	Pulse
Fuel rod temperature	1	X	X
Linear power level	1	X	-
Log power level	1	X	-
Integrated pulse power	1	-	X

TS 3.2.2 helps ensure that, during the normal operation of the WSU NRCR in the specified mode of operation (i.e., steady-state or pulse), sufficient information is available to the operator to ensure safe operation of the reactor. The minimum number of operable measuring channels shown in Table 3.1 of the TS will provide the operator with fuel temperature displayed at the control console giving continuous information on this parameter, which has a specified safety limit, and with linear, log, and integrated pulse power-level monitors to ensure that the reactor power level is adequately monitored for both steady-state and pulsing modes of operation.

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

5.3.2.3 TS 3.2.3 Reactor Safety System

The TS 3.2.3 specification is as follows:

> The reactor shall not be operated unless the safety channels described in Table 3.2 and interlocks described in Table 3.3 are operable.

Table 3.2 Minimum Reactor Safety Channels

Safety Channel	Function	Number operable in specified mode: Steady State	Number operable in specified mode: Pulse
Fuel temperature	Scram if fuel temperature exceeds 500 °C	1	1
Power level	Scram if power level exceeds 125% of full licensed power	2	-
Manual scram	Manually initiated scram	1	1
High-voltage monitor	Scram on loss of high voltage to power channels	1	1
Preset timer	Transient rod scram 15 seconds or less after pulse	-	1
Pool level alarm	Alarm if primary coolant level drops more than 8 inches below normal	1	1
Bulk primary coolant temperature	Manual scram if primary coolant temperature reaches 50 °C	1	1

TS 3.2.3, Table 3.2 ensures that, during the normal operation of the WSU NRCR 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 channels shown in Table 3.2 of the TS will provide the following safety measures:

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

- Manual scram allows the operator to shut down the system if an unsafe or abnormal condition occurs.

- In the event of failure of the power supply for the safety chambers, operation of the reactor without adequate instrumentation is prevented.

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

- Pool level alarm is provided to alert the operator for pool water loss to ensure that there is 16 feet of water above the top of the core.

- Bulk primary coolant temperature indication is provided to alert the operator to scram the reactor because the limit on bulk primary coolant temperature has been reached.

Table 3.3 Minimum Interlocks

Interlock	Function	Number operable in specified mode Steady State	 Pulse
1 kW pulse interlock	Prevent initiation of a pulse above 1 kW	-	1
Startup count rate inhibit on the log power level channel	Prevent control rod withdrawal when neutron count rate is less than 2 counts per second	1	-
Pulse-mode switch	Prevent withdrawal of standard control and regulation rods in pulse mode	-	1
Transient rod control	Prevent application of air pressure unless fully inserted	1	-
Control element selector	Prevent withdrawal of more than one control rod at a time	1	1

TS 3.2.3, Table 3.3 specifies the following interlocks:

- 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 rod temperature safety limits to be exceeded.

- Interlock to prevent startup of the reactor with less than 2 counts per second ensures that sufficient neutrons are available for proper startup.

- Interlock to prevent withdrawal of the standard or regulating control rods in the pulse mode is to prevent the reactor from being pulsed while on a positive period.

- Interlock to ensure proper operation of the transient control rod.

- Interlock to limit the rate of reactivity addition.

The NRC staff finds TS 3.2.3 to be acceptable because it ensures that the reactor will not be operated unless the required minimum number of reactor safety channels and interlocks are operable to ensure safe operation of the reactor.

5.3.2.4 TS 3.2.4 Pool Level Alarm

See Section 2.3 of this SER.

5.3.3 TS 3.3 Primary Coolant Conditions

See Section 2.3 of this SER.

5.3.4 TS 3.4 Ventilation System

The TS 3.4 specifications are as follows:

(1) The reactor shall not be operated unless the facility ventilation system is operable and operating, except for periods of time not to exceed 48 hours to permit repair or testing of the ventilation system. The ventilation system is operable when flow rates, dampers and fans are functioning normally. The normal, dilute and isolation modes shall be operable for the ventilation system to be considered operable.

   (a) The exhaust flow rate of the ventilation system in the normal mode, from the reactor pool room, shall be not less than 4000 cfm.

   (b) The exhaust flow rate of the ventilation system in the dilute mode, from the reactor pool room, shall be not less than 300 cfm.

(2) The reactor pool room atmospheric pressure shall be maintained negative with respect to the areas outside the pool room when the ventilation system is in the normal or dilute mode.

(3) The ventilation system shall automatically switch to dilute mode upon a high activity alarm from the Continuous Air Monitor.

(4) The ventilation system shall be switched to the isolate mode upon initiation of a reactor scram.

(5) The dilute mode air filter shall be changed whenever the pressure drop across the filter increases by 1 in. of water above the initial level.

The licensee stated that the purpose of this specification is to ensure that the ventilation system is operable and in operation to mitigate the consequences of possible releases of radioactive materials resulting from reactor operation. TS 3.4 establishes the ventilation flow rates during the normal and dilute modes of operation. TS 3.4 helps ensure that, during the normal operation of the WSU NRCR, the concentration of gaseous radioactive isotopes Ar-41 and N-16 are kept below limits in Appendix B to 10 CFR Part 20 for occupational and public dose. TS 3.4 also helps ensure that, in case of accidental release of radioactivity in the reactor room, the ventilation system in dilution mode provides sufficient airflow to dilute the radioactivity concentration in the effluent released from the stack. In addition, TS 3.4 allows the operator to secure the reactor, allowing airborne radioactive material to be retained in the reactor building, if needed. The NRC staff finds that TS 3.4 ensures that the ventilation 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 is acceptable.

5.3.5 TS 3.5 Radiation Monitoring System and Effluents

5.3.5.1 TS 3.5.1 Radiation Monitoring Systems

See Section 3.1.4 of this SER.

5.3.5.2 TS 3.5.2 Effluents

See Section 3.1.1.1, 3.1.1.2, and 3.1.3 of this SER.

5.3.6 TS 3.6 Limitations on Experiments

The TS 3.6 specifications are as follows:

> The reactor shall not be operated unless the following conditions governing experiments exist:
>
> (1) The reactivity worth of a moveable experiment shall be less than $1.00.
>
> (2) The reactivity worth of a secured experiment shall not exceed $2.00.
>
> (3) The sum of the absolute values of all individual experiments shall not exceed $5.00.
>
> (4) Explosive materials, such as TNT, or its equivalent, shall be limited to 25 mg, for irradiation in the reactor or experimental facilities. Explosive materials in quantities less than 25 mg may be irradiated in the reactor or experimental facilities, provided prior testing of explosive material encapsulation is shown to ensure no reactor damage in the event of a detonation and that the pressure produced upon detonation of the explosive has been demonstrated to be less than half the design pressure of the container.
>
> (5) Experimental materials, except fuel materials, which could off-gas, sublime, volatilize, or produce aerosols under:
>
> (a) normal operating conditions of the experiment or reactor;
>
> (b) credible accident conditions in the reactor;
>
> (c) possible accident conditions in the experiment;
>
> shall be limited in radioactivity so that if 100% of the gaseous radioactivity or radioactive aerosols produced escaped to the reactor room or the atmosphere of the unrestricted area outside the facility, the airborne concentration of radioactivity would not exceed the limits of 10 CFR 20, Appendix B, Table 1 or Table 2 averaged over one year. An atmospheric dilution factor of 3.4×10^{-3} for gaseous discharges from the facility shall be used in calculations of unrestricted area effluent discharges.

(6) Pursuant to specification (5) above, the following conditions shall be shown to exist:

   (a) at least 90% of the particles will be retained if the effluent from an experiment is designed to exhaust through a filter installation designed for greater than 99% filtration efficiency for 0.3 micrometer particles;

   (b) at least 90% of the vapors will be retained in the experiment or in the reactor pool for materials whose boiling point is above 60 °C and the materials are exposed to conditions in which the material can boil, and vapors formed by boiling this material can escape only through an undisturbed column of water above the core.

(7) Each fueled experiment shall be controlled so that the total radioactive inventory of iodine isotopes 131 through 135 in the experiment is less than 1.5 Ci.

(8) The experimental material and potentially damaged components shall be inspected to determine the consequences and need for corrective action if a capsule fails and releases material that could damage the reactor fuel or structure by corrosion or other means.

(9) Corrosive materials shall be doubly encapsulated. All liquid and gas samples shall be analyzed to determine whether they require double encapsulation.

In TS 3.6(1), the reactivity limit of less than $1.00 for a single unsecured experiment is designed to prevent prompt criticality from occurring and is substantially below the analyzed maximum allowable pulse size of $2.31 (see the conversion startup report Ref. 16).

TS 3.6(2) establishes the reactivity limit of $2.00 for secured experiments. Because the experiment is held stationary in the reactor, the likelihood that it would fall away from the core to produce an undesirable step increase in reactivity is minimized. In TS 3.6(2), the reactivity limit of $2.00 for secured experiments is designed to be below the analyzed maximum allowable pulse size of $2.31. The energy release, after removing a $2.00-worth experiment from a cold critical reactor, will not result in fuel temperatures exceeding the safety limit.

In TS 3.6(3), the sum of the absolute value reactivity worths of all individual experiments shall be less than $5.00. This value is permissible if it does not violate the TS on excess reactivity and shutdown margin. Because the experimental reactivity limits in TS 3.6(1) through (3) keep the reactivity of experiments within bounds are shown to be safe, TS 3.6(1) through (3) are acceptable to the NRC staff.

