Safety Evaluation Report

Renewal of the Facility Operating License for the Purdue University Research Reactor, PUR-1

License No. R-87
Docket No. 50-182

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

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ABSTRACT

This safety evaluation report summarizes the findings of a safety review conducted by the staff of the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Reactor Regulation. The NRC staff conducted this review in response to a timely application filed by Purdue University (the licensee) for a 20-year renewal of Facility Operating License No. R-87 with a requested thermal power increase from 1 kilowatt (kWt) to 12 kWt for the Purdue University Research Reactor (PUR-1). 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 Purdue University can continue to operate the PUR-1 for the term of the renewed license and at an increased power level, in accordance with the license, without endangering public health and safety, facility personnel, or the environment.

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ABBREVIATIONS AND ACRONYMS

$	dollar (of reactivity) or U.S. currency
% Δk/k	reactivity in percent
10 CFR	Title 10 of the *Code of Federal Regulations*
ADAMS	Agency-Wide Documents Access and Management System
AEA	Atomic Energy Act of 1954, as amended
AEC	Atomic Energy Commission
ALARA	as low as low as reasonably achievable
ALI	annual limit on intake
ANS	American Nuclear Society
ANSI	American National Standards Institute
Ar-41	argon-41
C	celsius
CAM	continuous air monitor(ing)
Ci	Curie
cm	centimeter
cm^3	cubic centimeter
cm^3/s	cubic centimeter per second
CORO	Committee on Reactor Operations
DAC	derived air concentrations
DCF	Dose Conversion Factor
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DOE	Department Of Energy
EP	emergency plan
F	fahrenheit
FGR	Federal Guidance Report
FOCD	foreign ownership, control and domination
ft	feet
FY	fiscal year
gpm	gallons per minute
HEPA	High-efficiency particulate air
HEU	high-enriched uranium
HVAC	heating, ventilation and air conditioning systems
I&C	instrumentation and control
in	inch
IR	inspection report
ISG	interim Staff Guidance
km/h	kilometer per hour
kW	kilowatt
kWt	kilowatt thermal
l/min	liters/minute
LCO	limiting condition for operation
LEU	low-enriched uranium
LOCA	loss-of-coolant accident
LRA	license renewal application
LSSS	limiting safety system setting

μCi/mL	microcuries per milliliter
m^3/s	cubic meter per second
m	meters
MCNP	Monte Carlo N-Particle Code
mg	milligram
MHA	maximum hypothetical accident
mhos/cm	microhmos per centimeter
mm	millimeter
mph	miles per hour
mrem	millirem
mrem/hr	millirem per hour
MTR	Materials Testing Reactor
μs	microsecond
MW	megawatt
MWt	megawatt thermal
N-16	nitrogen-16
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
ONB	onset of nucleate boiling
ONBR	ONB ratio
Pa	pascal
PDR	public document room
PUR-1	Purdue University Research Reactor
RAI	request for additional information
RAM	remote area monitors
RG	NRC Regulatory Guide
RO	reactor operator
RR	regulating control rod
RSO	radiation safety officer
RTR	research and test reactor
SAR	safety analysis report
SER	safety evaluation report
SL	safety limit
SOI	statement of intent
SRM	staff requirements memorandum
SRO	senior reactor operator
SRP	standard review plan
ss	shim-safety control rod
TEDE	total effective dose equivalent
TLD	thermoluminescent dosimeter
TS	technical specification(s)
U-235	uranium-235

1 INTRODUCTION

1.1 Overview

By letter and supporting documentation dated July 7, 2008 (Ref. 1), as supplemented by letters dated June 4, 2010 (Ref. 59); November 15, 2011 (Ref. 10); January 4 (Ref. 11), January 30 (Ref. 12), January 31 (Ref. 13), June 1 (Ref. 14), June 15 (Ref. 15), June 29 (Ref. 16), July 13 (Ref. 17), and August 11, 2012 (Ref. 18); April 10, 2013 (Ref. 19); July 24, 2015 (Ref. 20); and January 29 (Ref. 53), February 26 (Ref. 54), March 31 (Ref. 55), May 9 (Ref. 56), July 7 (Ref. 62), July 19 (Ref. 21), September 19, 2016 (Ref. 22), and September 29, 2016 (Ref. 63), Purdue University (the licensee) submitted to the U.S. Nuclear Regulatory Commission (NRC) an application for a 20 year renewal for the Class 104c Facility Operating License No. R-87, NRC Docket No. 50-182, for the Purdue University Research Reactor (PUR-1). The application also requested a power increase from a thermal power of 1 kilowatt (kWt) to 12 kWt.

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.51(a) (Ref. 2) states that "[e]ach license will be issued for a period of time to be specified in the license but in no case to exceed 40 years from date-of-issuance." The PUR-1 original license was issued on August 16, 1962, and subsequently was renewed on August 8, 1988, for a period of 20 years expiring on August 8, 2008. Because of the timely renewal provision contained in 10 CFR 2.109(a), the licensee is permitted to continue operation of the PUR-1 under the terms and conditions of the current license until the NRC staff completes action on the renewal request. A renewal would authorize the licensee to continue operation of PUR-1 for an additional 20 years. In its September 29, 2016, letter, the licensee stated that the renewal "and power uprate will allow the PUR-1 facility to expand its research space through a tenfold increase in neutron flux and enhance its teaching mission through the demonstration of more reactor principles."

The PUR-1 is located on the campus of Purdue University, in the city of West Lafayette, Tippecanoe County, Indiana and is licensed to operate at maximum power level of 1 kWt. PUR-1 became operational in 1962.

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 (RTRs) to convert from the use of high enriched uranium (HEU) fuel to low-enriched uranium (LEU) fuel, unless specifically exempted. In a letter dated August 13, 2006 (Ref. 4), the licensee submitted its proposal to convert the fuel from HEU to LEU requesting approval of the fuel conversion and of changes in the Technical Specifications (TSs). After an initial review, the NRC staff issued a request for additional information (RAI) to the licensee on March 13, 2007 (Ref. 5), and in a letter dated May 3, 2007 (Ref. 6), the licensee submitted responses to the RAI.

The NRC issued the Order for the licensee to convert the PUR-1 HEU fuel to LEU fuel on August 9, 2007 (Ref. 7). The Order included a Safety Evaluation Report (SER), as Enclosure 4 of the Order to convert, that provided the results of the NRC staff's evaluation of the licensee's conversion request. The Order also included changes in the TSs that would be required for operation of the facility with the LEU fuel. PUR-1 reached initial criticality using the LEU fuel in September 2007. The Order also required the licensee to submit a startup report to the NRC within six months of the completion of the conversion. In this SER for the current license

renewal request, 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 based its review of the request to renew the PUR-1 operating license and increase the steady-state operating power from 1 kWt to 12 kWt on the information contained in the license renewal application as well as supporting supplements and licensee responses to RAIs. Specifically, the renewal application included the Safety Analysis Report (SAR) (Ref. 3) and a revision of the PUR-1 physical security plan. In the SAR, the licensee stated that the emergency plan, the operator requalification plan, and the technical specifications (TS) are on file with the NRC. The licensee has since updated these plans and the TSs in response to RAIs issued by the NRC staff as part of the license renewal review process.

As part of its review, the NRC staff also reviewed annual reports of the facility operation (Ref. 8) submitted by the licensee and inspection reports (Ref. 9) prepared by NRC personnel. The NRC staff issued RAIs in letters dated March 24, 2010 (Ref. 58); July 6 (Ref. 23), July 8 (Ref. 24), and July 14, 2011 (Ref. 25); August 29, 2014 (Ref. 26); December 23, 2015 (Ref. 52); and January 19 (Ref. 27) and July 25, 2016 (Ref. 28). Several site visits were conducted at the facility to observe facility conditions and to discuss RAIs and responses.

The licensee provided responses to the RAIs in letters dated June 4, 2010; November 15, 2011; January 4, January 30, January 31, June 1, June 15, June 29, July 13, and August 11, 2012; April 10, 2013; July 24, 2015; and January 29, February 26, March 31, May 9, July 7, July 19, September 19, and September 29, 2016.

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

The "References" section of this document contains the dates and associated ADAMS Accession Numbers of the licensee's renewal application and associated supplements.

In conducting its safety review, the NRC staff evaluated the facility against the requirements of the regulations including 10 CFR Part 20, "Standards for Protection against Radiation," 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material." The recommendations of applicable regulatory guides (RGs) and relevant accepted industry standards, such as those of the American National Standards Institute/American Nuclear Society (ANSI/ANS) 15 series, are also considered. The NRC staff also specifically referred to

the recommendations contained in NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,” issued in February 1996 (Ref. 29). 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. 30), the NRC staff provided the Commission with information on plans to streamline the review of license renewal applications (LRAs) for RTRs. The Commission issued its staff requirements memorandum (SRM) for SECY-08-0161, dated March 26, 2009 (Ref. 31). The SRM directed the staff to streamline the renewal process for RTRs, 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 used in this SER, the NRC staff considered the results of past NRC staff reviews. A basic requirement, as contained in the SRM, is that licensees must be in compliance with applicable regulatory requirements.

The NRC staff developed the RTR Interim Staff Guidance (ISG)-2009-001, “Interim Staff Guidance on the Streamlined Review Process for License Renewal for Research Reactors,” (Ref. 32) to assist in the review of LRAs. The streamlined review process is a graded approach based on licensed power level. The streamlined review process divides the RTR facilities into two tiers. Facilities with a licensed thermal power level of 2 megawatts (MWt) and greater, or requesting a power level increase, undergo a full review using NUREG-1537. Facilities with a licensed power less than 2 MWt, undergo a focused review that centers on the most safety-significant aspects of the renewal application and relies on past NRC reviews for certain findings. The NRC staff made a draft of the ISG available for public comment and considered public comments in its development of the final ISG.

The ISG directs NRC staff to review license renewal applications that also include a request for an increase in licensed power level using a full review based on NUREG-1537. The NRC staff performs a complete license renewal at the increased power level because the requested power increase has the potential to impact almost all safety aspects of the reactor operation as reflected in the SAR. The NRC staff reviewed the PUR-1 LRA using the guidance in the final ISG, dated October 15, 2009 (Ref. 32), and because the PUR-1 LRA requested a power level increase, the NRC staff performed a full review of the licensee’s LRA, in accordance with the guidance of NUREG-1537.

The NRC staff separately evaluated the environmental impacts of the renewal of the license for PUR-1 in accordance with 10 CFR Part 51. The NRC staff published an Environmental Assessment and Finding of No Significant Impact in the *Federal Register* on October 27, 2016 (81 FR 74822), which concluded that renewal, including the power increase of the PUR-1 operating license will not have a significant effect on the quality of the human environment.

This SER summarizes the NRC’s staff findings of the PUR-1 safety review of the LRA and delineates the technical details that the NRC staff considered in reviewing and evaluating the safety aspects of continued operation. This SER also provides the basis for renewing the PUR-1 license at steady-state power level up to and including 12.0 kWt.

This SER was prepared by Alexander Adams Jr., Chief, and Cindy Montgomery, project manager and in the NRC's Office of Nuclear Reactor Regulation (NRR), Division of Policy and Rulemaking, Research and Test Reactors Licensing Branch; and Nicole Newton and Emil Tabakov, financial analysts in the NRC's NRR, Division of Inspection and Regional Support, Financial Analysis and International Projects Branch. Brookhaven National Laboratory, the NRC's contractor, provided input to this SER.

1.2 Summary and Conclusions on Principal Safety Considerations

The NRC staff's review and evaluation considers the information submitted by the licensee, including past operating history recorded in the licensee's annual reports to the NRC, as well as IRs prepared by the NRC staff. On the basis of this evaluation and resolution of the principle issues reviewed for the PUR-1, the NRC staff concludes the following:

- The design and use of the reactor structures, and systems and components important to safety during normal operation discussed in the PUR-1 SAR (Ref. 3), as supplemented, in accordance with the TSs are safe, and safe operation can reasonably be expected to continue.

- The licensee considered the expected consequences of a broad spectrum of postulated credible accidents and a maximum hypothetical accident (MHA), emphasizing those that could lead to a loss of integrity of fuel plate 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 for the facility staff, and members of the public, would not exceed doses in 10 CFR Part 20.

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

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

- The licensee's TSs, 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 the PUR-1 SAR (Ref. 3), as supplemented, and the TSs will continue to help ensure that there will be no significant degradation of safety-related equipment.

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

- The licensee maintains a program for providing for the physical protection of the facility and its special nuclear material in accordance with the requirements of 10 CFR Part 73, which reasonably ensures that the licensee will continue to provide the physical protection of the facility and its special nuclear material.

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

- The licensee’s procedure for training its reactor operators and the operator requalification plan give reasonable assurance that the licensee will continue to have qualified personnel who can safely operate the reactor.

On the basis of these findings, the NRC staff concludes that there is reasonable assurance that the licensee can continue to operate the PUR-1 at a power level up to and including 12 kWt 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, facility staff, or the environment. The issuance of the renewed license will not be inimical to the common defense and security.

1.3 General Description of the Facility

The PUR-1 is a heterogeneous, pool-type non-power reactor. The reactor is fueled with Materials Testing Reactor (MTR) type LEU fuel plates. The reactor core is cooled by natural convection of light water, moderated by light water, and reflected by water and graphite. The reactor is located near the bottom of a water-filled tank surrounded and supported by a concrete shielding structure.

The reactor core is submerged in a 17 feet-4 inches (5.3 m) deep in-ground tank with a stainless steel liner below floor level. A three-foot (0.9 m) high concrete wall above floor level serves as a biological shield. The core is located to one side of the pool to provide additional experimental space. Heat generated from the reactor core is directly transferred to the pool water by natural convection. Reactor pool water temperature is maintained on the average at 26 °C (79 °F) by a closed loop cooling system. This system provides the heat removal capability for the water in the reactor pool. The cooling system has three loops. A pump with a design flow rate of 30 gpm (114 l/min) 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. Heat is removed from the shell side of this heat exchanger by a Freon refrigerant loop. The third coolant loop uses the chilled campus water system to remove heat from the Freon loop, limiting the potential for radioactive contamination from the pool water of the campus chilled water system.

The PUR-1 experimental facilities include experiment locations within the graphite reflector and drop tubes next to the reflector boundary. The drop tubes are dry air tubes, while in-reflector facilities are aluminum tubes normally filled with graphite or experimental capsules.

The reactor is controlled by three blade-type control rods, one regulating, and two shim-safety rods. The regulating rod is a hollow stainless steel blade operated with a direct drive and no scram capability. The shim-safety rods are made with borated stainless steel with a magnetic clutch and screw operated drive mechanism. The shim-safety rods can be disengaged to drop by gravity into the core to scram the reactor.

The initial design power level for PUR-1 was 10.0 kWt, while the reactor licensed power level is 1.0 kWt. The analyses provide with the license renewal application supports the continued operation of the PUR-1 and a power increase to a power level of 12.0 kWt.

1.4 Shared Facilities and Equipment

The PUR-1 is contained within a separate annex to the Electrical Engineering Building complex. The reactor building previously was used as a high voltage laboratory that was converted to house general classrooms and laboratories. The reactor room contains minimal penetrations for air conditioning and water supply. Offices for reactor program personnel and some laboratories are located in the building at various elevations. Electrical power and potable water are supplied to the PUR-1 from the Electrical Engineering building. The campus water supply, which provides the ultimate cooling for the secondary cooling system contains a backflow preventer. Since commencing operation, there has been no adverse impact on the operation or safety of the reactor resulting from this facility design.

1.5 Comparison with Similar Facilities

The PUR-1 research reactor is similar to other MTRs with plate type fuel. MTR-type reactors have common features, such as light-water moderation, natural convection cooling, open pools, and plate-type fuel. Many reactors with similar design, construction, and operational characteristics have operated safely and reliably for more than four decades. PUR-1 is most similar to the Ohio State University Reactor (600 kWt) and University of Missouri at Rolla (200 kWt) in operating characteristics and facility features. Instruments and controls used in the PUR-1 facility are similar in principle to most non-power reactors licensed by the NRC. The safe operating histories of these reactors and PUR-1 demonstrate the reliability and safety of these systems. Both the Ohio State and Rolla reactors operate at a much higher power levels supporting the licensee’s request for a power increase to 12 kWt steady-state power level. There are no unique features of the PUR-1 that would preclude applying knowledge and experience gained in the operation of other comparable reactors.

1.6 Summary of Operations

The current usage of the PUR-1 is primarily education, training students in reactor engineering theory and operation, and nuclear research. The reactor is integrated into several core courses in the nuclear engineering curriculum. It also has research programs that utilize irradiation facilities, and accommodate visitors and other public outreach programs. The utilization of the PUR-1 over the past 20 year period was moderate, operating about 50-90 times per year on average. In the period of 2006-2008, the reactor averaged 40-70 operating runs with an integrated running time ranging between 110-160 hours per year. Most operating runs are at low power not exceeding 100 watts. Expectations for the upcoming license renewal period are to at least maintain or improve the present utilization rate with the expanded capabilities (Ref. 22).

The NRC staff review considered PUR-1 annual reports and NRC IRs from 2005 through 2015. The annual report summaries did not indicate any significant degradation of fuel element integrity, control rod operability issues, or radiological exposure concerns. The scram circuits required for operation are calibrated regularly. The IRs identified no findings of significance.

1.7 Compliance with the Nuclear Waste Policy Act of 1982

Section 302(b)(1)(B) of the Nuclear Waste Policy Act of 1982, 42 U.S.C. §10222(b)(1)(B) specifies that the NRC may require, as a precondition to issuing or renewing an operating license for a research or test reactor, that the applicant 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 the DOE informed H. Denton of NRC that DOE has determined that universities and other government agencies operating non-power reactors have entered into contracts with the DOE providing that the DOE retains title to the fuel and is obligated to take the spent fuel and/or high-level waste for storage or reprocessing (Ref. 33). An e-mail sent from Kenny Osborne of DOE to Duane Hardesty (NRC), dated January 15, 2014 (Ref. 49), reconfirms this contractual obligation with respect to the fuel at the PUR-1 (DOE Contract No. 78286), valid from March 1, 2009, to December 31, 2017. Additionally, DOE renews these contracts prior to their expiration to ensure that the contracts remain valid. By entering into such a contract with the DOE, Purdue University has satisfied the requirements of the Nuclear Waste Policy Act of 1982 (Ref. 33).

1.8 Facility Modifications and History

In 1962, the U.S. Atomic Energy Commission (AEC) issued an operating license to Purdue University for operation of the PUR-1 on its campus located in West Lafayette, Indiana. Facility Operating License R-87 authorized the PUR-1 to operate at steady-state power levels up to 1.0 kWt. Review of the facility modifications made during the last 20 years indicates no major changes.

In a letter dated August 13, 2006, as supplemented on May 3, 2007, Purdue University submitted its proposal to convert to LEU fuel, requesting approval of the fuel conversion and changes in the TSs (Ref. 4). The Order to convert was issued as License Amendment 12 on August 9, 2007, which authorized the conversion from HEU fuel to LEU fuel (Ref. 7). The conversion Order modified the license, including the TSs and EP, in accordance with 10 CFR 50.64, “Limitations on the use of highly enriched uranium (HEU) in domestic non-power reactors,” which requires that non-power reactors, such as the PUR-1, convert to LEU fuel under certain conditions. The licensee stated that there have been no changes to the PUR-1 facilities since the submittal of the license renewal and power increase application dated June 7, 2008 (Refs. 1, 14).

During this LRA review, most modifications to the PUR-1 involved technological upgrades to instrumentation and minor changes to the existing design that either enhanced its capability 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 PUR-1. Furthermore, the NRC staff reviewed the licensee’s annual operating reports from 2005 to 2015 (Ref. 8) and NRC IRs from 2005 to 2015 (Ref. 9) that documented these changes. The results of these reviews indicated that the changes were performed, as required, in accordance with the requirements of 10 CFR 50.59. The NRC staff concludes that all PUR-1 facility changes appear to be reasonable and the licensing actions taken over the years seem appropriate.

2 SITE CHARACTERISTICS

2.1 Geography and Demography

Chapter 2 of the licensee's SAR discusses the characteristics of the Purdue University Research Reactor (PUR-1) site. The follow sections describe the geography of the PUR-1 site including the location of the PUR-1 and the demography of the site.

2.1.1 Reactor Site

Section 2.1 of the licensee's SAR discusses the reactor site. The PUR-1 is located on the campus of Purdue University, in the city of West Lafayette, Tippecanoe County, Indiana. The PUR-1 is located in the Duncan Annex of the Electrical Engineering Building that was originally built as a high energy research laboratory. The building annex was reconstructed to house the reactor as well as various laboratories and class rooms.

TS 1.28 and 5.1 provide the reactor location description as follows:

TS 1. DEFINITIONS

(…)

1.28 Reactor Facility - The reactor facility is that portion of the ground floor of the Duncan Annex of the Electrical Engineering Building occupied by the School of Nuclear Engineering used for activities associated with the reactor.

(…)

TS 5.1 Site Description

a. The reactor shall be located on the ground floor of the Duncan Annex of the Electrical Engineering Building, Purdue University, West Lafayette, Indiana.

b. The School of Nuclear Engineering shall control approximately 5000 square feet of the Duncan Annex ground floor, which includes the reactor room. Access to the Nuclear Engineering controlled area shall be restricted except when classes are held there.

c. The licensed areas shall include the reactor room, and the fuel storage room. Both of these areas shall be restricted to authorized personnel, or those escorted by authorized personnel.

d. The reactor room shall remain locked at all times except for the entry or exit of authorized personnel or those escorted by authorized personnel, equipment, or materials.

(…)

TS 1.28, and TS 5.1, Specifications a through d, describe the PUR-1 facility site boundary, state important design features of the facility physical design, and specify the licensed and restricted areas. The restricted area defined above is a geometric arrangement that is used in calculations that could affect safety such as public doses from radiation. TS 5.1, Specification c, defines the restricted area and all activities performed within this area fall under the jurisdiction of the reactor license. TS 5.1, Specification d, describes the licensed and restricted areas that should remain locked except as described in the TS. The licensee modified TS 5.1, Specification b, to better define the restricted area and access control (Ref. 12, 22).

The NRC staff reviewed the facility boundary as described in the SAR, and toured the facility during license renewal site visits, and finds that TS 1.28 and TS 5.1, Specifications a through d, help ensure that important features associated with the licensed and restricted areas of the PUR-1 facility are defined. The NRC staff finds that TS 5.1, Specifications a through d, accurately describe the facility site's licensed and restricted areas consistent with the SAR, as supplemented, and the PUR-1 emergency plan (EP). The NRC staff finds that TS 5.1, Specifications a through d, are consistent with the guidance in NUREG-1537 and ANSI/ANS 15.1-2007. Based on the information provided above, the NRC staff concludes that TS 1.28 and TS 5.1, Specifications a through d, are acceptable.

2.1.2 Demography

According to the U. S. Census Bureau 2010 data, Tippecanoe County has a population of 172,780. The majority of the population, 79 percent, resides within 5 miles (8 km) from the reactor location. The Lafayette area is about 60 miles (100 km) northwest of Indianapolis, and about 140 miles (230 km) south-east of Chicago, Illinois.

On campus, the closest residence halls (Hawkins and Young Hall) are located about 1000 feet (304 m) to the south of the PUR-1. The typical population for all on-campus residence halls is about 20,000 persons (between the months of September and June). There are a number of adjacent office buildings to the Duncan Annex occupied by students and administrative staff during daytime. Because of the university's large transient population, the campus student population varies from approximately 40,000 during the school's fall and spring academic sessions to a significantly lower population during the summer sessions due to reduced summer enrollment. The total population of the campus is about 55,000 with Purdue University faculty and staff numbering about 15,000.

2.1.3 Conclusions

The licensee has provided a sufficiently detailed and accurate description of the geography surrounding the PUR-1. The demographic information is sufficient to allow accurate assessments of the potential radiological impact on the public from the continued operation of the PUR-1. Based on the information provided above, the NRC staff concludes that there is reasonable assurance that the geographic or demographic features pose no significant risk to the continued safe operation of the PUR-1 facility.

2.2 Nearby Industrial, Transportation and Military Facilities

Section 2.2 of the licensee's SAR discusses nearly industrial, transportation and military facilities. The areas surrounding the PUR-1 site consist primarily of the university campus, residential areas, apartment complexes and small businesses, such as restaurants and retail stores that cater to the student population. There are no significant industrial activities in the

immediate areas that could lead to potential accidents affecting the PUR-1. There are no major railroad routes or military installations near the campus.

Several commuter routes border the campus. Route 231, running about 200 feet (61 m) to the north-east, and Route 26, located 1,500 feet (152 m) south of the reactor site, are primary commuting transportation routes crossing the campus. These roads are mainly used for local transportation by students, university staff, and residents. The distance of the PUR-1 from these routes and nearby university buildings surrounding the reactor building makes it unlikely that a transportation incident would impact the reactor facility.

Purdue University operates an airport that is located at the southwest edge of the West Lafayette Campus. The airport averages about 130,000 aircraft operations annually. The majority of air operations are conducted by the University Aviation Technology program with an average of 750 takeoffs and landings a day during the spring and fall semesters. During the summer, the airport sees an average of 30 flight operations per day.

Most training flights use a runway that is about 6,300 feet (1.9 km) to the southwest from the Duncan Annex with a landing approach that passes near the Electrical Engineering Building. However, due to runway length restrictions only light aircraft can use this runway, while larger aircraft use another runway that has a takeoff/ landing approach not directed near the Electrical Engineering Building.

The licensee has stated that the potential crash of a small aircraft into the reactor building is extremely unlikely and any damage would be limited to the heavy concrete construction of the building. In addition, the reactor is located underground below grade level. The only event, however unlikely, that could lead to the release of radioactivity is a direct hit on the reactor core resulting from an aircraft crash through the roof of the reactor building and into the reactor pool. The reactor core is located in a tank of water 13 feet (4.0 m) below the water surface that would limit any damage and potential radioactivity release. The NRC staff considers the likelihood of such an event extremely low and concludes that members of the public are not subject to undue radiological risk in the unlikely event of an aircraft crash.

With the exception of the Purdue University airport, which has been satisfactorily considered, there are no major transportation routes and no significant military or industrial facilities in the vicinity of the reactor site. Based on the NRC staff review of these facilities, the NRC staff concludes that there is reasonable assurance that normal operations at these facilities will not affect the continued safe operation of the PUR-1.

2.3 Climatology and Meteorology

Section 2.3 of the licensee's SAR describe the climatology and meteorology of the PUR-1 site.

2.3.1 Climatology

The overall climate at the PUR-1 site is considered to be continental, generally characterized by cold winters and hot summers with frequent changes. The normal annual precipitation (primarily rain) for the Lafayette area is 35.7 inches (91 cm). On a monthly basis, rainfall amounts range from 4 inches (10 cm) in July to 1.4 inches (3.6 cm) in February.

Based on a historical survey of temperatures (1971—2000), monthly normal temperature ranges from 73.8 °F (23.2 °C) in July to 23.3 °F (-4.8 °C) in January. Daily extremes over this

time period varied from a high of 104 °F (40 °C) to a low of -24 °F (-31 °C). On average, for 137 days per year the temperature drops below freezing 32 °F (0 °C).

Average wind data collected at the Purdue University Airport show that, for the period 1977-2006, the average, mean wind speed was 8.8 mph (14.2 km/h) primarily with the average wind direction of 190-210 degrees (0°=north). The university buildings including the Duncan Annex were designed to withstand or exceed the wind load, 17 pounds per square foot (814 Pa), established by the Uniform Building Code in existence at the time of construction for the Lafayette area corresponding to a maximum wind zone of 80 mph (129 km/h).

2.3.2 Severe Weather

From 1950 through 2008, there were 38 reported tornadoes in Tippecanoe County. Among these, a total of three tornadoes were of magnitude F4 (Fujita Scale), with wind speeds from 207-260 mph (333-418 km/h), one was magnitude F3 (158-206 mph/254-331 km/h), 10 magnitude F2 (113-157 mph/181-252 km/h), 13 magnitude F1 (73-112 mph/118-180 km/h), and the remaining 11 tornadoes were magnitude F0 (winds from 40-72 mph/64-116 km/h). Using a conservative statistical method the NRC staff estimated the occurrence of a tornado striking the PUR-1 site is about 6×10^{-3}/year, which is considered a credible natural disaster that would affect the PUR-1 site. The licensee stated that due to the heavy construction of the Duncan Annex building the probability of damage due to a tornado hitting the building directly is estimated to be minimal (see further discussion in Section 13.2.8 of this SER).

2.3.3 Conclusions

Based on a review of the historical data submitted by the licensee, the NRC staff concludes that the meteorological information provided for the region around the PUR-1 is sufficiently documented. The information on average temperatures, wind speed and directions are sufficient to support dispersion analyses for postulated airborne releases which may occur during operations, including emergency situations. The NRC staff concludes that there is a relatively high tornado strike probability, but the effect of tornadoes should be minimal due to design features of the reactor building (see Section 13.2.8 of this SER). The NRC staff also concludes that there are no meteorological-related events or consequences at the site that would pose unacceptable risk to the continued safe operation of the PUR-1 facility.

2.4 <u>Hydrology</u>

Section 2.4 of the licensee's SAR discusses the hydrology of the PUR-1 site. Tippecanoe County is mostly covered by a glacial drift. Ground water occurs within the drift layers. The Wabash River near the PUR-1 (3000 feet (914 m) east from the reactor building) flows through an alluvial valley that was formed during the glacial times. The river is about 180 feet (55 m) lower than the campus insuring that both surface and ground water flows are in the general easterly direction toward the Wabash River. The West Lafayette and Purdue University water supply would not be affected by any leakage of contaminated water from the reactor since the ground water flows are away from known water wells.

In addition, the licensee has demonstrated that all liquid radioactive release is analyzed to assure compliance with regulatory requirements before release to the sanitary sewer system. There is no direct discharge to the surrounding waterways. In the unlikely event of an inadvertent release or leakage of primary coolant, significant dilution would take place before any of the affected water would be used for potential human consumption. The impact on public

safety and the environment would not be significant due to a small leakage. The amount of water would be small, and the radioactivity level would be low. The typical radioactivity concentrations in the primary coolant are very low and the equilibrium concentrations of predominant radionuclides in the primary coolant are within the limits found in 10 CFR Part 20 for release to the sewer.

There are no lakes or dams near the PUR-1 and therefore, seismically induced flooding due to a dam failure or seiches is not a risk for the PUR-1. Since the PUR-1 is not located near lakes or dams and it is very far from the coastline (no tsunami risk), there is very little risk that hydrology issues will significantly impact the safety at PUR-1. Based on the information provided above, the NRC staff concludes that there are no credible hydrologic events, which would affect the safe operation of the PUR-1 during the renewal period.

2.5 Geology and Seismology

Section 2.5 of the licensee's SAR describes the geology and seismology of the PUR-1 site. Tippecanoe County is located within the Tipton Till Plains with most of the soils being glacial deposits. The glacial drift covers the underlying bedrock ranging from a few feet to 300 feet (91 m). The underlying bedrock is primarily limestone and sandstone. The Purdue University campus is located above extensive glacial deposits of sand and gravel. Tippecanoe County is part of the drainage basin of the Wabash River with the land sloping generally to the southwest with the streams and the Wabash River flowing west-southwest.

The licensee reports that there are three seismic zones at a distance ranging from 240 to 120 miles (400 to 200 km): the New Madrid fault system in Missouri (260 miles (440 km distant)), the Wabash Valley fault system (120 miles (200 km)), and the Anna seismic zone system in Ohio (120 miles (200 km)). The New Madrid seismic zone had four of the largest North American earthquakes in recorded history, all occurring within a three-month period between December 1811 and February 1812. The biggest earthquake since 1811-1812 was a 6.6-magnitude (estimated Richter-scale) earthquake on October 31, 1895, with an epicenter at Charleston, Missouri.

The licensee estimated the maximum magnitude of peak horizontal acceleration at West Lafayette is 0.05-0.15 g, which would occur after maximum magnitude events at the three seismic zones with 7.4, 6.6 and 6.3 magnitude earthquakes. The acceleration includes consideration for the distance and attenuation. Due to the free standing construction of the reactor pool, it is highly unlikely that reactor primary coolant would be lost during a severe seismic event.

The NRC staff concludes that the licensee has provided sufficient information about geologic features and potential seismic activity at the PUR-1 site. Seismic activity has not been significant in the vicinity of the PUR-1. The NRC staff concludes that it is highly unlikely that the reactor pool and core would be damaged in a seismic event leading to radiological consequences that would exceed analyzed conditions (see Chapter 13 of this SER).

2.6 Conclusions

The NRC staff concludes that the reactor site has experienced no significant geographical, meteorological, or geological change since initial licensing, and therefore, the site remains suitable for continued operation of the PUR-1. The low frequency of earthquakes and the robustness of the facility in case of tornadoes, continue to make the site suitable for operation of

the PUR-1. Hazards related to industrial, transportation, and military facilities pose no significant risk to the continued safe operation of the PUR-1. The demographics of the area surrounding the reactor have not changed in any way that significantly increases the risk to public health and safety from continued operation of the PUR-1. Based on the information provided above regarding both natural and man-made hazards, the NRC staff concludes that there is no significant risk associated with the site that would make it unacceptable for continued operation of the PUR-1 facility at the increased power level of 12 kWt.

3 DESIGN OF STRUCTURES, SYSTEMS AND COMPONENTS

3.1 Design Criteria

The licensee's discusses the design criteria of the PUR-1 in section 3.1 of the SAR. The PUR-1 is housed in the Duncan Annex of the Electrical Engineering Building, which was built using brick, concrete block and reinforced concrete. The building was originally designed as a high voltage laboratory with a large open space that was modified to include the reactor on the ground floor in the high bay area while the rest of the building was subdivided to include offices, laboratories and classrooms.