TS 3.6(4) states that no explosive material in excess of 25 milligrams (mg) is allowed to be irradiated in the reactor. Explosive material up to 25 mg may be irradiated, provided the pressure produced on detonation of the explosive has been calculated or experimentally demonstrated to be less than the design pressure of the irradiation container. The licensee has modified TS 3.6(4) by the insertion of the word “half,” requiring the design pressure of the irradiation container to be twice the pressure produced on the detonation of the explosive. This modification allows the TS to be consistent with the recommendations of Regulatory Guide 2.2, “Development of Technical Specifications for Experiments in Research Reactors,” issued November 1973 (Ref. 49). The 25-mg limit is a longstanding limit also discussed in Regulatory Guide 2.2. The TS 3.6(4) requirement for experiments containing explosive materials is consistent with the recommendation of Regulatory Guide 2.2, and, therefore, it is acceptable to the NRC staff.

TS 3.6(5) and (6) place a limit on the radioactive products produced in experimental materials that may release airborne radioactive particles and provide conditions to be used in the safety analysis of the experiment. The purpose of TS 3.6(5) and (6) is to help ensure that potential releases of radioactive material from experiments are bounded by the exposure limits in 10 CFR Part 20 for WSU NRCR staff and members of the public. This includes failures under normal reactor operations, credible reactor accident conditions, and accident conditions in the experiment. The assumptions in TS 3.6(5) and (6) are standard for research reactor TS and ensure that source term calculations are conservative. TS 3.6(5) and (6) are acceptable to the NRC staff because they limit doses from potential experiment failure or malfunction to 10 CFR Part 20 limits.

TS 3.6(7) limits the iodine content of a fueled experiment to 1.5 Ci. TS 3.6(7) is acceptable to the NRC staff because it limits exposure from potential fueled experiment failure or malfunction to 10 CFR Part 20 limits.

TS 3.6(8) requires the removal of potentially corrosive material from the failure of an irradiation capsule. The TS also requires performing a physical inspection of the fuel and structures to determine the consequences of the failure and the need for corrective action.

TS 3.6(8) is acceptable to the NRC staff because it does not allow the operation of the reactor with damaged fuel or structures and requires the operator to take appropriate corrective action.

TS 3.6(9) requires double encapsulation of corrosive materials to help to ensure that the encapsulation does not fail.

The NRC staff has reviewed the licensee’s limitations on experiments. The TS areas recommended by NUREG-1537 and ANSI/ANS-15.1-2007are covered by the WSU NRCR TS along with associated surveillance requirements (see Section 5.4.6 of this SER). The technical content of the TS is consistent with guidance and provides an envelope of performance against which proposed experiments can be evaluated. Therefore, the licensee’s limitations on experiments in TS 3.6 are acceptable to the NRC staff.

5.3.7 TS 3.7 Sealed Sources in the Reactor Pool

See Section 3.1.1.3 of this SER.

5.3.8 TS 3.8 Boron Neutron Capture Facility

The TS for the BNC facility, as originally written, contained definitions, limiting conditions of operation, and surveillance requirements in a single specification. The licensee has modified the TS and placed the definitions in Section 1, the limiting conditions of operation in Section 3, and the surveillance requirements in Section 4 for these TS. The overall requirements remain unchanged.

The TS 3.8 specifications are as follows:

(1) It shall be possible to initiate a scram of the reactor from a control panel located in the BNC facility area. In the event that the BNC facility scram is inoperable, it shall be acceptable to use one of the control room scrams via communication with the reactor operator as a temporary means of satisfying this provision. Use of this temporary provision is limited to seven consecutive working days.

(2) Access to the BNC facility shall be controlled by means of a single access door located at the entrance to the BNC facility.

(3) The following BNC facility features and controls shall be operable:

   (a) The reactor bridge shall automatically move to the retracted position upon initiation of a reactor scram.

   (b) A BNC experiment may not be conducted if the BNC room radiation monitor reading exceeds 50 mrem/hr 30 seconds or longer after the initiation of the scram and bridge retraction.

   (c) An interlock shall prevent movement of the reactor bridge from the retracted position unless the BNC facility access gate is closed.

   (d) The reactor shall scram and the reactor bridge shall move to the retracted position automatically upon opening the BNC facility room access door.

   (e) The reactor bridge shall move to the retracted position automatically upon failure of BNC facility electric power or low voltage on backup batteries that are used to power the reactor bridge movement motor.

   (f) There shall be a means to manually move, without the use of electric power, the reactor bridge to the retracted position.

   (g) It shall be possible to initiate a signal from within the BNC facility to cause the reactor bridge to move to the retracted position.

   (h) A key operated switch near the BNC facility access door shall prevent reactor control rod withdrawal when the key is not inserted and turned to the locked position.

(4) The reactor bridge shall be equipped with a sensor that shows the position of the reactor bridge. The BNC facility control panel shall include a reactor bridge position indicator.

(5) It shall be acceptable to use an alternate means of verifying reactor bridge position, such as a video camera in the reactor pool room providing a signal to a video monitor at the control panel of the BNC facility in the event of a bridge position readout malfunction at the BNC

facility control panel. Use of an alternate means of bridge position verification is limited to seven consecutive working days.

(6) The BNC facility shall be equipped with a display of the reactor linear power and log power channels on a BNC facility control panel.

(7) The BNC facility shall be equipped with a display that provides an indication of the radiation level within the facility that indicates both within the facility and at the local control panel and provides an audible alarm within the facility and at the BNC facility control panel.

(a) The radiation monitor shall be equipped with a backup power supply.

(b) The radiation monitor audible alarm shall be set at or below 50 mrem/hr. The monitor and/or its alarm may be disabled once the BNC beam room has been searched and secured by closing and locking the BNC facility access door. If the radiation monitor and/or the audible alarm are disabled, both the monitor and the audible alarm shall automatically become functional upon opening of the BNC facility access door.

(c) Personnel entering the BNC facility shall use portable radiation detection instruments and audible alarm personal dosimeters if the radiation monitor becomes inoperable during use of the BNC facility. Use of portable radiation detection instruments and audible alarm personal dosimeters as a temporary means of satisfying this provision shall be limited to seven consecutive working days.

(8) An intercom or other means of two-way communication shall be operable between the BNC facility control panel and the reactor control room, and also between the BNC facility control panel and the interior of the BNC facility radiation shielding.

(9) It shall be possible for personnel monitoring a BNC experiment to open the BNC facility access door.

(10) It shall be possible to observe the BNC experiment by means of two independent closed-circuit television (CCTV) cameras.

(a) Each camera shall be operable at the beginning of a BNC experiment. A BNC experiment may be continued at the discretion of the experimenter if one camera fails during a BNC experiment. The BNC experiment shall be immediately stopped if both cameras fail during a BNC experiment.

(b) Emergency lighting and backup power shall be provided for one BNC facility CCTV camera.

(11) Maintenance, repair, and modification of the BNC facility shall be performed under the supervision of a Senior Reactor Operator. All modifications shall be reviewed pursuant to the requirements of 10 CFR 50.59.

(12) Personnel who are not licensed to operate the WSU research reactor may operate the controls for the BNC facility provided compliance is maintained with all technical specifications and that:

(a) instructions have been posted at the BNC facility control panel to ensure that only the appropriate target is in the irradiation facility before turning on the primary beam of radiation to begin an irradiation;

(b) training has been provided, proficiency satisfactorily demonstrated and documented on the design of the facility, the controls, and the use of the controls;

(c) the procedure for conduct of the BNC experiment shall be posted at the control panel of the BNC facility with instructions to notify the reactor operator if the BNC facility operator is unable to turn the BNC neutron beam off with BNC facility controls, or if any abnormal condition occurs; a directive shall be included with this procedure to notify the reactor console operator if an abnormality occurs;

(d) Personnel who are not licensed on the WSU research reactor but who have been trained under this provision may initiate bridge movement provided that verbal permission is requested and received from the reactor console operator immediately prior to such action. Emergency scrams causing a bridge retraction are an exception and may be made without first requesting permission.

(13) Personnel who are not licensed to operate the WSU research reactor shall not take any action that affects the reactivity of the research reactor without approval of a senior reactor operator.

(14) The following characterizations of the BNC neutron beam shall be carried out to prepare the BNC facility for a BNC experiment:

(a) the intensity of the beam shall be measured;

(b) the neutron energy spectrum shall be determined;

(c) the beam diameter and divergence shall be determined;

(d) the dose vs. depth profile in phantoms, evaluated from the surface of the phantom to a depth at least equivalent to the total thickness of the BNC experimental target as irradiated on a central axis.

Thermal and fast neutrons and gamma ray components shall be determined in this characterization.

TS 3.8(1) requires that it be possible to initiate a scram from a control panel located in the BNC facility area, or, as a temporary provision, from one of the control room scrams via direct communication with the reactor operator. This requirement helps ensure that the experimenter has the capability to terminate the irradiation immediately should the need arise and is consistent with the guidance in Section 16.2 of NUREG-1537, Part I. TS 3.8(1) is acceptable to the NRC staff because it provides for a BNC facility scram capability to support the safe operation and use of the BNC facility for BNC experiments.

TS 3.8(2) helps ensure that access to the BNC facility is controlled by means of the access gate located at its entrance. Limiting access to a single gate ensures that there will be no inadvertent entries during operation of the facility beam and is consistent with the guidance in Section 16.2 of NUREG-1537, Part I. TS 3.8(2) is acceptable to the NRC staff because it helps ensure that access to the BNC facility during operation is controlled.

TS 3.8(3) requires safety features and/or interlocks to ensure that exposure levels in the BNC facility are minimal before personnel enter. These features and/or interlocks help keep radiation exposure to facility personnel ALARA and are consistent with the guidance in Section 16.2 of NUREG-1537, Part I. TS 3.8(3) is acceptable to the NRC staff because it provides safety features and/or interlocks for the positioning and movement of the bridge that controls beam delivery in order to ensure that exposure levels in the BNC facility are minimal before facility personnel enter.

TS 3.8(4) and (5) ensure that the bridge that controls beam delivery is provided with a bridge position readout, visible at the BNC facility's local control panel. In the event of a bridge position readout malfunction, the specification allows for the temporary use of an alternate means of verifying bridge position. The bridge position indicator and status lights, or temporary alternate means for bridge position indication, serve to notify personnel of the beam's status and are consistent with guidance in Section 16.2 of NUREG-1537, Part I. TS 3.8(4) and (5) are therefore, acceptable to the NRC staff.

TS 3.8(6) helps ensure that the BNC facility is equipped with a readout display of the reactor log power and the linear power on the BNC facility control console just outside of the shielding to ensure that personnel will have information available on potential radiation levels in the BNC facility before entering. The readout display of the reactor log power and the linear power level is consistent with the guidance in Section 16.2 of NUREG-1537, Part I. TS 3.8(6) is therefore, acceptable to the NRC staff.

TS 3.8(7) requires a radiation monitor with backup power supply, calibration and functional checks, audible radiation alarm and setting, portable radiation survey instruments, and personal dosimetry devices to ensure that personnel will have available a monitor that provides a visual indication of the radiation levels and an audible alarm that indicates both within the facility and at the local control panel. The purpose of this radiation monitor's audible alarm is to alert personnel to the presence of elevated radiation levels. This monitor and its alarm may be disabled once the BNC facility has been searched and secured so that it will not distract attending personnel. The monitor and its alarm are interlocked with the access gate so that they are made functional upon opening that gate, and hence before any possible entry to the BNC facility. The radiation monitor and audible alarm will help to keep radiation exposure to facility personnel ALARA and is consistent with the guidance in Section 16.2 of NUREG-1537, Part I. TS 3.8(7) is therefore, acceptable to the NRC staff.

TS 3.8(8) ensures that the BNC facility is equipped with an operable intercom or other means of two-way communication between the individuals at the BNC facility control panel and the reactor control room, and also between the BNC facility control panel and the interior of the BNC facility shielding in order to provide a means for the prompt exchange of information between the experimenters and the reactor operators. The communication capability provided by the intercoms will support safe operation of the BNC facility, will help to keep radiation exposure to facility personnel ALARA, and is consistent with the guidance in Section 16.2 of NUREG-1537, Part I. TS 3.8(8) is therefore, acceptable to the NRC staff.

TS 3.8(9) helps ensure that the personnel monitoring a BNC experiment have the capability to manually open the BNC facility access gate to permit access to the experimental area in the event of a loss of electrical power. This capability supports safe operation of the BNC facility and is consistent with the guidance in Section 16.2 of NUREG-1537, Part I. TS 3.8(9) is therefore, acceptable to the NRC staff.

TS 3.8(10) provides BNC experimenters with the capability to visually monitor any BNC experiment at all times. The closed-circuit TV (CCTV) cameras enable the experimenters to monitor the target area visually as well as through the use of various instruments. The emergency lighting and the backup power for a TV camera and monitor will permit visual surveillance of the target area in the event of a power failure. The redundant CCTV cameras, system backup power, and emergency lighting provide the capability for continuous visual monitoring of BNC experiments to support safe operation of the BNC facility and are consistent with the guidance in Section 16.2 of NUREG-1537, Part I. TS 3.8(10) is therefore, acceptable to the NRC staff.

TS 3.8(11) helps ensure that any maintenance, repair, and modification of the BNC facility shall be performed under the supervision of a senior reactor operator who is licensed by the NRC to operate the WSU NRCR and that all modifications will be reviewed pursuant to the requirements of 10 CFR 50.59. This specification is consistent with the guidance in Section 16.2 of NUREG-1537, Part I. TS 3.15(11) is therefore, acceptable to the NRC staff.

TS 3.8(12) provides the training and proficiency requirements for operation of the controls of the BNC facility by personnel who are not licensed by the NRC, along with the requirements for posting information at the BNC local control panel. These requirements are consistent with the guidance in Section 16.2 of NUREG-1537, Part I. TS 3.8(12) is acceptable to the NRC staff because it helps ensure that all personnel who are not licensed to operate the WSU NRCR but who are responsible for either the BNC or the beam's design may operate the controls for the BNC facility beam provided that (1) adequate training has been provided, (2) their proficiency has been demonstrated and documented annually, and (3) instructions have been posted at the BNC facility local control panel about procedures to be followed in the event of abnormal conditions.

TS 3.8(13) requires that persons who are not licensed to operate the reactor take no part in activities that affect the reactivity of the reactor without approval of a senior reactor operator. TS 3.8(13) is acceptable to the NRC staff because it limits the activities of unlicensed personnel.

TS 3.8(14) helps ensure that the beam is accurately characterized when it is being used for BNC experiments. TS 3.8(14) is therefore, acceptable to the NRC staff because it requires characterization of the beam prior to performance of a BNC experiment.

5.4 Surveillance Requirements

5.4.0 TS 4.0 General

TS 4.0 controls surveillance requirements and changes to certain system designs as follows:

> 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 control rod drive mechanisms, or the reactor safety systems 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 Safeguards Committee. A system shall not be considered operable until after it has been successfully tested.

TS 4.0 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 conduct of surveillance required to allow operational flexibility that does not impact safety. Because the TS maintains rigor in the design control process, it is acceptable to the NRC staff.

5.4.1 TS 4.1 Reactor Core Parameters

5.4.1.1 TS 4.1.1 Steady-State Operation

See Section 2.5.1 of this SER.

5.4.1.2 TS 4.1.2 Pulse Mode Operation

TS 4.1.2 presents the surveillance requirement to control peak fuel temperature during pulsing as follows:

> The surveillance requirements in this section may be postponed during periods of reactor shutdown. If the surveillance requirement occurs during a period of reactor shutdown the surveillance shall be completed upon resumption of reactor operation.
>
> (1) The maximum safe allowable reactivity insertion shall be calculated annually for an existing core and prior to pulsing a new or modified core arrangement.
>
> (2) The reactor shall be pulsed semiannually to compare fuel temperature measurements and peak power levels with previous pulses of the same reactivity.

TS 4.1.2(1) and (2) require that, if the surveillance is postponed during periods of reactor shutdown, it must be accomplished before pulsing operation is resumed. The specification also requires that the maximum allowed pulse be recalculated annually and that the reactor be pulsed semiannually to compare fuel element temperatures and peak power levels with previous pulses. TS 4.1.2 provides reasonable assurance not only that the maximum temperature during pulsing will not be exceeded but also that operability and performance will be reproducible. TS 4.1.2(1) and (2) are therefore, acceptable to the NRC staff.

5.4.1.3 TS 4.1.3 Shutdown Margin

See Section 2.2 of this SER.

5.4.1.4 TS 4.1.4 Maximum Excess Reactivity

See Section 2.2 of this SER.

5.4.1.5 TS 4.1.5 Core Configuration Limitation

See Section 2.2 of this SER.

5.4.1.6 TS 4.1.6 Fuel Parameters

See Section 2.2.1 of this SER.

5.4.2 TS 4.2 Reactor Control and Safety System

5.4.2.1 TS 4.2.1 Control Rods

See Section 2.2.2 of this SER.

5.4.2.2 TS 4.2.2 Reactor Measuring Channels

TS 4.2.2 presents the surveillance requirements for the reactor measuring channels as follows:

(1) A channel test of each of the required operable measuring channels listed in Table 3.1 for the intended mode of operation shall be performed before each day's operation or before each operation extending more than one day.

(2) A channel check of the fuel rod temperature measuring channel shall be made each time the reactor is operated in the steady state mode by comparing the indicated instrumented fuel rod temperature with previous indicated temperature values for the same core configuration and power level.

TS 4.2.2 provides for a periodic check of the fuel rod temperature measuring channel and tests of the reactor's other measuring channels before each startup. These surveillance requirements and their intervals are consistent with the guidance in ANSI/ANS-15.1-2007 and with the requirements and intervals used at similar research reactors. This specification provides reasonable assurance that component degradation and failure will be detected in a timely manner and that operability and performance will be as required. TS 4.2.2(1) and (2) are therefore, acceptable to the NRC staff (see also Section 2.2.1 of this SER).

5.4.2.3 TS 4.2.3 Reactor Safety System

See Sections 2.2.1, 2.3, and 2.5.1 of this SER.

5.4.2.4 TS 4.2.4 Pool Level Alarm

See Section 2.3 of this SER.

5.4.3 TS 4.3 Primary Coolant Conditions

See Section 2.3 of this SER.

5.4.4 TS 4.4 Ventilation System

TS 4.2.4 presents the surveillance requirements for the ventilation system as follows:

(1) The operation of the pool room ventilation system shall be checked monthly by cycling the system from the "normal" to the "isolate" and "dilution" modes of operation. The positions of the associated dampers, indicator display, and fan operation shall be visually checked to ensure correspondence between the device performance and selected mode of operation.

(2) The pressure drop across the absolute filter in the pool room ventilation system shall be measured semiannually.

(3) The air flow rates in the ventilation system shall be measured biennially.