The reactor is a pool-type reactor with a current maximum licensed operating power level of 1.0 kWt. The reactor core sits on a support structure at the bottom of a pool. The pool consists of steel cylinders containing compacted magnetite sand and a carbon steel tank, which has a stainless steel liner. The reactor pool is built below floor level except for the three foot wall that serves as a biological shield at the top of the pool. The reactor instrumentation and control (I&C) system allows operation of the control rod drives and control rods to control the reactivity of the reactor. The I&C system is also a safety system that monitors key reactor parameters and can quickly shut down the reactor by scram if predetermined parameter limits are exceeded. The reactor core is cooled by natural convection of the pool water through the core. A cooling system removes heat from the reactor pool to the secondary cooling system that eventually transfers heat to the campus water system. The ventilation system controls the movement of air through the facility to a defined monitored path. The system can be shut down if predetermined radiological conditions are met. A number of experimental facilities exist to facilitate the conduct of research, education, and service operations.

TS 5.1, Specifications a and b, contain the design features applicable to the reactor building (see Section 2.1.1 of this SER). The purpose of these TSs is to require prior NRC review and approval before a change is made to the basic arrangement of the reactor facility. Changing this design feature could impact dose calculations and basic assumptions of the safety analysis.

The original reactor installation in the 1960s used components manufactured by Lockheed Nuclear Products of Lockheed Aircraft Corp., the reactor designer. The Electrical Engineering Building Duncan Annex was designed and constructed including later modifications in conformance with the local building codes in existence at the time. The licensee stated that buildings at Purdue University are designed to withstand maximum wind loads in a wind zone of 80 miles per hour (129 km/hr) satisfying the requirements of the Uniform Building Code in existence of the original design and construction (1940).

The PUR-1 was designed and constructed in accordance with the license issued by the Atomic Energy Commission. The design of the fuel and control rods are discussed in Chapter 4 of this SER. The design of the reactor safety system is discussed in Chapter 7 of this SER. The design of the reactor confinement is discussed in Chapter 6 of this SER.

Changes to the PUR-1 have been made by license amendments or the licensee review process under 10 CFR 50.59, "Changes, Tests, and Experiments," in accordance with the Commission's rules and regulations. The NRC staff previously evaluated all amendments to the facility license, and the NRC inspection program verified that the licensee conducted the proper

reviews. The application for license renewal under review includes changes made to the facility since initial licensing. Chapter 16 of this SER discusses age-related issues.

The reactor was originally designed to a 10 kWt power level, but is licensed to operate at 1 kWt. The licensee requested a power increase to 12 kWt, and submitted analyses demonstrating that the facility could withstand any credible accident with no hazard to the public, without reliance on engineered safety systems at the proposed 12 kWt power level. The fuel design and reactor operation are such that the plate-type fuel can be adequately cooled by natural convection of coolant through the reactor core.

The reactor building is designed to function as a confinement-type structure. The reactor building has three personnel doors with foam rubber seals.

TS 5.1, Specification f, specifies the minimum reactor building volume as follows:

> TS 5.1 <u>Site Description</u>
>
> <u>Specifications</u> -
>
> (…)
>
> f. The minimum free volume of the reactor room shall be approximately 15,000 cubic feet.
>
> (…)

TS 5.1, Specification f, helps ensure that the PUR-1 is housed in a facility designed with a minimum free volume sufficient to capture and control any radioactive effluents before release. TS 5.1, Specification f, specifies the minimum free volume that must be maintained by the facility providing a dilution capability limiting the dose from any airborne radioactivity released within personnel occupied areas of the PUR-1 from reactor operations. The NRC staff finds that the minimum free volume is used to calculate the radioactive concentration used in the maximum hypothetical accident (MHA) dose calculation for an exposed worker in the PUR-1 reactor building.

The NRC staff reviewed TS 5.1, Specification f, and finds that this specification helps ensure the design features supporting the assumptions used in the SAR, as supplemented, and provides for the confinement and controlled release of diluted radioactive materials by the ventilation system (see Section 9 of this SER). The NRC staff finds that TS 5.1, Specification f, is consistent with the guidance in NUREG-1537 and ANSI/ANS 15.1-2007. Based on the information provided above, the NRC staff concludes that TS 5.1, Specification f, is acceptable.

3.2 Meteorological Damage

The licensee discusses meteorological damage in section 3.2 of the SAR. The PUR-1 is located in the reactor building, which was constructed to local codes to withstand local wind, rain, snow and ice loads. The licensee stated that the reactor building is designed and constructed to withstand a wind load corresponding to a maximum wind zone of 80 mph (129 km/h) as per the Unified Building Code designation in existence at the original design and construction (1940).

The licensee stated that the original design required a design wind load corresponding to an 80 mph (129 km/h) wind with an appropriate safety factor greater than 1.5 (Ref. 17). The present Indiana Building Code design wind speed is now 90 mph (145 km/h) that would increase the load factor by 12.5 percent, which is still within the original design capability. The licensee also stated that the building was constructed with steel reinforced concrete, concrete blocks, and bricks, and the reactor pool is also constructed with steel reinforced concrete. The core is located in a pool underground, below grade level and therefore it is unlikely that high winds or snow loads would lead to reactor core damage (Ref. 17).

The structural integrity of the reactor building and the reactor pool is unlikely to be affected by hurricanes or tornadoes due to the robust design of the building and the below ground floor location of the reactor.

The PUR-1 reactor pool is located below ground level protected by the biological shield, which is a thick reinforced concrete structure surrounding the top three feet (0.9 m) of the pool tank above floor level. The NRC staff concludes that, given the meteorological data for the PUR-1 site, the location of the reactor below ground level within the biological shield, and reactor building help ensure that significant meteorological damage is very unlikely.

3.3 Water Damage

The licensee's SAR section 3.3 discusses water damage to the PUR-1 facility. The PUR-1 and the Purdue University campus is located well above the Wabash River flood plain. The Lafayette area has a relatively low probability of experiencing extreme wind conditions such as tornados or storms. The major consequence of these storm conditions is an increase in precipitation draining to the nearby Wabash River due to the higher elevation of the site. Storms are not expected to result in flooding around the reactor building. Historical records have indicated no record of any flood or standing water in the reactor operation floor. In the unlikely event that the biological shield around the top three feet (0.9 m) of the reactor tank would fail, the amount of water spilling to the operating floor would result in 3 inches (7.6 cm) of standing water in the reactor building without causing any damage to the reactor control and instrumentation. Therefore, the NRC staff concludes that there is reasonable assurance that the probability of potential damage to the reactor from water damage is extremely small.

3.4 Seismic Damage

Section 3.4 of the licensee's SAR discusses seismic damage to the PUR-1 facility. Available information on past seismic activity and the likelihood of future earthquakes in the West Lafayette area indicates that the PUR-1 facility is located in a relatively inactive seismic area. The licensee has concluded that the likelihood of a significant seismic event at the site is low. The Electrical Engineering Building Duncan Annex was built according to the then current building standard. The AEC staff determined, in the original licensing evaluation, that the site was suitable for the proposed reactor. There is no evidence to suggest that the bases for the NRC staff's conclusion have changed. In a seismic event, the reactor pool water would be contained in the reactor tank, which is a free standing unit not tied to the building structure. The NRC staff concludes that in the unlikely event of an earthquake causing catastrophic damage to the reactor building, it is extremely unlikely to cause damage to the fuel and that the radiological effects would be bounded by the licensee's MHA analysis (see Chapter 13 of this SER).

3.5 Systems and Components

Systems and components are discussed in section 3.5 of the licensee's SAR. The control rod drive assemblies, the reactor building ventilation system, and the reactor building confinement features have been identified by the licensee as part of the safety systems and components important for safe operation of the facility. The control rod drive assemblies for all control rods are mounted on the reactor bridge structure, which has been performing its intended function as designed for many years. The key components of the system are accessible for inspection and testing. Important design aspects are controlled in the TSs. The reactor control system is subject to TS-required surveillance and is part of the licensee's maintenance program. The reactor control system is discussed further in Chapters 4 and 7 of this SER.

The reactor building and the ventilation system act as a confinement system. The ventilation system reduces the consequences of releases from postulated accidents and controls the release of routine effluents. The system provides a controlled pathway for release of air from the reactor building and is manually shut down in emergencies. The reactor building serves as a barrier to release of effluents and allows a controlled pathway for release with the ventilation system. All doors are normally closed during operation per TS 5.1, Specifications d and e (see Section 6.2.1 of this SER), which, in conjunction with the ventilation system, maintains proper ventilation airflow. The ventilation system is also subject to TS-required surveillance and is part of the licensee's maintenance program.

Review of IRs and reports of reportable occurrences has shown that malfunctions are rare. Therefore, the NRC staff concludes that there appears to be no significant uncompensated deterioration of this equipment with time or operation. Based on the information provided above, the NRC staff concludes that there is reasonable assurance that the systems and components will continue to perform as designed at the increased power level and pose negligible risk to the health and safety of the public or to the PUR-1 staff.

3.6 Conclusions

Based on the information provided above, the NRC staff concludes that the PUR-1 is adequately designed and constructed to withstand all credible and likely wind, water, and seismic damage associated with the site. The design and performance of the reactor systems and components have been verified through many years of safe operation during the current license period. Accordingly, the NRC staff concludes that the reactor systems and components are adequate to provide reasonable assurance that continued operation at the increased power level will not cause undue radiological risk to the health and safety of the public, the facility staff, or the environment.

4 REACTOR DESCRIPTION

4.1 Summary Description

The PUR-1 is a heterogeneous, pool-type research reactor that operates at a licensed power level of 1 kWt. The licensee has applied for renewal of its operating license for a period of 20 years and, at the same time, for a power increase from 1 kWt to 12 kWt continuous steady state operations.

The PUR-1 reactor pool and core is composed of fuel elements in a plate-type geometric configuration that was first designed for and used in the Materials Testing Reactor (MTR). The MTR fuel plates are fabricated using low-enriched uranium (LEU) U_3Si_2-Al alloy and are clad with 6061 aluminum alloy. The reactor operates at steady-state power and has no pulsing capability. The core is moderated and cooled by natural circulation of water, and is reflected by water and graphite. The reactor core is located at the bottom of a water-filled tank and is supported by an aluminum grid plate, with additional support provided by a fixture at the top of the pool. The reactivity of the core is controlled by the operator moving the control rods that are suspended from electromagnets. Ionization chambers are used to monitor neutron and gamma-ray fluxes and these are located adjacent to the core. The reactor control console is located near the reactor and consists of standard research reactor control and monitoring instrumentation.

The PUR-1 reactor has been operating since 1962 and the core was converted from high enriched uranium (HEU) to LEU fuel in 2007. On August 13, 2006, PUR-1 submitted a proposal to convert the fuel from HEU to LEU fuel, requesting approval of the fuel conversion and of changes in the Technical Specifications (TS) (Ref. 4). After an initial review by the NRC, a request for additional information (RAI) was issued to the licensee (Ref. 5) and in a letter dated May 3, 2007, the licensee submitted responses to the RAI (Ref. 6). The order to convert was issued on August 9, 2007 (Ref. 7). The order included a SER providing the results of the NRC staff's evaluation of the licensee's conversion proposal. The order also included the changes in the TSs that would be required for operation of the facility with the LEU fuel. The only consequential changes made to the reactor during the conversion process were to the fuel assemblies. Similar reactors using the LEU MTR fuel are the University of Missouri (at Rolla), Ohio State University and the University of Florida.

The PUR-1 facility is used for teaching and research in association with the Purdue University Nuclear Engineering program and other university staff.

4.2 Reactor Core

The licensee discusses the PUR-1 core in section 4.2 of the SAR. The PUR-1 reactor core is located in the PUR-1 reactor pool and consists of a 4 by 4 square array of fuel assemblies surrounded by graphite assemblies on three sides and an irradiation facility on the fourth side. Thirteen of the fuel assemblies are normal fuel assemblies and three are control assemblies. Each of the 13 standard fuel assemblies can hold up to 14 fuel plates or a combination of fuel and dummy plates. Dummy plates are made of aluminum and contain no uranium. The three control assemblies can each hold up to eight fuel plates and can accommodate a control rod. The handle for the fuel assembly, being normal to the plates, will restrict their possible

movement. The fuel and control assemblies, as well as the startup source are contained in 6061 aluminum containers.

The reactor is controlled by three control rods located in the core region of the reactor: two shim-safety rods and one regulating rod.

The primary reactor water is de-ionized 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.

The following definitions delineate the reactor core and reactivity conditions as follows:

TS 1.0 DEFINITIONS

(…)

1.6 Core Configuration - The core configuration includes the number, type, or arrangement of fuel assemblies (elements), reflector elements, reflector element configuration, and regulating/control rods occupying the core grid.

(…)

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

(…)

1.35 Reference core condition - The condition of the core when it is at ambient temperature (cold) and the reactivity worth of xenon is negligible (<0.003 $\Delta k/k$).

(…)

1.37 Rod, control - A control rod is a device fabricated from neutron-absorbing material that is used to establish neutron flux changes and to compensate for routine reactivity losses. A control rod can be coupled to its drive unit allowing it to perform a safety function when the coupling is disengaged.

1.38 Rod, regulating - The regulating rod is a low worth control rod used primarily to maintain an intended power level that need not have scram capability. Its position may be varied manually or by a servo-controller.

1.39 Rod, Shim-Safety: The control rods used in PUR-1 as described in the definition for Rod, control.

(…)

1.43 Shutdown Margin - The shutdown margin is the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means

of the control and safety systems starting from any permissible operating condition and with the most reactive rod in the most reactive position, and the nonscrammable rods in their most reactive positions and that the reactor will remain subcritical without further operator action.

(…)

The licensee modified the definition of “core configuration” based on a RAI from the NRC staff concerning the handling of graphite plugs in aluminum tubes used for sample irradiations (See Section 10.2 of this SER)

These are standard definitions used in research reactor TSs and are therefore, acceptable to the NRC staff. It is also consistent with the guidance provided in ANSI/ANS-15.1-2007, “The Development of Technical Specifications for Research Reactors,” (Ref. 34).

TS 5.3, Specification e, states the following:

TS 5.3 Reactor Core and Fuel

Specifications -

(…)

e. The core configuration shall consist of 13 standard fuel assemblies as described in b, and 3 control fuel assemblies as described in c, and two shim-safety rods and one regulating rod.

(…)

TS 5.3, Specification e, helps ensure that the reactor core consists of thirteen standard MTR fuel assemblies and three control fuel assemblies. MTR cores have been in use for years and their characteristics are well documented. TS 5.3, Specification e, helps ensure that only MTR fuel elements are authorized to be used in the PUR-1. This design feature information is important to help ensure that the PUR-1 core consists of MTR fuel elements that have been evaluated in the SAR, and approved for use.

The NRC staff finds that TS 5.3, Specification e, characterizes the PUR-1 design features for the reactor core and helps ensure that the reactor core loading conforms to and is limited to the analysis provided in the PUR-1 SAR (Ref. 3). The NRC staff finds that TS 5.3, Specification e, is consistent with the guidance in NUREG-1537, and helps ensure that the PUR-1 will be operated consistent with the design criteria for the reactor core. Based on the information provided above, the NRC staff concludes that TS 5.3, Specification e, is acceptable.

The reactor core reactivity limitations are defined as follows:

TS 3.1 Reactivity Limits

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

a. The shutdown margin, relative to the reference core condition with the most reactive shim rod fully withdrawn, and the regulating rod fully withdrawn shall be at least 0.010 Δk/k.

b. The reactor shall be subcritical by more than 0.03 Δk/k during core loading changes.

c. No shim-safety rod shall be removed from the core if the shutdown margin is less than 0.01 Δk/k with the remaining shim-safety rod fully withdrawn.

d. The reactor shall be shutdown if the maximum positive excess reactivity of the core and any installed experiment exceeds 0.006 Δk/k.

(…)

TS 3.1, Specification a, defines the shutdown margin ensuring that the reactor can be shut down by an acceptable margin (negative reactivity) even if the highest worth shim rod is stuck out of the core and the non-scrammable regulating rod is fully withdrawn. The minimum shutdown margin of 0.01 Δk/k or greater is a value that can be easily determined by the licensee and is a sufficient amount of shutdown reactivity to give reasonable assurance that the licensee will be able to shut down the reactor under anticipated conditions and the reactor will remain shut down.

Shutdown margin is normally determined with non-secured experiments (called moveable and unsecured experiments in the PUR-1 TS) in their most reactive position. This is to help ensure that the reactor cannot be made critical by the removal of an experiment that is readily moveable. The licensee discussed not having non-secured experiments as part of the definition in RAI responses (Ref. 10). TS 3.1, Specification f, limits the total absolute value of all moveable and unsecured experiments to 0.003 Δk/k. Reducing the shutdown margin in TS 3.1, Specification a, by the value for total moveable and unsecured experiments leaves a shutdown margin of 0.007 Δk/k, which is greater than the guidance of NUREG-1537. The NRC staff finds that not having the requirement of non-secured experiments in the determination of shutdown margin is acceptable. The NRC staff notes that the value of xenon in the definition of reference core condition (TS 1.35) (0.003 Δk/k) is larger than the guidance provided in ANSI/ANS-15.1 (0.0021 Δk/k). The purpose of limiting xenon, a neutron poison that is produced by operation of the reactor and decays with time after reactor shutdown, is to help ensure that the reactor is not shut down because of a poison that will decay away. The NRC staff finds that the licensee has proposed a large shut down margin such that even after considering the reactivity value of xenon in the definition of reference core condition, the reactor would still be shut down by an acceptable margin. The NRC staff finds that the 0.01 Δk/k negative reactivity required for the shutdown margin is consistent with the guidance provided in NUREG-1537, Part 1, Appendix 14.1.

The licensee calculated the control rod worth using the MCNP model for the LEU core (Ref. 3). The worth of shim-safety control rod SS-1 is calculated to be 0.0377 Δk/k, and the worth of SS-2 is calculated to be 0.0189 Δk/k. The regulating rod (RR) worth is calculated to be 0.0023 Δk/k. The calculated worth of each control rod was compared with measured values showing that the calculated worths were consistently lower than the measured worth. Shim-safety rod SS-1 was 3.11 percent lower, SS-2 was 6.74 percent lower and the RR was 3.57 percent lower. These differences between calculated and measured control rod reactivity worths are within typical bounds of uncertainties in calculations.

The shutdown margin for the LEU core was calculated with SS-2 (the lower-worth rod) inserted and SS-1 and the RR withdrawn. Using the worth of SS-2 of 0.0189 $\Delta k/k$ and the calculated excess reactivity of 0.00351 $\Delta k/k$, the shutdown margin is 0.0158 $\Delta k/k$, which is greater than the limit of 0.01 $\Delta k/k$ specified by TS 3.1, Specification a. The shutdown margin based on measured values was 0.0018 $\Delta k/k$ higher than the calculated value.

TS 3.1, Specification a, also helps ensure proper and consistent core reference conditions for deriving the shutdown margin. The reactivity state of the reactor can be affected by the shutdown margin. The fission product xenon, which is a neutron poison, and the temperature of the reactor can affect the reactivity state 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 satisfying the shutdown margin requirement.

The NRC staff performed a simplified confirmatory analysis of the PUR-1 shutdown margin using information provided in the PUR-1 SAR (Ref. 3). These results confirm that the actual core shutdown reactivity is greater than the TS shutdown margin requirement with the maximum worth shim-safety and the regulating control rods removed from the core.

TS 3.1, Specification b, helps ensure that the reactor is subcritical by a large factor during core loading operations.

TS 3.1, Specification c, helps ensure subcriticality of the core when removing a shim-safety rod. The reactor core is subcritical by at least 0.01 $\Delta k/k$ with the other shim-safety rod fully withdrawn.

TS 3.1, Specifications b and c, help ensure that activities involving core loading and shim-safety rods are preceded by placing the core in a subcritical condition that prevents inadvertent criticality.

TS 3.1, Specification d, establishes a limit on maximum excess reactivity allowing operational flexibility while limiting the reactivity available for reactivity addition accidents conforming to the value assumed in the SAR (Ref. 3). By limiting the maximum excess reactivity it is ensured that adequate shutdown margin is available by control rod insertion. The excess reactivity for the LEU core is calculated as 0.0035 $\Delta k/k$, which is less than the TS 3.1, Specification d, limit of 0.006 $\Delta k/k$. The maximum positive excess reactivity limit assures that the consequences of reactivity transients will remain below the transient scenarios analyzed in the SAR. In addition, the maximum positive excess reactivity limit assures that the reactor period is adequate allowing the reactor to be shut down without exceeding the safety limit.

The NRC reviewed the shutdown margin established in TS 3.1, Specification a, and determined that the minimum shutdown margin is sufficient and proper controls are established to ensure that the shutdown margin requirements are satisfied when the highest worth shim-safety and regulating rods are withdrawn and the PUR-1 is in the reference core condition. These requirements provide sufficient negative reactivity to ensure that the PUR-1 can be shut down and remain in a subcritical condition from any operating condition described and analyzed in the SAR.

The NRC staff reviewed the maximum positive reactivity established in TS 3.1, Specification d, and finds that the shutdown margin and maximum positive excess reactor core reactivity value,

respectively, are supported by analyses in the SAR. Furthermore, the NRC staff finds that the use of the reference core conditions provided in TS 3.1, Specification a, is consistent with the guidance in NUREG-1537, Part 1, Appendix 14.1, Section 3.1 and ANSI/ANS-15.1-2007. On the basis of the above information, the NRC staff finds that TS 3.1, Specifications a through d, are acceptable.

The corresponding surveillance requirement for the shutdown margin is presented in TS 4.1 as follows:

TS 4.1 <u>Reactivity Limits</u>

<u>Specification -</u>

a. The shim-safety rod reactivity worths shall be measured and the shutdown margin calculated biennially with no interval to exceed 2½ years and whenever a core configuration is loaded for which shim-safety rod worths have not been measured. This may be deferred with CORO approval during any extended reactor shutdown. Additionally, if a new rod is used, its worth must be measured on the first start-up following installation. In the case of a deferred measurement, the measurement must be performed prior to resuming routine reactor operations.

(…)

TS 4.1, Specification a, helps ensure that the shutdown margin and the reactivity worth of the control rods are determined biennially to assure that the required shutdown margin is available and to provide an accurate means for determining the reactivity worth of experiments inserted in the core. The licensee stated that TS 4.1 includes an implicit requirement for excess reactivity and is determined as part of the shim-safety rod worth measurement (Ref. 3). Experience with low power level MTR-type reactors gives assurance that measurement of the control rod reactivity worth on a biannual basis is adequate to insure no significant changes occur in the shutdown margin. The licensee also performs this surveillance whenever a core configuration is loaded for which the shim-safety rod worths have not been measured or if a new control rod is used. The low power level of the reactor results in small reactivity changes due to burnup of uranium. Deferring the surveillance during reactor shutdown is acceptable because control rod worths and shut down margin do not change if the reactor is not operated. Deferred measurements must be performed prior to resuming reactor operations. Because the reactor must be operated to perform these measurements, the measurement will be performed before normal operations are resumed. These surveillance intervals are sufficient to detect changes in core behavior and help ensure that the core parameters are within their TS limits.

Before the conversion to LEU fuel, the NRC staff evaluated the fuel and core design for the LEU core including physical and chemical composition, the nuclear design, calculation methodology, core parameters, criticality, kinetic parameters, reactivity coefficients, neutronic behavior, thermal behavior, effects of burnup and temperature, and power peaking. The NRC staff also reviewed an LEU startup plan designed to experimentally determine some of these parameters. The NRC staff found that the changes in nuclear design of the core resulting from the fuel conversion were acceptable (Ref. 7).

The above TS requirements control important aspects of the design and the basic overall characteristics of the PUR-1 reactor core that are defined in more detail in other specifications

dealing with material composition and surveillance. TS 3.1 relates to the normal operating conditions of the reactor core including limits on the shutdown margin and excess reactivity. TS 4.1, Specification a, relates to the corresponding surveillance requirements. The NRC staff finds TS 3.1 and TS 4.1, Specification a, consistent with NUREG-1537 and ANSI/ANS-15.1-2007. The NRC staff also finds that the analysis presented in the SAR (Ref. 3) justifies TS 3.1 and TS 4.1, Specification a, and shows that normal operation will not lead to the release of fission products from the fuel. Based on the information provided above, the NRC staff concludes that the licensee has adequately analyzed the expected normal operation during the period of the renewed facility operating license. On the basis of the above information, the NRC staff finds that TS 3.1 and TS 4.1, Specification a, are acceptable.

4.2.1 Reactor Fuel

In Section 4.2.1 of the SAR, the licensee discusses reactor fuel. The reactor core is composed of 13 fuel assemblies containing fourteen standard MTR LEU plates installed during the conversion of PUR-1 from HEU to LEU fuel. The core also includes dummy aluminum plates identical in size to the fuel plates which are installed in both the standard and control assemblies. The dummy plates are used to control core reactivity.

Some of the major fuel parameters are shown in Table 4.1. The fuel is of MTR plate design, where the fuel meat is a dispersion fuel consisting of U_3Si_2 in aluminum. The nominal enrichment of the LEU fuel is 19.75 percent. The fuel clad material is 6061 aluminum. The LEU silicide has been approved by the NRC for use in non-power reactors with slab fuel plates (Ref. 35).

Table 4.1: Summary of Key Nominal Design Parameters LEU Core

Design Data	**LEU**
Fuel Type	MTR Plate
Fuel “Meat” Composition	U_3Si_2-Al
Fuel Enrichment U-235 (nominal)	19.75%
Cladding Composition	6061 Al
Cladding Thickness (mm)	0.381
Dummy Plate Composition	6061 Al
Dummy Plate Dimensions	Same as Fuel
Standard Fuel Assemblies Number of standard assemblies Number of plates per standard assembly	 13 14
Control Fuel Assemblies Number of control assemblies Number of plates per control assembly	 3 8

The fuel characteristics and configuration are defined in TS 5.3 as follows:

TS 5.3 Reactor Core and Fuel

Specifications -

a. The fuel assemblies shall be MTR type consisting of U_3Si_2-Al, 6061 Aluminum clad plates enriched less than 20% in the U-235 isotope.

b. A standard fuel assembly shall consist of up to 14 fuel plates containing a maximum of 180 grams of U-235.

c. A control fuel assembly shall consist of up to 8 fuel plates containing a maximum of 103 grams of U-235.

d. Partially loaded fuel assemblies in which some of the fuel plates are replaced by aluminum plates containing no uranium may be used.

(…)

TS 5.3, Specification a, specifies the maximum enrichment of the MTR fuel elements and helps ensure that the fuel is consistent with the SAR and the analysis provided in the Conversion SAR (Ref. 4). In response to RAI 11 (Ref. 22), the licensee added the composition of the fuel and the cladding to TS 5.3, Specification a. This allows a complete description of the fuel design characteristics that need prior NRC approval in the form of a license amendment to change. The Conversion SAR indicates that the enrichment may be higher by about one percent of the nominal (design) value of 19.75 (maximum 19.95 percent) percent used in the Conversion SAR and that this small increase would result in a corresponding increase in the power density of about one percent.

TS 5.3, Specification b, limits the total amount of fuel plates in a standard assembly, which also limits the maximum amount of U-235 contained in the assembly.

TS 5.3, Specification c, specifies the configuration for a control assembly ensuring that there is sufficient space available for inserting control blades with neutron absorbing material.

The U-235 LEU loading specified in TS 5.3, Specifications b and c, for the operating and control fuel assemblies are reasonable for the intended purpose of the reactor and are consistent with other LEU conversions, when configuration and power level are considered.

TS 5.3, Specification d, specifies that partially loaded assemblies would contain dummy fuel plates. The dummy plates are fabricated from aluminum. Since dummy fuel plates contain no fuel, the partially loaded fuel assembly has a lower heat generation rate than a standard fuel assembly, and therefore lower power density.

TS 5.3, Specifications a through d, describe important design aspects of the reactor core and fuel used in the PUR-1. Review of the specifications shows the important physical parameters of the fuel have been defined and adequately incorporated into the reactor control systems to ensure that the fuel temperature safety limit and excess reactivity are not exceeded. The design and configuration requirements stated in TS 5.3, Specifications a through d, meet the PUR-1 safety basis as stated in the SAR. The fuel characteristics were accepted by NRC in the

process of conversion from HEU to LEU fuel and as such are acceptable to the NRC staff (Ref. 7).

The NRC staff reviewed the information regarding the constituents, materials, and components of the fuel elements provided in the SAR (Ref. 3), RAI response (Ref. 22), and the analysis provided in the Conversion SAR (Ref. 4). On the basis of its review, the NRC staff finds that the licensee has adequately described the fuel elements used in the PUR-1, including design limits, and the technological and safety-related bases for these limits. The NRC staff concludes that compliance with TS 5.3, Specifications a through d, limits will ensure uniform characteristics and compliance with design bases and safety-related requirements.

In response to RAI 12 (Ref 22), the licensee moved fuel inspection requirements in TS 5.3, Specifications f through h, from the design feature section of the TSs and proposed a new limiting condition for operation, TS 3.6 (content of TS 5.3, Specifications g and h), Fuel Parameters, and related surveillance requirement, TS 4.6 (content of TS 5.3, Specification f), Fuel Parameters, to align fuel inspection actions with the requirements of the TSs. The operational limits on fuel are given in TS 3.6:

TS 3.6 Fuel Parameters

Specification - The reactor shall only be operated when the following specifications have been met:

a. The inspection of fuel assemblies shall be performed to identify any abnormal or previously undocumented defect present on a fuel plate. These defects may include but are not limited to blistering of the cladding on the fuel plate from elevated temperatures beyond the design of the cladding, deep scratches or gouges on the plate due to debris in the coolant flowing along the face, scratches on the edges of the plate due to insertion and removal from the assembly, discoloration from the deposition of particulates within the coolant or corrosion of the plate itself.

b. The reactor shall not operate with fuel plates that have been determined to be unsound for use as outlined above in 3.6.a. These plates shall be removed from service and the manufacturer consulted to determine possible causes.

The associated surveillance requirement is given in TS 4.6:

TS 4.6 Fuel Parameters

Specification - Representative fuel plates shall be inspected annually, with no interval to exceed 15 months. Representative is set forth to mean at least one plate from the assembly expected to have the highest burnup as well as a plate from one of the 12 remaining, non-control assemblies.

The licensee modified TS 5.3, Specification f (now TS 4.6), and added additional requirements in TS 5.3, Specifications g and h (now TS 3.6), in a response to an RAI (Ref. 22) to further clarify the requirements on fuel plate inspections.

TS 3.6, Specification a, establishes inspection requirements to detect gross failure or visual deterioration of the PUR-1 fuel plates. The fuel element attributes inspected include clad

blistering, deep scratching, and discoloration of the plates. The NRC staff finds that TS 3.6, Specification a, provides limits on fuel plate damage, which are consistent with the guidance provided in NUREG-1537, Part 1, Appendix 14.1. Based on the information provided above, the NRC staff concludes that TS 3.6, Specification a, is acceptable.

TS 3.6, Specification b, helps ensure that the PUR-1 fuel plates are not operated in a damaged condition and the licensee must remove fuel plates with damage or deterioration from the core as determined in accord with TS 3.6, Specification a. The NRC staff concludes that the licensee has used the standard definition of damaged fuel for MTR-type fuel plates and has a sufficient surveillance interval to help insure fuel plate integrity. The NRC staff review of PUR-1 annual reports from 2005 to 2015 has not identified any reported fuel failures or damaged fuel. The NRC staff finds that TS 3.6, Specification b, is typical of MTR type reactors, and consistent with the guidance in NUREG-1537, Part 1, Appendix 14.1. The NRC staff finds that TS 3.6, Specification b, helps ensure that the quality of the PUR-1 reactor fuel is maintained.

TS 4.6 establishes inspection requirements for the fuel to detect gross failure or visual deterioration by performing periodic visual inspection for bulges or other cladding defects. The periodic fuel inspection helps ensure that the fuel continues to operate with effective barriers to prevent the inadvertent release of fission products. The surveillance frequencies are consistent with NUREG-1537, ANSI/ANS-15.1-2007, and other MTR facilities. The intervals proved to be sufficient to ensure that the facility maintains fuel element integrity and that it can detect any deterioration in cladding integrity. The inspections of the fuel elements required by TS 4.6 provide adequate oversight of the physical condition of the fuel and are acceptable to NRC staff. The NRC staff reviewed the surveillance intervals provided in TS 4.6 and finds that these specifications are sufficient to help ensure that fuel element integrity is maintained and any deterioration in cladding integrity will be detected.

The NRC staff also finds that TS 3.6 and TS 4.6 are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 3.6 and TS 4.6 are acceptable.

The NRC staff asked the licensee about the need for a TS requirement on fuel burnup as recommended in the guidance of NUREG-1537. In its response (Ref. 22), the licensee demonstrated that given the low power level of the reactor the reactor would need to operate continuously for over 150 years to approach the limit. Based on the licensee's information, the NRC staff concludes that a TS limit on fuel burnup is not needed

The licensee conducts fuel plate inspections by removing fuel from the reactor pool. With the requested increase in reactor power level, the NRC staff asked the licensee about increases in dose rates from fuel undergoing inspection. In its response (Ref. 22), the licensee stated that removal of fuel from the pool for inspection will be controlled by the existing procedural limit that items leaving the pool have a dose of less than 1 R/hr at 1 foot (0.3 meters). If the fuel does not meet that requirement, it will be allowed additional time in the pool for decay. Because the existing procedural limit continues to be used, the licensee's plans for conducting fuel inspections are acceptable to the NRC staff.

The NRC staff reviewed the PUR-1 SAR (Ref. 3) that describes the fuel elements used in the PUR-1, the design limits, and the technological and safety-related bases for these limits. The NRC staff finds that the licensee adequately discussed the fuel elements in detail (materials, components, fabrication) in the SAR (Ref. 3). The design limits are identified for use in

applicable design bases to support the technical specifications. The NRC staff also finds that compliance with the applicable TSs helps ensure uniform core operating characteristics and adherence to the design bases and safety-related requirements. Based on the information provided above, the NRC staff concludes that the PUR-1 fuel elements and their associated TSs are acceptable.