TS 4.2.4(1) and (2) require that the system to be tested monthly and the absolute filter be checked semiannually. These requirements for system cycling and visual checks help to ensure that the system will operate in accordance with the design features. The requirement to measure the pressure drop across the absolute filter ensures that the filter is not saturated and is capable of removing air particulates. Therefore, these surveillance requirements are acceptable to the NRC staff.

The licensee has added an additional surveillance requirement in TS 4.2.4(3) for biennial ventilation flow rate measurements to verify that the system flow rate satisfies the design requirements established as part of the MHA scenario in Section 9.1 of the revised conversion SAR (Refs. 7 and17).

5.4.5 TS 4.5 Radiation Monitoring System and Effluents

5.4.5.1 TS 4.5.1 Radiation Monitoring System

See Section 3.1.4 of this SER.

5.4.5.2 TS 4.5.2 Effluents

See Sections 3.1.1.1, 3.1.1.2, and 3.1.3 of this SER.

5.4.6 TS 4.6 Limitations on Experiments

TS 4.6 presents the surveillance requirements for the limitations on experiments as follows:

(1) The reactivity worth of moveable experiments shall be shown by measurement, testing, calculation, or comparison to other experiments, to be less than $1.00. This surveillance requirement shall be considered to be satisfied for subsequent movable experiments that exhibit the same characteristics as a previously analyzed moveable experiment.

(2) The reactivity worth of a secured experiment shall be shown by measurement, testing, calculation, or comparison to other experiments, to be less than $2.00. This surveillance requirement shall be considered to be satisfied for subsequent secured experiments when a measurement of an initial secured experiment is applied to subsequent secured experiments that are the same as the initially analyzed secured experiment.

(3) The sum of absolute values of all individual experiments shall be shown to be less than $5.00.

(4) The following surveillance requirements apply to use of explosive materials in experiments:

 (a) The quantity of explosive materials (if any) used in an experiment shall be documented and shown to be less than 25 mg.

 (b) Testing of explosive material encapsulation shall be documented and shown to be in accordance with Section 3.6 (4).

(5) A safety analysis shall document conformance to the requirements of Section 3.6(5).

(6) A safety analysis shall document conformance to the requirements of Section 3.6(6).

(7) A safety analysis shall document that the total radioactive inventory of iodine isotopes 131 through 135 in a fueled experiment is less than 1.5 Ci.

(8) The results of an inspection and any corrective action taken following a sample failure that releases material that could damage reactor fuel or the reactor structure shall be reviewed by the facility Director and the Reactor Safeguards Committee and shall be determined to be satisfactory before operation of the reactor is resumed.

(9) Minor modifications to a reviewed and approved experiment may be made at the discretion the Reactor Supervisor, provided that the hazards associated with the modifications have been reviewed and a determination has been made and documented that the modifications do not create a significantly different, a new, or a greater hazard than the

originally approved experiment.

TS 4.6 requires that, before installation of an experiment in the reactor, a safety analysis be performed to show that the requirements of TS 3.6, "Limitations on Experiments," are met. Minor modifications to an experiment may be made by the Reactor Supervisor, since compliance with TS 4.6 helps ensure that the requirements of TS 3.6 for experiments are met, the NRC staff finds TS 4.6 acceptable.

5.4.7 TS 4.7 Sealed Sources in the Reactor Pool

See Section 3.1.1.3 of this SER.

5.4.8 TS 4.8 Boron Neutron Capture Facility

TS 4.8 provides the surveillance requirements for the BNC facility as follows:

(1) Operability of the BNC facility reactor scram mechanism shall be checked and documented before operation of the experiment commences each day that the BNC facility is used to carry out a BNC experiment. A scram test shall be carried out from full power before BNC facility use if more than 6 months have elapsed during which the BNC facility was not used for BNC experiments, or after a component or system modification which could affect the scram system.

(2) Single door access to the facility shall be confirmed before each day of operation of the BNC facility.

(3) The operability of each system listed in Specifications 3.8(3)(a) through 3.8(3)(h) shall be checked and documented before operation of the experiment commences each day that the BNC facility is used to carry out a BNC experiment.

(4) Operability of the reactor bridge position sensor shall be checked and documented before operation of the experiment commences each day that the BNC facility is used to carry out a BNC experiment.

(5) Operability of an alternate means of monitoring reactor bridge position in the event of a failure of the bridge position readout, as permitted in Technical Specification 3.8(5), shall be checked and documented before operation of the experiment commences each day that the BNC facility is used to carry out a BNC experiment.

(6) Operability of the display of the reactor linear power and log power display channels shall be checked and documented before operation of the experiment commences each day that the BNC facility is used to carry out a BNC experiment.

(7) Operability of the radiation monitor, radiation monitor alarm, portable radiation detection instruments and personal dosimeters shall be checked and documented before operation of the experiment commences each day that the BNC facility is used to carry out a BNC experiment. The

radiation monitor shall be calibrated quarterly for every calendar quarter that the BNC facility is in use to perform a BNC experiment.

(8) Operability of the intercom system or two way communication system shall be checked and documented before operation of the experiment commences each day that the BNC facility is used to carry out a BNC experiment.

(9) Operability of the BNC facility access door shall be checked and documented before operation of the experiment commences each day that the BNC facility is used to carry out a BNC experiment.

(10) Operability of the closed circuit television monitoring system and emergency lighting and backup power system shall be checked and documented before operation of the experiment commences each day that the BNC facility is used to carry out a BNC experiment.

(11) Maintenance, repair and modification of the BNC facility, including identification of the supervising Senior Reactor Operator, shall be documented.

(12) Before each day's operation it shall be verified that:

   (a) the instructions have been posted;

   (b) training of personnel who operate the controls of the BNC facility has been documented before being permitted to operate the BNC facility controls.

   (c) the procedure for conduct of the BNC experiment has been posted.

   (d) instruction has been given to BNC facility personnel that the reactor bridge shall be moved only with permission of a Reactor Operator or Senior Reactor Operator and under the direct supervision of a Senior Reactor Operator.

(13) It shall be documented before use of the BNC facility commences that the approval of a senior reactor operator is required before performing any action that affects the reactivity of the research reactor.

(14) Characterization of the BNC neutron beam and beam monitors shall be carried out before initiation of use of the neutron beam for BNC experiments. The following surveillances apply:

   (a) beam intensity, diameter and divergence shall be measured according to the following schedule:

      (i) before initial use in a BNC experiment, weekly for any week that the beam will be used and semiannually for any six month interval that the BNC neutron beam will be used for BNC experiments;

(ii) prior to performance of a BNC experiment if a beam component has been modified or replaced in the interim since a prior beam characterization was carried out;

(b) beam monitors shall be calibrated at least once every two years for any two year interval that the neutron beam is used for BNC experiments. An initial calibration shall be made prior to carrying out a BNC experiment if more than two years have elapsed during which the BNC facility was not used to perform BNC experiments. The beam monitors shall be calibrated by a means that is traceable to the National Institute of Standards and Technology, and shall measure dose or dose rate.

The licensee states that TS 4.8(1) through (10) and (12) create mechanisms for ensuring that the BNC facility and its beam will perform as originally designed by providing operability checks before each day's operation of the BNC facility. The surveillance requirement checks operability of the reactor and BNC facility scram functions, safety interlocks and inhibits, instrumentation and control channels, radiation monitoring instrumentation and alarms, CCTV systems, and emergency lighting as listed in TS 3.8. TS 4.8(11) requires documentation for maintenance, repair, and modification of the BNC facility, including identification of the senior reactor operator supervising the maintenance, repair, and modification. TS 4.8(13) requires documentation that a senior reactor operator has approved all operations that affect the reactivity of the reactor. TS 4.8(14) provides for periodic characterization of the BNC beam.

Performance of these surveillance tests will support safe operation of the BNC facility and is consistent with the guidance of NUREG-1537, Part 1, Chapter 16 Section 2, the guidance of ANSI/ANS-15.1-2007, and the intervals used at other research reactors for comparable activities. These surveillance frequencies will help ensure the performance and operability of the BNC facility. The specified intervals provide reasonable assurance that radiation monitor component failure and degradation will be detected in a timely manner and the specified beam calibration frequencies are adequate to prevent significant drift in fluence. TS 4.8 is therefore, acceptable to the NRC staff.

5.5 Design Features

5.5.1 TS 5.1 Site and Facility Description

TS 5.1 presents the site and facility description as follows:

(1) The site is that area bound by the perimeter that encloses the Nuclear Radiation Center building, (also known as the Dodgen Research Facility), the fenced area immediately outside the east pool room loading dock door and the fenced area immediately outside the beam room west loading dock door.

(2) The Washington State University research reactor shall be located in the licensed area of the Dodgen Research Facility.

(3) The facility shall be the following:

(a) the room in which the WSU research reactor is located, also known as Room 201, the reactor control room which is within Room 201, the pump room, primary coolant water purification room, primary coolant and makeup water valve manifold room;

(b) the research reactor beam room, also known as Room 2.

(4) The facility shall be a restricted area.

The NRC staff finds that TS 5.1 is as described in the license renewal SAR (Ref. 1) and the WSU NRCR emergency plan. On this basis, the NRC staff concludes that TS 5.1 is acceptable.

5.5.2 TS 5.2 Reactor Fuel

See Section 2.2.1 of this SER.

5.5.3 TS 5.3 Reactor Core

See Section 2.2 of this SER.

5.5.4 TS 5.4 Control Rods

See Section 2.2.2 of this SER.

5.5.5 TS 5.5 Fuel Storage

TS 5.5 presents the design criteria for fuel storage as follows:

(1) All fuel rods and fueled devices shall be stored in a geometrical array where the k_{eff} is less than 0.8 for all conditions of moderation and reflection.

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

TS 5.5(1) limits the k_{eff} value to 0.8, which is lower than the commonly used k_{eff} value 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 therefore acceptable to the NRC staff.

TS 5.5(2) is the basic design requirement to ensure adequate cooling by natural convection cooling, either by water or air, of stored irradiated fuel rods and fueled devices. Because criticality and fuel temperature are controlled to acceptable limits, TS 5.5 is acceptable to the NRC staff.

5.5.6 TS 5.6 Radiation Monitoring System

TS 5.6 presents the design criteria for the radiation monitoring system as follows:

(1) The area radiation monitors shall be sensitive to gamma radiation, shall monitor radiation fields in key locations, and shall alarm and readout at the reactor control console.

(2) The Continuous Air Monitor shall:

(a) be capable of particulate collection, and detection of beta and gamma radiation;

(b) monitor particulate radioactivity in the pool room air, alarm and readout at the reactor control console;

(c) be capable of causing the building ventilation system to switch from the normal mode into the dilution mode upon initiation of a high continuous air monitor alarm signal when the reactor is operating.

(3) The exhaust gas monitor shall be capable of detecting gamma radiation, and shall monitor ^{41}Ar content in ventilation system exhaust air, and shall alarm and readout at the reactor control console.

The WSU NRCR facility places constraints on radioactive materials that may be released into the reactor area and the environment. TS 5.6 helps ensure that operating personnel are aware of radioactivity levels inside the facility and of radioactivity levels being released to the environment through the stack. This specification also requires that the ventilation be switched automatically from the normal (automatic) mode into the dilution mode upon initiation of a high continuous air monitor alarm signal when the reactor is operating. The licensee states that analysis has shown that, when ventilation is operating in the dilution mode, the release of radioactivity to the environment is minimized. The NRC staff finds TS 5.6 acceptable because this design feature helps to ensure that the radioactivity levels inside the facility are monitored and alarmed at the control console and radioactivity being released through the stack is minimized, monitored, and alarmed at the console.

5.5.7 TS 5.7 Reactor Building and Ventilation System

TS 5.7 presents the design criteria for the reactor building and ventilation system as follows:

(1) The reactor shall be housed in a facility designed to restrict leakage. The minimum free volume in the facility shall at least 10^9 cm^3.

(2) The reactor building shall be equipped with a ventilation system designed to filter and exhaust air or other gases from the reactor building and release them from a stack at a height of 46 ± 2 feet from the ground level in the front of the Dodgen Research Facility building.

(3) A set of controls for the ventilation system shall be located outside the reactor pool room and control room areas. The controls shall be capable of changing the ventilation system mode of operation into the dilute or isolate mode.

(4) The reactor pool room ventilation system shall have a dilution mode of operation with the following characteristics:

(a) air from the reactor pool room shall be mixed and diluted with outside air before being discharged from the facility when the ventilation system is operated in the dilution mode;

(b) the exhaust air from the reactor pool room shall pass through a filter before being discharged from the facility when the ventilation system is operated in the dilution mode.

TS 5.7(1) provides the requirement to house the reactor in an enclosed building with a minimum free volume that restricts leakage.

TS 5.7(2) provides the requirement to have a ventilation system with a filtered and controlled air pathway release point. As discussed in Chapter 11 of the license renewal SAR, the height of the exhaust stack helps to ensure dispersion and dilution of effluents released from the stack before they reach the ground.

TS 5.7(3) helps ensure that the reactor operator can control the system from areas outside the control room and pool room area if the control room and pool room are inaccessible.

TS 5.7(4) provides the requirement for a dilution mode operation that allows the system to mix air from the reactor bay area with outside air, thereby diluting the potential release of radioactive material and ensuring that any release satisfies the limits specified in 10 CFR Part 20, Appendix B.

TS 5.7(1) through (4) specify design requirements for the ventilation system. The NRC staff finds that these design aspects are acceptable and therefore, concludes that these TS are acceptable.

5.5.8 TS 5.8 Reactor Pool Water System

See Section 2.3 of this SER.

5.6 TS 6.0 Administrative Control

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

The primary guidance for the development of administrative control TS for research reactors is currently ANSI/ANS-15.1-2007. The licensee’s TS were based on the 1990 and 2007 (Ref. 38) versions of the ANSI/ANS 15.1 standard. However, for the revision of the TS for the license renewal, the licensee used the guidance of ANSI/ANS 15.1-2007 (Ref. 39). The NRC staff used the 2007 version of ANSI/ANS-15.1 in its review of the licensee’s administrative controls. In some cases, the wording of the WSU NRCR is not identical to that of ANSI/ANS-15.1-2007 and NUREG-1537. However, the NRC staff reviewed these cases and determined that the licensee’s TS met the intent of the guidance and are acceptable.

5.6.1 TS 6.1 Responsibility and Organization

TS 6.1 presents the responsibilities and organization of the individuals identified in the organizational chart as follows:

(1) The Washington State University research reactor shall be operated by the Nuclear Radiation Center of Washington State University. The organization of the research reactor facility management and operation shall be as shown in Figure 6.1. The responsibilities and authority of the Level 2, Level 3, and Level 4 operating staff shall be defined in writing in these Technical Specifications.

(2) The following organizational levels and responsibilities shall exist:

(a) Vice President for Research (Level 1): The Vice President for Research is the head of the WSU Office of Research.

(b) Director of the Nuclear Radiation Center (Level 2): The Director of the Nuclear Radiation Center shall report to the Vice President for Research. The Director is responsible for ensuring that regulatory requirements and implementation are in accordance with requirements of the U.S. Nuclear Regulatory Commission, the Code of Federal Regulations, the State of Washington, and Washington State University regulations and the requirements of the WSU Reactor Safeguards Committee.

(c) Reactor Supervisor (Level 3):

(i) The Reactor Supervisor shall report to the Director of the Nuclear Radiation Center and is responsible for guidance, oversight, and technical support of reactor operations.

(ii) The Reactor Supervisor shall report to the Director of the Nuclear Radiation Center and to the Reactor Safeguards Committee in matters of radiation protection.

(d) Reactor Operating Staff (Level 4): The reactor operating staff shall report to the Reactor Supervisor. Reactor operating staff shall include one or more licensed Senior Reactor Operator, Reactor Operator or Reactor Operator trainee.

(e) Radiation Protection

(i) Radiation protection activities shall be carried out by Level 3 or Level 4, with supervisory function performed by the Level 3, Reactor Supervisor.

(ii) The Reactor Safeguards Committee shall perform the review and audit function over the radiation protection activities within the facility.

(iii) The Director of the Radiation Safety Office, as an ex-officio member of the Reactor Safeguards Committee, shall provide communication regarding radiation safety to the Director of the Nuclear Radiation Center.

(iv) The Director of the Radiation Safety Office shall have oversight, through the Reactor Safeguards Committee, of activities utilizing radioactive material.

(3) Responsibilities of one level may be assumed by higher levels or by alternates designated by a higher level, conditional upon meeting all requirements for the position.

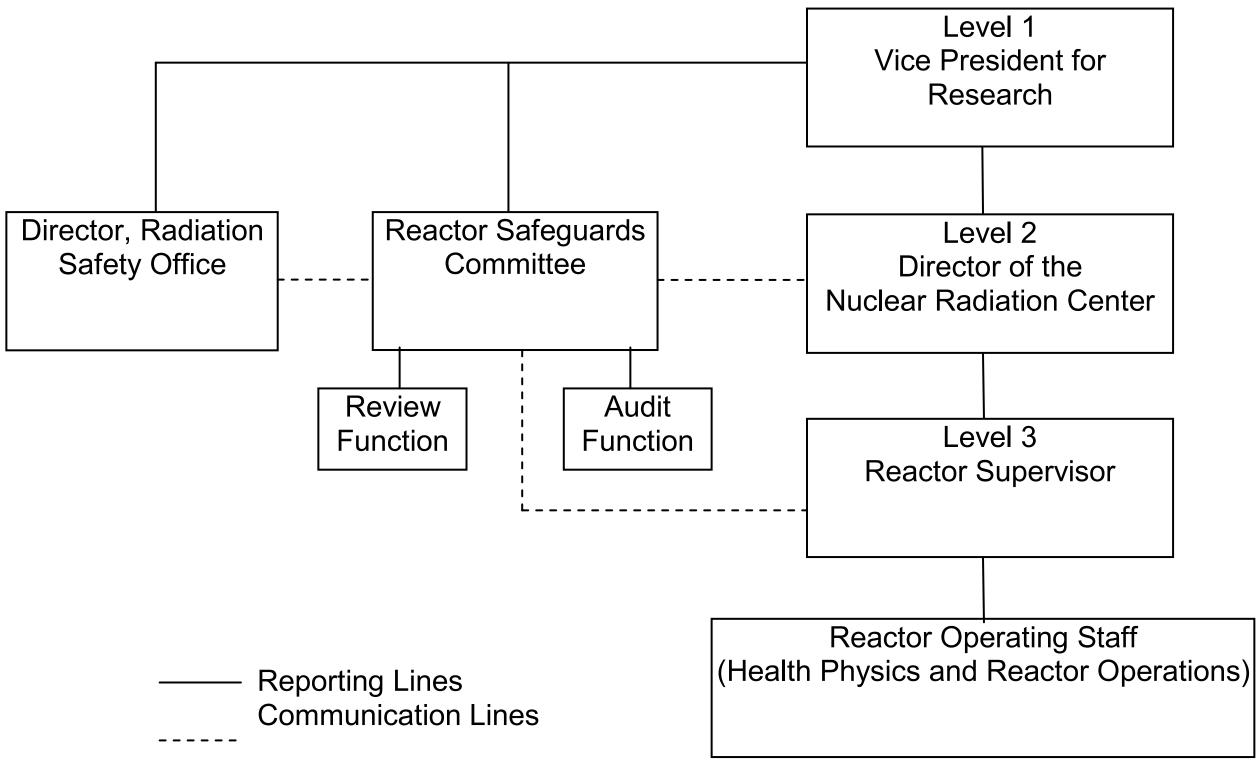

Figure 6.1 Facility organization

The licensee has modified TS 6.1 to include both the responsibilities and organization of individuals associated with the facility. TS 6.1 has been modified to clearly show the organizational arrangement for the radiation protection function. The licensee also modified Figure 6.1 to correct typographical errors and to identify the reporting and communication relations between the organizational units. TS 6.1 describes the organization and responsibilities of individuals in direct control of the facility and is consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007 and is therefore, acceptable to the NRC staff.