4.2.2 Control Rods

Control rods are discussed in Chapter 4 of the SAR with a summary of characteristics in Section 4.2.2. Reactor power at the PUR-1 is controlled by three blade-type control rods: two shim safety rods and a regulating rod. The control rods are located within standard 6061 aluminum containers.

Table 4.2 presents the basic characteristic features of the control rods.

Table 4.2: Summary of Control Rod Characteristics

CONTROL RODS	
Number of Regulating Rods	1 – 304 stainless steel, hollow
Number of Shim Safety Rods	2 – Boron-stainless steel, solid
Operating Rates	
Regulating Rod	17.7 in/min
Shim Safety Rod	4.4 in/min
Scram	Less than 1 second (from signal to complete insertion)

The two shim-safety rods are oblong in shape and are fabricated of solid borated stainless steel, and utilize a magnetic clutch between the blades and the lead screw operated drive mechanisms. The regulating rod is made of 304 stainless steel, has an oblong shape, and a screw-operated direct drive. The regulating rod does not have scram capability. Each of the control blades is protected by an aluminum guide plate on each side within the control fuel assembly. The shim-safety and regulating rods are controlled with raise-lower switches on the reactor console, which operate mercury relays that control the rod drive motors. The control rod withdrawal speed is a fixed and the reactor operator cannot alter this value. The licensee demonstrated (Ref. 20, 22) that the maximum possible reactivity insertion rate due to control rod withdrawal is well below the maximum reactivity insertion rate analyzed in Chapter 13.

The NRC staff asked (RAI 15 of Ref. 22) the licensee about the need for a TS to control important aspects of the control rods. The licensee proposed a new TS 5.3, Specification f:

TS 5. Reactor Core and Fuel

Specifications -

(…)

f. The core shall include two shim-safety rods and one regulating rod placed within a control assembly. The two shim-safeties shall be made of solid borated 304 stainless steel. The Regulating Rod shall be stainless steel in composition.

Each control blade shall be protected by an aluminum guide plate on each side within the control fuel assemblies.

TS 5.3, Specification f, contains the important design features of the control rods that cannot be changed without a license amendment approved by the NRC. TS 5.3 Specification f, helps ensure that the design specifications and requirements for the shim safety and regulating rods are maintained as discussed in the SAR. The NRC staff reviewed TS 5.3, Specification f, and finds that the material characteristics provided in TS 5.3, Specification f, are consistent with those of other MTR-type reactors, will help ensure that the important aspects of the design of the control rods are maintained, and will help ensure that the control rods will perform their safety function. Based on the information provided above, the NRC staff concludes that TS 5.3, Specification f, are acceptable.

A scram is achieved as a result of cutoff of current to two electromagnets, each of which holds a shim-safety rod. The shim-safety rods drop into the core when the current in the two magnet amplifiers is quickly reduced to zero. The scram time is specified as one second from signal to complete insertion. The operating rates of the regulating and shim safety rods are 17.7 in/min and 4.4 in/min, respectively. The following TS helps ensure that the control rods will promptly shut down the reactor upon a scram signal:

TS 3.2 <u>Reactor Safety System</u>

<u>Specifications</u> - The two shim-safeties shall not be moved more than 6 cm from the fully inserted position unless the following conditions are met:

(…)

b. Both shim-safety and the regulating rod shall be operable.

c. The time from the initiation of a scram condition in the scram circuit until the shim-safety rod reaches the rod lower limit switch shall not exceed one second.

In response to RAIs (Ref. 22), the licensee proposed that TS 3.2 allow movement of the shim-safety rods up to 6 cm (2.4 in) from the fully inserted position without the requirements of the TS being met. This was to allow maintenance and start-up checklist activities to be performed. This shim-safety rod movement represents a small amount of positive reactivity and the reactor will still be significantly subcritical. The NRC staff has reviewed this condition and concludes that based on the small amount of reactivity added to the reactor and the limited circumstances in which the licensee will use this condition, it is acceptable to the NRC staff.

TS 3.2, Specification b, helps ensure that the appropriate conditions exist to establish shim-safety and regulating rod operability. These conditions include verification that no visible damage to the shim-safety and regulating rods is evident, the control rods are operable, and that the rod scram time meets TS 3.2, Specification c, requirement.

TS 3.2, Specification c, helps ensure that the time required for the scrammable shim-safety control rods to be fully inserted from the instant that a safety channel variable reaches the safety system setting is rapid enough to prevent fuel damage. The one second scram insertion time is typical of MTR-type research reactors as provided in the guidance in NUREG-1537, Part 1, Appendix 14.1, Section 3.2(1). The shim-safety rod insertion time was used as an input

parameter in the analysis of the insertion of excess reactivity transient presented in the SAR. The acceptability of the scram time was established by the analysis discussed in Section 13.1.2 of this SER. Based on the information provided above, the NRC staff concludes that TS 3.2, Specification c, is acceptable.

TS 3.2, Specifications b and c, help ensure that, during the normal operation of the PUR-1, 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 the PUR-1 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, Specifications b and c, support the basic design requirements to prevent reactor fuel damage and, therefore, the NRC staff concludes that TS 3.2, Specifications b and c, are acceptable.

The surveillance requirements for the control rods are given in TS 4.1 as follows:

TS 4.1 Reactivity Limits

Specification -

(…)

b. The shim-safety rods shall be visually inspected biennially with no interval to exceed 2½ years, which may be deferred with CORO approval during any reactor shutdown. If the rod is found to be deteriorated, it shall be replaced with a rod of approximately equivalent or greater worth, meeting the limiting conditions of operation specified in 3.1. In the case of a deferred measurement, the measurement must be performed prior to resuming routine reactor operations.

(…)

TS 4.1, Specification b, helps ensure that a visual inspection of the shim-safety control rods is made to evaluate corrosion and wear characteristics caused by operation of the reactor. Not having a requirement for inspection of the regulating rod is acceptable because that rod is not considered in the determination of the shutdown margin. Based on experience at reactor facilities using similar control rods, the inspections times are acceptable.

TS 4.1, Specification b, helps ensure that deferred shim-safety rod measurements are accomplished in a planned and organized manner. Experience has shown that these intervals are sufficient to help ensure operability. The NRC staff finds that TS 4.1, Specification b, helps ensure that the shim-safety rods are properly maintained and is consistent with the guidance in NUREG-1537, Part 1, Appendix 14.1, Section 4.0, and therefore, is acceptable.

Adequate response time of the shim-safety control rods is monitored by a testing program specified in TS 4.2, Specification c, as follows:

TS 4.2 Reactor Safety System

Specification -

(…)

c. Shim-safety rod drop times shall be measured annually, with no measurement's interval to exceed 15 months. These drop times shall also be measured prior to operation following maintenance which could affect the drop time or cause movement of the shim-safety rod control assembly. Drop times may be deferred with CORO approval during periods of reactor shutdown, but shall be performed prior to startup.

TS 4.2, Specification c, specifies surveillance intervals to help ensure the operability of the shim-safety rods and requires verification that shim-safety rods meet the scram time requirement specified in TS 3.2, Specification c. Annual measurement of the scram time not only checks the scram system electronics, but also indicates the capability of the shim-safety rods to perform properly. The one-second value is provided as an assumption in the safety analysis and TS 4.2, Specification c, helps ensure the operability of the control rods. The NRC staff finds that TS 4.2, Specification c, is consistent with the guidance in NUREG-1537, Part 1, Appendix 14.1, and helps ensure the operability of the control rod drives, and preserves the design assumptions used in the PUR-1 SAR for shim-safety rod scram times.

TS 3.2, Specifications b and c, TS 4.1, Specification b, and TS 4.2, Specification c, help ensure that the shim-safety control rods will promptly shut down the reactor upon a scram signal. The NRC staff finds that the intervals for control rod inspection and scram time determination are sufficient to help ensure operability and are consistent with the surveillance intervals recommended by NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 3.2, Specifications b and c, TS 4.1, Specification b, and TS 4.2, Specification c, are acceptable to help ensure the performance of the shim-safety control rods.

Based on a review of the information that the licensee provided in the SAR, as supplemented, the NRC staff concludes that the control rods conform to the applicable design bases and can shut down the PUR-1 reactor from any operating condition. Specifically, the NRC staff concludes that there is reasonable assurance that the control rods will perform as required during the renewal period to ensure fuel integrity and protect public health and safety at the requested steady-state power operation of 12 kWt. The control rod design for the PUR-1 includes reactivity worth that can control the excess reactivity planned for the PUR-1, including the assurance of an acceptable shutdown reactivity and margin.

The NRC staff finds that the licensee has evaluated appropriate design limits, limiting conditions for operation, and surveillance requirements for the control rods. Based on the information provided above, the NRC staff concludes that the TS requirements related to the PUR-1 control rods are acceptable.

4.2.3 Neutron Moderation and Reflector

The licensee discusses the PUR-1 neutron moderator and reflector in Section 4.2.3 of the SAR. The PUR-1 reactor is moderated by the light water of the reactor pool. A graphite reflector surrounds the reactor and light water reflects neutrons at the top and bottom of the core. The graphite is nuclear grade and is less than 1 part per million boron.

The water level is maintained at least 13 feet (4.0 m) above the top of the reactor core. Prior to each reactor startup the level is checked visually using a ruler mounted on the reactor tank. An alarm would be sounded and a reactor scram would occur, if the water level decreased to the point where the radiation monitors would signal an elevated dose rate.

The water level monitoring system assures that the reactor pool water inventory will be maintained. This provides assurance that water will always be available for neutron moderation. Loss of moderation, however, is not a safety issue, since that event would lead to reactor shutdown. The licensee performs visual inspections, during core configuration changes, on the graphite blocks to ensure the graphite reflectors maintain their structural integrity, and to ensure that any degradation is observed and evaluated.

The NRC staff reviewed the licensee's neutron reflector and moderator, as discussed in the SAR, and finds that they are consistent with those used in other research reactors. Based on its review of the SAR, as supplemented, the NRC staff concludes that there is reasonable assurance that the neutron moderator and reflectors will function safely in the PUR-1 core for the renewal period without adversely affecting public health and safety.

4.2.4 Neutron Startup Source

The licensee discusses the neutron startup source in Section 4.2.4 of the SAR. The primary function of the neutron source is to provide sufficient counts such that the instrumentation will function properly during startup. The PUR-1 uses a 5 Curie plutonium-beryllium neutron source, enclosed in stainless steel cladding, for startup operations. The source is located in a peripheral core position and may be removed from the core upon reactor criticality. A drive mechanism is used to lower and raise the source into and out of the core as needed.

A neutron-source clad failure would be detected during the routine analysis of pool water as required by TS 4.3, Specification a, to periodically measure the conductivity content of the reactor pool. In addition, TS 4.3, Specification b, requires measuring the radioactivity content of the reactor pool monthly. The NRC staff finds that the surveillance requirements specified in TS 4.3, Specifications a and b, are acceptable for limiting the radioactivity content of the pool water, reducing personnel exposure, and detecting potential damage to the source cladding (See Section 5.4 of this SER).

The NRC staff reviewed the information provided in the SAR, as supplemented. Based on its review, the NRC staff concludes that the PUR-1 neutron startup source is adequate to allow controlled reactor startup, and therefore, is acceptable.

4.2.5 Core Support Structure

The core support structure is discussed by the licensee in Section 4.2.5 of the SAR. The PUR-1 reactor is supported on an aluminum grid plate structure that sits on the bottom of the reactor tank. The grid plate controls the placement of fuel within the core and allows locating experiments that may be placed outside the core. The plate is fabricated of 6061 aluminum and has not shown signs of degradation during past operation.

The licensee indicates that the core support structure maintains its structural integrity and visual inspections during reactor core changes is sufficient to recognize significant degradations.

On the basis of its review of the information provided in the SAR, as supplemented, the NRC staff concludes that there is reasonable assurance that the reactor core support structure will function safely for the renewal period without adversely affecting public health and safety.

4.3 Reactor Tank

The licensee discusses the reactor pool (tank) in Section 4.3 and Chapter 5 of the SAR. The PUR-1 reactor tank consists of a 17-foot (5.2 m) deep pool that is normally filled with 6,400 gallons (24,226 liter) of water and contains the submerged reactor core. The reactor core is located in that part of the pool that is built below floor level. A 3-foot (0.9 m) concrete wall around the pool above the floor level serves as a biological shield for the operators and experimenters. The normal water level is 4 inches (10.2 cm) below the top of the tank. Heat generated from the reactor core is directly transferred to the pool water by natural convection of water through the reactor core. The pool is made watertight by a welded stainless steel liner that is affixed to the walls and floor of the concrete.

A plate at the top of the tank provides support for control rod drive mechanisms, a fission chamber, a neutron source and neutron detectors. From 1993 to 2006, the average temperature of the pool water was 26 ˚C (79 °F). The water level is maintained at least 13 feet (4.0 m) above the top of the reactor core. An alarm would be sounded and a reactor scram would occur, if the water level decreased to the point where the radiation monitors would signal an elevated dose rate.

Leakage pathways through primary coolant system piping, fittings, and other components would result in leakage into observable areas and would be discovered in a 24–72-hour time period; potential leakage would not result in an unmonitored and uncontrolled release. The licensee stated that the pool level is monitored by comparing water level to a standing ruler at the top of the pool and is observed at least once weekly (RAI 57 in Ref. 15). Because the pool water level is periodically monitored, pool water leakage will be detected.

In the event that leakage is not observable and water leaks to the environment through the steel liner and tank wall, the TS coolant radiation limit of TS 3.3, Specification d, and periodic monitoring of the pool water by TS 4.3, Specification b, helps ensure that the radioactivity level of the released water would be within the 10 CFR Part 20 limits for effluent releases. Based on the analysis presented above, the NRC staff concludes that the licensee has adequate measures in place to maintain the water level above the core.

As discussed in section 5.4 of this SER, conductivity, pH, and radioactivity content of the primary water system are routinely measured. The licensee stated and the NRC staff confirmed during an NRC site visit that there is no visible signs of degradation and no evidence of external leakage.

TS 3.3 and TS 4.3 specify the LCO and related surveillance requirement for the pool water level and temperature as follows:

TS 3.3 Primary Coolant Conditions

Specification -

(…)

b. The primary coolant shall be maintained at least 13 feet above the core whenever the reactor is operating. The primary coolant shall be maintained at least 13 feet above the top of the core or at a level sufficient for the pool top radiation monitor to indicate less than 1 mRem/hour during non-operational periods.

c. The primary coolant (bulk pool volume) temperature shall be maintained at or below 30 $°C$ while the reactor is operating.

(…)

TS 3.3, Specification b, specifies the minimum water level to be maintained above the core, which is sufficient to provide adequate shielding for personnel near the reactor during operation and provide adequate cooling to the reactor core using natural circulation. It also helps to ensure adequate shielding from pool water during periods when the reactor is shut down by limiting the radiation level at the pool top radiation monitor. This gives the licensee flexibility by allowing the water level to drop lower than 13 feet (4 m) above the core if radiation levels are maintained at an acceptable level. It ensures compliance with 10 CFR Part 20 requirements for occupational radiation exposure. An analysis of radiation doses was provided in the response to RAI 6 (Ref. 20), which demonstrated that acceptable doses are calculated for this depth of water at reactor operation at 18 kWt (12 kWt plus 50 percent margin). An acceptable DNBR analysis was provided in a response to RAI 68 (Ref. 13), which supported the adequacy of 13 feet (4.0 m) of water for cooling purposes. Based on the information provided above, the NRC staff concludes that TS 3.3, Specification c, is acceptable.

TS 3.3, Specification c, helps ensure that the pool water temperature is consistent with the assumptions used in the thermal-hydraulic calculations and specifies the limit for pool water temperature as 30 °C (86 °F). The NRC staff finds that the 30 °C (86 °F) limit specified in TS 3.3, Specification c, is consistent with the design assumptions provided in the licensee's SAR. In addition, the licensee provided an analysis in a response to RAI 5 (Ref. 21), demonstrating that the bulk pool temperature over a four hour time interval would increase less than 3 °C (5 °F) with the cooling system off. The bulk pool temperature increase would result in a reduction of the onset of nucleate boiling (ONB) power level to 96 kWt from 98.6 kWt at the maximum licensed power level with a very small change in the safety margin of the facility. The instrument check interval is four hours limiting the unobserved pool water temperature increase to less than 3 °C (5 °F). Based on the information described above, the NRC staff concludes that TS 3.3, Specification c, is acceptable.

TS 4.3 <u>Primary Coolant System</u>

<u>Specification</u> -

(…)

c. During reactor shutdown, the primary coolant level or radiation level shall be monitored monthly with an interval not to exceed six weeks. Primary coolant height shall be measured prior to reactor operation.

d. The Primary Coolant temperature shall be recorded in the log book at no interval to exceed four hours if any shim-safety or regulating rod is at a height greater than 6 cm.

TS 4.3, Specification c, specifies that the water level must be maintained 13 feet (4.0 m) above the top of the reactor core to assure adequate shielding of the reactor (response to RAI 6 in Ref. 20). The weekly level monitoring requirement (established in response to RAI 1 in Ref. 20) ensures that the assumptions in the safety analysis, Ar-41 analysis, ALARA commitments, and transient analysis are maintained and that radiological consequences are limited to below 10 CFR Part 20 limits. The weekly inspection of the reactor pool water level is used by the operators to detect an unusual change in the makeup of the pool water, which could indicate a potential leak. The NRC staff finds that TS 4.3, Specification c, is consistent with the guidance in NUREG-1537, Part 1, Appendix 14.1, and helps ensure that an acceptable inventory of the reactor pool water is maintained.

TS 4.3, Specification d, contains a requirement for the primary coolant temperature to be recorded in the log book at an interval not to exceed four hours when a control rod is at a height greater than 6 cm (0.19 feet). This helps to ensure that the primary coolant operating temperature limit of 30 °C (86 °F) given in TS 3.3, Specification c, is not exceeded during operation. Even at the proposed increased power limit of 12 kWt, the heat up of the reactor pool with the cooling system off is a slow process, less than 1 °C (1.8 °F) per hour. Taking a log of the temperature every four hours will allow the reactor operator to maintain primary coolant temperature within the limit of 30 °C (86 °F). The NRC staff finds that TS 4.3, Specification d, helps maintain primary coolant temperature within TS limits and concludes that TS 4.3, Specification d, is acceptable.

The NRC staff reviewed the information provided by the licensee and finds that the periodic inspections of the water level are adequate to help ensure that the pool water level is 13 feet (4.0 m) above the core during operation and at a level during reactor shut down periods sufficient to protect public health and safety by limiting radiation levels at the pool top. The NRC staff finds that this surveillance interval is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Therefore, the NRC staff finds that TS 3.3, Specification b, and TS 4.3, Specification c, are acceptable.

Pool water makeup is provided from the campus water system through a wall mounted water supply tank. The frequency and quantity of makeup water added to the reactor tank is recorded in a log. Any off-normal changes in the analysis of these parameters would be noticed by the operating staff and investigated (Ref. 12). The minimum detectable leak rate would depend on the location of the leak and the means by which the leak would be detected. A loss of approximately 10-15 gallons (37.8-56.8 liter) per week would be the minimum detectable leakage rate, based on PUR-1 historical primary makeup rates that typically run less than 40 gallons (151 liter) per week for normal operations (Ref. 15).

If a small leak is detected, additional water would be added or the fuel removed to storage and the tank drained to facilitate repairs. In the event of a catastrophic failure of the biological shield and the top of the reactor tank, water would spill onto the floor resulting in 3 inches (7.6 cm) of standing water without damaging any safety equipment. The accumulated water would be collected and analyzed before disposal.

In the event of an unknown failure of the primary water tank liner, the most likely leakage path would be from the primary joint failure to the gap between the tank liner and the reactor water tank, ultimately leaking into the surrounding soil.

In the unlikely event a small, unobservable leak due to a failure that releases pool water to the soil surrounding the reactor tank occurs, the impact on public safety and the environment would not be significant. The periodic monitoring of the pool water helps ensure that the radioactivity concentrations in the primary coolant are within the limits found in Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DAC) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage," to 10 CFR Part 20 for release to the sewer. Because the pool is primarily below grade and surrounded by soil, transport of water from a leak through the soil would be slow allowing for radionuclides in the water to further decay. In addition, as discussed in Section 2.4 of this SER, groundwater flow at the PUR-1 site would not take water from a leak to known wells.

Based on the information provided above, the NRC staff concludes that the likelihood of a significant release to the environment resulting from pool leakage is low. If a limited amount of primary coolant was released to the environment from the pool, the periodic testing of pool water ensures that primary coolant radioactivity levels are below 10 CFR Part 20 limits and ensures that doses to the public will be within acceptable limits.

The NRC staff concludes that the PUR-1 reactor tank provides adequate heat removal from the fuel to prevent loss of integrity under both normal operating and accident conditions. This is accomplished passively by natural convection of coolant through the core. Calculations performed in Chapter 13 of the SAR for the maximum credible loss of coolant accident demonstrate this ability. Therefore, there is reasonable assurance that the reactor pool provides a sufficient ultimate heat sink for the proposed licensed power level of 12 kWt.

The NRC staff reviewed the information provided in the PUR-1 license renewal SAR, as supplemented, regarding pool water level and quality (Ref. 3). The NRC staff finds that the water-level instrumentation and the water quality program are adequate to help ensure that the water level exceeds 13 feet (4.0 m) 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 PUR-1 reactor staff. The NRC staff concludes that the possibility of a significant release of primary coolant to the environment resulting from pool leakage is unlikely.

The PUR-1 reactor pool has withstood all mechanical loads and stresses from operation without any loss of coolant or other indications, which would impact safe reactor operation or ability to safely shut down. A review of the design and materials of the reactor pool provides reasonable assurance that it can continue to perform as designed for the period of the license renewal. The combination of the stainless steel liner of the concrete pool and the maintenance of water chemistry will minimize the potential for corrosion. The design of the reactor pool helps to ensure sufficient radiation shielding to protect operating personnel and other facility components. The design features of the pool offer reasonable assurance of its reliability and integrity for its anticipated life. The design of the pool is acceptable to avoid undue risk to the health and safety of the public.

Based on a review of the information provided in the SAR, as supplemented, the NRC staff concludes that the design of the PUR-1 reactor tank will provide adequate cooling and shielding for the reactor core. The NRC staff also finds that the PUR-1 reactor tank, including the

constituents, materials, and components of the reactor tank, are typical of other RTR reactors. Based on the information provided above, the NRC staff concludes that the design of the PUR-1 reactor tank is acceptable.

4.4 Biological Shield

The licensee discusses the PUR-1 biological shield in Section 4.4 of the SAR. The reactor biological shield consists of the concrete pool structure and the pool water. The reactor tank is located in a 14 feet (4.3 m) deep pit with the top 3 feet 4 inches (1.0 m) above floor level. A 15 inches (38.1 cm) thick concrete shell surrounds the reactor tank from the floor level to the top of the tank. The reactor tank rests on a concrete pad and is enclosed below ground level by 2 feet (0.6 m) of sand between the tank and earth. A 13 feet (4.0 m) depth of water above the core provides shielding over the core. The NRC inspection program routinely reviews the licensee's radiation protection program and performs independent measurements of radiation levels in the facility. The NRC staff reviewed PUR-1 annual reports and NRC IRs from 2005 through 2015, and finds that the annual releases and worker doses were below the limits of 10 CFR 20.1201 and 20.1301.

Based on a review of the information provided in the SAR, as supplemented, operational experience, and results from the NRC inspection program, the NRC staff concludes that there is reasonable assurance that during the renewal period, the PUR-1 biological shield will limit exposures from the reactor and reactor-related sources of radiation.

4.5 Nuclear Design

The licensee discusses nuclear design in Section 4.5 of the SAR. The information discussed in this section establishes the design bases for the content of other chapters in this SER. The following sections provide an assessment of the nuclear design analysis of the PUR-1 core at the current operating licensed power of 1 kWt and the requested increase to the steady-state thermal power level of 12 kWt.

The licensee performed its neutronics analysis with the Monte Carlo N-Particle Code (MCNP-5) that has been used extensively for nuclear reactor applications. The MCNP model solves the Boltzman transport equation using a Monte Carlo technique. MCNP5 is a state-of-the-art program used for many nuclear reactor analyses and is accepted by the NRC staff for the PUR-1 analysis. An MCNP5 model was developed for a fresh LEU core with the ENDF-VI.5 cross section library. Only fresh cores were considered since there is negligible burnup at the nominal operating power for the PUR-1. The MCNP5 model had previously been validated during the HEU-LEU fuel conversion review for the earlier HEU core by comparing calculated to measured values of k_{eff} for the core with the control rods at various positions (Ref. 4).

The licensee also used MCNP to calculate the PUR-1 core power distribution. This technique is extensively benchmarked and widely used in the RTR community for evaluating the reactor core power distribution including detailed fuel plate axial power density profiles. The information from the power distributions analysis was used as input to the PUR-1 thermal-hydraulic analysis (see Section 4.6 of this SER). The NRC staff reviewed the licensee's use of MCNP for the PUR-1 neutronic core analyses and concludes that the analysis in the PUR-1 SAR, as supplemented, satisfied the guidance in NUREG-1537.

Based on the information provided in the SAR, as supplemented, the NRC staff concludes that the computational methodology used by the licensee in the nuclear analysis of the PUR-1 is acceptable. The information discussed in this section establishes the design bases for the content of other chapters, especially the safety analysis and some of the TSs.

4.5.1 Normal Operating Conditions

Normal operating conditions are discussed by the licensee in Section 4.5.2 of the SAR. The PUR-1 nominally operates at a steady-state power level of 1 kWt with a proposed increase to 12 kWt. The following definitions delineate the operational state of the reactor:

TS 1.0 DEFINITIONS

(…)

1.25 Power Level – There are three important and separately defined power levels.

a. Instantaneous Power Level shall be the power level of the reactor at any given moment, as indicated by the reactor instrumentation.

b. The Operating Level shall be the power level from which setpoints for scram and setback shall be calculated. The Operating power level shall be 10 kW or less.

c. The Maximum Power Level shall be the maximum instantaneous power level allowed by the PUR-1 License. The Maximum Power Level shall be 12 kW, which shall not be exceeded.

(…)

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

(…)

1.32 Reactor Secured – A reactor is secured when:

a. *Either* there is insufficient moderator available in the reactor to attain criticality or there is insufficient fissile material present in the reactor to attain criticality under optimum available conditions of moderation and reflection;

b. *Or* the following conditions exist:

1. Both shim-safeties and the regulating rod shall be fully inserted

2. Electrical power to the control rod circuits shall be switched off

3. The reactor key shall be out of the key switch and under control of a licensed operator or locked in an approved location

4. No work shall be 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 shall be moved or serviced that have, on movement, a reactivity worth exceeding the maximum value allowed for a single experiment

1.33 Reactor Shutdown – The subcritical condition of the reactor where the negative reactivity, with or without experiments in place, is equal to or greater than the shutdown margin.

(…)

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

(…)

In a response to RAI 3 (Ref. 21) and RAI 20 (Ref. 22), the licensee modified TS 1.25 providing a definition for the power levels used in the PUR-1 operation. It defines a maximum power level that as that allowed by the PUR-1 license, 12 kWt. The maximum power level serves as the basis for TS 2.2 as described in Section 4.5.3 of this SER. The operating power level (10 kWt) is the power used for calculating setpoints for the power scram and setback. The reactor can be operated above the operating power level. Instantaneous power level is the power as indicated by the reactor instrumentation. The NRC staff finds that these definitions are consistent with the assumptions used in the PUR-1 SAR accident analyses and are acceptable to the NRC staff because these definition help ensure that the setpoints for the LSS and reactor safety channels are appropriately selected.

The definitions above are either standard definitions used in research reactor TSs or are facility-specific definitions that NRC staff finds to be consistent with the definitions in NUREG-1537 and ANSI/ANS-15.1-2007.

The current licensed steady-state operating power of 1 kWt and LSSS of 1.2 kWt were accepted in the safety evaluation of the conversion from HEU to LEU fuel (Ref. 7). For the requested power increase, the requested licensed power level is stated as 12 kWt, with an LSSS of 12 kWt (Ref. 1).

The steady-state power limit ensures that adequate cooling is provided for the fuel plates by natural convection of pool water. As discussed in the thermal-hydraulic analysis (Section 4.6 of this SER), operation of the PUR-1 at 12 kWt would allow for sufficient safety margins. A scram, moreover, would occur at a power level of 12 kWt or less based upon TS 2.2 and TS 3.2.

The SAR (Ref. 3), as updated in the response to RAI 5 (Ref. 20), shows by analysis that a power level of 12 kWt corresponds to a peak fuel temperature of 43.2 °C (110 °F)

(licensee's analysis was completed at 18 kWt which is 12 kWt with an additional 50 percent margin). The analysis further indicated that even at a power level of 98.6 kWt corresponding to the onset of nucleate boiling, the peak fuel temperature would be 112.6 °C (235 °F) well below the safety limit of 530 °C (986 °F).

The NRC staff finds that the licensee adequately analyzed the reactivity effects of individual core components. TSs related to the normal operating conditions of the reactor core include limits on excess reactivity, minimum shutdown margin, allowable core configurations, and surveillance requirements for the core reactivity parameters and reactivity worth of the control rods (see discussions in Section 4.2 of this SER). These TSs are consistent with the requirements of ANSI/ANS-15.1-2007 and NUREG-1537. The NRC staff finds that the analysis presented in the PUR-1 SAR adequately justify these TSs and show that normal reactor operation will not lead to the release of fission products from the fuel. Based on the information provided above, the NRC staff concludes that the licensee has adequately analyzed expected normal reactor operation during the period of the renewed license at 12 kWt. The NRC staff further concludes that the TSs provide reasonable assurance that normal operation of the PUR-1 core will not pose a significant risk to the health and safety of the public or the environment.

4.5.2 Reactor Core Physics Parameters

The licensee discusses reactor core physics parameters in Section 4.5.3 of the SAR. The major reactor physics core parameters are shown in Table 4.3. The delayed neutron fraction, β_{eff} was calculated as 0.00784 for the core. The neutron lifetime was calculated using the 1/v method with a result of 81.3 μs.

The temperature and void coefficients of reactivity were calculated by the licensee with the MCNP5 model. The temperature regimes studied were between 20 °C and 100 °C (68 °F and 212 °F) for the water and between 20 °C and 127 °C (68 °F and 260 °F) for the fuel. In all cases the temperature and the void coefficients were negative for the heating of water within the core. Heating the water only in the reflector has a small positive temperature coefficient. However, no scenarios were identified where only the water in the reflector increases in temperature. It would take a long time for any transient to heat water in the reflector and this would also lead to heating water in the core where the temperature coefficient is negative. The overall effect of the water heating would be negative. Hence, this is not considered to have a safety impact. Therefore, the NRC staff concludes that the temperature and void coefficients of reactivity in the core are acceptable.

Table 4.3: Summary of Key Reactor Parameters for PUR-1

REACTOR PARAMETERS	Calculated
Fresh Core excess reactivity (%Δk/k)	0.351
Shutdown Margin (%Δk/k)	-1.58
Control rod worth (%Δk/k) Shim-safety 1 Shim-safety 2 Regulating Rod	 3.77 1.89 0.23
Maximum reactivity insertion rate $\left[\frac{\%\Delta k}{k \cdot s}\right]$ Shim-safety 1	 1.75E-02

Shim-safety 2 Regulating Rod	8.75E-03 4.66E-03
Avg. coolant void coefficient $\left[\frac{\%\Delta k}{k \cdot \%void}\right]$	-1.93-1±7%
Coolant temperature coefficient $\left[\frac{\%\Delta \mathrm{k}}{\mathrm{k} \cdot {}^{\circ}\mathrm{C}}\right]$	-9.05E-3±9%
Fuel temperature coefficient $\left[\frac{\%\Delta \mathrm{k}}{\mathrm{k} \cdot {}^{\circ}\mathrm{C}}\right]$	-8.05E-4±10%
Effective delayed neutron fraction (%)	0.784
Neutron lifetime (µs)	81.3

The average thermal neutron flux in the PUR-1 fuel region of the core is calculated by MCNP5 as 1.38×10^{10} n/cm^{2}s, which is consistent with other MTR-type research reactors of this power level.

The values of the reactor core physics parameters (fuel temperature, coolant void, and coolant temperature coefficients) were compared to values published in the literature for MTR fuel and also with values at other MTR-type reactors and were found in the same range.

The NRC staff reviewed the licensee's analyses and finds that the licensee considered appropriate core physics parameters. The methods used to determine the values of the core physics parameters are acceptable and the values of the core physics parameters are similar to those found acceptable at other MTR fueled reactors. Based on the information provided above, 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.

4.5.3 Operating Limits

Operating limits are discussed in Section 4.6 of the SAR by the licensee. 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 upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. LSSSs for nuclear reactors are defined as settings for automatic protective devices related to those variables having significant safety functions. Where an LSSS is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded.

TS 2.1 specifies the safety limit for the PUR-1 as follows:

TS 2.1 <u>Safety Limit</u>

<u>Specification</u> - The fuel and cladding temperatures shall not exceed 530°C (986°F).

TS 2.1 specifies the maximum fuel and cladding temperatures for the MTR-type fuel plates, to ensure fuel and cladding integrity and to prevent damage to the fuel. TS 2.1 is consistent with the guidance provided in NUREG-1537, Part 1, Appendix 14.1, for MTR fuel elements that

states that NRC finds 530 °C (986 °F) an acceptable fuel and cladding temperature limit not to be exceeded under any conditions of operation.

Safety limits for nuclear reactors are based on important process variables that are necessary to reasonably protect the integrity of certain physical barriers that are meant to protect against an uncontrolled release of radioactivity. The principal physical barrier in nuclear reactors is the fuel cladding. The safety limit of the PUR-1 fuel is based on the temperature at which the fuel element would release radioactivity from the fuel plates. While the 6061 aluminum alloy has an incipient melting temperature of 582 °C (1080 °F), experiments have shown that fission products are released from irradiated fuel plates near the “blister temperature” of the cladding, which has been defined approximately as 550 °C (1022 °F) (Ref. 35). Using an element of conservatism, the safety limit for PUR-1 has been specified as 530 °C (986 °F).