5.6.2 TS 6.2 Staffing

5.6.2.1 TS 6.2.1 Minimum Staffing Levels

TS 6.2.1 presents the minimum staffing levels as follows:

(1) When the reactor is not secured, the minimum staffing level shall consist of:

(a) a licensed Reactor Operator or Senior Reactor Operator in the control room;

(b) a second designated person present at the facility complex able to carry out written instructions;

(c) a designated Senior Reactor Operator who shall be readily available in the Dodgen Research Facility or on call.

(2) A Senior Reactor Operator who is “on call” shall be defined as an individual who:

(a) has been specifically designated and this designation is known to the Reactor Operator on duty;

(b) keeps the Reactor Operator on duty informed of where he/she can be rapidly contacted and the contact telephone number;

(c) is capable of getting to the reactor facility in less than 30 minutes and shall remain within a 15 mile radius of the facility;

(3) It is not necessary to have a Senior Reactor Operator on call if the Reactor Operator in the control room is a Senior Reactor Operator. If the Reactor Operator in the control room is a Senior Reactor Operator a second person shall be present in the facility as described in Section 6.2(1)(b).

5.6.2.2 TS 6.2.2 Contact Information

TS 6.2.2 presents contact information as follows:

> (1) A list of personnel including name and telephone number shall be readily available in the control room for use by the Reactor Operator. The list shall include:
>
> > (a) facility Director;
> >
> > (b) Reactor Supervisor;
> >
> > (c) all licensed Reactor Operators and Senior Reactor Operators.

The licensee added TS 6.2.2 to the TS to incorporate contact information for the facility. TS 6.2.2 is consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007 and is therefore, acceptable to the NRC staff.

On the basis of its review as discussed above, the NRC staff concludes that TS 6.2 is acceptable.

5.6.2.3 TS 6.2.3 Events Requiring the Direction of a Senior Reactor Operator

TS 6.2.3 presents the events requiring the direction of a senior reactor operator as follows:

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

> (1) Initial startup and approach to power;
>
> (2) All fuel movement or relocation;
>
> (3) All control rod relocations within the core region;
>
> (4) Relocation of any in-core experiments or irradiation facilities with a reactivity worth greater than $1.00;
>
> (5) Recovery from unplanned or unscheduled shutdown; or
>
> (6) Recovery from unplanned or unscheduled significant power reduction.

The licensee added TS 6.2.3 to the TS to incorporate the events which require the direction of a senior reactor operator. TS 6.2.3 is consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007 and is therefore, acceptable to the NRC staff.

5.6.3 TS 6.3 Selection and Training of Personnel

TS 6.3 presents the requirements for the selection and training of personnel as follows:

> The selection, training and requalification of each member of operations personnel shall meet or exceed the requirements of ANSI/ANS.4 – 2007,

“Standard for the Selection and Training of Personnel for Research Reactors,” for comparable positions.

TS 6.3 establishes the criteria for the training and requalification program for operations personnel. The licensee uses ANSI/ANS-15.4-2007, “Selection and Training of Personnel for Research Reactors” (Ref. 49), as guidance for the selection and training of personnel. The NRC staff finds that TS 6.3 is consistent with the guidance provided in NUREG-1537, ANSI/ANS 15.1-2007 and ANSI/ANS-15.4-2007. On this basis, the NRC staff concludes that TS 6.3 is acceptable.

5.6.4 TS 6.4 Reactor Safeguards Committee

The function, composition and qualification, operation, reviews, audits, and records and experiment review and approval of the RSC are described in Section 12.2 of the license renewal SAR (Ref. 1).

TS 6.4 describes these attributes of the RSC as follows:

5.6.4.1 TS 6.4.1 Function

TS 6.4.1 presents the functions of the RSC as follows:

> The Reactor Safeguards Committee shall function to provide an independent review and audit of the Nuclear Radiation Center activities including:
>
> (1) reactor operations;
>
> (2) radiological safety;
>
> (3) general safety;
>
> (4) testing and experiments;
>
> (5) licensing and reports;
>
> (6) quality assurance.

5.6.4.2 TS 6.4.2 Composition and Qualification

TS 6.4.2 presents the requirements for the composition and qualification of the RSC as follows:

> (1) The Reactor Safeguards Committee shall be composed of at least five members knowledgeable in fields that relate to nuclear reactor safety.
>
> (2) The members of the Committee shall include:
>
> > (a) one Senior Reactor Operator who may be the Director of the Nuclear Radiation Center. The presence of Nuclear Radiation Center staff members shall not be counted to constitute a quorum. Nuclear Radiation Center staff members shall not be voting members of the Committee.

(b) WSU faculty and staff members designated to serve on the Committee in accordance with the procedures specified by the WSU committee manual.

(3) The Director of the WSU Radiation Safety Office shall be an ex-officio member of the Committee.

(4) The Reactor Safeguards Committee is a WSU Presidential Committee which performs reviews and audits of the WSU Nuclear Radiation Center. The Reactor Safeguards Committee reports to the WSU Vice President for Research.

5.6.4.3 TS 6.4.3 Reactor Safeguards Committee Operation

TS 6.4.3 presents the requirements for the operation of the RSC as follows:

The Reactor Safeguards Committee shall operate in accordance with a written charter, including provisions for:

(1) semiannual meetings of the full committee;

(2) voting rules;

(3) quorums: the committee chair or a designate and two voting members;

(4) method of submission and content of presentations to the committee;

(5) use of subcommittees;

(6) review, approval and dissemination of minutes.

5.6.4.4 TS 6.4.4 Reviews

TS 6.4.4 presents the requirements for reviews by the RSC as follows:

The responsibilities of the Reactor Safeguards Committee or designated subcommittee shall include the following:

(1) review and approval of new experiments utilizing the research reactor;

(2) review and approval of proposed changes to the following:

(i) the operating license (R-76) by amendment;

(ii) Standard Operating Procedures;

(iii) Technical Specifications.

(3) review of the operation and operational records of the Nuclear Radiation Center;

(4) review of operating abnormalities or deviations from normal and expected performance of equipment with safety significance;

(5) review in accordance with 10 CFR 50.59 whether proposed changes in equipment, systems, tests, experiments or Standard Operating Procedures would be allowed without prior authorization by the U.S. Nuclear Regulatory Commission.

(6) review of reportable occurrences and the reports filed with the U.S. Nuclear Regulatory Commission for reportable occurrences;

(7) biennial review and approval of all standard operating procedures and changes to the standard operating procedures;

(8) biennial review of the emergency plan and the security plan;

(9) annual review of the radiation protection program;

(10) review audit reports.

5.6.4.5 TS 6.4.5 Audits

TS 6.4.5 presents the requirements for audits by the RSC as follows:

(1) The RSC or a subcommittee shall audit reactor operations semiannually. The semiannual audit shall include at least the following:

    (a) review of the reactor operating records;

    (b) inspection of the reactor operating areas;

    (c) review of reportable occurrences;

    (d) radiation exposures within and outside the facility;

    (e) operations for conformance to the Technical Specifications and license conditions.

(2) The RSC or a subcommittee shall audit the following at biennial intervals:

    (a) emergency plan and implementing procedures;

    (b) retraining and requalification program;

    (c) security plan.

5.6.4.6 TS 6.4.6 Records

TS 6.4.6 presents the requirements for recordkeeping by the RSC as follows:

The activities of the RSC shall be documented by the secretary of the Committee and distributed as follows:

(1) A written report of all audits performed under Section 6.4.5 shall be prepared and forwarded to Level 1 and Level 2 management within 3 months after the audit has been complete.

(2) A written report of all reviews performed under Section 6.4.4 shall be prepared and forwarded to the Level 1 and Level 2 management within 30 days following the completion of the review.

(3) The secretary of the Reactor Safeguards Committee shall maintain a file of the minutes of all meetings.

5.6.4.7 TS 6.4.7 Experiment Review and Approval

TS 6.4.7 presents the requirements for experiment review and approval as follows:

Approved experiments shall be carried out in accordance with established and approved procedures. The following provisions shall be stated in a Standard Operating Procedure for review and approval of experiments:

(1) All new experiments or classes of experiments shall be:

a. installed in the reactor or in its irradiation facilities only after a safety analysis has been performed;

and

b. reviewed and approved by at least 2 Senior Reactor Operators, including written approval by Level 2 or Level 3 management for compliance with the Technical Specifications;

and

c. reviewed and approvad by the Reactor Safeguards Committee.