The fuel and clad temperature limit is in agreement with the NRC recommendation in NUREG-1537 Part 1 (Ref. 29). The licensee has described the fuel elements in detail (materials, components, fabrication). Design limits have been discussed along with technological and safety-related bases for all applicable limits. The design limits are identified for use in applicable design bases to support the TS. The NRC staff concludes that the design and development program for the LEU MTR fuel offers reasonable assurance that the fuel will function safely for the period of the renewal operation at 12 kWt without adversely affecting the health and safety of the public (Ref. 35).

An additional consideration is the need to provide adequate cooling relative to the maximum heat flux to prevent departure from nucleate boiling and the resulting rapid increase in clad temperature which will lead to failure of the clad. A power-level limit is calculated that ensures that fuel temperatures will not be exceeded and that film boiling will not occur. The design bases analysis has shown that operation at 12 kWt with natural convection flow, will not lead to film boiling that results in high fuel and clad temperatures and the attendant loss of clad integrity (see Section 4.6 of this SER).

Review of the operating experience and the HEU to LEU conversion demonstrated that TS 2.1 is adequate to maintain the fuel and fuel cladding below temperatures at which fuel degradation would occur (Ref. 7).

The NRC staff finds that TS 2.1 is consistent with the guidance in NUREG-1537, Part 1, Section 2.1 of Appendix 14.1, and therefore, helps ensure that the fuel element cladding integrity is maintained to lessen the likelihood for the potential of cladding failure and the release of fission products. Based on the information provided above, the NRC staff concludes that TS 2.1 is acceptable.

The limiting safety system setting (LSSS) is the measured power level that, if exceeded, will initiate a scram to prevent the fuel and cladding temperatures from exceeding the safety limit.

For the PUR-1 the LSSS is set at the power level of 12 kWt.

TS 2.2 specifies the LSSS scram setting as follows:

TS 2.2 Limiting Safety System Setting

Specification - The measured value of the power level scram shall be no higher than 12.0 kW.

NUREG-1537 (Ref. 29) defines the LSSS as the calculated setpoint for a protective action which provides the minimum acceptable safety margin and includes measurement uncertainty. The licensee has shown that for steady-state power operation at 12 kWt, the margins to fuel failure are large and the LSSS of 12 kWt would conservatively protect the fuel from reaching its safety limit.

The LSSS for the PUR-1 reactor core is based upon measurement of the reactor power level. The power level is controlled so that adequate margin is maintained between the measured power level and the power at which the onset of nucleate boiling (ONB) would occur. The ONB power level is a conservative limit for the degradation in heat removal capability that would actually occur at significantly higher power as a result of the departure from nucleate boiling. As long as the power is below the predicted ONB power for any fuel element in the core, the fuel temperature would be maintained well below the blister temperature safety limit of 530 °C (986 °F).

The licensee demonstrated that at a steady-state power level of 12 kWt, an LSSS of 12 kWt conservatively maintains the fuel at a steady-state power level well below the ONB heat flux, and the fuel at a temperature well below the fuel blister temperature of 530 °C (986 °F) (see Section 4.6 of this SER). The NRC staff reviewed the licensee's methodology and calculated results and concluded that the PUR-1 TS 2.2 LSSS of 12 kWt provides acceptable margin to the safety limit of 530 °C (986 °F).

The NRC staff finds that TS 2.2 helps ensure that an LSSS is established to protect the fuel cladding integrity and prevent the release of fission products. The TS 2.2 LSSS value is supported by the SAR, as supplemented, and provides a substantial margin of safety.

Based on the information provided above, the NRC staff finds that the safety limit and LSSS for the PUR-1 are based on acceptable analytical and experimental investigations and are consistent with those approved by the NRC and used at other MTR-type reactors. On this basis, the NRC staff concludes that the LSSS power level of 12 kWt is sufficient to protect the fuel safety limit of 530 °C (986 °F) and is therefore, acceptable.

4.5.4 Conclusions

The NRC staff reviewed the licensee's nuclear design of the PUR-1 core and found that the PUR-1 core contains all the components for an operable reactor core. The NRC staff found that the licensee used input parameters justified by analysis presented in the SAR. The NRC staff reviewed the licensee's analyses and found that the licensee considered appropriate core physics parameters. The methods used to determine values of the core physics parameters are in general use for analysis of research reactors and are acceptable. The values are acceptable and are similar to those found acceptable at other MTR-fueled reactors.

Furthermore, the NRC staff finds that the PUR-1 nuclear design as described in the SAR is typical of MTR-fueled reactors, was properly documented in the SAR, and important design features were properly implemented in the appropriate portions of the PUR-1 TSs as described

above. Based on the information provided above, the NRC staff concludes that the PUR-1 nuclear design is acceptable.

4.6 Thermal-Hydraulic Design

The licensee is requested by NUREG-1537, Part 1, to present the thermal-hydraulic analyses establishing the limiting safety system settings that ensure fuel integrity under all conceivable operating conditions. The licensee performed thermal-hydraulic analyses for steady state operation of the reactor and presented the corresponding limiting safety system settings for the steady state mode of operation. The thermal-hydraulic (T-H) design of the PUR-1 was presented by the licensee in the SAR Section 4.6 and a related responses to RAIs (Ref. 3 and Ref. 20).

The licensee states that the important parameter in T-H design is the critical heat flux, which describes the heat flux associated with the departure from nucleate boiling. PUR-1 relies on natural circulation of pool water to cool the fuel plates. For steady-state thermal-hydraulic analysis the calculations were performed by a computer code, NATCON v2.0, developed at Argonne National Laboratory (Refs. 3, 4, and 20). NATCON calculates the buoyancy driven flow between the plates in an assembly, axial temperatures in the coolant and fuel plate surface and centerline, and the approach to onset of nucleate boiling (ONB). NATCON assumes incompressible laminar single-phase liquid flow that is thermally expandable. Coolant flow is determined from a balance between buoyancy and wall friction. The code calculates the power level at which the peak power plate reaches an ONB ratio (ONBR) of unity with no margin to incipient boiling. At the ONBR of unity the single-phase liquid temperature reaches the saturation temperature at the fuel plate surface leading to localized vapor-bubble formation and a two-phase flow regime with reduced heat transfer properties.

The licensee analyzed the power density distribution for the fuel plates using the MCNP code. The MCNP code calculates individual plate power levels and relative radial and axial power profiles of the individual fuel plates that were used in the thermal-hydraulic analysis.

The NATCON code uses six hot channel factors representing systematic and local uncertainties related to fuel plate and components manufacturing processes as well as other parameters that affect thermally-hydraulic performance. In addition, uncertainties were also included for the friction factor in developing flow, the effect of temperature on coolant viscosity and power variation along the width of a fuel plate.

The licensee performed T-H analyses to predict the steady-state performance of the peak power fuel plate (Plate-1348) at the current licensed power level of 1 kWt and also at 18 kWt corresponding to the proposed increased power level of 12 kWt and a 50 percent margin accounting for uncertainties. The NATCON code is widely used in research reactor analyses, and the MCNP5 code, used to calculate the plate power density distribution, is also a widely accepted computer code for reactor core analyses. The NRC staff reviewed the licensee's methodology for calculating the peak plate power and the T-H behavior and finds that the licensee's methodology is typical for analyzing the performance of research reactors with MTR type fuel plates, and is acceptable.

A pool temperature of 30 °C (86 °F) was used in the T-H analysis. This is also the temperature assumed for the coolant inlet temperature. The sensitivity of the ONB power to the pool temperature was evaluated by repeating the ONB power determination with a hypothetical pool

temperature of 35 °C (95 °F) and a hypothetical inlet loss coefficient increased by a factor 20. The ONB power was reduced from 98.6 to 79.3 kWt under the hypothetical conditions, indicating a large margin compared to the proposed PUR-1 operating power of 12 kWt. Data from the licensee shows that since 1993, the maximum pool temperature was always below 30 °C (86 °F), and therefore using a pool temperature of 35 °C (95 °F) in the calculations provides a bounding limit.

The LSSS for the reactor core were established by the licensee based upon (1) the margin between computed fuel temperature at operating power and the fuel safety limit temperature of 530 °C (986 °F), and (2) the margin between operating power and computed power at the onset of nucleate boiling.

The MCNP power density analysis indicates that Plate 1348, located in one of the central fuel assemblies (Assembly 4-4) next to one of the shim-safety rods, is the peak power plate and was evaluated in the T-H analyses. The licensee performed analyses for the most limiting, peak power plate 1348 at a number of reactor core operating power levels: 1.5 kWt (1 kWt+50 percent margin), 18 kWt (12 kWt+50 percent margin), and at the ONB power corresponding to a power level with no margin to incipient boiling. The ONB reactor power is calculated by NATCON as 98.6 kW based upon a plate-to-plate (P-T-P) spacing. The conditions at the analyzed power levels are shown in Table 4.4 (provided in RAI-5 response in Ref. 20).

Table 4.4: Operating Conditions for PUR-1, Plate 1348

Parameters	Reactor Core Power		
	1.5 kW (1 kW+50%)	18 kW (12 kW+50%)	ONB 98.6 kW
Max. Fuel Temp. (ºC)	32.3	43.2	112.6
Max. Clad Temp. (ºC)	31.3	43.2	112.5
Coolant Inlet Temp. (ºC)	30.0	30.0	30.0
Coolant Outlet Temp. (ºC)	31.6	35.4	44.8
Margin to incipient boiling (ºC)	78.5	68.1	0
Coolant Velocity (mm/s)	5.4	19.2	56.2
Coolant Mass Flow rate (kg/m^{2}s)	5.4	19.0	55.7

The T-H analysis shows that the maximum temperatures for fuel, clad, and coolant under the nominal operating power of 1.5 kWt will only be a few degrees above the coolant channel inlet temperature, while at 18 kWt the temperature rise is less than 10 °C (50 °F).

The guidance provided in NUREG-1537 suggests a ratio of two or larger for departure of nucleate boiling ratio (DNBR) for research reactor cores. The ONB power value provided by the licensee is more limiting than the power corresponding to DNB, and therefore, the DNBR is larger than the ONBR at any power level. The PUR-1 ONBR is 5.4 at 18 kWt, which is above the guidance provided in NUREG-1537 for DNBR. The steady-state T-H calculations demonstrate large safety margins to the thermal limits of ONB power and fuel temperature at both 1 kWt and 12 kWt operating power.

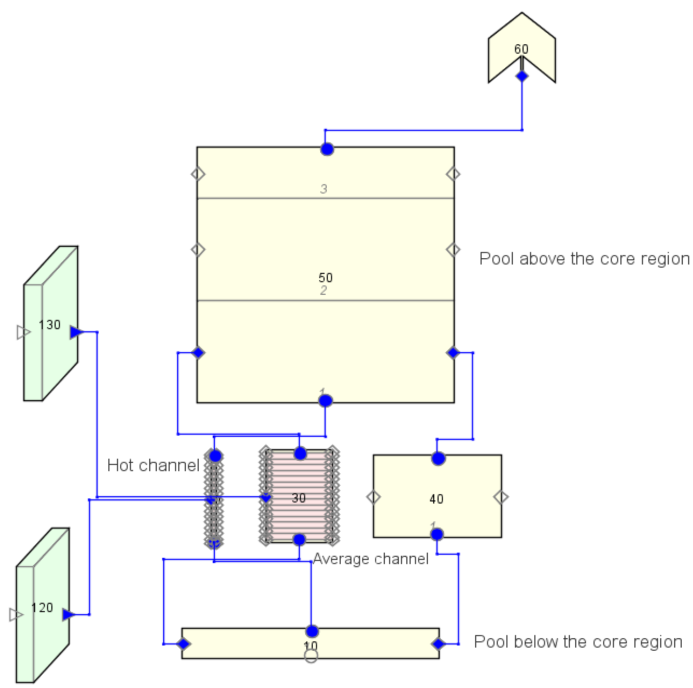
The LSSS setpoint is established at the power of 12 kWt. The licensee's T-H analysis demonstrated the acceptability of this setpoint. The 12 kWt power level was used consistently for safety analysis, ONBR analysis, and transient analysis. Each scenario resulted in acceptable consequences with the fuel temperature remaining below the fuel safety limit of

530 °C (986 °F). The calculated maximum fuel temperature was 43.2 °C (110 °F) for the LSSS at 12 kWt. Based on the information provided above, the NRC staff concludes that the proposed LSSS setpoint of 12 kWt is acceptable.

NRC Confirmatory Analysis

The NRC staff performed confirmatory calculations to verify the magnitude of the ONB conditions using the TRACE computer code (Ref. 37). These results demonstrate that at the analyzed operating power conditions of 1 kWt and 12 kWt large margins exist to the core power corresponding to the onset of nucleate boiling. Additionally, large temperature margins exist to the incipient boiling and the fuel safety limit of 530 °C (986 °F). The model used by the NRC staff is displayed in Figure 4-1.

Figure 4-1: PURDUE University Training Reactor - TRACE Model



The TRACE model was developed using the physical characteristics from the PUR-1 SAR. The model consists of five pipe components, two heat structure components, two power components (#120 and #130), and one heat sink simulating the PUR-1. The pipe components represent the reactor pool (Pipe 10, 40, 50), the hot channel (Pipe 20, Plate-1348) together with a heat structure and power component (#120). The average channel represents the remaining 190 flow channels (Pipe 30) connected to another heat structure with a power component (#130).

The licensee's MCNP core power distribution analysis indicated that the peak power plate, plate-1348 has a total plate power of 80.7 W at a total core power of 10 kWt. The licensee's NATCON T-H analysis is performed at 18 kWt. The 18 kWt total core power corresponds to 145.3 W (80.7x1.8) peak plate power that is also used in the TRACE simulations. The model

also included the axial power distribution profile in the hottest plate as calculated by the MCNP code and a 3 percent reduction in the hot channel flow area for conservatism.

The steady-state results at 18 kWt with a 145.3 W peak plate power are presented in Table 4.6.

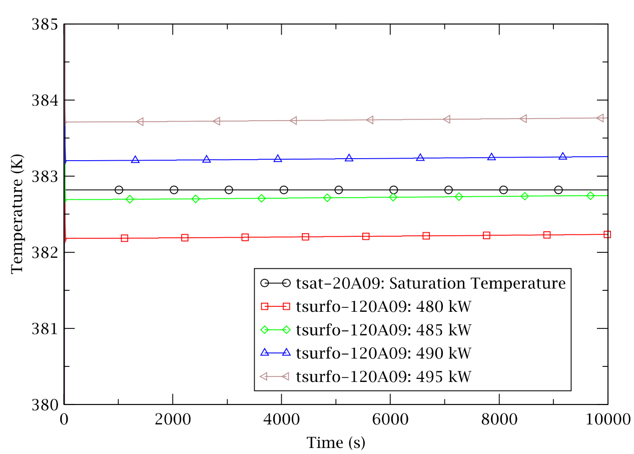
Table 4.6: PUR-1 Steady-State Conditions at 18 kWt TRACE vs. NATCON

Variable	TRACE	NATCON
Maximum fuel temperature (°C)	40.7	43.2
Maximum clad temperature (°C)	40.7	43.2
Coolant inlet temperature (°C)	30.0	30.0
Coolant outlet temperature (°C)	38.9	35.4
Margin to incipient boiling (°C)	--	68.1
Coolant velocity (mm/s)	13.2	19.2
Coolant mass flux (kg/m²·s)	13.2	19.0

The NRC's TRACE model predicts a lower coolant velocity resulting in a slightly higher exit coolant temperature. The maximum fuel temperatures as calculated by TRACE are somewhat lower than in the NATCON model due to differences in the fuel heat conduction model. The predicted maximum fuel and clad temperatures are much lower than the blister temperature of 550 °C (1022 °F). TRACE doesn't have the capability to evaluate the NATCON calculated margin to incipient boiling, which is 68.1 °C (155 °F) at 18 kWt.

The ONB conditions are simulated by TRACE by increasing the peak plate power until the peak cladding temperature equals the fluid saturation temperature corresponding to the onset of boiling. Figure 4.2 compares the peak cladding temperature at different reactor core powers with the fluid saturation temperature. The results indicate that the ONB does not occur in the hot channel with the total power level of up to 485 kWt.

Figure 4.2: Peak Cladding vs. Saturation Temperature



Evaluation of DNBR

The licensee presented analysis using the ONB power level, which is more limiting than the power level corresponding to the DNB. However, for completeness, the NRC staff also calculated the DNBR for the PUR-1 using the TRACE calculated parameters. The DNBR values were evaluated under steady-state conditions with the reactor power of 18 kWt. Two Critical Heat Flux (CHF) correlations, which are appropriate to plate-type fuel assemblies, are used: a) the Bernath correlation, and b) the Sudo-Kaminaga correlation that was experimentally developed for vertical rectangular channels (Ref. 38 and Ref. 39).

Table 4.7 shows the DNBRs evaluated using the two correlations under normal operating conditions with the reactor power of 18 kWt.

Table 4.7: PUR-1 DNBRs at 18 kWt, Steady State

Variable	Bernath	Sudo-Kaminaga
Highest heat flux of hottest plate (W/m^2)	2,905.4	2,905.4
Critical Heat Flux (W/m^2)	1,028,302	159,500
DNBR	354	55

The DNBR values at the proposed operating power level are significantly greater than the guidance in NUREG-1537 (DNBR greater than 2) and also well above than the ONBR value calculated by the licensee (ONBR = 5.4).

Sensitivity Analysis

Additional confirmatory analyses were performed to see the effects of a) increase in wall friction due to the actual rectangular flow channels modeled as circular pipes and b) an increase in the hot channel flow area by 25 percent from the base calculations. The changes in wall friction were modeled by adding a loss coefficient (K-factor) to the hot and average channel internal junctions (K_{base}=0.0 vs. K_{sens}=1.5). The results are shown in Table 4.8.

Table 4.8: Effects of Increasing Wall Friction and Flow Area on DNBR

Variable	Base calculation	Increased wall friction (K=1.5)	Increased flow area (25%)
Maximum fuel temperature (°C)	40.7	41.3	38.8
Maximum clad temperature (°C)	40.7	41.3	38.8
Coolant inlet temperature (°C)	30.0	30.0	30.0
Coolant outlet temperature (°C)	38.9	39.6	36.7
Coolant velocity (mm/s)	13.2	12.2	14.0
Coolant mass flow rate (kg/s)	0.0039	0.0036	0.0051
DNBR - Bernath	354	352	462
DNBR - Sudo-Kaminaga	55	55	69

The effect of higher wall friction on the T-H behavior (especially DNBR) is negligible under normal operating conditions at 18 kWt total reactor power. Increasing the hot channel flow area increases the DNBR value leading to less severe conditions.

The NRC staff reviewed the licensee's analyses in the SAR as described above. The NRC staff finds that the PUR-1 T-H design analysis was consistent with other MTR-type reactors, adequately described in the PUR-1 SAR, and properly controlled and implemented in the PUR-1 TSs. The NRC staff finds that the licensee used qualified analytical methods and conservative or justifiable assumptions. The NRC staff confirmatory calculation of the PUR-1 ONB and DNBR values provided comparable results. Based on the information provided above, the NRC staff concludes that the T-H analysis in the SAR, as supplemented, demonstrates that the PUR-1 reactor can operate at 12 kWt with acceptable T-H safety margins.

4.7 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 PUR-1 core without undue risk to the health and safety of the PUR-1 staff, 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, and its operational limitations as identified in the TSs. The NRC staff concludes that the PUR-1 SAR, as supplemented, demonstrates that the PUR-1 core has acceptable safety margins for thermal-hydraulic conditions.

The NRC staff also finds that the licensee's analyses use qualified calculation methods and conservative or justifiable assumptions. The applicability of the analytical methodology is demonstrated by comparing analytical results with independent confirmatory analysis. The NRC staff reviewed the analysis of the steady-state operation of the PUR-1 core at a power level of 12 kWt and finds that the maximum core fuel temperature remains below the limit set by the known mechanical and thermal properties of the fuel. The NRC staff finds that the PUR-1 TSs on the reactor design, reactor core components, reactivity limits, and related surveillance requirements provide reasonable assurance that the reactor will be operated safely in accordance with the TSs. Based on the information provided above, the NRC staff concludes that there is reasonable assurance that continued operation of the PUR-1, up to 12 kWt, as limited by the TSs, would not pose undue radiological risk to the facility staff, the public, or the environment.

5 REACTOR COOLANT SYSTEMS

5.1 Summary Description

The licensee discusses the reactor coolant systems in Chapter 5 of the SAR. Reactor coolant systems at the PUR-1 consist of those systems that remove heat from the reactor pool, maintain water purity, water clarity, and provide makeup water. The design of the reactor coolant systems at the PUR-1 is typical of other pool type research reactors.

The reactor coolant system design characteristics and configuration are defined in TS 5.2 as follows:

TS 5.2 Reactor Coolant System

Specifications -

a. Primary Cooling System – The PUR-1 primary cooling system shall be a pool containing approximately 6,400 gallons of water.

b. Process Water System – The process water system shall be assembled in one unit and contains a pump, filter, demineralizer, valves, flow meters, and a heat exchanger (see Section 5.2.d). The demineralizer shall contain a removable cartridge that is monitored continuously for radioactivity buildup.

c. Primary Coolant Makeup Water System – Makeup water for the pool shall be taken batchwise from the Purdue University water line and is passed through the demineralizer enroute to the pool. A vacuum breaker shall exclude any possibility of siphoning pool water into the supply line. The pool makeup water system, in addition to the demineralizer, also shall include a normally closed manual shutoff, solenoid valve and a check valve.

d. Primary Coolant Chiller System – The chiller shall be designed with three loops. Pool water shall pass through the primary loop, a Freon refrigerant in the secondary loop, and water from the building water supply shall be used to remove heat, which shall then be discharged to the building sewer system. The heat-removal capacity of the heat exchanger shall be at least 10.5 kW or greater.

TS 5.2, Specification a, describes the primary coolant system at the PUR-1 consisting of a 17-foot (5.2 m) deep concrete pool, normally filled with 6,400 gallons (24,226 liter) of water, in which the reactor core is submerged. Heat generated from the reactor core is directly transferred to the pool water by natural convection. The cooling system is designed to remove sufficient heat from the reactor pool to allow all licensed operations without exceeding the established bulk coolant temperature limit of 30 °C (86 °F) as required by TS 3.3, Specification c (see Section 4.3 of this SER). The primary coolant above the reactor also provides shielding reducing gamma ray exposures to the operating staff at the operating floor level.

TS 5.2, Specification b, describes the process water cleanup system that controls primary coolant quality and maintains pool water conductivity and controls corrosion rate. The NRC

staff has reviewed the design of the process water cleanup system and concludes that the system is sufficient to maintain and control the quality of the primary coolant water.

TS 5.2, Specification c, describes the primary coolant makeup water system that provides replacement for primary water that is lost through evaporation. The NRC staff has reviewed the design of the primary coolant makeup water system and concludes that the system is sufficient to replace normal primary water loss and will protect against the entry of primary coolant into the city water system.

TS 5.2, Specification d, describes the primary coolant chiller system that provides the heat removal capability for the water in the reactor pool. Heat is removed from the shell side of a heat exchanger and includes additional coolant loops. Design features of this system allow transfer of reactor heat from the primary system under all operating conditions. The NRC staff has reviewed the design of the primary coolant chiller system and concludes that the system is sufficient to remove heat generated by the operation of the reactor core.

TS 5.2, Specifications a through d, describe the facility cooling systems. These systems are discussed in greater detail below. The NRC staff finds that TS 5.2, Specifications a through d, help ensure important features of the physical design of the PUR-1 cooling system are maintained, describes the PUR-1 cooling systems, and is consistent with the license renewal SAR (Ref. 3). The NRC staff finds that TS 5.2, Specifications a through d, are consistent with the guidance in NUREG-1537 and ANSI/ANS 15.1-2007. Based on the information provided above, the NRC staff concludes that TS 5.2 is acceptable.

5.2 Primary Coolant System

The primary coolant system is discussed in Section 5.2 of the licensee's SAR. The primary coolant system at the PUR-1 consists of a 17-foot (5.2 m) deep concrete pool, normally filled with about 6,400 gallons (24,226 liter) of water, in which the reactor core is submerged. Heat generated from the reactor core is directly transferred to the pool water by natural convection. The T-H analysis discussed in Section 4.6 of this SER shows that the heat produced by the reactor can be safely dissipated to the primary coolant by natural convective water flow.

Water temperature in the reactor pool is kept between 16 - 27 °C (60 - 80 °F) by a closed loop cooling system with a design flow rate of 30 gpm (114 l/min). A pump takes water from a 1.5-inch (3.8-cm) pipe located in the reactor pool, passes it through the tube side of a stainless steel heat exchanger, and returns it through a 1.5-inch (3.8-cm) pipe to the reactor pool, 4 feet (1.2 m) below the surface.

A flow indicator measures the primary coolant flow rate and water purity is measured by a conductivity instrument installed in the primary coolant piping. The primary coolant system is controlled remotely from the control console in the reactor room.

Pool water is extracted through a scupper drain and PVC suction line by a circulating pump, then passed through a micron filter and demineralizer, and returned to the reactor pool. The scupper drain penetration is above the operating floor level that is well above the reactor core elevation.

Water level in the tank is normally maintained at least 13 feet (4.0 m) above the reactor core as specified by TS 3.3, Specification b, and TS 4.3, Specification c (see Section 4.3 of this SER).

The water level is controlled to provide a minimum level of radiation shielding of the reactor core. Radiation surveys with the reactor at full power show that a sufficient level of shielding is provided above the core.

The NRC staff has reviewed the primary coolant system and concludes that the system is designed in accordance with the design bases derived from the T-H analysis in the SAR. The system is designed to remove sufficient heat from the reactor pool to allow all licensed operations without exceeding the established bulk coolant temperature limit. The NRC staff also concludes that sufficient shielding is provided by the coolant above the reactor when coolant levels are controlled in accordance with the TS. Based on the information provided above, the NRC staff concludes that the design and operation of the primary coolant system is acceptable.

5.3 Secondary Coolant System

The licensee discusses the secondary coolant system in Section 5.3 of the SAR. The secondary coolant system provides the heat removal capability for the water in the reactor pool. Heat is removed from the shell side of a heat exchanger by a 36,000-Btu/hr (10.6-kW) water chiller compressor-condenser unit. The chiller is a three loop design with Freon refrigerant circulated in the shell side of the heat exchanger and campus water removing heat from the refrigerant and discharging to the campus sewer system. In addition, the secondary side of the heat exchanger is operated at the higher pressure than the primary side. The licensee provided additional information about system leaks in the response to RAI 25 (Ref. 22). Any tube leak would result in Freon in the primary system which would be detected by the licensee. Because Freon is not soluble in water, Freon would be visible on the pool surface. This three loop system prevents the spread of contamination in the case of a tube failure in the heat exchanger, since the chance of contamination passing through a three loop system to the environment is small. The secondary coolant system design allows the transfer of reactor heat from the primary system under all operating conditions. Malfunctions of this system will not lead to fuel failure or a subsequent uncontrolled release of radioactivity to the environment.

Although the secondary coolant system only has an operating capacity of approximately 10.6 kW of thermal energy, and the proposed steady-state operating condition of the PUR-1 is 12 KWt, the NRC staff finds that the pool temperature limit imposed by TS 3.3, Specification c, of 30 °C (86 °F) will ensure that the licensee suspends reactor operation if the secondary coolant system is unable to maintain the bulk primary pool coolant temperature at the TS limit. Based on the information provided above, the NRC staff concludes that PUR-1 operation with the bulk primary pool coolant temperature limit imposed by TS 3.3, Specification c, is acceptable.

Because the secondary coolant system is maintained at a higher pressure than the primary coolant system when in operation and is designed as a three-loop system, the NRC staff concludes that there is reasonable assurance that in the event of an internal failure of the heat exchanger primary coolant would be contained within the PUR-1 facility. Based on the information provided above, the NRC staff concludes that the design and operation of the secondary coolant system is acceptable.

5.4 Primary Coolant Cleanup System

The primary coolant cleanup system is discussed in Section 5.4 of the licensee's SAR. Primary coolant discharged from the primary coolant pump is passed through a cartridge-type filter for

purification, then through a demineralizer to control conductivity. A flow meter and conductivity indicator provide a check on flow rate and water purity discharging from the demineralizer and returning to the reactor pool.

TS 3.3 specifies the reactor pool water quality requirements that minimize corrosion of the fuel element cladding and minimize neutron activation of any dissolved materials in the pool water as follows:

TS 3.3 Primary Coolant Conditions

Specification -

a. The primary coolant resistivity shall be maintained at a value greater than 330,000 ohm-cm.

(…)

d. The primary coolant radiation levels shall not exceed the levels for water in 10 CFR 20 Appendix B, Table 2.

TS 3.3, Specification a, helps ensure that the resistivity of the bulk pool water in the tank is maintained at or higher than 330,000 omh-centimeter to control the potential corrosion of reactor components. Limiting resistivity helps extending the longevity and integrity of the fuel clad. The licensee stated in the SAR, as supplemented, 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. The licensee also states that the corrosion limits help ensure that oxide buildup on the cladding will not reduce the heat transfer between the cladding and coolant.

The licensee proposed eliminating the LCO and surveillance requirement on primary coolant pH. The NRC staff completed an analysis (Ref. 51) that demonstrated that a conductivity limit no greater than 5×10^{-6} mhos/cm will ensure that the pH range is limited to 5.6 to 5.8, which is consistent with the guidance provided in NUREG-1537 to maintain the pH range of 5.0 to 7.5. Since the licensee chose a conductivity limit within 5×10^{-6} mhos/cm, there was no need for a TS requirement to limit the reactor pool water pH.

TS 3.3, Specification d, helps ensure that the radioactive content of the primary coolant is maintained at low level. The radioactive content of the primary coolant shall not exceed the levels for water in 10 CFR Part 20, Appendix B, Table 2. The effluent concentrations given in Table 2 for water are concentrations that can be released to the environment resulting in doses within the public dose limits in 10 CFR Part 20.

The licensee proposed a resistivity limit that is used by the other MTR fueled reactor licensees as a longstanding value for research reactors, which has been shown to be effective in controlling corrosion in aluminum and stainless steel systems. These limits also help to maintain the concentration of activation products to acceptably low levels such that this level does not pose a significant radiological hazard.

The licensee proposed a radioactive limit on primary water that meets the water effluent limits in 10 CFR Part 20. This helps ensure that if primary coolant is released to the environment, the

resulting doses will be within the limits of 10 CFR Part 20. Because primary coolant radioactivity content is limited to 10 CFR Part 20 limits, TS 3.3, Specification d, is acceptable to the NRC staff.

The NRC staff reviewed TS 3.3, Specifications a and d, and finds that TS 3.3, Specifications a and d, are consistent with the guidance in NUREG-1537, Section 5.4 and the design assumptions described in the PUR-1 SAR. Based on the information provided above, the NRC staff concludes that TS 3.3, Specifications a and d, are acceptable.

The surveillance requirement for primary coolant conductivity, and measurement of radionuclides is given in TS 4.3 as follows:

TS 4.3 Primary Coolant System

Specification –

a. The conductivity of the primary coolant shall be recorded monthly, not to exceed six weeks. This cannot be deferred during reactor shutdown.

b. The primary coolant shall be sampled monthly, not to exceed six weeks, and analyzed for gross alpha and beta activity. This cannot be deferred during reactor shutdown.

(…)

TS 4.3, Specifications a and b, apply to the surveillance requirements for monitoring the pool water and the water-conditioning system. The objective is to assure the integrity of the water purification system, thus maintaining the purity of the reactor pool water, eliminating possible radiation hazards from activated impurities in the water system, and limiting the potential corrosion of fuel cladding and other components in the primary water system. The licensee has modified TS 4.3, Specifications a and b, to indicate that the measurements cannot be deferred during reactor shutdown (Ref. 12).

TS 4.3 Specification a, helps ensure periodic monitoring of the primary coolant water conductivity to alert the operators of any changes in the primary coolant water chemistry. TS 4.3, Specification a, helps maintain a suitable chemical environment for the core components. The NRC staff finds that TS 4.3, Specification a, is consistent with the guidance in NUREG-1537, Part 1, Appendix 14.1, and helps ensure that the quality of the reactor water is maintained.

TS 4.3, Specification a, specifies the monthly surveillance requirement for the primary coolant conductivity. The surveillance is not postponed, even if the reactor is shut down. It provides for periodic monitoring of primary coolant water conductivity to provide timely information of possible changes in primary coolant water chemistry. Experience has shown the surveillance interval to be acceptable for ensuring acceptable water conductivity. Therefore TS 4.3, Specification a, is acceptable to the NRC staff.

TS 4.3, Specification b, helps ensure that the radionuclide content of the pool water is measured periodically in order to detect increased levels of radioactivity content. In addition, periodic monitoring of the pool water helps ensure that the water is within 10 CFR Part 20 release limits under any circumstances or condition of operation. The NRC staff finds that TS 4.3,

Specification b, is consistent with the guidance in NUREG-1537, Part 1, Appendix 14.1 and ANSI/ANS-15.1, and therefore, acceptable to the NRC staff.

The NRC staff finds that the surveillance intervals specified in TS 4.3, Specifications a and b, are consistent with guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 4.3, Specifications a and b, and the PUR-1 design and operation of the primary coolant cleanup system are acceptable.

5.5 Primary Coolant Makeup Water System

The primary coolant makeup water system is discussed in Section 5.5 of the licensee's SAR. Additional information was provided in response to a RAI from the NRC staff (Ref. 22). Reactor pool level is maintained at least 13 feet (4.0 m) above the reactor core during operation. Makeup for pool evaporation and potential minor, undetectable leakages from the reactor coolant system is provided by a 20 gallon (76 liter), gravity driven reservoir filled from the campus water system as necessary after being processed through a mixed bed demineralizer. This tank is open to the atmosphere and has an overflow pipe which eliminates the possibility of over pressurizing the tank or flowing water into the university water supply line. Approximately 40 gallons (151 liter) of makeup water a week are required to offset evaporation losses. Tank discharge to the reactor pool is controlled by a solenoid valve controlled by a float switch on the reactor pool surface which also turns on an LED that indicates water needs to be added to the pool. Water additions are made by turning on the solenoid valve manually in order to track water additions. The automatic pool water level control is only activated when needed to replace evaporated coolant inventory, and the automation is solely to prevent overfill of the pool.

The NRC staff has reviewed the design of the primary coolant makeup water system and concludes that the system is sufficient to replace normal primary water loss and will protect against the entry of primary coolant into the campus water system.

5.6 Nitrogen-16 Control System

The licensee discusses the Nitrogen-16 (N-16) control system in Section 5.6 of the SAR. N-16 at the PUR-1 is controlled through decay by travel time from the reactor core to the surface of the reactor pool. The reactor core is cooled by natural convection of the demineralized water in the reactor pool. The reactor exit coolant velocity at 18 kWt power level is less than 2 cm/sec (see Table 4.6 in Section 4.6 of this SER), and even without any mixing, a water column would take more than 198 sec to reach the pool surface. Since the half-life of N-16 is 7.14 seconds, the N-16 activated in the core would decay before reaching the pool surface. There has been no N-16 activity observed in the reactor room since PUR-1 was commissioned.

As discussed in Chapter 11 of this SER, TS 3.2, Specification a, requires the operation of the reactor top area radiation monitor and TS 4.2, Specification a, requires a daily a channel check will be performed daily. The NRC staff concludes that the use of the reactor top area radiation monitor helps ensure that the radioactivity associated with N-16 generated in the reactor coolant is monitored by the reactor operators, who can respond to any unusual increase. Based on the information provided above, the NRC staff finds that the use of the reactor top radiation monitor acceptable to detect N-16 activity produced by the reactor.

The NRC staff concludes that the decay from the long traveling time of N-16 to the pool surface along with the licensee's radiation protection and ALARA programs provide sufficient reduction

of radiation fields at the top of the reactor tank from N-16 to maintain personnel exposures below the limits in 10 CFR Part 20.

5.7 Auxiliary Systems Using Primary Coolant

There are no auxiliary systems at the PUR-1 that use primary coolant.

5.8 Conclusions

The NRC staff reviewed and evaluated the design and operation of the PUR-1 reactor coolant systems at the proposed increased power level as part of its review of the licensee's SAR. Based on the information provided above, the NRC staff concludes that the design and operation of the reactor coolant systems are acceptable for the following reasons:

- The PUR-1 reactor coolant systems are adequate to remove heat from the fuel and prevent loss of fuel integrity under normal full-power operating conditions at 12 kWt.

- There is reasonable assurance that any accidental leakage from the primary coolant system would be contained within the PUR-1 facility. As a result, there would not be significant radiation exposure to the public in the event of such leakage.

- The TS provide reasonable assurance that the cooling system will operate as designed and be adequate for reactor operations as described in the SAR.

- The water purification system will control chemical quality of the primary coolant to limit corrosion of the reactor fuel and other systems that contact primary coolant.

- The design of the N-16 control system along with the licensee's radiation protection and ALARA programs provides an effective method to reduce the level of radioactivity at the top of the reactor tank from N-16, and helps to maintain personnel exposure below the limits in 10 CFR Part 20.

6 ENGINEERED SAFETY FEATURES

6.1 Summary Descriptions

The licensee discusses engineered safety features in Chapter 6 of the SAR. The licensee has demonstrated that no postulated accident analyzed in Chapter 13 of the SAR would result in fuel damage significant enough to require an active emergency core cooling system. The low operating power level (12 kWt) of the PUR-1 ensures that fission product inventories are minimal and would be well contained in the fuel with just natural convection cooling from the 6,400 gallons (24,226 liters) of demineralized water in the reactor pool or air cooling if all water were to be lost.

Any airborne radioactivity from an operational event would be below 10 CFR Part 20 limits, and further reduced through release through a controlled path with dilution by the reactor room free air volume prior to exhausting via the reactor room exhaust stack on the building roof.

6.2 Description

6.2.1 Confinement

Confinement is discussed in Section 6.2.1 of the licensee's SAR. The PUR-1 reactor building and reactor building ventilation system act as a confinement. The reactor room air is supplied by the building's commercial grade ventilation system discussed in Section 9 of this SER. There are three area radiation monitors within the reactor room that alarm at a predetermined level of airborne radiation in the room. If an alarm occurs, the reactor operator has a switch on the control console that will shut down the exhaust fan and close dampers in the air ducts isolating the reactor room. Dilution with the reactor room free air volume is used to ensure operational and offsite dose rates are within regulatory limits during both normal operations and under postulated accident conditions.

ANSI/ANS-15.1 defines a confinement as an enclosure of the overall facility that is designed to limit the release of effluents between the enclosure and its external environment through controlled or defined pathways. As shown in Chapter 13 of this SER, the confinement and ventilation systems are not required to keep doses during potential accident conditions within regulatory limits. The confinement and ventilation systems help to control releases from the reactor building and serve to reduce doses to members of the public during normal operation and potential accident conditions. The NRC staff concludes that the confinement and ventilation systems as used at the PUR-1 are not engineered safety features.

The licensee defines confinement as follows:

> TS 1.0 DEFINITIONS
>
> (…)
>
> 1.5 Confinement – Confinement is an enclosure of the overall facility that is designed to limit the release of effluents between the enclosure and its external environment through controlled or defined pathways.

(…)

This definition is the standard definition of confinement used in research reactor TSs, consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007, and is therefore acceptable to the NRC staff.

TS 5.1 states important design features of the confinement as follows:

TS 5.1 Site Description

(…)

e. The PUR-1 reactor room shall be a closed room designed to restrict leakage.

(…)

h. Openings into the reactor room shall consist of the following:

1. Three personnel doors.
2. One door to a storage room with no outside access.
3. Air intake
4. Air exhaust
5. Sewer vent

TS 5.1, Specifications e and h, are the requirements to house the reactor in an enclosed building that restricts leakage and defines the openings in the confinement. TS 5.1, Specifications e and h, support the accident analysis by providing parameters necessary to support the assumptions used to demonstrate compliance with 10 CFR Part 20 requirements.

The licensee modified TS 5.1, Specifications c through e, providing a better description on reactor access and authorization (Ref. 12) (TS 5.1, Specification c, is discussed in Section 2.1.1 of this SER). The NRC staff review finds that the design of the reactor building is adequate to ensure the health and safety of the facility workers and the public.

TS 5.1, Specifications e and h, help ensure important features of the physical design of the facility used to house the PUR-1 are maintained by the licensee. These specifications support the accident analysis by providing parameters necessary to support the assumptions used to demonstrate compliance with 10 CFR Part 20 requirements. The NRC staff finds that TS 5.1, Specifications e and h, are consistent with the guidance in NUREG-1537 and ANSI/ANS 15.1-2007. Based on the information provided above, the NRC staff concludes that TS 5.1, Specifications e and h, containing the basic design requirements for the confinement system, are acceptable.

TS 3.4 specifies the requirements to maintain confinement and provide restrictions on the release of airborne radiation to the environment as follows:

TS 3.4 Confinement

Specification -

a. During reactor operation and when radioactive material is being handled with potential for airborne release, the following conditions shall be met:

    1. (see Section 9.1 of this SER)

    2. All exterior doors in the reactor room shall remain closed except as required for personnel, equipment, or materials access.

b. All inlet and exhaust air ducts and the sewer vent shall contain a HEPA filter or its equivalent.

(…)

TS 3.4, Specification a, item 2, specifies one of the criteria required to establish an operable confinement, which limits the consequences of a potential release to below 10 CFR Part 20 limits. TS 3.4, Specification a, item 2, helps ensure that the confinement of the reactor building is in effect during reactor operation and when material is being handled that may have the potential for airborne release. If a potential radioactivity release occurs, the consequences would be minimized through the control of air flow into and out of the reactor building.

TS 3.4, Specification b, ensures that during reactor operation any potential particulate radioactivity release is filtered resulting in limited consequences to the public.

The NRC staff reviewed TS 3.4, Specification a, item 2, and TS 3.4, Specification b, and finds that TS 3.4 helps provide additional barriers to limit the spread of airborne radioactive material and helps ensure that the potential radiological consequences are below the limits of 10 CFR Part 20. Hence, the NRC staff concludes that TS 3.4, Specification a, item 2, and TS 3.4, Specification b, are acceptable.

TS 4.4 (see Section 9.1 of this SER) states the surveillance requirements for maintaining the integrity of the confinement. It specifies that the ventilation system including the inlet and outlet dampers, as well as the air conditioning system are to be tested periodically. These surveillance requirements with the visual checks help ensure that the system will operate in accordance with the design features. The surveillance intervals are based on experience, which shows that they are acceptable to detect degradation of components and help ensure that the system is operating properly.

6.2.2 Containment

Most research reactors can be designed, sited, and operated such that containment is not required for normal operation or accident mitigation. ANSI/ANS-15.1 defines containment as an enclosure of the facility designed to be at a negative internal pressure to ensure in-leakage, control the release of effluents in the environment, and mitigate the consequences of certain analyzed accidents or events (these accidents usually have the potential to pressurize the containment). Containments are much more robust than confinements. If a facility does not need a confinement engineered safety feature it also does not need containment. Staff review

of the accidents analyzed in Chapter 13 of the SAR demonstrated that containment is not necessary for the PUR-1.

6.2.3 Emergency Core Cooling System

Section 13 of this SER discusses a potential loss-of-coolant accident. Evaluations performed by the licensee show that air cooling after a loss-of-coolant accident is sufficient to remove decay heat from the fuel and prevent loss of fuel element integrity. Therefore, the NRC staff concludes that an emergency core cooling system is not necessary.

6.3 Conclusions

Based on the information provided above, the NRC staff concludes that there is no need for any engineered safety features to mitigate the consequences of the potential accidents analyzed in Chapter 13 of the SAR. The PUR-1 reactor facility is in a confinement. The NRC staff concludes that the licensee's description of the confinement features along with purposed TSs give reasonable assurance that the confinement will operate as described and provide a defined pathway for the flow of air out of the reactor facility.

7 INSTRUMENTATION AND CONTROL SYSTEM

7.1 Summary Description

The PUR-1 facility houses a research reactor fueled with MTR-type fuel, moderated by light water and graphite reflectors, and cooled by light water. The PUR-1 is currently licensed for 1.0 kWt steady state power and proposed to increase the power level to 12 kWt.

The purpose of the instrumentation and control (I&C) system is to provide the reactor operator with the required information to keep the reactor within its operational safety envelope. The I&C system automatically trips the reactor if it begins to operate outside of pre-described safety conditions for operations and prevents the reactor from operating if required support systems are not in the proper operating configuration.

The licensee stated that the power increase to 12 kWt licensed power may require the adjustment of the power detector location and the input impedance of the associated signal preamplifier, which will be detailed in the start-up plan for the increased power level of 12 kWt. The I&C system was designed for even higher power levels and would be capable of measuring operational parameters with appropriate accuracy at the 12 kWt power level (Ref. 16 and Ref. 22).

The I&C system employed at the PUR-1 is similar to those used by other research reactors operating in the United States. Control of the nuclear fission process is achieved using three control rods: two shim-safety control rods and one regulating control rod. The rods are moved in and out of the reactor core by mechanical drives, or in the event of power failure or receipt of a scram signal, the shim-safety rods are disconnected from their drives by means of electromagnetic clutches and are allowed to fall, by gravity, into the reactor. The instrumentation provides indication of process variables, reactor core nuclear parameters, radiation levels at various locations throughout the facility, effluent activity levels, alarms, and other parameters necessary to allow safe operation and shutdown of the reactor and protection of personnel. The control systems provide flexible and reliable control of the reactor during all regimes of operation and shutdown.

The following definitions in TS 1.0 are related to the instrumentation and control system:

TS 1.0 DEFINITIONS

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

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

1.3 Channel Check – A channel check is a qualitative verification of acceptable performance by observation of channel behavior. This verification may include

comparison of the channel with other independent channels or methods of measuring the same variable.

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

(…)

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

(…)

1.22 Operable – A system or component is operable when it is capable of performing its intended function in a normal manner.

1.23 Operating – A system or component is operating when it is performing its intended function.

(…)

1.26 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.

(…)

1.31 Reactor Safety System – The reactor safety system is that combination of measuring channels and associated circuitry which forms the automatic protective system of the reactor, or provides information which requires manual protective action to be initiated.

(…)

1.44 Surveillance and Test Intervals – These are intervals established for periodic surveillance and test actions. Established intervals shall be maintained on the average. Maximum intervals are allowed to provide operational flexibility, not to reduce frequency.

(…)

1.46 True Value –The true value of a parameter is its exact value at any instant.

These definitions are either standard definitions used in research reactor TSs or are facility-specific definitions that NRC staff finds to be consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007, and are therefore acceptable to the NRC staff.

7.2 Design of Instrumentation and Control System

The design of the I&C system is discussed in Section 7.2 of the licensee's SAR. The instrumentation at PUR-1 consists of a four channel system. There are three operational channels and one safety channel. The operational channels include a startup channel, a Log N and reactor period channel, and a linear power channel. The fourth channel is the safety channel, which in conjunction with the safety circuit of the Log N-period channel, initiates an automatic reactor trip if the reactor power exceeds the normal operating power of 12 kWt or the reactor period becomes less than seven seconds. Scram systems are interconnected into these channels to initiate reactor shutdown in the event of an emergency or abnormal condition. An annunciator and alarm system are included to indicate specific trouble.

The PUR-1 I&C system monitoring the status of the reactor include three operational and one safety channel:

- Startup Channel (Log count rate and period) – The startup channel is used to monitor the neutron flux. This channel uses the output of the fission chamber and provides indication of the neutron flux and the reactor period. Two setpoints, based on the reactor period, provide for a reactor setback and trip (slow scram) in the event of a short period (12 and 7 seconds, respectively). An additional interlock, based on the reactor period at 15 seconds or neutron counts above a certain level, prohibits the simultaneous withdrawal of the control rods. When the counting rate channel is near the limit of its counting range, the Log N-period and linear servo channels become the principal means of controlling the reactor.

- Log N and Period Channel – This channel indicates the reactor power level over the range from 0.0001 to 300 percent power level using the output from a compensated ionization chamber. If power level reaches 12 kWt, this channel initiates a slow reactor scram. Two additional setpoints, based on the reactor period, provide for a reactor setback and trip (both slow and fast scram) in the event of a short period. An additional interlock, based on the reactor period at 15 seconds, prohibits the simultaneous withdrawal of the control rods.

- Linear Power Channel – This channel is capable of measuring neutron flux in the reactor operating range from shutdown to >100 kWt using the output from a BF_3 ionization chamber. The channel is capable of measuring the power level from startup to full power through an adjustable range switch. The linear channel also provides a signal to the servo control unit which, through a drive unit, drives the control rod maintaining the power level set at the controller. The servo control unit is automatically deactivated, if the control or regulating rods are manually moved. This channel has two setpoints that will initiate a reactor set back at either zero or 110 percent range. These setpoints insure that the instrument is kept on range at all times during reactor operation. In addition, at the 120 percent range setpoint a reactor trip is initiated (slow scram).

- Safety Channel – This channel also utilizes the BF_3 ionization chamber with a range from a few percent to at least 150 percent of power. It initiates a reactor setback at 110 percent power level, 11 kWt (with normal operating power level of 10 kWt), and a reactor trip (fast scram) at 120 percent power level, 12 kWt.

7.3 Reactor Control System

The licensee discussed the reactor control system in Section 7.3 of the SAR. The PUR-1 has four control channels that make up its reactor control system; three operational channels and one safety channel. The operational channels include a counting rate channel, a log-N period channel, and a linear level channel. Reactor trip (or fast scram) is provided by the log-N channel based on the reactor period, while the safety channel initiates a reactor trip based on the reactor power level. There are also rod position indicators located on the reactor console that allow the reactor operator to continuously monitor all control rod positions. An annunciator and alarm system are also included to indicate specific conditions.

The TS requirements on the reactor control and safety system are specified in TS 3.2 specifying the requirement for the safety measurement channels that must be available during reactor operation as follows:

TS 3.2 Reactor Safety System

Specification - The two shim-safeties shall not be moved more than 6 cm from the fully inserted position unless the following conditions are met:

a. The reactor safety channels and safety-related instrumentation shall be operable in accordance with Tables I and II including the minimum number of channels and the indicated maximum or minimum setpoints.

(…)

TABLE I. SAFETY CHANNELS REQUIRED FOR OPERATION

Channel	Minimum Number Required	Setpoint	Function
Log count rate and period	1[(a)]	2 cps or greater 12 sec. or greater 7 sec. or greater 15 sec. or greater	2 cps rod withdrawal interlock Setback Slow Scram Rod withdrawal interlock
Log N and period	1[(b)]	12 sec. or greater 7 sec. or greater 7 sec or greater 15 sec or greater 12 kW, 120% Operating power level or less	Setback Slow Scram Fast Scram Rod withdrawal interlock Slow Scram
Linear	1	0% Selected Range, or greater 110% Selected Range or less 120% Selected Range or less	Setback Setback Slow Scram
Safety	1[(b)]	11 kW, 110% Operating power level, or less 12 kW, 120% Operating power level or less	Setback Fast Scram
Manual Scram (console) (hallway)	1 1		Slow Scram Slow Scram

(a) Not required after Log N-Period channel comes on scale.
(b) Required to be operable but not on scale at startup.
(c) All percentage based setpoints shall be tripped when the measured value is greater than or equal to the specified value. Period and counts per second (cps) setpoints are at values less than or equal to the specified value. Exception: Trip point for 0% shall happen as the value goes from the positive to negative value.
(d) Setbacks shall be set such that they will be initiated prior to a Scram

(…)

TS 3.2, Specification a, Table I, helps ensure that during the normal operation of the PUR-1 reactor the minimum number of reactor safety system channels required for safe operation of the reactor are operable and sufficient information is available to the operator to ensure safe operation of the reactor. The minimum number of operable measuring channels shown in Table I of the TSs will provide the operator with the following:

- Log count rate, Log N and period, linear, and power level monitors to ensure that the reactor power level is adequately monitored for steady-state operation.
- The power level scram (safety channel) provides protection to ensure that the reactor can be shut down before the safety limit has been exceeded.
- Manual scram allows the operator to shut down the system if an unsafe or abnormal condition occurs either using the manual scram button or the reactor “ON” key switch.

The scram setpoint is established at a power of 12 kWt or less and helps ensure that the licensed power limit and the value of the LSSS (TS 2.2) are maintained during operation.

The licensee modified some of the setpoint definitions in a response to a RAI provided in Ref. 22. The 110 percent and 120 percent setpoints are referenced to the operating power level as defined in TS 1.25 that should be 10 kWt or less. Transient deviations above 10 kWt are allowed provided that the requested maximum licensed power level of 12 kWt is not exceeded. The scram setpoints are calculated based on the operating power level that is expected to be less than or equal to 10 kWt.

The NRC staff finds that TS 3.2, Specification a, Table I, is consistent with the assumptions used in the insertion of excess reactivity and loss-of-coolant accident (LOCA) analysis presented in the PUR-1 SAR and the NRC staff's review in Section 13.2.3 of this SER. TS 3.2, Specification a, Table I, is acceptable to the NRC staff because it ensures that the reactor will not be operated unless the required minimum number of measuring channels, reactor safety channels and interlocks are operable to ensure safe operation of the reactor and that the reactor remains inside the analyzed safety envelope.

The NRC staff reviewed TS 3.2, Specification a, Table I, and finds the safety channels provided in TS 3.2, Specification a, Table I, are consistent with the guidance in NUREG-1537 and ANSI/ANS 15.1-2007. The safety channels provide a comprehensive and diverse method to help ensure that the PUR-1 will be operated safely. Based on the information provided above, the NRC staff finds that TS 3.2, Specification a, Table I, establishes operability requirements for the safety channels that are consistent with the analyses in the PUR-1 SAR. Therefore, the NRC staff concludes that PUR-1 TS 3.2, Specification a, Table I, is acceptable.

TS 4.2 specifications apply to the surveillance requirements for measurements, tests, and calibrations of the control and safety systems to verify the performance and operability of the systems and components that are directly related to reactor safety as follows:

TS 4.2 Reactor Safety System

Specification -

a. A channel calibration of the reactor safety channels as described in Table I shall be performed as follows:

   1. An electronic calibration will be performed annually, with no interval to exceed 15 months. The electronic calibration may be deferred with CORO approval during periods of reactor shutdown, but shall be performed prior to startup.

   2. A power calibration by foil activation shall be performed annually, with no interval to exceed 15 months. The power calibration may be deferred with CORO approval during periods of reactor shutdown, but shall be performed prior to startup.

(…)

d. A channel check of each of the Scram capabilities specified in Table I shall be performed prior to each day's startup.

e. A channel check of the pool top radiation monitoring equipment's off-site alarm capability shall be done biannually, not to exceed 7 ½ months.

TS 4.2, Specification a, helps ensure the operability of the reactor safety system channels by requiring an electronic calibration annually. In addition, the power measuring channels also have to be calibrated by an independent, foil activation method to help maintain accuracy. The NRC staff finds that TS 4.2, Specification a, is consistent with the guidance in NUREG-1537, Part 1, Appendix 14.1, and helps ensure that the scram and power measuring channels accurately indicate the PUR-1 reactor power level. The surveillance frequencies are consistent with NUREG-1537, ANSI/ANS-15.1-2007, and other research reactor facilities. The NRC staff concludes that TS 4.2, Specification a, provides reasonable assurance that component degradation and failure will be detected in a timely manner.

TS 4.2, Specification d, helps ensure that the scrams in TS 3.2, Table I are operable by requiring the performance of a channel check prior to the initial startup of the day. The specified channel checks of the scram signals, 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 the licensed power level. The channel checks help ensure that the scram channels will perform properly when required.

TS 4.2, Specification e, helps ensure that the remote high radiation alarm in TS 3.2, Specification d, is operable by requiring the performance of a channel check biannually. The channel check provide assurance that the remote high radiation alarm will perform its function to alert reactor staff that the pool water level has dropped when the facility is secure without the PUR-1 staff.

Redundancy in the important ranges of power measurements by nuclear instrumentation is ensured by overlapping ranges of the log and linear power channel and the safety channels. All important nuclear process variables are monitored and displayed at the reactor console. During a site visit, the NRC staff observed the operation of the I&C system and found that the system provides the types of information necessary to allow the reactor operator to safely and reliably control the reactor power. Based on the above considerations and years of safe operation with the current systems, the NRC staff finds the I&C designs to be acceptable for operation of the PUR-1 during the period of the renewed license at the proposed steady-state power level of 12 kWt.

The NRC staff reviewed the information in the SAR, as supplemented, and finds that the scope and frequency of performing tests on the reactor safety channels are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 4.2, Specifications a, d, and e, are acceptable.

7.4 Reactor Protection System

The licensee discusses the reactor protection system in Section 7.4 of the SAR. There are two types of scrams incorporated in the PUR-1 control system that effect shutdown of the reactor if an emergency occurs. They are the "fast" and "slow" scram. The fast scram occurs when either a high reactor power or a short reactor period is sensed. The slow scram is initiated when the manual scram button on the console or in the hallway is depressed, a high radiation alarm is detected, a high voltage power supply failure in the compensated ion chamber occurs,

or a failure in the circuits of the composite safety amplifiers occurs. These scrams are initiated by different types of circuits. The more complex electronics in the fast scram circuit cuts power to the control rod magnets in less time (difference is a fraction of a second) than in a slow scram circuit. Although small, this difference in scram time can result in a measurable difference in peak power reached during reactivity addition accidents.

7.5 Control Console and Display Instruments

The control console and display instruments are discussed by the licensee in Section 7.5 of the SAR. The reactor console contains all indicators and controls necessary to implement startup and shutdown operations. Indicator lights on the reactor console are color coded to enhance the operator's ability to determine reactor status. Control rods are interlocked with warning indicators to prevent their withdrawal when a potential problem exists with the reactor.

The NRC staff concludes that the indications on the control console give assurance that the status of systems important to adequate and safe operation will be presented to the reactor operator. The NRC staff compared the general arrangement and types of controls and displays provided by the control console to those at similar research reactors and found that the designs are similar. The NRC staff observed the control console during a site visit and found that the control console provides the reactor operator with the types of information and controls necessary to facilitate reliable and safe operation of the reactor. Based on the above considerations and years of safe operation with the current systems, the NRC staff finds the I&C designs to be acceptable for operation of the PUR-1 during the period of the renewed license.

7.6 Radiation Monitoring Systems

Radiation monitoring systems are discussed in section 7.6 of the SAR. Radiation levels are monitored at various locations throughout the reactor facility. There are three scintillation-type area monitors installed in the vicinity of the reactor: at the top of the pool, adjacent to the skid mounted primary cooling water cleanup system, and at the reactor console. Three remote area monitor meters are mounted on the reactor instrument racks to provide the operator with an indication of the radiation level at the area monitor. As discussed in Section 11.1.4 of this SER, there are four radiation monitoring channels required by TS 3.2, Specification a, Table II, three remote area monitors (RAM) and one continuous air monitor (CAM).

Based on its review of the SAR and observations made during an onsite visit, and as discussed in Chapter 11.1.4 of this SER, the NRC staff concludes that the radiation monitoring systems described in the SAR provide reasonable assurance that all anticipated sources of radiation will be identified and accurately evaluated.

7.7 Conclusions

On the basis of its evaluation of the information presented above, the NRC staff concludes as follows:

- The reactor protection system is designed to prevent or mitigate hazards to the reactor and to prevent the escape of radiation. The protection channels and protective responses are sufficient to ensure that no safety limit or LSSSs specified in the TSs will be exceeded, and that the full range of reactor operation poses no undue radiological risk to the health and safety of the public, the facility staff, or the environment. The

design of the reactor control system interlocks ensures that it can maintain the reactor in a safe condition as derived from the SAR analysis and in accordance with the TSs.

- The I&C system at the PUR-1 is well designed and maintained. Redundancy in the important ranges of power measurements by nuclear instrumentation is ensured by overlapping ranges of the log and linear power channel and the safety channels. All important nuclear process variables are monitored and displayed at the reactor console.

- The design of the reactor control system ensures that it can maintain the reactor in a shutdown condition, change reactor power, and maintain operation at a fixed power level.

- The radiation monitoring systems described in the SAR provide reasonable assurance that all anticipated sources of radiation will be identified and accurately evaluated.

8 ELECTRICAL POWER SYSTEMS

8.1 Normal Electrical Power Systems

The licensee discusses normal electrical power systems in Section 8.1 of the SAR. The PUR-1 reactor building has a standard 120 volt electrical system connected to the campus distribution system. Power for reactor control and instrumentation is provided by a dedicated 120 volt circuit. The water process system uses three-phase power at 240 volt. The HVAC and reactor room light are also supplied by the building power on their own circuits.

As discussed below, electrical power is not required to safely shut down the reactor and maintain the reactor in a safe-shutdown condition.

8.2 Emergency Electrical Power Systems

The licensee discusses the need for emergency electrical power systems in Section 8.2 of the SAR. The PUR-1 does not have any requirements for emergency electrical power. In the event of the loss of normal ac electrical power, the magnet amplifiers on the shim-safety rods are de-energized, and the control rods are then fully inserted by gravity into the reactor core, shutting down the reactor. Consequently, no ac power sources of power are required to safely shutdown the reactor and maintain it in a safe condition. Confirmation of control rod insertion can be accomplished by visual observation of the reactor core.

The primary and secondary cooling system pumps will also stop, but the water in the reactor tank is a sufficient heat sink for decay heat from the reactor. The licensee stated that on loss of electrical power the exhaust fan shuts down and the air duct dampers close isolating the reactor room (Ref. 14). Battery powered portable equipment are available in the reactor room to ensure that operations staff can monitor radiation levels following a power outage (Ref. 15).

8.3 Conclusions

Based on the information provided above, the NRC staff finds that the normal electrical power system at the PUR-1 facility provides reasonable assurance of adequate operation. In addition, the NRC staff concludes that the loss of normal electrical power will lead to safe shutdown of the facility and that emergency power is not required to maintain the reactor in a safe-shutdown state.

9 AUXILIARY SYSTEMS

9.1 Heating, Ventilation, and Air Conditioning Systems

Section 9.1 of the licensee's SAR discusses the heating, ventilation and air conditioning (HVAC) system. Air is supplied to the PUR-1 reactor room by a commercial grade HVAC system. The licensee stated that the reactor room HVAC system is isolated from the rest of the building HVAC (Ref. 15). Air is exhausted by an individual fan to the monitored roof stack approximately 50 feet (15.2 m) above ground level. This exhaust fan also maintains the reactor room at a negative pressure of about 0.05 inch (1.3 mm) of water. If a high radiation alarm occurs, the reactor operator has a switch on the control console that will shut down the exhaust fan and close dampers in the air ducts, which isolates the reactor room. There are HEPA filters in both the supply and the exhaust duct. No emergency power for the HVAC system is necessary since a loss of electrical supply will scram the reactor and natural convection of the primary pool water is sufficient to prevent fuel damage.

TS 5.1, Specification g, contains the fundamental design requirement for the ventilation system as follows:

TS 5.1 Site Description

(…)

g. The ventilation system shall be designed to exhaust air or other gases from the reactor room through an exhaust vent at a minimum of 50 feet above the ground.

(…)

TS 5.1, Specification g, is the basic requirement to have a ventilation system with a controlled air pathway release point. TS 5.1, Specification g, helps ensure that the ventilation system is designed to exhaust air from the reactor building and release it from an exhaust vent a minimum of 50 feet (15.2 m) above the ground. The NRC staff finds that the proposed height of the exhaust stack helps ensure dispersion and dilution of effluents released from the vent before they reach the ground. The NRC staff finds that TS 5.1, Specification g, helps ensure a ventilation system with a controlled air pathway release point.

The NRC staff reviewed the ventilation system design as presented in the SAR, as supplemented, and finds that these design aspects are acceptable 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 5.1, Specification g, is acceptable.

The operation of the ventilation system is specified in TS 3.4 as follows:

TS 3.4 Confinement

a. During reactor operation and when radioactive material is being handled with potential for airborne release, the following conditions shall be met:

    1. The reactor room shall be maintained at a negative pressure of at least 0.05 inches of water with the operation of the room exhaust fan.

    (…)

(…)

c. Dampers in the ventilation system inlet and outlet ducts shall be capable of being closed.

(…)

TS 3.4, Specification a, item 1, helps ensure that the exhaust system has the capability to maintain negative pressure in the confinement building. TS 3.4, Specification a, item 1, specifies the criteria required to establish an operable confinement, which limits the consequences of a potential release to below 10 CFR Part 20 limits. TS 3.4, Specification a, item 1, helps ensure that the ventilation system is in operation to mitigate the consequences of the release of airborne radioactive materials. Operation of the room exhaust fan will achieve confinement of the reactor building during normal conditions.

TS 3.4, Specification c, helps ensure that during reactor operation including any movement of irradiated fuel, potential radioactivity releases results in acceptable doses to the public. It requires the operability of the isolation dampers limiting the release of airborne radioactive material. TS 3.4, Specification c, helps ensure that the potential radiological consequences are limited to below 10 CFR Part 20 limits.

TS 3.4, Specification a, item 1, and TS 3.4, Specification c, help ensure that the ventilation system is maintained operable when the potential exists for the release of airborne radioactivity. These specifications are consistent with the assumptions used in the PUR-1 SAR dose calculations for both occupational and public doses. Additionally, TS 3.4, Specification a, item 1, and TS 3.4, Specification c, support the PUR-1 ALARA program by reducing the potential exposure to Ar-41 (see Section 11.1.1.1 in this SER). The NRC staff finds that TS 3.4, Specification a, item 1, and TS 3.4, Specification c, establish conditions required to consider the ventilation system operable that are reasonable and consistent with the analysis outlined in the PUR-1 SAR, as supplemented. The NRC staff finds that TS 3.4, Specification a, item 1, and TS 3.4, Specification c, help ensure the operability of the ventilation system and are consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 3.4, Specification a, item 1, and TS 3.4, Specification c, are acceptable.

TS 4.4 state the surveillance requirements for the ventilation system as follows:

TS 4.4 Confinement

Specification -

a. The negative pressure of the reactor room shall be recorded weekly.

b. Operation of the inlet and outlet dampers shall be checked semiannually, with no interval to exceed 7 ½ months.

c. Operation of the air conditioner shall be checked semiannually, with no interval to exceed 7 ½ months.

TS 4.4, Specifications a through c, help ensure the operability of the ventilation system by requiring periodic testing of the ventilation system, the inlet and outlet dampers, as well as the air conditioning system. The licensee stated that if the negative room pressure requirement is not satisfied as per TS 3.4, Specification a, item 1, then the HEPA filter is changed in the exhaust line (Ref. 11). The licensee also stated that TS 4.4, Specifications a and b, are the surveillance requirements for the ventilation system (Ref. 11). These surveillance requirements with the visual checks and HEPA filter change requirement help ensure that the system will operate in accordance with the design features. The surveillance intervals are based on experience, which shows that they are acceptable to detect degradation of components and help ensure that the system is operating properly.

The NRC staff finds that TS 4.4 is consistent with the guidance in NUREG-1537, Part 1, Appendix 14.1 including the surveillance and testing requirements. On the basis of the above information the NRC staff concludes that TS 4.4 helps ensure that the performance of the ventilation system is maintained, and therefore, TS 4.4 is acceptable.

The NRC staff reviewed the design, operation, and TS requirements of the HVAC system. The NRC staff concludes that the HVAC system is adequate to provide controlled release of airborne radioactive effluents during normal operations and in the event of abnormal or accident conditions (see Sections 11 and 13 of this SER). The reactor staff, researchers, and the public will be adequately protected from airborne radioactive hazards related to reactor operations. Based on the NRC staff's review of the operational experience of the facility and TS requirements for operability and testing of the system, the NRC staff concludes that degradation of components will be detected; therefore, there is reasonable assurance that the HVAC system can continue to operate safely, as limited by the TSs for the proposed license renewal period at a power level of 12 kWt.