(2) Substantive Changes to previously approved experiments shall be made only after review by the Reactor Safeguards Committee and approved in writing by Level 2 or designated alternates. Minor changes that do not significantly alter the experiment may be approved by Level 3 or higher.

(3) An experiment shall not be installed in the reactor or its irradiation facilities until after a safety analysis has been performed and reviewed for compliance with Section 3.6 by the Reactor Safeguards Committee in accordance with Section 6.4.7 of these Technical Specifications.

TS 6.4.1(1) through (7) describe WSU RSC activities. The facility organizational structure (as shown in Figure 6.1 of TS 6.1) shows that the RSC reports to the Vice President for Research with communication to the Director and Reactor Supervisor. The RSC's function to perform an independent audit of the activities listed in TS 6.4.5 is consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007 and, therefore, is acceptable to the NRC staff.

TS 6.4.2 describes the composition of the RSC and requires the RSC to include operating staff and experts who are not directly involved with the operation of the WSU NRCR facility. TS 6.4.2 requires that the director of the WSU Radiation Safety Office be a member of the RSC. This is consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007 and, therefore, is acceptable to the NRC staff.

TS 6.4.3 describes the operation of the RSC, which is responsible for an independent audit of the WSU NRCR activities and conducts its review and audit functions in accordance with a written charter. The charter includes provisions for meeting frequency, voting rules, quorums, method of submission and content of presentations to the RSC, use of subcommittees, and minutes. NUREG-1537 and ANSI/ANS-15.1-2007 specify that the purpose of the review committee is to provide independent oversight and that the operating staff should not constitute the majority of a quorum. The RSC charter establishes a quorum of three members, which guarantees that operations personnel will not be a majority. The rules of the RSC, as outlined in TS 6.4.3 and the RSC charter, are consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007 and, therefore, are acceptable to the NRC staff.

TS 6.4.4(1) requires that the RSC must review and approve all experiments. TS 6.4.4(1) together with TS 6.4.1(4) contain the requirements for the review and approval of experiments. TS 6.4.4(1) is consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007 and, therefore, is acceptable to the NRC staff.

TS 6.4.4(2) through (8) describe the items specified for review by the RSC, which are consistent with the guidance provided in NUREG-1537 and ANSI/ANS 15.1-2007 and, therefore, are acceptable to the NRC staff.

TS 6.4.4(9), requiring the RSC to perform an annual review of the radiation protection program, is consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007 and, therefore, is acceptable to the NRC staff.

TS 6.4.4(10), requiring the RSC to review audit reports, is consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007 and, therefore, is acceptable to the NRC staff.

TS 6.4.5 requires the RSC to perform audits of reactor operations on a semiannual basis and audits of the emergency plan, training and requalification program, and security plan biennially. TS 6.4.5 is consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007 and, therefore, is acceptable to the NRC staff.

TS 6.4.6, containing the requirements for documentation of RSC activities, is consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007 and, therefore, is acceptable to the NRC staff.

TS 6.4.7 presenting the requirements for experiment review and approval is consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007 and, therefore, is acceptable to the NRC staff.

5.6.5 TS 6.5 Radiation Safety

TS 6.5 states the following requirements for radiation safety:

(1) The Reactor Supervisor (Level 3) shall have responsibility for implementing the radiation protection program using guidelines of ANSI/ANS-15.11-1993 (R2004). The Reactor Supervisor shall report to Level 2 management and shall communicate with the Reactor Safeguards Committee on matters of radiation safety.

(2) Radiation Protection

(a) The Reactor Safeguards Committee shall have oversight responsibility as defined in Section 6.1(2)(e)(ii) and 6.1(2)(e)(iv).

(b) The Reactor Operating Staff (Level 4) shall conduct radiation protection procedures in licensed areas, and shall report to the Reactor Supervisor (Level 3).

TS 6.5 is acceptable to the NRC staff because it states the responsibilities and requirements for the radiation protection, including the requirement that the program use the guidance of ANSI/ANS-15.11.

5.6.6 TS 6.6 Action To Be Taken if a Safety Limit Is Exceeded

TS 6.6 states the following required actions in the case of a safety limit violation:

The following actions shall be taken if a safety limit is exceeded:

(1) The safety limit violation shall be reported within 24 hours by telephone to the U.S. Nuclear Regulatory Commission Operations Center.

(2) The reactor shall be shut down and reactor operation shall not be resumed until authorized by the U.S. Nuclear Regulatory Commission.

(3) The safety limit violation shall be promptly reported to Level 1 management or designated alternates, to Level 2 management or designated alternates, to Level 3 management and to the Chair of the Reactor Safeguards Committee.

(4) A safety limit violation report shall be prepared. The report shall describe the following:

(a) applicable circumstances leading to the violation, the cause and contributing factors;

(b) impact of the violation upon reactor facility components, systems, or structures and on the health and safety of personnel and the public;

(c) corrective action to be taken to prevent recurrence.

(5) The report shall be submitted to the Reactor Safeguards Committee for review.

(6) A report shall be submitted in writing, within 10 days, to the U.S. Nuclear Regulatory Commission Document Control Desk.

TS 6.6(1) requires that the violation be reported to the NRC within 24 hours.

TS 6.6(2) requires the facility to shut down when the safety limit is exceeded and not resume operations until authorized by the NRC.

TS 6.6(3) requires that a report of the violation be made to the Vice President for Research, the Director of the Nuclear Radiation Center, and the Chair of the RSC.

TS 6.6(4) specifies the content of the report, including the evaluations and corrective actions to be taken.

TS 6.6.5 requires that the report be reviewed by the RSC.

TS 6.6(6) expands on the requirements for reporting to the NRC.

The NRC staff finds that TS 6.6 meets the requirements of 10 CFR 50.36(c)(1) for actions to be taken if a safety limit is exceeded. The NRC staff also finds that TS 6.6 is consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. On this basis, the NRC staff concludes that TS 6.6 is acceptable.

5.6.7 TS 6.7 Required Actions for Reportable Occurrences other than Safety Limit Violation

TS 6.7 states the actions to be taken in the event of a reportable occurrence other than a safety limit violation as follows:

> The following actions shall be taken as required by regulations or for a Reportable Occurrence, as defined in Section 1 for events that are reportable to the U.S. Nuclear Regulatory Commission within 24 hours NRC. Reports are to be made to the U.S. Nuclear Regulatory Commission Operations Center for:
>
> (1) Reactor conditions shall be returned to normal, or the reactor shall be shut down. If it is necessary to shut down the reactor to correct the occurrence, operation of the reactor shall not be resumed unless authorized by Level 2 or designated alternates and the Chair of the Reactor Safeguards Committee.
>
> (2) The occurrence shall be reported to Level 1 management, Level 2 management or designated alternates.
>
> (3) The occurrence shall be reviewed by the Reactor Safeguards Committee at its next scheduled meeting.

(4) An immediate report of the occurrence shall be made to the Chair of the WSU Reactor Safeguards Committee.

(5) A report shall be prepared that includes an analysis of the causes and extent of possible damage, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence. The report shall be submitted to the Reactor Safeguards Committee for review.

(6) A report shall be submitted in writing to the U.S. Nuclear Regulatory Commission Document Control Desk within 10 days.

The NRC staff finds that TS 6.7 is consistent with the guidance provided in NUREG-1537 and in ANSI/ANS-15.1-2007 and, therefore, is acceptable.

5.6.8 TS 6.8 Standard Operating Procedures

In Section 12.3 of the license renewal SAR (Ref. 1), the licensee discussed the requirement for written procedures that must be prepared and approved before use.

TS 6.8 specifies the following requirements for standard operating procedures:

(1) Written procedures shall be prepared, reviewed, and approved prior to initiating any of the activities listed in this section. The procedures shall be reviewed by at least 2 Senior Reactor Operators. The procedures shall be reviewed and approved by the Reactor Safeguards Committee after approval by Level 2 management or designated alternates, and such reviews and approvals shall be documented.

(2) Written operating procedures shall be adequate to ensure the safe operation of the reactor, but shall not preclude the use of independent judgment and action if required to protect the health and safety of the public. Operating procedures shall be in effect for the following:

   (a) startup, operation, and shutdown of the reactor;

   (b) fuel loading, unloading, and movement within the reactor;

   (c) maintenance of major components of systems which could influence reactor safety;

   (d) surveillance checks, calibrations, and inspections required by Technical Specifications or those that could have an influence on reactor safety;

   (e) personnel radiation protection, consistent with applicable regulations or guidelines. The procedures shall include management commitment and programs to maintain exposures and releases as low as reasonably achievable in accordance with guidelines of ANSI/ANS-15.11-1993 (R2004).

(f) performing irradiations and experiments using the reactor;

(g) implementation of emergency and security plans;

(h) use, receipt, and transfer of radioactive material;

(i) control rod removal and replacement;

(j) reactor power calibration;

(k) performing maintenance and/or calibration on the reactor and associated equipment.

(3) Substantive changes to the previous procedures shall be made effective only after documented review by the review group of the Reactor Safeguards Committee and approval by Level 2 or designated alternates or if necessary, by a review under the regulations established by 10 CFR 50.59. Modifications to the procedures that do not change their original intent may be made by Level 3 or higher, but the modifications shall be approved by Level 2 or a designated alternate. Minor changes, such as corrections of typographical errors, editing for clarity or formatting that do not change the execution of the procedure may be made by any Senior Reactor Operator but the modifications shall be approved by Level 2 or a designated alternate. Temporary deviations from the original procedures may be made by the responsible Senior Reactor Operator or higher individual present to deal with special or unusual circumstances or conditions. Such deviations shall be documented and reported within 24 hours to the Level 2 or designated alternate.