9.2 Fuel Storage and Handling

Fuel storage and handling are discussed in Section 9.2 of the licensee's SAR. The low power rating of the PUR-1 minimizes depletion of the fuel. All fuel manipulations are performed underwater using hand tools by senior reactor operators. There are sufficient spaces in the pool to totally off-load the core if necessary. Spent fuel is kept in two racks located on the interior wall of the pool. Extra fuel plates can be stored in a secured dry storage facility. The licensee performed calculations using the MCNP computer code to determine the k_{eff} of the in-pool storage racks with all positions filled with standard fuel assemblies and the dry storage facility

with varying stored fuel plate spacing in a hypothetical flooded condition (Ref. 4). Results for the in-pool storage racks showed the maximum k_{eff} less than 0.41 and for the dry storage facility hypothetically flooded with water, the maximum k_{eff} less than 0.26. In both cases, the maximum k_{eff} was well below the TS 5.4.1 limit of k_{eff} less than 0.8.

TS 5.4 specifies the storage of reactor fuel at times when it is not in the reactor core to help ensure that fuel that is being stored will not become critical and will not reach an unsafe temperature.

TS 5.4 Fuel Storage

Specifications -

a. All reactor fuel and fueled devices shall be stored in a geometric array where k_{eff} is less than 0.8 for all conditions of moderation and reflection.

b. Irradiated fuel assemblies and fueled devices shall be stored in an array which will permit sufficient natural convection cooling by water or air such that the fuel integrity is maintained per the Safety Analysis Report.

TS 5.4, Specification a, helps ensure that the k_{eff} value is limited to less than 0.8, which is less than then k_{eff} value of 0.9 that is provided as guidance in NUREG-1537 and ANSI/ANS-15.1-2007. The NRC staff finds that the lower value of 0.8 is more conservative and thus acceptable to the NRC staff.

TS 5.4, Specification b, helps ensure that adequate cooling by natural convection, either by water or air, of stored irradiated fuel elements and fueled devices will be incorporated into the design of any storage array. Stored fuel must have sufficient cooling such that the stored fuel elements will not exceed the fuel temperature safety limit. The NRC staff finds that this design feature is acceptable to protect the fuel element cladding and fission product barrier.

The NRC staff reviewed TS 5.4 and finds that the k_{eff} value required in TS 5.4 is less than the values in guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Additionally, the NRC staff finds that TS 5.4 cooling requirements are adequate to ensure protection of the fuel cladding integrity. Based on the information provided above, the NRC staff finds that TS 5.4 is acceptable.

9.3 Fire Protection Systems

Fire protection systems are discussed in Section 9.3 of the licensee’s SAR. Manual pull stations are located throughout the building that houses the PUR-1. They alarm at the Purdue University Fire Department which is a continuously staffed, onsite fire protection service (Ref. 17). The reactor room has limited combustible material and fire extinguishers maintained by the fire department appropriate for the types of fires that could be encountered. The licensee stated that the two fire extinguishers located in the reactor room are inspected annually by the Purdue University Fire Department (Ref. 17). Training on the use of fire extinguishers is done at initial employment of staff and annually thereafter. Purdue’s safety program is designed to OSHA Standards and Requirements. This training is done for all reactor staff (Ref. 22). The fire department personnel receive training in radiological hazards and PUR-1 specific familiarization training.

The NRC staff has reviewed the fire protection systems at the PUR-1 which are typical for a small research reactor. The NRC staff concludes that the fire protection systems are capable of detecting, alarming, and responding to fires. Based on the information provided above, the NRC staff concludes that fire protection at the PUR-1 is acceptable.

9.4 Communication Systems

The licensee discusses communications systems in Section 9.4 of the SAR. A telephone is located at the reactor console in the reactor room. Since the reactor room is small, voice communications is sufficient to alert personnel working there from the control panel area. The telephone allows the control room operator to report the existence of an emergency condition to Emergency Services (onsite 911) and communicate with all personnel required by the TSs during normal and off-normal operating conditions.

The communications systems help ensure that the PUR-1 staff can carry out the communication requirements of the emergency and physical security plan. Based on the information provided above, the NRC staff concludes that the design and operation of the communication systems are acceptable.

9.5 Possession and Use of Byproduct, Source, and Special Nuclear Material

Special nuclear materials and experimental irradiations are considered part of the PUR-1 operating license, since no separation of fuel byproduct material is performed at the reactor facility. Other radioactive byproduct materials are generated during reactor operation, such as radwaste from experiments. Designated storage areas for the possession and use of byproduct material produced by the reactor as well as source material are set aside within the facility. Handling of these materials is controlled by existing operational and health physics procedures. These procedures are written to comply with 10 CFR Part 20 and the facility ALARA program. This provides reasonable assurance that no uncontrolled release of radioactive material to unrestricted areas will occur.

The licensee has proposed two changes to the material possession limits in the license. License condition 2.B.(2) authorizes the receipt, possession and use of special nuclear material under the Atomic Energy Act and 10 CFR Part 70. The licensee has added that the material is not to be separated. This reflects the current wording acceptable to the NRC staff for this license condition. It helps ensure that the licensee does not engage in activities associated with production facilities.

License condition 2.B.(4) is being removed from the license. This license condition authorized the possession, but not use, of high enriched uranium fuel. This is the reactor core that was removed from use when the reactor was converted from the use of high enriched uranium to low enriched uranium fuel. The license condition is no longer needed because the Department of Energy has removed the fuel from the site to a Department of Energy facility. The NRC staff has reviewed the revised possession limits of the license and concludes that they are acceptable for continued operation of the reactor.

Based on the NRC staff’s review as discussed above and the acceptable results of the NRC inspection program, the NRC staff concludes that the licensee has procedures and equipment in place to safely receive, possess, and use the materials authorized by the reactor license.

9.6 Conclusions

The NRC staff reviewed the design and impact of auxiliary systems on the safe operation of the PUR-1 at an increased power level of 12 kWt. Based on the information presented above, the NRC staff concludes the following:

- The HVAC and confinement systems are designed so that the release of airborne radioactive effluent will be controlled and in compliance with the regulations.
- There is adequate assurance that fuel elements and fueled devices will be stored and handled in a safe manner.
- The fire protection systems are capable of acceptably detecting, alarming and responding to fires.
- Communications systems are adequate to meet emergency plan and physical security plan requirements.
- License conditions for receipt, possession, and use of nuclear material is acceptable.

10 EXPERIMENTAL FACILITIES AND UTILIZATION

10.1 Summary Description

Experiment facilities and utilization are discussed in Chapter 10 of the licensee's SAR. The PUR-1 is primarily used as a teaching facility and serves as a source of neutron radiation for research. Experimental facilities include locations within the graphite reflector on one side of the reactor, and drop tubes located next to the reflector boundary. In accordance with Subsection 104c of the Atomic Energy Act, as amended (AEA), the facility license allows the licensee to conduct widespread research through the development, review, and approval of new experiments and experimental facilities. The facility license allows the licensee to conduct research experiments in accordance with the requirements in 10 CFR 50.59. The facility license contains specifications on experiment maximum reactivity worth, design, and materials in TS 3.1, TS 3.5, TS 4.1, and TS 4.5, which provide the basis for the licensee in determining whether the experiment and the experimental facility satisfy the requirements of 10 CFR 50.59.

10.2 Experimental Facilities

The PUR-1 experimental facilities allow gamma and neutron irradiation of materials. The licensee stated that the general scope of the experimental program is routine teaching, performing basic nuclear reactor measurements such as, control rod worth, reactivity periods, activation of materials and testing (Ref. 17). The design and location of the different facilities provide a spectrum of neutron energies and fluxes. Experimenters can perform irradiations in the graphite reflector or the moderator near the reflector boundary. The experimental facilities are comparable in design, construction, utilization, and purpose to experimental facilities at other similar research reactors. The experimental facilities have been successfully and safely utilized during the period of the current facility license.

Accidents such as loss-of-coolant and reactivity insertion that experimental facilities could be subject to are discussed below and in Section 13 of this SER. The design, construction, and utilization of the experimental facilities are such that these accidents are extremely unlikely. Access to experimental facilities is controlled by the use of operating and radiation protection procedures. Use of appropriate radiation detection equipment, radiation protection practices (including the ALARA program), and established experiment review procedures provide reasonable assurance that doses from the use of experimental facilities will meet the requirements of 10 CFR Part 20 for operation personnel and members of the general public.

The following definitions in TS 1.0 are related to the experimental facilities and experiments as follows:

TS 1.0 DEFINITIONS

(…)

1.7 Core Experiment - A core experiment is one placed in the core, in the graphite reflector, or within six inches (measured horizontally) of the reflector. This includes any experiment in the pool directly above or below the core.

(…)

1.10 Experiment - Any operation, hardware, or target (excluding devices such as detectors, foils, etc.) that is designed to investigate non-routine reactor characteristics or that 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 so as to be a part of its design to carry our experiments is not normally considered an experiment.

1.11 Experimental Facility - Experimental facilities are:

a. those regions specifically designated as locations for experiments or

b. systems designed to permit or enhance the passage of a beam of radiation to another location.

1.12 Experiment With Movable Parts (Secured and Nonsecured) - An experiment with movable parts is an experiments that contains parts that are intended to be moved while the reactor is operating.

1.13 Explosive Material - Explosive material is any solid or liquid which is categorized as a Severe, Dangerous, or Very Dangerous Explosion Hazard in "Dangerous Properties of Industrial Materials" by N. I. Sax, Tenth ed. (2000), or is given an Identification of Reactivity (Stability) index of 2, 3, or 4 by the National Fire Protection Association in its publication 704, "Identification System for Fire Hazard of Materials."

1.14 Fueled Experiment - A fueled experiment is any experiment planned for irradiation of uranium 233, uranium 235, plutonium 239, or plutonium 241.

(…)

1.19 Movable Experiment - A movable experiment is one where it is intended that all or part of the experiment may be moved in or near the core or into and out of the reactor while the reactor is operating.

1.20 New Experiment - A new experiment is one whose nuclear characteristics have not been experimentally determined.

1.21 Non-secured Experiment - See Unsecured Experiment.

(…)

1.24 Pool Experiment - A pool experiment is one positioned within the pool more than six inches (measured horizontally) from the graphite reflector.

(…)

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

(…)

1.36 Removable Experiment - A removable experiment is any experiment, experimental facility, or component of an experiment, other than a permanently attached appurtenance to the reactor system, which can reasonably be anticipated to be moved one or more times during the life of the reactor.

(…)

1.40 Secured Experiment - Any experiment, experimental facility, or component of an experiment is deemed to be secured, or in a secured position, if it is held in a stationary position relative to the reactor by mechanical means. The restraining forces must be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces which are normal to the operating environment of the experiment, or by forces which can arise as a result of credible malfunctions.

(…)

1.45 Tried Experiment - A tried experiment is :

a. An experiment previously performed in this facility, or

b. An experiment of approximately the same nuclear characteristics as an experiment previously tried. These nuclear characteristics include but are not limited to neutron activation cross-sections, absorption crosssections, and moderating ability.

(…)

1.48 Unsecured Experiment - Any experiment, experimental facility, or component of an experiment is considered to be unsecured when it is not secured as defined in this section.

The licensee updated the definition of “tried experiment” based on a RAI from the NRC staff (Ref. 22). Examples of nuclear characteristics were added to the definition. The NRC staff notes that experiments of approximately the same nuclear characteristics as an experiment previously tried may be considered tests or experiments not described in the SAR for purposes of 10 CFR 50.59. These tried experiments may still need a successful review under 10 CFR 50.59 before being performed.

These definitions are either standard definitions used in research reactor TS or are facility-specific definitions that NRC staff finds to be consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007, and are therefore acceptable to the NRC staff.

The irradiation facility in the graphite reflector region consists of aluminum tubes filled with graphite when not used for irradiation. All samples placed in either irradiation tube must conform to the material and reactivity requirements specified in the TS. Preparing to insert a sample in the tube replaces graphite with air reducing the core reactivity by about 1% Δk/k, if all six sample tubes are used (and increases reactivity when graphite replaces air). Because this change in reactivity is greater than the TS 3.1 reactivity limits for experiments, the NRC staff

asked the licensee an RAI, and the licensee replied (Ref. 22) that the definition of “core configuration” has been updated to read, “The core configuration includes the number, type, or arrangement of fuel assemblies (elements), reflector elements, reflector element configuration, and regulating/control rods occupying the core grid.” Reflector element configuration has been added to the definition. The tubes when filled with graphite are reflector elements. Therefore, the changing of the installation, or lack thereof, regarding the graphite plugs, will alter the core configuration which would then require a re-measurement of the rod worths and other associated TSs such as shutdown margin. The NRC staff finds that changing the irradiation tube graphite configuration slightly changes the power distribution and the T-H characteristic of the core. However, this effect is very small and bounded by the power distribution hot channel factor used in the T-H analysis. Because changing the graphite configuration of the aluminum tubes is now controlled as a change to the reactor core rather than a change in the configuration of an experiment, the NRC staff concludes that the issue of control of reactivity has been satisfactorily addressed.

There are a number of locations within the reactor, where dry air drop tubes may be inserted for irradiating samples. Two tubes are located next to the core (5/8 inch (1.6 cm) and 1.75 inch (4.4 cm) in diameter), a 3 inch (7.6 cm) PVC drop tube is located in the pool, and a 5 inch (12.7 cm) stainless steel drop tube that can be located in any grid position. The flooding of the 5 inch (12.7 cm) drop tube was analyzed in Section 13.1.6 of the SAR concluding that the tube failure is bounded by the maximum reactivity insertion accident analyzed in Section 13.1.2 of the SAR (Ref. 3).

10.3 <u>Experiment Review</u>

Restrictions are placed on experiments to limit the impact on fuel temperature and the amount of radiation that could be released, if a failure were to occur. Experimental limits are placed on reactivity, materials, and failures and malfunctions. Experiment malfunction is discussed in Section 13.8 of this SER.

Experiment types are defined above. Unsecured experiments are not moved in the reactor during operation, but are not restrained like secured experiments and therefore it is possible that an unsecured experiment may move during operation. Movable experiments can be loaded and unloaded from the reactor while the reactor is operating. They are not constrained and can be moved during reactor operation, and therefore are limited to reactivity values less than secured experiments. Secured experiments are held in position by mechanical means.

TS 3.1 limits the reactivity effect of experiments as follows:

TS 3.1 Reactivity Limits

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

(…)

e. The absolute value of the reactivity worth of each experiment shall be limited as follows:

Experiment	**Maximum Reactivity Worth**
Movable	0.003 Δk/k
Unsecured	0.003 Δk/k
Secured	0.004 Δk/k

f. The sum of the absolute value of the total worth of all movable and unsecured experiments shall not exceed 0.003 Δk/k.

g. The sum of the absolute value of the total worth of all secured experiments shall not exceed 0.005 Δk/k

TS 3.1, Specification e, establishes an upper limit on the absolute reactivity worth of experiments in the reactor and in the associated experimental facilities to provide assurance that the reactor cannot achieve a power level that could exceed the core temperature safety limit should they be inadvertently removed. TS 3.1, Specification e, limits the reactivity of a single movable and unsecured experiment to 0.003 Δk/k reactivity worth, which is substantially below the analyzed maximum allowable reactivity insertion of 0.006 Δk/k. Secured experiments are also limited to 0.004 Δk/k, which is still substantially below the analyzed maximum allowable reactivity insertion of 0.006 Δk/k. Since a secured 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. The removal of a movable, unsecured, or secured experiment from the reactor operating at full power would result in a relatively slow power increase such that the reactor protective systems would act to prevent high power levels from being attained. The NRC staff finds that the 0.003 Δk/k limit for movable or unsecured, and the 0.004 Δk/k limit for secured experiments are substantially below the analyzed maximum allowable 0.006 Δk/k reactivity insertion.

TS 3.1, Specification f, helps ensure that the 0.003 Δk/k absolute value reactivity worth limit imposed on the total worth of all movable and unsecured experiments will not result in an unacceptable power increase. TS 3.1, Specification f, limits the absolute value of the total reactivity of all movable and unsecured experiment to 0.003 Δk/k, which is below the analyzed maximum reactivity insertion limit of 0.006 Δk/k. The results of the supporting analysis, described in Section 13.1.2 of this SER, demonstrate that as much as 0.006 Δk/k of reactivity inadvertently inserted into the PUR-1 reactor would not result in a maximum fuel temperature in excess of the TS safety limit.

TS 3.1, Specification g, helps ensure the absolute value of the sum of the total reactivity limit of 0.005 Δk/k for all secured experiments. Secured experiments are held stationary in the reactor,

reducing the likelihood that they would fall away from the core producing an undesirable step increase in reactivity. The sum of the absolute value of the reactivity worth of all secured experiments is designed to ensure that the reactivity insertion is still less than the analyzed maximum reactivity insertion limit of 0.006 $\Delta k/k$ in the unlikely event of simultaneous removal of all experiments. The NRC staff finds that in TS 3.1, Specification g, the total reactivity limit of 0.005 $\Delta k/k$ for all secured experiments, is designed to be below the analyzed maximum allowable reactivity insertion limit of 0.006 $\Delta k/k$.

The NRC staff reviewed the reactivity limits established in TS 3.1, Specifications e through g, above, and determined that the specifications are based on adequate evaluations of reactivity insertions for the PUR-1. The NRC staff has evaluated the reactivity insertion scenario described in Section 13.2.2 of this SER and concluded that the results were acceptable and the experimental reactivity limits keep the reactivity of experiments within bounds shown to be safe. The NRC staff finds that TS 3.1, Specifications e through g, help ensure that excess reactivity introduced by experiments are properly controlled by the PUR-1 staff. The NRC staff also finds that TS 3.1, Specifications e through g, are consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 3.1, Specifications e through g, are acceptable.

TS 3.5 specifies the limits on the type of materials that may be used in experiments and the review and approval of new experiments as follows:

TS 3.5 Limitations on Experiments

Specification - The reactor shall not be operated unless the following conditions are met:

a. All experiments shall be constructed of material which will be corrosion resistant for the duration of their residence in the pool.

b. All experiments and experimental procedures shall receive approval by the Committee on Reactor Operations.

c. Known explosive materials shall not be placed in the reactor pool.

d. No experiment shall be placed in the reactor or pool that interferes with the safe operation of the reactor.

e. Any failure of an experiment shall not have a consequence that could exceed dose limits as set forth in 10 CFR Part 20, as analyzed and approved by the Reactor Supervisor and the Committee on Reactor Operations.

f. A fueled experiment shall not produce more than 0.5 Curies of radio-iodine.

TS 3.5, Specification a, requires experiments to be constructed of corrosion resistant material. This requirement provides protection for reactor components against experiment failure.

TS 3.5, Specification b, helps ensure that the Committee on Reactor Operations (CORO) must review and approve all experiments and experimental procedures. The CORO must approve any new experiment involving the reactor, as well as substantive changes to existing

experiments. TS 3.5, Specification b, helps ensure that the experiments will be conducted properly and with the appropriate supervision.

TS 3.5, Specification c, helps ensure that no explosive material may be placed in the reactor pool. This requirement eliminates the potential for a detonation and reactor damage during the irradiation of an experiment. Explosive material is defined in TS 1.13. In its response to an RAI (Ref. 22), the licensee stated that any explosive material placed and/or stored in the PUR-1 facility is subject to search, and the licensee implements administrative controls in order to limit these materials from being taken into locations such as the reactor confinement which may compromise reactor safety. The NRC staff finds that TS 3.5, Specification c, is consistent with the recommendations of NUREG-1537, Part 1, Appendix 14.1, Section 3.8.2. Therefore, the NRC staff concludes that TS 3.5, Specification c, is acceptable.

TS 3.5, Specification d, helps ensure that the placement of experiments will be conducted properly and with the appropriate supervision.

TS 3.5, Specifications e and f, set limits on the radioactive products produced in experiment materials that may release airborne radioactive material. TS 3.5, Specification e, limits the consequence from the failure of an experiment to the dose limits in 10 CFR Part 20 for the reactor staff and the public. TS 3.5, Specification f, imposes limitations on the allowed inventories of iodine isotopes in a fueled experiment to ensure the health and safety of the facility workers and the public in the event of an experiment failure. The licensee analyzed the maximum failure of a fueled experiment and showed that it is well bounded by the designated MHA (which is the failure of a fuel plate cladding). TS 3.5, Specifications e and f, are acceptable to the NRC staff because they limit exposure from potential experiment failure or malfunction to 10 CFR Part 20 limits.

The purpose of TS 3.5, Specifications e and f, is to help ensure that potential releases of radioactive material from experiments are bounded by the exposure limits in 10 CFR Part 20 for PUR-1 staff and members. This includes experiment failures under normal reactor operations and credible reactor accident conditions.

The NRC staff has reviewed the licensee's limitations on experiments. The technical content of TS 3.5 is consistent with guidance in NUREG-1537 and provides an envelope of performance against which proposed irradiations can be evaluated. The NRC staff has reviewed TS 3.5, Specifications e and f, and finds that these specifications help to limit doses from potential experiment failure or malfunction from exceeding 10 CFR Part 20 limits. Based on the information provided above, the NRC staff concludes that TS 3.5 is acceptable.

The surveillance requirements for experimental limits are given in TS 4.1 and 4.5 as follows:

TS 4.1 Reactivity Limits

Specification -

(…)

c. The reactivity worth of experiments placed in the PUR-1 shall be measured during the first startup subsequent to the experiment's insertion and shall be verified if core

configuration changes cause increases in experiment reactivity worth which may cause the experiment worth to exceed the values specified in Specification 3.1.

TS 4.1, Specification c, helps ensure that the requirements of TS 3.1, Specifications e through g, are met. All aspects of TS 3.1, Specifications e through g, are required to be considered by the licensee. TS 4.1, Specification c, helps ensure that the reactivity worth of an experiment shall be measured and verified, as appropriate at first startup or if the core configuration is changed affecting the reactivity worth of an experiment.

The NRC staff finds that TS 4.1, Specification c, helps ensure that the reactivity of an experiment is measured and meets the requirement of TS 3.1. TS 4.1, Specification c, is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1--2007. On the basis of its review of the information described above, the NRC staff concludes that TS 4.1, Specification c, is acceptable.

TS 4.5 Experiments

Specification - No experiments shall be performed unless:

a. It is a tried experiment.

b. The experiment has been properly reviewed and approved according to Section 6 of the technical specifications.

    1. Proposed experiments shall be approved by the Committee on Reactor Operations

    2. Submitted proposed experiments shall provide a comprehensive list of steps to be performed, quantities to be measured, hazards to be considered, limiting initial conditions of the reactor, and required available personnel.

TS 4.5, Specification a, helps ensure that only previously reviewed and approved experiments are performed.

TS 4.5, Specification b, helps ensure that all experiments are reviewed and approved as discussed in Section 12.2 of this SER. Proposed experiments are approved by the CORO. Proposed experiments are required by TS 4.5, Specification a, item 2, to contain specific information. The licensee states that the PUR-1 operation manual has detailed instruction for the experiment review and approval process. New experiments are reviewed by reactor staff using established safety criteria and the technical specification requirements (Ref. 19).

The NRC staff finds that TS 4.5, Specifications a and b, help ensure that experiments, including their reactivity worth, are reviewed and approved before implementation. The NRC staff finds that TS 4.5, Specifications a and b, are consistent with the guidance in NUREG-1537 and ANSI/ANS 15.1-2007. Therefore, based on the information provided above, the NRC staff concludes that TS 4.5, Specifications a and b, are acceptable.

The review and approval process for experiments is governed by TS 3.5, Specification b, as discussed above, and TS 6.2 and TS 6.5, which are discussed in Section 12 of this SER. The process requires that a safety analysis be developed for each experiment that demonstrates

compliance with the TS. The CORO must review and approve, and the Laboratory Director must also approve, in writing, any new experiment involving the reactor, as well as substantive changes to existing experiments. In addition, the CORO reviews new experiments and all changes made under 10 CFR 50.59 to experiments. Minor changes to experiments that do not significantly alter the experiment may be approved by the Laboratory Director or Reactor Supervisor. The NRC staff requested additional information about minor changes to experiments (Ref. 22). The licensee explained that small changes that do not significantly alter the experiment or the intent of the procedure are not automatically entered into the 10 CFR 50.59 process. The licensee would screen these changes to determine if they meet the criteria of the 10 CFR 50.59 process. If they screen in a 10 CFR 50.59 review would be performed. Experimental deviations which are permitted are outlined in the experiment itself and those deviations may only be made within the limits of the written approved framework.

TS 4.5 provides reasonable assurance that administrative oversight will preclude the experiment program from posing a significant risk to the health and safety of the public, facility personnel, experimenters, and the environment.

The NRC staff has reviewed the licensee's limitations on experiments. The TS areas recommended by NUREG-1537 and ANSI/ANS-15.1-2007 are covered by the PUR-1 TSs. The technical content of the TSs are consistent with guidance and provide an envelope of performance against which proposed experiments can be evaluated. Based on the information provided above, the licensee's limitations on experiments in TS 3.5, TS 4.1, and TS 4.5 are acceptable to the NRC staff.

10.4 <u>Conclusions</u>

The NRC staff has reviewed the experimental facilities associated with the PUR-1 facility, and finds that the review process for experiments, the use of the experimental facilities, and the governing TSs, provides reasonable assurance that appropriate precautions are taken to minimize the risk to personnel and the public. Based on the information provided above, the NRC staff concludes as follows:

- The design of the PUR-1 experimental facilities, combined with the review and TS requirements applied to experimental activities, give reasonable assurance that experiments are unlikely to fail, are unlikely to release significant radioactivity, and are unlikely to cause damage to either the reactor or its fuel.

- The licensee has a sufficient experimental review process.

- The TSs place acceptable limits on the use of experimental facilities and provide reasonable assurance that experiments are conducted in a safe and controlled manner, as provided by TS 3.1 and TS 3.5 (limitations on experiments), TS 4.1 and TS 4.5 (surveillance), and TS 6.2 (review and audit).

11 RADIATION PROTECTION AND WASTE MANAGEMENT

11.1 Radiation Protection

The licensee discusses radiation protection and waste management in Chapter 11 of the SAR. Activities involving radiation at the PUR-1 are controlled under a radiation protection program that meets the requirements of 10 CFR Part 20, “Standards for Protection Against Radiation” and the guidance in ANSI/ANS-15.11-2004, “Radiation Protection at Research Reactor Facilities,” (Ref. 41). 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 as low as reasonably achievable (ALARA).

The basic aspects of the radiation protection program include occupational and general public exposure limits, surveys and monitoring, and personnel dosimetry.

The NRC inspection program routinely reviews radiation protection and radioactive waste management at the PUR-1. The licensee stated that the PUR-1 has operated under the Purdue University Radiation Safety Program, which is under the responsibility of the University Radiation Safety Officer and the Institution Radiation Safety Committee (Ref. 11). The Radiation Safety Program is reviewed annually by the Radiation Safety Officer.

The NRC staff reviewed the licensee’s annual operating reports for the PUR-1 and the NRC inspection reports from 2005 to 2015 concerning the radiation protection program. The NRC staff finds that the licensee’s 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 following discussion, the NRC staff concludes that the radiation protection program at the PUR-1 is acceptable.

11.1.1 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. 3) and verification that the hazards were accurately depicted and comprehensively identified. Primary radiation sources are directly related to reactor operation that include radioactive isotopes, activated components and samples, and radiation associated with reactor operations.

11.1.1.1 Airborne Radiation Sources

In the SAR, the licensee states that during normal operations the primary airborne sources of radiation are argon-41 (Ar-41) and nitrogen-16 (N-16). Nitrogen-16 is produced when oxygen in the pool water is irradiated in the reactor core that then must diffuse to the pool surface before it is released to the atmosphere. There is no strong current of upward flow of heated water containing N-16 and with the very short half-life (7.14 seconds) the nitrogen essentially decays before reaching the pool surface. The licensee estimated that, at a bounding power level of

18 kWt, N-16 produced in the reactor core may reach the surface in about 206 seconds (Ref. 19). Any amount of N-16 would have long decayed before reaching the pool surface. Analysis of effluent samples in the reactor room has not detected any N-16 due to the short half-life of N-16, and therefore exposure to the PUR-1 staff and the public is negligible from N-16.

Ar-41 results from irradiation of the argon in air in experimental facilities and the dissolved air in the reactor pool water. The primary source of Ar-41 is the reactor pool. Other production sources include the air contained in the experimental tubes. At 1 kWt steady-state operation, effluent samples in the reactor room have not contained detectable traces of Ar-41. In the response to RAI-7 in Ref. 20 the licensee performed a bounding analysis of Ar-41 release at 18 kWt corresponding to the proposed operating power level, 12 kWt, with an additional 50 percent safety factor.

The Ar-41 mainly results from neutron capture by Ar-40 in the air that is dissolved in the reactor pool water. 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 reactor room. The analysis considered two potential sources for Ar-41 release, the experimental dry drop and movable tubes filled with air and the pool water that contains dissolved air from the reactor room. Activation of dissolved Ar-40 in the pool water accounts for almost 90 percent of the Ar-41 production.

The reactor area has a ventilation system that removes Ar-41 through the building exhaust system, thus minimizing the dose to workers from Ar-41. Since the Ar-41 generated in the reactor area is continuously removed by the ventilation system (as required by TS 3.4), public occupants in the area of the class rooms and building spaces adjacent to the reactor room are not subjected to any Ar-41 dose.

The Ar-41 saturation activity was derived by the licensee estimating the level of dissolved Ar-40 in the coolant and considering the activation time together with the core thermal neutron flux level at 18 kWt (Ref. 20 and Ref. 21). Assuming that the Ar-41 production is at steady state, an effective half-life is calculated considering radioactive decay and room purging by the ventilation system (air flow rate at 0.2 m^3/s). At 18 kWt reactor power level, the equilibrium concentration of Ar-41 is calculated to be 2.08×10^{-7} $\mu Ci/cm^3$ in the exhaust air and the reactor room.

The maximum air concentration limit for occupational workers is 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 at 3.0×10^{-6} $\mu Ci/cm^3$. The DAC concentration limit corresponds to a 5 rem occupational dose for an individual worker assumed to be exposed to the limiting DAC continuously for 2,000 hours. The bounding analysis for the maximum facility worker exposure shows that the total radioactivity concentration of 2.08×10^{-7} $\mu Ci/cm^3$ is below the DAC limit of 3.0×10^{-6} $\mu Ci/cm^3$ and, therefore the occupational radioactivity exposure levels are below the 10 CFR Part 201201 limit of 5 rem.

In its RAI response (Ref. 20), the license evaluated the dose rate to a worker given the assumption that the concentration of Ar-41 was generated continuously during a steady state operating level with a resulting dose rate of 0.167 mrem/hr (0.00167 mSv/hr). Using the calculated dose rate, the TEDE to a worker in the reactor room for the entire year would be less than 334 mrem (3.34 mSv), assuming a hypothetical 2000 hour steady state, full power

operation, since the reactor license contains no restriction on operating hours. The reactor normally operates for much less than the assumed 8 hours per day and the conservatively calculated dose is still well below the 5,000-mrem (50 mSv) limit established in 10 CFR 20.1201, "Occupational dose limits for adults."

The NRC staff finds that the licensee's dose estimates are conservative and satisfy the requirements of 10 CFR Part 20. Based on the information provided above, the NRC staff concludes that the licensee's occupational airborne dose estimates for the operation of the PUR-1 are acceptable.

TS 3.4, Specification d, limits the airborne radioactive releases from the confinement building to the analysis in the SAR as follows:

> TS 3.4 Confinement
>
> Specification -
>
> (…)
>
> d. Concentration of Ar-41 shall not exceed $2.08 \times 10^{-7}\ \mu Ci/cm^3$ at the top of the confinement exhaust stack.

In responses to RAIs (Ref. 20 and Ref. 21), the licensee calculated a bounding dose from normal operations to a person in the unrestricted area due to argon-41 released from the building ventilation opening. The release point is on the roof vent on the top of the building 15 meters above ground. The analysis assumed that the reactor operates continuously for a year, the wind is always directed to the point of exposure, and that the person stands at the point of maximum exposure continuously for the year. Using the Ar-41 concentration activity of 2.08×10^{-7} µCi/cm^3 in the exhaust air flow leaving the stack with an exhaust rate of 0.2 m^3/s and applying an appropriate atmospheric dispersion factor (see Section 13.1.1), the Ar-41 concentration at the maximally exposed location is calculated as 3.19×10^{-10} µCi/cm^3. The potential dose rate of a member of the public in the plume containing the Ar-41 activity was calculated as 3.17×10^{-4} mrem/hr (3.17×10^{-6} mSv/hr) or 2.8 mrem/yr (0.28 mSv/yr), which is well below the limit in 10 CFR 20.1301 of 100 mrem/yr (1 mSv/yr) and also meets the constraint of air emissions of radioactive material in 10 CFR 20.1101(d). A review of Ar-41 releases from the licensee's annual reports shows that the annual release of Ar-41 from 2005 through 2015 was well below the above calculated potential dose rate.

The NRC staff performed confirmatory analyses using the COMPLY computer code for the hypothetical, maximum allowable operational regime assuming that the reactor operates continuously throughout the year. The COMPLY analysis assumed a typical Gaussian atmospheric plume dispersion with site specific average meteorological data to derive downwind concentration and dose values due to measured Ar-41 effluent release rates. Other parameters such as stack height, terrain, and wind speed and directions were consistent with typical PUR-1 conditions. Using these conservative assumptions, the calculated public dose value from Ar-41 exposure was 0.7 mrem/yr (0.007 mSv/yr), comparable to the licensee's result. The bounding analysis for Ar-41 release demonstrates that the dose to the member of the public exposed to the Ar-41 stack effluents continuously for a year is well below the limit of 100 mrem/yr (1mSv/yr) given in 10 CFR 20.1301.