TS 6.8 specifies the items that must be covered by standard operating procedures. TS 6.8 also specifies that new procedures and substantive changes to procedures must be approved by the Director and the RSC. Modifications to procedures that do not change the original intent may be made by the Reactor Supervisor or Director. Minor changes such as typographical errors may be made by the Senior Reactor Operator.

The NRC staff finds that TS 6.8(1) and (2) are consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. The NRC staff finds that the required items, management of procedure control, and proper review of procedures provide reasonable assurance of the safe operation of the reactor and proper administration of the facility. On this basis, TS 6.8 is acceptable to the NRC staff.

5.6.9 TS 6.9 Facility Operating Records

TS 6.9 specifies the following requirements for records and record retention:

> Records may be in the form of logs, data sheets, or other suitable forms. The required information may be contained in single or multiple records, or a combination of single or multiple records. In addition to the requirements of applicable regulations, records and logs shall be prepared for at least the following items and retained for the periods of time indicated in sections 6.9.1, 6.9.2, and 6.9.3:

5.6.9.1 TS 6.9.1 5-Year Record Retention

TS 6.9.1 requires a 5-year retention for records as follows:

> Records of the following shall be kept for at least five years:
>
> (1) normal reactor operation, including supporting documents such as pre-startup checklists and reactor operating log sheets;
>
> (2) principal maintenance operations;
>
> (3) reportable occurrences;
>
> (4) surveillance activities required by the Technical Specifications;
>
> (5) experiments performed with the reactor;
>
> (6) approved changes in operating procedures;
>
> (7) facility radiation and contamination surveys;
>
> (8) Reactor Safeguards Committee meeting records and audit reports.

5.6.9.2 TS 6.9.2 Life-of-the-Facility Records Retention

TS 6.9.2 requires life-of-the-facility retention for records as follows:

> Records of the following components or items shall be kept for the life of the facility:
>
> (1) gaseous and liquid radioactive effluents released to the environs;
>
> (2) off-site environmental monitoring surveys required by the Technical Specifications;
>
> (3) radiation exposures for all personnel monitored;
>
> (4) updated, corrected and as-built drawings of the reactor facility;
>
> (5) fuel inventories, receipts, and shipments;
>
> (6) reviews and reports of violations of Safety Limits;
>
> (7) reviews and reports of violations of a Limiting Safety Systems Setting;
>
> (8) reviews and reports of violations of a Limiting Condition of Operation.

5.6.9.3 TS 6.9.3 Training Records

TS 6.9.3 requires the retention of training records as follows:

Record of training, retraining and requalification of licensed personnel shall be maintained at all times the individual is employed or until the operator license is renewed.

The NRC staff finds that TS 6.9 is consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007 and, therefore, is acceptable.

5.6.10 TS 6.10 Reports to the U.S. Nuclear Regulatory Commission

TS 6.10 presents the location for submitting reports to the NRC as follows:

> All reports in this Section shall be submitted to the U.S. Nuclear Regulatory Commission Document Control Desk.

5.6.10.1 TS 6.10.1 Written Reports within 10 Days

TS 6.10.1 presents the requirements for a written report to the NRC within 10 days as follows:

> Written reports of the following shall be submitted to the U.S. NRC within 10 days:
>
> (1) A release of radioactivity above permissible limits in unrestricted areas whether or not the release resulted in property damage, personal injury, or exposure. The written report (and, to the extent possible, the preliminary telephone report) shall describe, analyze, and evaluate safety implications, and outline the corrective measures taken or planned to prevent recurrence of the event.
>
> (2) a violation of a safety limit;
>
> (3) a reportable occurrence as defined in Section 1, "Reportable Occurrence," of these Technical Specifications.

5.6.10.2 TS 6.10.2 Written Reports Due within 30 Days

TS 6.10.2 presents the requirements for a 30-day written report to the NRC as follows:

> Written reports of the following shall be submitted to the U.S. NRC within 30 days:
>
> (1) a significant variation of measured values from a corresponding predicted or previously measured value of safety related operating characteristics occurring during operation of the reactor;
>
> (2) a significant change in the transient or accident analysis as described in the Safety Analysis Report;
>
> (3) permanent changes in the facility organization involving Level 1 or Level 2 management personnel.

5.6.10.3 TS 6.10.3 Written Reports Due within 60 Days

TS 6.10.3 presents the requirements for a 60-day written report to the NRC as follows:

> A report shall be submitted within 60 days after completion of startup testing of the reactor upon receipt of a new facility license or an amendment to the license authorizing an increase in reactor power level. The report shall describe the measured values of the operating conditions, including:
>
> (1) an evaluation of facility performance in comparison with design predictions and specifications;
>
> (2) a reassessment of the safety analysis submitted with the license application which discusses measured operating parameters when measurements indicate a substantial variation from prior analysis.

5.6.10.4 TS 6.10.4 Written Reports to the U.S. NRC within 60 Days after June 30 of Each Year

TS 6.10.4 presents the requirements for an annual report to the NRC as follows:

> The annual report shall provide the following information:
>
> (1) a brief narrative summary of
>
> (a) operating experience (including experiments performed),
>
> (b) changes in facility design, performance characteristics, and operating procedures related to reactor safety and occurring during the reporting period, and
>
> (c) results of surveillance tests and inspections;
>
> (2) tabulation of the energy output (in megawatt-days) of the reactor, the number of hours that the reactor was critical, the cumulative total energy output since initial criticality, and number of pulses greater than $1.00;
>
> (3) the number of emergency shutdowns and inadvertent scrams, including reasons for them and actions taken to prevent recurrence;
>
> (4) 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;
>
> (5) a brief description, including a summary of the safety evaluations of changes in the facility or in procedures and of tests and experiments carried out pursuant to 10 CFR 50.59;
>
> (6) 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 before the point of such release or discharge:

(7) liquid waste (summarized on a monthly basis):

(a) monthly radioactivity discharged;

(b) total estimated quantity of radioactivity released (in curies);

(c) an estimation of the specific quantity for each detectable radionuclide in the monthly release;

(d) fraction of 10 CFR 20 Table 3, Appendix B limit for each detectable radionuclide taking into account the dilution factor from the total volume of sewage released by the licensee into the sewage system;

(e) sum of the fractions for each radionuclide reported above;

(f) total quantity of radioactive material released by the facility into the sewage system during the reporting period.

(8) gaseous waste (summarized on a monthly basis) radioactivity discharged during the reporting period, including:

(a) total estimated quantity of radioactivity released (in curies) determined by an appropriate sampling and counting method;

(b) total estimated quantity of ^{41}Ar released (in curies) during the reporting period based on data from an appropriate monitoring system;

(c) estimated average atmospheric diluted concentration of ^{41}Ar released during the reporting period in terms of μCi/mL and fraction of the applicable DAC value;

(d) total estimated quantity of radioactivity in particulate form with halflives greater than 8 days (in curies) released during the reporting period as determined by an appropriate particulate monitoring system;

(e) average concentration of radioactive particulates with half-lives greater than 8 days released in μCi/mL during the reporting period;

(f) an estimate of the average concentration of other significant radionuclides present in the gaseous waste discharge in terms of μCi/mL and fraction of the applicable DAC value for the reporting period if the estimated release is greater than 20% of the applicable DAC.

(9) solid waste (summarized on an annual basis) including:

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

(b) total radioactivity in solid waste in curies,

(c) the dates of shipment and disposal (if shipped off-site).

(10) an annual summary of the radiation exposure received by facility personnel and visitors in terms of the average radiation exposure per individual and the greater exposure per individual in the two groups. Each exposure in excess of the limits of 10 CFR 20 shall be reported, including the time and date of the exposure as well as the circumstances that led to the exposure.

(11) an annual summary of the radiation levels including contamination levels observed during routine surveys performed at the facility including a summary of the average and highest levels;

(12) an annual summary of environmental surveys performed outside the facility.

The NRC staff finds that TS 6.10 is consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007 and, therefore, is acceptable.

5.6.11 TS 6.11 Written Communications

TS 6.11 presents the requirements for written communications with the NRC as follows:

> All written communications with the U.S. Nuclear Regulatory Commission shall be made in accordance with the requirements of 10 CFR 50.4 “Written Communications.”

The NRC staff finds that TS 6.11 is consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007 and, therefore, is acceptable.

5.7 Conclusions

The NRC staff reviewed and evaluated the TS as part of its review of the application for license renewal of Facility Operating License No. R-76, NRC Docket No. 50-27. The TS define certain features, characteristics, and conditions governing the operation of the WSU NRCR. The TS are explicitly included in the renewed license as Appendix A. The NRC staff reviewed and evaluated the content of the TS to determine whether or not the TS meet the requirements in 10 CFR 50.36, “Technical Specifications.” Based on its review, the NRC staff concludes that the WSU NRCR TS do meet the requirements of the regulations. The NRC staff also 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 finds that the TS are consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. 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 appropriate summary bases for the TS. Those summary bases are part of the TS but not required by the regulations.

- The WSU NRCR 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 operating license will include the TS. To satisfy the requirements of 10 CFR 50.36(b), the licensee provided TS derived from analyses in the WSU NRCR license renewal SAR, as supplemented.

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

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

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

6. CONCLUSIONS

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

- The application for license renewal dated June 24, 2002, as supplemented, complies with the standards and requirements of the AEA and the Commission's rules and regulations set forth in the regulations.

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

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

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

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

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