The NRC staff reviewed the licensee's analysis demonstrating the PUR-1 routine gaseous effluent releases and TS 3.4, Specification d, and determined that the analysis used conservative assumptions and acceptable methodology. The calculated dose to the member of the public is below the limits given in 10 CFR 20.1201, and the licensee's ALARA goal of 10 mrem/yr (0.1 mSv/yr), which satisfies the ALARA goals given in 10 CFR 20.1101(d). Based on the information provided above, the NRC staff concludes that the production and release of Ar-41 in accordance with TS 3.4, Specification d, poses little risk to the health and safety of the public and to the PUR-1 staff.

11.1.1.2 Liquid Radiation Sources

The reactor coolant system is a liquid radiation source. Because the pool and primary system piping contains water that has been circulated through the reactor core, radioactive corrosion products produced during normal operation may be capable of producing radiation exposure to personnel. Potential exposure to operating staff is minimized by the implementation of the ALARA policy (see Section 11.1.3 of this SER). As required by TS 4.3, Specification b, the licensee samples primary water for radioactive content monthly during periods of operation to help detect potential fission product leakage from the reactor fuel, leakage from sealed sources, or activation of materials in the coolant water.

The licensee stated that the air conditioner drain collected in the condensate holdup tank may also become contaminated. If contamination is confirmed, using identical procedures as for analyzing reactor water, the contaminated water will be disposed by the Department of Radiological and Environmental Management in accordance with federal, state, and local laws (Ref. 14).

The licensee also states that another potential low-level radiation source would be small quantities of liquid wastes that are accumulated from operations and discharged to the sanitary sewer (See Section 11.2.3 of this SER). Because radiation exposures from these liquid radiation sources at PUR-1 are small, they do not present a significant hazard to either operating personnel or the public. Samples from the discharged liquid are analyzed for radioactivity before discharge to ensure that releases are within the 10 CFR Part 20 limits.

Based on the discussion above including a review of the licensee's annual operating reports from 2005 through 2015, the NRC staff finds that liquid radioactive sources from the past normal operation of the PUR-1 are small, and access to the liquid sources and disposal of the liquid sources are controlled. The NRC staff finds that TS 3.3, Specifications a and d, provide acceptable monitoring and analysis, such that radioactive effluent releases are within the 10 CFR Part 20 limits. The NRC staff also finds that TS 4.3, Specification b, provides for analysis of potential liquid effluents from the pool prior to discharge. Based on the information provided above, the NRC staff concludes that the potential liquid discharges to the environment are properly monitored and analyzed before discharge, and do not represent a hazard to the public or environment.

11.1.1.3 Solid Radiation Sources

In the SAR, the licensee states that the fission products in the reactor fuel and the reactor core are the primary solid radiation sources at the PUR-1. The fission products in the reactor fuel constitute the most significant solid radiation source. However, the fission products in the fuel are protected by fuel cladding. The reactor pool water and concrete shielding around the pool

provide protection for personnel from this source of radiation exposure. Nonfuel sources include activated reactor components, resins from the primary water clean-up demineralizer, and irradiated samples. Because final radioactivity is estimated before experimental irradiations are performed, both shielding and storage duration requirements will be known. The radiation protection program controls solid radiation sources. A review of past NRC inspection and licensee annual reports from 2005 through 2015 has confirmed that solid radioactive waste handling has not resulted in any significant personnel exposure at the PUR-1 facility.

The reactor contains a stainless steel-clad plutonium-beryllium startup source located in core position C3 (discussed in Section 4.2.4 of this SER). TS 4.3, Specification b, requires monthly measurement of radionuclide content of the pool water to provide information as a means to detect a leak of radioactive material.

Based on the information provided above, the NRC staff finds that solid radioactive sources and wastes from continued operation of the PUR-1 are properly controlled, have resulted in no significant personnel exposures, and can be handled without endangering the safety of the PUR-1 staff. The NRC staff concludes that the control of solid radioactive sources at the PUR-1 facility is acceptable.

11.1.2 Radiation Protection Program

The radiation protection program is discussed in Section 11.1.2 of the licensee's SAR. The regulations in 10 CFR 20.1101(a) require that each licensee shall develop, document, and implement a radiation protection program. The NRC inspection program routinely reviews the radiation protection program at the PUR-1 facility for compliance. The licensee stated that PUR-1 utilizes the Purdue University radiation safety program established by university policy, which complies with applicable federal and state laws (Ref. 14). The radiation protection program at the PUR-1 facility is implemented by the Radiation Safety Officer (RSO) reporting to the Purdue University Radiation Safety Committee. The PUR-1 facility has a structured radiation protection program, which is implemented by qualified health physics staff that is equipped with radiation detection capabilities to determine, control, and document occupational radiation exposures at the facility.

Their responsibilities include maintaining radiation exposure files on facility personnel, supervising the environmental monitoring program, implementing and maintaining the ALARA program, and ensuring compliance with applicable requirements and regulations. The administrative requirements for radiation safety are given in TS 6.3 (See Section 12.3 of this SER). The RSO provides onsite advice concerning personnel and radiological safety. The RSO also provides technical assistance and review in the area of radiation protection. The program is implemented using written standard operating procedures (SOPs). The operating staff performs the day-to-day radiation protection activities at the facility, under the supervision of the Reactor Supervisor.

In Section 11.1.4 of the SAR, the licensee indicates that the PUR-1 radiation protection program establishes: exposure limits; procedures and record system for surveys and monitoring; and requirements and responsibilities for personnel dosimetry.

The PUR-1 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 an 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 Committee on Reactor Operations (CORO) of the PUR-1 periodically reviews the program. The NRC inspection program routinely reviews the radiation protection program. The NRC staff reviewed the information provided in the SAR, together with the licensee's annual operating reports and NRC inspection reports from 2005 through 2015, and finds that the licensee demonstrated that adequate measures are in place to minimize radiation exposure to personnel and provide adequate protection against operational releases of radioactivity to the environment.

The licensee's 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. The health physics staff is equipped with radiation detection capabilities to determine, control, and document occupational radiation exposures at the PUR-1 facility. All personnel entering the facility are issued the appropriate personnel monitoring devices.

General training topics 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 PUR-1 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. The licensee stated that an environmental monitoring program has been established at the PUR-1 facility using TLD badges located at the boundaries of the facility, which are checked for exposure every other month (Ref. 14).

The PUR-1 radiation protection program requires that all gaseous and liquid effluents are monitored as discussed in Sections 11.1.1.1 and 11.1.1.2 of this SER before release to comply with 10 CFR Part 20 limits. TS 6.2, Specification d, item 1, requires auditing the health physics and radiation protection program procedures on an annual basis. This includes all procedures, personnel radiation doses, radioactive material shipments, radiation surveys, and radioactive effluents released to unrestricted areas. The Laboratory Director 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 reviewed the PUR-1 radiation protection program, as described in the license renewal SAR, as supplemented, and finds that the program complies with 10 CFR 20.1101(a), is implemented in an acceptable manner, and provides reasonable assurance that, for all facility activities, the program will protect the PUR-1 staff, the environment, and the public from unacceptable radiation exposures. Based on the information provided above, the NRC staff concludes that the PUR-1 radiation protection program, as described, is acceptable.

11.1.3 ALARA Commitment

To comply with the regulations in 10 CFR 20.1101, the licensee 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 policies are determined by the Purdue University Radiation Safety Committee and administered by the Radiation Safety Officer. The program to implement this policy is based on the guidelines of ANSI/ANS-15.11-2004, “Radiation Protection at Research Reactor Facilities,” which is supported by the NRC staff (Ref. 41). The program is applied through written procedures and guidelines. All proposed experiments and operational procedures at the PUR-1 facility are reviewed for ways to minimize potential exposure to personnel. The PUR-1 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 to prevent recurrence.

The ALARA program is adequately supported by the upper levels of the Purdue University management. The review of controls for limiting access and personnel exposure in the PUR-1 facility provides reasonable assurance that radiation doses to the environment, the public, and facility personnel will be ALARA. The NRC staff concludes that the PUR-1 ALARA program complies with 10 CFR 20.1101, is acceptably implemented and provides reasonable assurance that for all facility activities, radiation exposure will be maintained ALARA.

The licensee submits the results of the environmental monitoring program to the NRC in the annual report. An examination of these reports from 2005 through 2015 by the NRC staff shows that the impact on the environment from the gaseous and liquid releases of the PUR-1 facility has indicated an insignificant radiation impact on the environment.

The NRC staff reviewed the annual operating reports and finds that these reports show that the impact on the environment from the gaseous and liquid releases at the PUR-1 is acceptable. In addition, the NRC inspection program routinely reviews the ALARA program and concludes that the program as implemented meets the requirements of the regulations. The NRC staff finds that the PUR-1 ALARA program complies with 10 CFR 20.1101 and provides reasonable assurance that radiation exposure will be maintained ALARA for all facility activities. On the basis of the information discussed above, the NRC concludes that the PUR-1 ALARA program is acceptable.

11.1.4 Radiation Monitoring and Surveying

The regulation 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 health physics staff at the PUR-1 facility regularly performs radiation and contamination surveys on a regular basis. The placement, use, and control of the radiation monitoring and surveying equipment are in accordance with applicable national standards, guidance, and regulations. The equipment selected is appropriate for detecting the types and intensities of radiation likely to be encountered within the facility at appropriate frequencies to help ensure compliance with 10 CFR Part 20 requirements and the facility ALARA program under all operating conditions.

In the SAR, as supplemented, the licensee indicates that the radiation monitoring system is intended to inform operating staff of any impending or existing danger from radiation so that there will be sufficient time to evacuate the facility and take the necessary steps to prevent the spread of radioactivity to the surroundings. SAR Chapter 11 describes a comprehensive set of radiation survey instrumentation that has sufficient range to cover the various types of radiation that may be encountered at the PUR-1 facility. Systems used at the PUR-1 facility for monitoring radiation, include the radiation area monitors (RAM) and a continuous air monitor (CAM) (called the continuous air sampler in TS 3.2, Specification a, Table 2).

The three radiation area monitors are located at the top of the pool, near the pool water flowing through the process system, and at the reactor console. The area radiation monitors have local audible and visual alarms and upon signal of exceeding preset radiation levels initiate a scram signal. The alarm is set at levels agreed upon by the reactor operations supervisor and the senior health physicist based on potentially abnormal radiation levels. Airborne activity in the reactor room is monitored by a CAM, which permits early detection of a cladding failure. The

licensee states that it can adjust the alert and alarm setpoints as necessary to account for changes in the background radiation levels.

The NRC staff reviewed the design characteristics and finds that the radiation monitoring system will provide reasonable assurance that airborne radioactivity will be properly detected to comply with 10 CFR Part 20 requirements. The alert and alarm setpoints will provide an advance warning to the operators that radiation levels have changed and an evaluation and assessment is needed to ensure the safety of the workers and the public.

TS 3.2 specifies the radiation monitoring channels that must be operable during reactor operation and the information that must be available to the reactor operator during reactor operation to ensure safe operation of the reactor.

The radiation monitoring systems are specified in TS 3.2 as follows:

TS 3.2 Reactor Safety System

The two shim-safeties shall not be moved more than 6 cm unless the following conditions are met:

a. The reactor safety channels and safety-related instrumentation shall be operable in accordance with Tables I and II including the minimum number of channels and the indicated maximum or minimum set points.

(…)

d. The pool top radiation monitor shall be capable of indicating an alarm to off-site reactor staff when a high limit is reached and the reactor has been secured. The alarm may be out of service up to thirty days. Loss of functionality beyond thirty days shall require a visual pool level inspection in intervals of 24 hours, not to exceed 30 hours.

TABLE II. SAFETY-RELATED CHANNELS (AREA RADIATION MONITORS)

Channel	**Minimum Number Required**[(c)]	**Setpoint**	**Function**
Pool top monitor	1	50 mR/hr or 2x full power background	Slow Scram
Water process	1	7 ½ mR/hr	Slow Scram
Console Monitor	1	7 ½ mR/hr	Slow Scram
Continuous air sampler	1	Stated on sampler	Air sampling
(e) For periods of one week or for the duration of a reactor run, a radiation monitor may be replaced by a gamma sensitive instrument which has its own alarm and is observable by the reactor operator.			

TS 3.2, Specification a, Table II, helps ensure that the PUR-1 radiation monitoring system is operable to support reactor operations. TS 3.2, Specification a, Table II, states the minimum operable number of area radiation monitors and continuous air monitor (CAM) the PUR-1 can

operate with, as well as the lowest acceptable level of performance for these monitors and detectors. The NRC staff finds that TS 3.2, Specification a, Table II, is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007 and helps ensure that the radiation monitoring systems will alert the operator in the event that alarm setpoints are exceeded during reactor operations. Footnote (c) provides an alternate monitoring method should a radiation monitor become inoperable.
The area radiation monitoring system and the continuous air monitor (CAM) provide information to the operator on radiation levels under the full range of operating conditions. Audible alarms indicate when corrective operator action is required, and in addition, the area radiation monitors have a warning light that indicates situations requiring special operator attention and evaluation. The licensee explained the bases for the monitor setpoints in a reply to a NRC staff RAI
(Ref. 22)

The substitution of one of the area radiation monitor detectors with a gamma sensitive instrument for a period not to exceed one week (or reactor run) is acceptable to the NRC staff. One week is a sufficient amount of time to return an inoperable monitor to operation.

The NRC staff reviewed the licensee's radiation monitoring equipment, as described in the SAR, as supplemented, and TSs, and finds that the licensee has adequate instruments and equipment for quantitative radiation measurements and TS 3.2, Specification a, Table II, requires sufficient monitors to evaluate potential radiation hazards. Routine effluent releases are within regulatory limits, and the discussion in Section 13 of this SER shows that the consequences of accidents are acceptable. Based on the information provided above, the NRC staff concludes that TS 3.2, Specification a, Table II, is acceptable.

TS 3.2, Specification d, requires that the pool top radiation monitor be capable of alerting off-site reactor staff when a high level radiation alarm occurs when the reactor is secured. The purpose of this alarm is to alert reactor staff in the unlikely event of a low pool water level condition when the facility is not staffed. The alarm may be out of service for up to 30 days if a visual inspection of the pool level is made at 24 hour intervals. The loss of coolant is evaluated in Section 13.2.3 of this SER and was found to be acceptable. The NRC staff finds that in the event of a water loss from the pool during periods of time where the PUR-1 facility is not staffed, TS 3.2, Specification d, will help ensure that the staff is aware of the event and can take appropriate action. The NRC staff finds that TS 3.2, Specification d, is consistent with the guidance in NUREG-1537. Therefore, the NRC staff concludes that TS 3.2, Specification d, is acceptable.

TS 4.2 specifies the surveillance requirements for the radiation monitoring systems:

TS 4.2 Reactor Safety System

(…)

b. A channel check on the radiation monitoring equipment shall be completed daily during periods when the reactor is in operation. Calibration of the Safety-Related Channels specified in Table II and hand held radiation survey instruments shall be performed annually, with no interval to exceed 15 months. Calibration may be deferred with CORO approval during periods of reactor shutdown, but shall be performed prior to startup.

(…)

TS 4.2, Specification b, helps ensure that the calibration of the radiation monitors is performed annually. Experience has shown that daily verification of area radiation and air monitoring system operation in conjunction with annual calibration is adequate to correct for any variation in the system due to a change of operating characteristics over time. The NRC staff reviewed the information provided in the SAR, as supplemented, and finds that these surveillance intervals are consistent with guidance in NUREG-1537 and ANSI/ANS-15.1-2007 and are sufficient to detect any changes in the operating characteristics of the monitoring systems. Based on the information provided above, the NRC staff concludes that the surveillance and calibration requirements in TS 4.2, Specification b, are acceptable.

Personnel exposure is monitored by beta-gamma and neutron film badges and pocket ionization chambers, which are assigned to individuals who have the potential to be exposed to radiation. Dose rates in radiation areas are measured using survey meters and the measured dose rates are posted where required. Tour groups do not have access to radiation areas within the PUR-1 facility where the dose rate exceeds 2 mrem/hr (0.02 mSv/hr). These provisions provide assurance that external and internal radiation monitoring of all individuals required to be monitored meets the requirements of 10 CFR Part 20 and the goals of the facility ALARA program. Occupational exposure of operations and maintenance personnel seldom exceed 0.1 rem total effective dose equivalent per year.

Radiation and contamination surveys are performed on a regular basis by the health physics staff at the PUR-1 facility, which provides adequate oversight of laboratory areas where work with radioactive materials is performed. This review showed that the placement, use, and control of the radiation monitoring and surveying equipment are in accordance with applicable national standards, guidance, and regulations. The review also verified that the selection of equipment used is appropriate for detecting the types and intensities of radiation likely to be encountered within the facility at appropriate frequencies to help ensure compliance with 10 CFR Part 20 requirements and the facility ALARA program under all operating conditions.

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

11.1.5 Radiation Exposure Control and Dosimetry

In the SAR, as supplemented, the licensee indicates that the reactor shielding is based on the combination of pool water and the concrete pool structure. The principal design feature for control of radiation exposure during operation is the column of water around and above the reactor core, plus the location of the reactor tank being partially below ground level. In response to RAI-6 in Ref. 20, the licensee performed an analysis at the proposed 12 kWt increase power level estimating the dose rate above the pool. The analysis used an acceptable shielding model with experimental buildup factors assuming that the reactor is at 18 kWt power (12 kWt plus 50 percent margin).

The dose rate for operating personnel is estimated at 0.032 mSv/hr (3.2 mrem/hr). This bounds dose rates that would be encountered during fuel handling operations, when the fuel is allowed to decay before handling. The dose to the member of the public in the unrestricted area outside the biological shield is estimated at $1x10^{-04}$ mSv/hr (0.01 mrem/hr), which is well below the limits

in 10 CFR Part 20.1301. The ventilation system maintains the reactor room at negative pressure with respect to outside areas and helps lower the concentration of Ar-41 and N-16 to levels that satisfy the dose limits in 10 CFR 20.1201, “Occupational Dose Limits for Adults.”

The regulations in 10 CFR 20.1502 require monitoring of workers likely to receive, in one year from sources external to the body, a dose in excess of 10 percent of the specified limits. In the SAR, as supplemented, the licensee states that personnel exposure is monitored by TLDs that are assigned to radiation workers who will be exposed to radiation to monitor whole body and extremity doses. Portable equipment is used to perform radiation surveys. Personnel protective equipment is used as needed. Facilities and equipment to decontaminate persons are available if needed. Procedures exist that govern the use of this equipment.

The licensee states that it uses survey meters to measure does rates from radiation fields, and these measured rates are posted where required. These provisions help ensure that external and internal radiation monitoring of all individuals required to be monitored meet the requirements of 10 CFR Part 20 and the goals of the facility ALARA program. The licensee states that it also maintains personnel exposure records and effluent and environmental monitoring readings for the life of the PUR-1 facility. The NRC staff reviewed the licensee’s annual operating records and NRC inspection reports from 2005 through 2015. The review showed that the highest annual dose equivalent incurred by PUR-1 staff complies with the facility’s ALARA program as well as the efficacy of the radiation exposure and control program. The NRC staff finds that all PUR-1 staff received significantly less radiation dose than the 10 CFR 20.1201 limits.

The NRC staff reviewed the information provided in the licensee’s SAR and the responses to RAI-6 in Ref. 20, and finds that the PUR-1 radiation exposure and control program is acceptable. The NRC staff finds historically low radiation doses and the application of the equipment and procedures used to be acceptable. The personnel exposures at PUR-1 facility are controlled through satisfactory radiation protection and ALARA programs. The NRC staff finds these conclusions are consistent with information provided in PUR-1 annual reports for the period 2005 through 2015. Based on the information provided above, the NRC staff concludes that the licensee’s exposure control and dosimetry programs are acceptable.

11.1.6 Contamination Control

In the PUR-1 SAR, the licensee indicates that it performs contamination surveys on a monthly basis; depending on the frequency that radioactive material is used or handled. Handling of any radioactive material within the PUR-1 facility is controlled by written procedure. Written procedure controls the handling of any radioactive material within the PUR-1 facility. Workers are trained in working with radioactive materials, including how to limit its spread when entering and exiting an area containing radioactive material. The facility surveys have routinely shown no detectable contamination in non-radiological areas of the facility. The NRC staff reviewed the licensee’s annual operating records and the NRC inspection reports from 2005 through 2015. This review showed that adequate controls exist to prevent the spread of radiological contamination within the facility.

Based on its review of the PUR-1 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 PUR-1 facility.

11.1.7 Environmental Monitoring

Environmental monitoring is discussed in Section 11.1.7 of the SER. The PUR-1 staff established an environmental radiation program that is conducted to measure the integrated radiation exposure in and around the environs of the facility on a bimonthly basis to help determine if radioactive effluents from the facility are in accordance with the ALARA criteria. In addition to monitoring all radioactive effluents released from the facility, thermoluminescent dosimeters, evaluated bimonthly are also used for environmental dose monitoring and assessment. Radioactive effluent measurements at the PUR-1 facility have not shown any detectable traces of Ar-41 and N-16 in the reactor room. A continuous air monitor (CAM) in the reactor room monitors any radioactive particulates released to the air.

The environmental monitoring program is reviewed and audited by the CORO as part of its review and audit of the radiation protection program as required by TS 6.2. The CORO review helps ensure that the environmental monitoring program contains an adequate number of locations, and sufficient frequency of collection such that the analysis of the data has sufficient sensitivity to ensure that the overall program complies with 10 CFR 20.1301, “Dose Limits for Individual Members of the Public,” and will provide an early indication of any environmental impact caused by the reactor facility operation. TS 6.7, Specification a, requires that the licensee include in its annual report to the NRC a description and summary of any environmental surveys performed outside the facility.

In response to a RAI from the NRC staff (Ref. 22), the licensee proposed a TS that requires environmental samples. The proposed TS 4.7 reads as follows:

> TS 4.7 Effluents
>
> Specification -
>
> a. Dosimetry shall be placed at the following locations
>
>   1. The location inside the reactor room which represents the hypothetical minimum distance a member of the public could reach to the reactor pool.
>
>   2. At the exhaust location of the reactor facility which is representative of effluent release from the reactor facility.
>
> b. Dosimetry shall be changed out according to the guidance of the Purdue Radiological Management on the same time period as facility personnel or semiannually, not to exceed 7 ½ months, whichever is lesser.

TS 4.7 contains the scope of environmental monitors and the frequency of dosimetry change out. The licensee explained in the RAI response that the pathways to the environment have been evaluated and they concluded that there is no credible pathway to the environment under normal conditions of operation. The reactor cooling water is sampled on a regular basis and no activity related to fuel leakage or activation is found. There is also no leakage of the reactor pool. Particulates that may be produced in the reactor room would be measured by the continuous air monitor but no activity related to reactor operations has been identified. Additionally, airborne emissions are filtered by a HEPA filter which would prevent the release of

particulate emissions. The only remaining pathway to the environment would be the release of gasses.

The dosimetry location near the top of the pool will indicate the dose at a point where the public has access during activities such as tours. The dosimetry location near the exhaust system of the reactor facility will determine the impact from Ar-41 releases. The NRC staff reviewed the licensee's environmental monitoring program and the results of the program as reported in the licensee's annual reports and NRC IRs. The reports indicated that the operation of the PUR-1 had not adversely affected the environment. The NRC staff finds that the environmental monitoring program can properly assess the operation of the facility to help minimize the radiological impact on the environment.

Based on the information provided above, the NRC staff concludes that the environmental monitoring program is sufficient to assess the radiological impact of the operation of the PUR-1 on the environment.

11.2 Radioactive Waste Management

Radioactive waste management is discussed in Section 11.2 of the licensee's SAR. The purpose of the radioactive waste (radwaste) management program is to minimize radioactive waste and ensure its proper handling, storage, and disposal. The licensee states that all radioactive waste handling operations are controlled by procedure and overseen by the Purdue University health physics staff. The PUR-1 generates radwaste in solid, liquid, and gaseous form. TS 6.7 specifies that annually the facility report to the NRC concerning waste management.

The NRC staff reviewed the specific annual reporting requirement related to radioactive waste releases and finds that they are consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. The licensee's annual operating reports summarize radioactive waste. The NRC staff finds that the reporting requirement and the results shown in the licensee's annual operating reports are acceptable (See Section 12.6 of this SER).

11.2.1 Radioactive Waste Management Program

Reactor operations at the PUR-1 reactor facility generate radwaste in gaseous, solid, and liquid form. Gaseous effluents, such as Ar-41 and N-16 are diluted and discharged through the facility stack. Liquid wastes are stored until released in accordance with 10 CFR Part 20. Solid wastes, such as routine laboratory wastes are properly packaged and stored until final disposition. All radwaste handling operations are controlled by procedure to help ensure compliance with the requirements of 10 CFR Part 20 and other appropriate NRC regulations.

The NRC staff reviewed the facility radioactive waste release practices and finds that these practices demonstrate reasonable assurance that radiological releases from the facility will not exceed applicable regulatory limits nor pose unacceptable radiation risk to the environment and the public. The NRC staff also finds that the licensee has adequate controls in place to prevent uncontrolled personnel exposures from radioactive waste operations and to provide the necessary accountability to prevent any potential unauthorized release of radioactive waste. Based on the information provided above, the NRC staff concludes that the licensee's radioactive waste management program is acceptable.

11.2.2 Radioactive Waste Controls

The licensee states that low-level solid radioactive waste from laboratory experiments or disposable protective clothing items are accumulated and stored in plastic-lined waste containers. Activated equipment and activated irradiation samples are stored in the reactor pool or in high level waste storage areas for reuse or to decay to low-level activity limits. When filled, the low-level waste container liner is sealed and stored in the radioactive waste storage building until the final disposition is determined under the Purdue University Broadscope license (By-product License, 13-02812-04) (Ref. 22). The licensee also states that 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. Adequate controls are in place to prevent uncontrolled personnel exposures from radwaste operations and provide the necessary accountability to prevent any potential unauthorized release of radwaste.

Based on the information provided above, the NRC staff finds that acceptable procedures are in place to monitor the radiation exposure from radioactive waste, and to perform required handling operations. Furthermore, the NRC staff concludes that the PUR-1 facility has adequate radioactive waste controls in place to monitor the radiation exposure from radioactive waste, to perform required handling operations, and prepare the material for transfer to offsite disposal.

11.2.3 Release of Radioactive Waste

The licensee states that normal operation of the PUR-1 reactor facility does not produce significant liquid radioactive waste. However, small quantities of liquid waste are periodically generated by sampling of the reactor pool and the primary coolant cooling water. Reactor building floor drains, laundry liquid effluent, laboratory sinks and drains, demineralizer regeneration, and decontamination shower water are collected and stored until determination of final disposition is determined. Sampling requirements ensure that all discharges are consistent with the PUR-1 ALARA program and ensure that they are within the limits stated in 10 CFR Part 20, Appendix B, Table 3 and 10 CFR 20.2003. The specific annual reporting requirement related to radioactive releases is consistent with NUREG-1537 and ANSI/ANS-15.1-2007.

As described in SAR Section 11, gaseous effluents are discharged through the roof vent while under the surveillance of the continuous air sampling system. This continuous monitoring helps ensure that effluents do not exceed 10 CFR Part 20 limits, and helps ensure the health and safety of the public and the environment.

Low level solid waste that decays to free release criteria is disposed of in the local landfill. High level radwaste remains onsite until it decays to the free release criteria or is shipped to a licensed disposal facility for disposition. The licensee adequately described the movement process and release practices of radioactive waste from controlled to uncontrolled areas.

Based on the information provided above, the NRC staff concludes that the licensee's controls and techniques for release of radioactive waste are acceptable. Furthermore, the NRC staff concludes that the PUR-1 facility has adequate controls in place to minimize releases of radioactive material into the environment.

11.3 Radiation Protection - Conclusions

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

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

- The licensee's ALARA program complies with the requirements of 10 CFR 20.1101(b). Review of controls for radioactive material at the PUR-1 provides reasonable assurance that radiation doses to the PUR-1 staff, the public, and the environment will be ALARA.

- The results of radiation surveys carried out at the PUR-1, doses to the persons issued dosimetry, and the results of the environmental monitoring program verify that the radiation protection and ALARA programs are effective, and in compliance with the requirements of 10 CFR 20.1501(a).

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

- Facility design and procedures control the potential exposures from N-16 and Ar-41 to the PUR-1 staff, the public and the environment. Conservative calculations of the quantities of these gases released into restricted and unrestricted areas provide reasonable assurance that doses to PUR-1 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 public and the environment.

The NRC staff reviewed the PUR-1 radiation protection program and radioactive waste management program summary as described in the SAR, as supplemented. The NRC staff finds that the licensee implemented adequate and sufficient measures to minimize radiation exposure to workers and the public. Furthermore, the NRC staff concludes that there is reasonable assurance that the PUR-1 radiation protection and radioactive waste management programs will provide acceptable radiation protection to its workers, the public, and the environment.

12 CONDUCT OF OPERATIONS

Chapter 12 of the licensee's SAR discusses the conduct of operations. Section 6 of the TSs, "Administrative Controls," includes requirements for the conduct of operations for the PUR-1. The administrative controls presented in TS 6 include responsibilities, facility organization, staff qualifications, training, the safety committee, operational review and audits, procedures, required actions, and reports and records. In response to RAIs from the NRC staff, the licensee has proposed a number of changes to Section 6 of the TSs to be consistent with the guidance of NUREG-1537 and ANSI/ANS-15.1-2007 (Ref. 29, 31).

The primary guidance for the development of administrative controls for research reactor operation is NUREG-1537 and ANSI/ANS-15.1-2007. The licensee's TS 6 is based on these guidance documents and the licensee used the guidance of ANSI/ANS 15.1-2007 (Ref. 31). 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 PUR-1 TSs is not identical to that of ANSI/ANS-15.1-2007 and NUREG-1537. However, this review considered these cases and determined that the licensee's proposed administrative controls met the intent of the guidance and are acceptable.

The following definitions are related to the conduct of operations as follows:

TS 1.0 DEFINITIONS

(…)

1.8 Direct Supervision – In visual and audible contact.

(…)

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

1.16 Licensed – See licensee.

1.17 Licensee – An individual or organization holding a license.

(…)

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

(…)

1.34 Readily Available on Call - Readily available on call shall mean the licensed senior operator shall be within a reasonable driving time (1/2 hour) or less than 15 miles from the reactor building, and the operator on duty is currently informed, and can rapidly contact the senior reactor operator by phone.

(…)

1.41 Senior Reactor Operator – An individual who is licensed to direct the activities of reactor operators. Such an individual is also a reactor operator.

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

(…)

These are standard definitions used in research reactor TS conforming to NUREG-1537 and ANSI/ANS-15.1-2007 (Ref. 29 and Ref. 31) and are therefore acceptable to the NRC staff.

12.1 Organization

TS 6.1 discusses the organizational responsibilities identified in the organizational chart as follows:

TS 6.1 Organization

The PUR-1 Facility is managed and run by members of the university’s College of Engineering, specifically the School of Nuclear Engineering. The Dean of the College of Engineering shall be the final authority on all PUR-1 matters. The Laboratory Director is responsible to the Dean for the administration and proper and safe operation of the facility. Figure 6.1 shows the administration chart for the PUR-1. The Committee on Reactor Operations advises the director of the PUR-1 on all matters or policy pertaining to safety. The Radiological Safety Officer provides advice concerning personnel and radiological safety and provides technical assistance and review in the area of radiation protection.

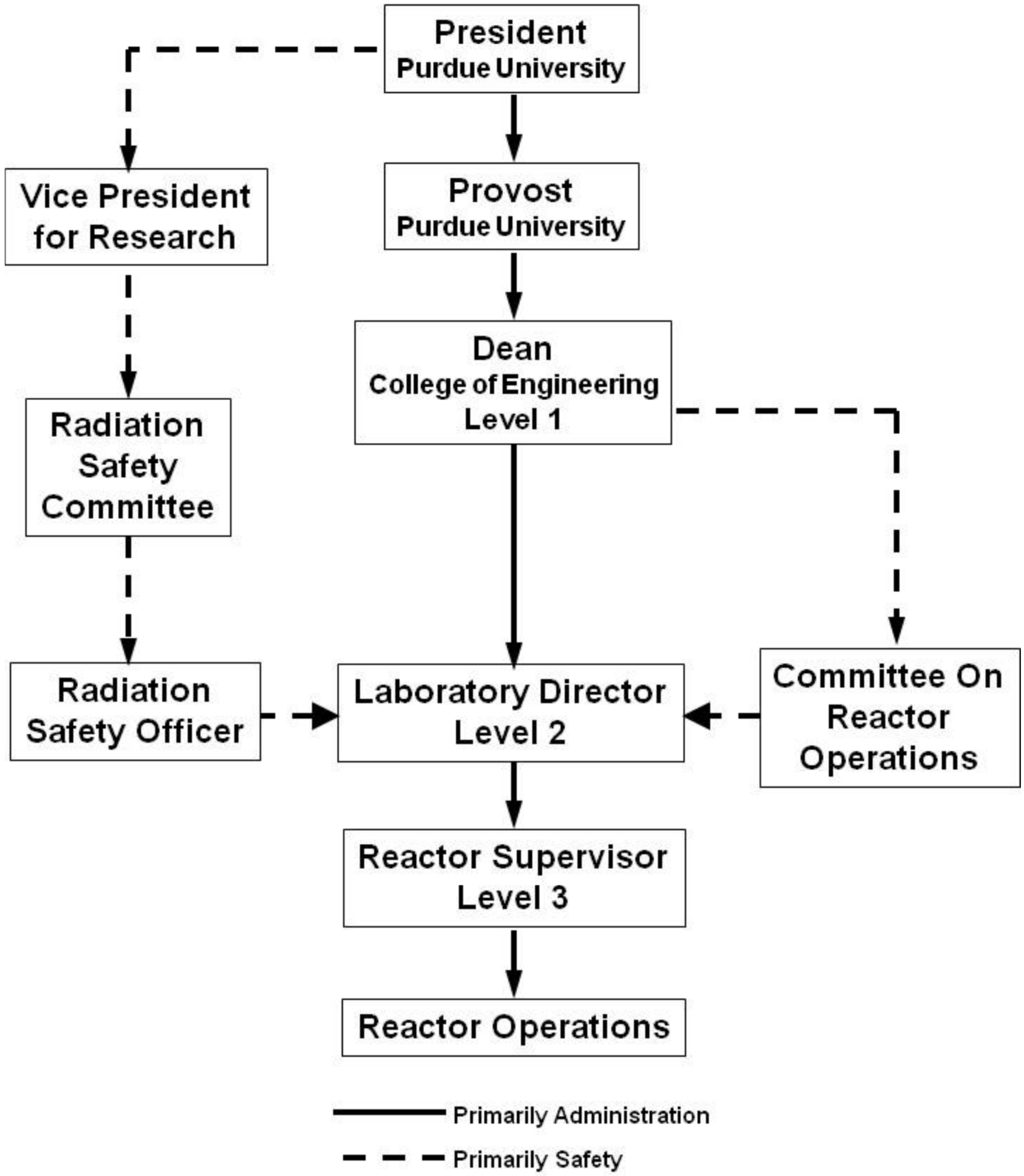


Figure 6.1: Organization Chart for Reactor Administration

TS 6.1 helps ensure that PUR-1 organizational responsibilities are delineated. The NRC staff finds that the organization described in TS 6.1 is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of the information provided above, the NRC staff concludes that TS 6.1 is acceptable.

a. Structure

1. A line management organizational structure provides for personnel who shall administrate and operate the reactor facility.

2. The Dean and the Facility Director shall have line management responsibility for adhering to the PUR-1 license and Technical Specifications and for safeguarding the public and facility personnel from undue radiation exposure.

3. Management Levels:

   a) Level 1: Dean of the College of Engineering: Responsible for the PUR-1.

   b) Level 2: PUR-1 Facility Director: Responsible for reactor facility operation and shall report to Level 1.

   c) Level 3: Reactor Supervisor: Responsible for the day-to-day operation of the PUR-1 including shift operation and shall report to Level 2.

   d) Level 4: Reactor Operating Staff: Licensed reactor operators and senior reactor operators and trainees. These individuals shall report to Level 3.

   e) The reporting structure of Figure 6.1 is such that those personnel below shall report up and those personnel listed above may communicate down.

4. Committee on Reactor Operations (CORO):

   The CORO shall be responsible to the licensee for providing an independent review and audit of the safety aspects of the PUR-1.

TS 6.1, Specification a, helps ensure that the PUR-1 organization structure is delineated. TS 6.1, Specification a, specifies the individuals at all levels being responsible for safeguarding the health and safety of the public and adhering to all requirements of the license, TS, and NRC regulations. The NRC staff reviewed TS 6.1, Specification a, and finds that the PUR-1 organizational structure described in TS 6.1, Specification a, and shown in Figure 6.1, is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of its review of the information provided above, the NRC staff concludes that TS 6.1, Specification a, is acceptable.

b. Responsibility

Responsibility for the safe operation of the reactor facility shall be in accordance with the line organization established in Section 6.1.a. In all instances, responsibilities of one level may be assumed by designated alternates or by higher levels, conditional upon appropriate qualifications.

The reactor facility shall be under the direct control of the Reactor Supervisor, a Senior Reactor Operator, or Reactor Operator (RO). The RO shall be responsible for ensuring that all operations are conducted in a safe manner and within the limits prescribed by the

facility license, procedures and requirements of the Radiation Safety Officer and the CORO.

TS 6.1, Specification b, helps ensure that the PUR-1 specific organization levels and responsibilities are delineated. The NRC staff reviewed TS 6.1, Specification b, and finds that the organizational responsibilities stated in TS 6.1, Specification b, are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of the information provided above, the NRC staff concludes that TS 6.1, Specification b, is acceptable.

c. Staffing

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

a) At least two individuals shall be present at the facility complex and shall consist of at least a licensed reactor operator and a second person capable of calling 911. Unexpected absence for as long as 2 hours to accommodate a personal emergency are acceptable provided immediate action is taken to obtain a replacement. During periods when the reactor is not secured, it shall be under the direct control the of the reactor operator

b) During periods of reactor maintenance the two individuals who shall be present at the facility complex shall consist of a licensed senior reactor operator and a second individual capable of calling 911.

c) A licensed reactor operator or senior reactor operator shall be in the reactor room;

d) A Senior Reactor Operator shall be readily available for emergencies or on call (the individual can be rapidly reached by phone or radio and is within 30 minutes or 15 miles of the reactor facility); and

e) A list of reactor facility personnel by name and telephone number shall be readily available for use in the reactor room. The list shall include:

i. Senior Reactor Operator on Call,
ii. Radiation Safety Officer
iii. Other operations personnel, as deemed appropriate by the Facility Director

2. Events requiring the presence at the facility of the senior reactor operator:

a) initial startup and approach to power,

b) A Senior Reactor Operator shall direct any loading or unloading of fuel or control rods within the reactor core region,

c) A senior reactor operator shall direct the recovery from an unplanned shutdown, unscheduled shutdown, or unplanned power reduction of more than 5%.

TS 6.1, Specification c, helps ensure the minimum staffing requirements and availability of an SRO for events specified in the regulation in 10 CFR 50.54(m)(1), which state “A senior operator licensed pursuant to part 55 of this chapter shall be present at the facility or readily available on call at all times during its operation, and shall be present at the facility during initial start-up and approach to power, recovery from an unplanned or unscheduled shut-down or significant reduction in power, and refueling, or as otherwise prescribed in the facility license.”

TS 6.1, Specification c, helps ensure that a reactor or senior reactor operator is at the reactor controls during operations, satisfying regulation in 10 CFR 50.54(k), which state “An operator or senior operator licensed pursuant to part 55 of this chapter shall be present at the controls at all times during the operation of the facility.” TS 6.1, Specification b, was modified in response to RAI 2 (Ref. 20), indicating that there are no exceptions for the required presence of a senior operator for recovery from an unplanned or unscheduled shutdown in accordance with the requirements listed in 10 CFR 50.54m)(1). The NRC staff finds that TS 6.1, Specification c, satisfy the requirements of 10 CFR 50.54(m)(1), and are also consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of the information provided above, the NRC staff concludes that TS 6.1, Specification c, is acceptable.

d. Selection and Training of Personnel

The selection and training of operations personnel shall be in accordance with the following:

1. Responsibility: The Reactor Supervisor is responsible for the selection, training, and requalification of the facility reactor operators and senior reactor operators.

2. Selection: The selection of operations personnel shall be consistent with the standards related to selection in ANSI/ANS-15.4-2007

3. Training Program: The Training Program shall be consistent with the standards related to training in ANSI/ANS-15.4-2007.

4. Requalification Program: The Requalification Program shall be consistent with the standards related to requalification in ANSI/ANS-15.4-2007.

TS 6.1, Specification d, describes the selection and training of personnel for key positions and helps ensure that the criteria for the training and requalification programs for operations personnel are consistent with the guidance in ANSI/ANS-15.4-2007, “Selection and Training of Personnel for Research Reactors” (Ref. 42). Requalification of licensed senior reactor operators and reactor operators also must follow the NRC approved Senior Reactor Operator and Reactor Operator Requalification Program. The qualification program meets the applicable regulations of 10 CFR Part 55, “Operators’ License”. The licensee uses ANSI/ANS 15.4 as guidance for the selection and training of personnel (See Section 12.10 of this SER).

The reactor supervisor serves as the training coordinator and is responsible for the implementation, coordination, and operation of the requalification program including the training of new operators. TS 6.1, Specification d, is consistent with the guidance in ANSI/ANS 15.4-2007 and NUREG-1537. The NRC staff supports the use of ANSI/ANS 15.4 by licensees for selection and training of personnel. Based on the information provided above, the NRC staff concludes that TS 6.1, Specification d, is acceptable.

The NRC staff evaluated the requirements of TS 6.1, Specifications a through d, and finds that they are consistent with the guidance in ANSI/ANS-15.1-2007 and NUREG-1537, and that they satisfy the requirements of 10 CFR Part 50.54(k) and 10 CFR Part 50.54(m)(1). On the basis of the information provided above, the NRC staff concludes that TS 6.1, Specifications a through d, are acceptable.

12.2 Review and Audit Activities

NUREG-1537 and ANSI/ANS 15.1 specify that the purpose of a reactor safety committee such as the licensee's Committee on Reactor Operations (CORO) is to provide independent oversight of reactor operations. The CORO exists to review and audit matters relating to the safe operation of the facility and the health and safety of the reactor staff, the public, and the environment.

TS 6.2 describes the overall responsibility, composition and qualification of members of the CORO and the associated rules as follows:

TS 6.2 Review and Audit

a. Committee on Reactor Operations (CORO)

The CORO shall be comprised of at least 3 voting members knowledgeable in fields which relate to Nuclear Safety. One of these members, the Radiation Safety Officer, will serve as the Chair. If the Chair is unable to attend one or a number of committee meetings, then the Chair may designate a committee member as Chair pro tem. The members are appointed by the Dean of the College of Engineering to serve three year terms. It is expected that the members will be reappointed each term as long as they are willing to serve so that their experience and familiarity with the past history of the PUR-1 will not be lost to the committee.

TS 6.2, Specification a, helps ensure that the CORO composition, qualifications, and operation are properly delineated. TS 6.2, Specification a, describes the composition of the CORO and requires the CORO to include experts who are not directly involved with the operation of the PUR-1.

The NRC staff reviewed TS 6.2, Specification a, and finds that the requirements in TS 6.2, Specification a, are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of the information provided above, the NRC staff concludes that TS 6.2, Specification a, is acceptable.

b. CORO Charter and Rules

The operations of the CORO shall be in accordance with a written charter, including provisions for:

1 Meeting Frequency: The CORO shall meet annually at intervals not to exceed 15 months. (Note: The facility license requires a meeting at least once per year and as frequently as circumstances warrant consistent with effective monitoring of facility activities);

2. Quorum: A quorum shall be comprised of not less than one-half of the voting membership where the operating staff does not constitute a majority;

3. Voting Rules: On matters requiring a vote, if only a quorum is present a unanimous vote of the quorum shall be required; otherwise a majority vote shall be required;

4. Subcommittees: The Chair may appoint subcommittees comprised of members of the CORO to perform certain tasks. Subcommittees or members of the CORO may be authorized to act for the committee; and

5. Meeting Minutes: The Chair shall designate one individual to act as recording secretary. It shall be the responsibility of the secretary to prepare the minutes which shall be distributed to the CORO, including the Dean of the College of Engineering, within three months. The CORO shall review and approve the minutes of the previous meetings. A complete file of the meeting minutes shall be maintained by the Chair of the CORO and by the Facility Director.

TS 6.2, Specification b, helps establish the CORO charter and rules. TS 6.2, Specification b, describes the operation of the CORO, which is responsible for an independent audit of the PUR-1 activities and conducts its review and audit functions in accordance with a written charter. The charter includes provisions for meeting frequency, voting rules, quorums, 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 CORO charter establishes a quorum of not less than half the voting membership, where the operating staff does not constitute a majority.

The NRC staff reviewed TS 6.2, Specification b, and finds that the composition and qualifications for the CORO, as stated in TS 6.2, Specification b, are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information described above, the NRC staff concludes that TS 6.2, Specification b, is acceptable.

c. CORO Review Function

The review responsibilities of the CORO or a designated subcommittee shall include, but are not limited to the following:

1. Review and evaluation of determinations of whether new tests and experiments and proposed changes to equipment, systems, or procedures can be made under 10 CFR 50.59 or would require a change in Technical Specifications or license conditions;

2. Review of new procedures, major revisions of procedures, and proposed changes in reactor facility equipment or systems which have significant safety impact to reactor operations

3. Review of new experiments or classes of experiments that could affect reactivity or result in the release of radioactivity;

4. Review of proposed changes to the Technical Specifications and U.S. NRC issued license;

5. Review of the PUR-1 radiation protection program;

6. Review of violations of Technical Specifications, U.S. NRC issued license, and violations of internal procedures or instructions having safety significance;

7. Review of operating abnormalities having safety significance;

8 Review of reportable occurrences listed in Section 6.6.a and 6.6.b of these Technical Specifications; and

9. Review of audit reports.

TS 6.2, Specification c, helps ensure that the CORO review functions are properly delineated. The NRC staff reviewed TS 6.2, Specification c, and finds that the CORO review functions specified in TS 6.2, Specification c, are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 6.2, Specification c, is acceptable.

d. CORO Audit Function

The audit function shall include selective (but comprehensive) examination of operating records, logs, and other documents. Discussions with cognizant personnel and observation of operations should be used also as appropriate. In no case shall the individual immediately responsible for an area perform an audit in that area. Audits shall include but are not limited to the following:

1. Facility operations, including radiation protection, for conformance to the Technical Specifications, applicable license conditions, and standard operating procedures: at least every 12 months (interval between audits not to exceed 15 months);

2. The results of action taken to correct those deficiencies that may occur in the reactor facility equipment systems, structures, or methods of operations that affect reactor safety: at least once every 12 months (interval between audits not to exceed 15 months);

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

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

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

   Deficiencies uncovered that affect reactor safety shall immediately be reported to the Dean of the College of Engineering (Level 1 Management). A written report of the findings of the audit shall be submitted to the Dean of the College of Engineering (Level 1 Management) and the review and audit group members within 3 months after the audit has been completed.

TS 6.2, Specification d, helps ensure that the CORO audit functions are properly delineated. The licensee specifies that no individual responsible for an area may conduct the audit of that area. Written reports of audit findings are submitted to Level 1 management within three months after completion of the audit. The NRC staff reviewed TS 6.2, Specification d, and finds that the CORO audit functions, as stated in TS 6.2, Specification d, are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 6.2, Specification d, is acceptable.

e. Audit of ALARA Program

> The Chair of the CORO or designated alternate (excluding anyone whose normal job function is within the operating staff) shall conduct an audit of the reactor facility ALARA program annually. The auditor shall transmit the results of the audit to the CORO at the next scheduled meeting for its review and approval.

TS 6.2, Specification e, helps ensure that the Chair of the CORO or designated alternate performs an annual audit of the ALARA program. The NRC staff reviewed TS 6.2, Specification e, and finds that TS 6.2, Specification e, is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 6.2, Specification e, is acceptable.

The NRC staff reviewed TS 6.2 and finds the licensee's review and audit functions are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 6.2 is acceptable.

12.3 <u>Radiation Safety</u>

TS 6.3 contains the administrative requirements for radiation safety. The radiation safety program is discussed in Section 11.1.2 of this SER.

TS 6.3 Radiation Safety

> The Radiation Safety Officer shall be responsible for implementing the radiation safety program for the PUR-1. The requirements of the radiation safety program are established in 10 CFR 20. The Program should use the guidelines of the ANSI/ANS-15.11-1993; R2004, "Radiation Protection at Research Reactor Facilities."

As discussed in Section 11.1.2 of this SER, the NRC staff reviewed the PUR-1 radiation protection program, as described in the SAR, as supplemented, and finds that the program complies with 10 CFR 20.1101(a), is implemented in an acceptable manner, and provides reasonable assurance that, for all facility activities, the program will protect the PUR-1 staff, the environment, and the public from unacceptable radiation exposures. The NRC staff concludes that the PUR-1 radiation protection program, as described, and TS 6.3 are acceptable.

12.3 <u>Procedures</u>

Operating procedures are written by the reactor staff and reviewed by the CORO. Changes to existing procedures are also reviewed by the CORO for safety significance impact.

TS 6.4 specifies the type of written procedures that must be prepared and approved prior to use as follows:

TS 6.4 Procedures

Written operating procedures shall be prepared, reviewed, and approved before initiating any of the activities listed in this section. The procedures shall be reviewed and approved by the Facility Director, the CORO, and shall be documented in a timely manner. Procedures shall be adequate to ensure the safe operation of the reactor but shall not preclude the use of independent judgment and action should the situation require such. Operating procedures shall be used for the following items:

a. Startup, operation, and shutdown of the reactor;

b. Fuel loading, unloading, and movement within the reactor;

c. Control rod removal or replacement;

d. Routine maintenance of the control rod, drives and reactor safety and interlock systems or other routine maintenance of major components of systems that could have an effect on reactor safety;

e. Surveillance checks, calibrations, and inspections of reactor instrumentation and controls, control rod drives, area radiation monitors, facility air monitors, the central exhaust system and other systems as required by the Technical Specifications;

f. Administrative controls for operations, maintenance, and conduct of irradiations and experiments, that could affect reactor safety or core reactivity;

g. Implementation of required plans such as emergency or security plans;

h. Radiation protection program to maintain exposures and releases as low as reasonably achievable (ALARA);

i. Use, receipt, and transfer of by-product material, if appropriate; and

j. Surveillance procedures for shipping radioactive materials.

TS 6.4 helps ensure the items that must be covered by standard operating procedures are identified. TS 6.4 helps ensure that procedures are written, reviewed, and approved before performance of the group of important activities listed in the specifications. 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. The NRC staff reviewed TS 6.4, and finds that TS 6.4 is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 6.4 is acceptable.

12.5 Experiment Review and Approval

TS 6.5 states the following regarding the administrative aspects of experiment review and approval:

TS 6.5 Experiment Review and Approval

Approved experiments shall be carried out in accordance with established and approved procedures.

a. All new experiments or class of experiments shall be reviewed by the CORO as required by TS 6.2.c and implementation approved in writing by the Facility Director or designated alternate.

b. Substantive changes to previously approved experiments shall be made only after review by the CORO and implementation approved in writing by the Facility Director or designated alternate.

TS 6.5 helps ensure acceptable management control over PUR-1 experiments. TS 6.5 provides requirements for the review and approval of different types of experiments before they are performed at the PUR-1 and specifies the extent of the analysis needed to be submitted for CORO review. The NRC staff reviewed TS 6.5, and finds that TS 6.5 is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of the information provided above, the NRC staff concludes that TS 6.5, Specifications a and b, are acceptable.

12.6 Required Actions

The licensee has defined the required actions for events in TS 6.6 including those actions to be taken in case of a safety limit violation and a reportable occurrence, as described below:

TS 6.6 Required Actions

a. Action to be Taken in the Event of a Safety Limit Violation

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

2. An immediate notification of the occurrence shall be made to the CORO Chair and the Facility Director, and reports shall be made to the U.S. NRC in accordance with Section 6.7.b of these specifications; and

3. A report shall be prepared which shall include:

   a) Applicable circumstances leading to the violation including, when known, the cause and contributing factors,

   b) Effect of the violation upon reactor facility components, systems, or structures and on the health and safety of personnel and the public,

   c) Corrective action to be taken to prevent recurrence.

This report shall be submitted to the CORO for review and then submitted to the U.S. NRC when authorization is sought to resume operation of the reactor.

TS 6.6, Specification a, helps ensure that the proper actions are taken if an SL violation occurs. TS 6.6, Specification a, requires the facility to shut down if a SL is exceeded. The facility may not resume operation without authorization from the NRC. The violation also must be reported to the CORO and to the NRC. TS 6.7, Specification b, details the reporting requirement, specifying that the NRC must be notified within the next working day by telephone and confirmed in writing and a report must be submitted to the NRC within 14 days. TS 6.6, Specification a, item 3.c, specifies that the report should include the corrective actions to be taken to prevent recurrence. The NRC staff finds that TS 6.6, Specification a, is in conformance with the requirements of 10 CFR 50.36, and is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of its review of the information provided above, the NRC staff concludes that TS 6.6, Specification a, is acceptable.

TS 6.6, Specification b, reads as follows:

b. Action to be Taken in the Event of a Reportable Occurrence Other Than A Safety Limit Violation

    1. PUR-1 staff shall return the reactor to normal operating via the approved PUR-1 procedure or shut down conditions. If it is necessary to shut down the reactor to correct the occurrence, operations shall not be resumed unless authorized by the Facility Director or a designated alternate;

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

    3. The Facility Director or designated alternate shall notify the CORO Chair who shall arrange for a review by the CORO;

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

    5. A report shall be made to the U.S. NRC in accordance with Section 6.7.b of these specifications.

TS 6.6, Specification b, helps ensure that actions are taken if a reportable occurrence other than a SL violation occurs. TS 6.6, Specification b, helps ensure that the reactor is returned to normal operation or shut down, the PUR-1 Director is notified and corrective actions are taken, the CORO Chairman is notified to arrange a review by the CORO, and the NRC is notified within the next working day by telephone and confirmed in writing and by written report within 14 days. The NRC staff finds that TS 6.6, Specification b, provides acceptable notification of the PUR-1 Director and the NRC, provides for a review by the CORO, provides corrective actions, and is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of its review of the information provided above, the NRC staff concludes that TS 6.6, Specification b, is acceptable

12.7 Reports

TS 6.7 contains reporting requirements for annual reports and special reports as follows:

TS 6.7 Reports

a. Annual Operating Report

An annual report covering the operation of the reactor facility during the previous calendar year shall be submitted to the NRC before March 31 of each year providing the following information:

1. A narrative summary of (1) reactor operating experience (including experiments performed), (2) changes in facility design, performance characteristics, and operating procedures related to reactor safety and occurring during the reporting period, and (3) results of surveillance tests and inspections;

2. Tabulation of the energy output of the reactor, hours reactor was critical, and the cumulative total energy output since initial criticality;

3. The number of unscheduled shutdowns and inadvertent scrams, including, where applicable corrective action to preclude 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 Section 50.59 of 10 CFR Part 50;

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. The summary shall include to the extent practicable an estimate of individual radionuclides present in the effluent. If the estimated average release after dilution or diffusion is less than 25% of the concentration allowed or recommended, a statement to this effect is sufficient:

   a) Liquid Waste (summarized on a monthly basis)

      i. Radioactivity discharged during the reporting period.

         I. Total radioactivity released (in Curies),

         II. The effluent concentration used and the isotopic composition if greater than 1 x 10-7 µCi/cc for fission and activation products,

         III. Total radioactivity (in Curies), released by nuclide during the reporting period based on representative isotopic analysis, and

IV. Average concentration at point of release (in µCi/cc) during the reporting period.

ii. Total volume (in gallons) of effluent water (including dilution) during periods of release.

b) Airborne Waste (summarized on a monthly basis)

i. Radioactivity discharged during the reporting period (in Curies) for:

I. 41Ar, and

II. Particulates with half-lives greater than eight days.

c) Solid Waste

i. The total amount of solid waste transferred (in cubic feet),

ii. The total activity involved (in Curies), and

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

7. A summary of radiation exposures received by facility personnel and visitors, including dates and time where such exposures are greater than 25% of that allowed or recommended; and

8. A description and summary of any environmental surveys performed outside the facility.

TS 6.7, Specification a, helps ensure that adequate annual reporting information is maintained. TS 6.7, Specification a, provides requirements for the status of the facility, major changes, radiation exposures, and other pertinent information to be provided to the NRC. The NRC staff finds that TS 6.7, Specification a, provides PUR-1 facility annual operating report requirements that are consistent with guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of its review of the information provided above, the NRC staff concludes that TS 6.7, Specification a, is acceptable.

TS 6.7, Specification b, contains the requirements for special reports as follows:

b. Special Reports

In addition to the requirements of applicable regulations, reports shall be made to the NRC Document Control Desk and special telephone reports of events should be made to the Operations Center as follows:

1. There shall be a report not later than the following working day by telephone and confirmed in writing by fax or similar conveyance to the NRC Headquarters Operation Center, and followed by a written report that describes the circumstances of the event and sent within 14 days to the U.S. Nuclear

Regulatory Commission, Attn: Document Control Desk, Washington, DC 20555, of any of the following:

a) Violation of safety limit (see TS 6.6.a);

b Any release of radioactivity from the site above allowed limits; and

c) Any of the following:

   i. Operation with actual safety system settings for required systems less conservative than the limiting safety system settings specified in the technical specifications

   ii. Operation in violation of limiting conditions for operation established in the technical specifications.

   iii. A reactor safety system component malfunction that renders or could render the reactor safety system incapable of performing its intended safety function. If the malfunction or condition is caused by maintenance, then no report is required.

   Note: Where components or systems are provided in addition to those required by the technical specifications, the failure of the extra components or systems is not considered reportable provided that the minimum numbers of components or systems specified or required perform their intended reactor safety function.

   iv. An unanticipated or uncontrolled change in reactivity greater than 0.006 $\Delta k/k$.

   v. Abnormal and significant degradation in reactor fuel or cladding, or both, coolant boundary, or confinement boundary (excluding minor leaks).

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

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

   a) Permanent changes in the facility organization involving Level 1 and Level 2; and

   b) Significant changes in the transient or accident analysis as described in the Safety Analysis Report.

TS 6.7, Specification b, helps ensure that special reporting requirements are adequately delineated. The NRC staff reviewed TS 6.7, Specification b, and finds that the special report requirements are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of the information provided above, the NRC staff concludes that TS 6.7, Specification b, is acceptable.

12.8 Records

Records maintained for the life of the facility and for a five year time frame are properly stored in the Reactor Supervisor's office. More recent documents are kept on electronic media and stored on existing network servers. Security related documentation is kept in a secure repository.

TS 6.8 lists the type of required records that must be retained at the facility and specifies the duration of the retention period. This ensures that important documents remain available for various administrative purposes, consistent with guidance set forth in ANSI/ANS-15.1-2007 and NUREG-1537. TS 6.8 reads as follows:

TS 6.8 Records

Records of facility operations in the form of logs, data sheets, or other suitable forms shall be retained for the period indicated as follows:

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

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

2. Principal maintenance operations,

3. Reportable occurrences,

4. Surveillance activities required by the Technical Specifications,

5. Reactor facility radiation and contamination surveys where required by applicable regulations,

6. Experiments performed with the reactor,

7. Fuel inventories, receipts, and shipments,

8. Approved changes in operating procedures, and

9. Records of meeting and audit reports of the CORO.

b. Records to be Retained for at Least One Certification Cycle

Records of retraining and requalification of licensed operations personnel shall be maintained at all times the individual is employed or until the license is renewed.

c. Records to be Retained for the Lifetime of the Reactor Facility

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

2. Radiation exposure for all personnel monitored,

3. Drawings of the reactor facility, and

4. Reviews and reports pertaining to a violation of the safety limit, the limiting safety system setting, or a limiting condition of operation.

TS 6.8 specifies the type of required records that must be retained and specifies the duration of that retention for various functions and also situations that may occur during PUR-1 operations. This ensures that important information is made available in a timely manner for proper management oversight, which is consistent with guidelines set forth in ANSI/ANS 15.1-2007.

TS 6.8, Specification a, helps ensure that the PUR-1 facility records retention requirements for records to be retained for at least five years are properly delineated in the TSs. The NRC staff finds that the record requirements in TS 6.8, Specification a, are consistent with the guidance in NUREG-1537 and ANSI/ANS 15.1-2007. On the basis of its review of the information provided above, the NRC staff concludes that TS 6.8, Specification a, is acceptable.

TS 6.8, Specification b, helps ensure that the licensee maintains training records of licensed operators while employed or until the license is renewed. The NRC staff reviewed TS 6.8, Specification b, and finds that the record retention requirements stated in TS 6.8, Specification b, are consistent with the guidance in NUREG-1537 and ANSI/ANS 15.1- 2007. On the basis of the information provided above, the NRC staff concludes that TS 6.8, Specification b, is acceptable.

TS 6.8, Specification c, helps ensure that the PUR-1 facility records retention requirements for records that need to be retained for the lifetime of the PUR-1 facility are appropriately delineated. The NRC staff reviewed TS 6.8, Specification c, and finds that TS 6.8, Specification c, for record retention requirements is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of the information provided above, the NRC staff concludes that TS 6.8, Specification c, is acceptable.

The licensee has modified TS 6.8, Specification c, for completeness and conformity to 10 CFR 50.36. The NRC staff finds that TS 6.8 is consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 6.8 is acceptable.

12.9 <u>Emergency Planning</u>

The license renewal request referenced the current version of the PUR-1 Emergency Plan on file with the NRC. In the SAR, the licensee provided a revision of the PUR-1 Emergency Plan

as changed under 10 CFR 50.54(q). The NRC staff reviewed the PUR-1 Emergency Plan and finds that it meets the applicable regulations, and based on that finding, concludes that the PUR-1 Emergency Plan is acceptable. Additionally, the licensee is required to maintain the EP, in compliance with 10 CFR 50.54(q) and Appendix E to 10 CFR Part 50, “Emergency Planning and Preparedness for Production and Utilization Facilities,” which reasonably ensures that the licensee will continue to be prepared to assess and respond to emergency events. The NRC staff routinely inspects the licensee’s compliance with the requirements of the emergency plan, and no violations have been identified in recent years.

12.10 <u>Security Planning</u>

The NRC staff reviewed the PUR-1 physical security plan entitled, “Security Plan for the Purdue University Reactor, the Fast Breeder Blanket Facility and the Nuclear Fuel Storage Areas,” Revision 4, dated October 24, 2000 (Ref. 51), and as changed under 10 CFR 50.54(p). The NRC staff issued RAIs to the licensee in a letter dated December 23, 2015 (Ref. 52), and the licensee responded by letters dated January 29 (Ref. 53), February 26 (Ref. 54), March 31 (Ref. 55), and May 9, 2016 (Ref. 56), including a revised PUR-1 physical security plan.

The NRC staff reviewed the revised PUR-1 physical security plan entitled, “Purdue University Reactor-1 Physical Security Plan” dated March 2016, and finds that it meets the applicable regulations, and based on that finding, concludes that the “Purdue University Reactor-1 Physical Security Plan,” dated March 2016, is acceptable. The licensee maintains the program to provide the physical protection of the facility and its SNM in accordance with the requirements of 10 CFR Part 73. Changes to the physical security plan can be made, by the licensee, in accordance with 10 CFR 50.54(p), as long as those changes do not decrease the effectiveness of the plan.

In addition, the NRC staff performs routine inspections of the licensee’s compliance with the requirements of the physical security plan. The NRC staff’s review of the NRC Security IRs from years 2010 through 2015 (Ref. 57) for the PUR-1 facility for the past six years has identified no violations of the security plan requirements.

12.11 <u>Quality Assurance</u>

The Laboratory Director and the Reactor Supervisor are responsible for quality assurance activities at the PUR-1. This includes monitoring reactor operator performance, ensuring testing of reactor equipment and instrument calibrations are carried out properly, and repairs are properly performed. Appropriate documentation of these activities is reviewed by the CORO, which has the technical specification oversight responsibility.

12.12 <u>Operator Training and Requalification</u>

The license renewal request referenced the current version of the operator requalification plan entitled, “Operator Requalification Program for the PUR-1 Reactor Facility.” The NRC staff reviewed the operator requalification plan and issued RAIs to the licensee in letters dated July 8, 2011 (Ref. 24) and August 29, 2014 (Ref. 26). The licensee provided its responses in letters dated January 31, 2012 (Ref. 13) and July 24, 2015 (Ref. 20), including a revised operator requalification plan. The NRC staff reviewed the revised operator requalification plan entitled, “Operator Requalification Program for the PUR-1 Reactor Facility,” dated March 29, 2015 and finds that it meets the applicable requirements of Title 10 of the *Code of*

Federal Regulations, Part 55 “Operators’ Licenses,” and is consistent with the guidance provided in ANSI/ANS-15.4, “Selection and Training of Personnel for Research Reactor.” Based on this finding, the NRC staff concludes that the licensee’s operator requalification program and training program provide reasonable assurance that the licensee will have technically qualified reactor operators and senior reactor operators.

12.13 Conclusions

Based on the information provided above, the NRC staff concludes that the licensee has sufficiently experienced oversight, management structure, and procedures to provide reasonable assurance that the reactor will continue to be managed in a way that will cause no significant risk to the health and safety of the public. The NRC staff has reviewed SAR Chapter 12 and Section 6 of the TS, which discuss the licensee’s proposed organization, training including operator requalification, review and audit activities, administration of radiation protection activities, procedures, experiment review, required actions, and records and reports against the guidance given in ANSI/ANS-15.1-2007 and NUREG-1537. The licensee's proposed conduct of operations in the areas reviewed is consistent with the guidance of the ANSI/ANS-15.1-2007 standard and NUREG-1537. The staff also reviewed Section 6 of the TSs against 10 CFR 50.36 “Technical Specifications,” and concludes that the TSs meet the requirements of the regulation.

The staff has reviewed the Purdue University Research Reactor Physical Security Plan, and concludes that the licensee’s security plan meets the requirements of 10 CFR Part 73.67 and 10 CFR Part 73.60 for protecting the special nuclear material associated with the facility.

The staff has reviewed the Purdue University Research Reactor Emergency Response Plan and concludes that the licensee’s emergency plan meets the requirements of the regulations in 10 CFR Part 50, Appendix E and is therefore acceptable.

Based on the above discussion, the NRC staff concludes that the licensee has the appropriate organization, experience levels, and adequate controls through the TS, to provide reasonable assurance that the PUR-1 is managed and operated in a manner which will not cause significant radiological risk to facility staff or to the public.

13 ACCIDENT ANALYSIS

13.1 Accident Initiating Events

The accident analysis presented in the SAR, as supplemented, for the PUR-1 helped establish the safety limit (SL) and limiting safety system settings (LSSSs) that are imposed on the PUR-1 through the TSs. The licensee analyzed potential reactor transients and hypothetical accidents including the potential effects of natural hazards. The NRC staff reviewed the licensee's analytical assumptions, methods, and results. In addition, the NRC staff performed limited, independent calculations and obtained independent analysis of accidents with other MTR reactors and compared those results with accidents analyzed by the licensee. As will be demonstrated below, none of the potential accidents considered in the SAR, as supplemented, would lead to significant occupational or public exposure.

The PUR-1 SAR, in conjunction with the licensee's response to RAIs, provided accident analyses to demonstrate that the health and safety of the public and workers are protected during analyzed reactor transients and other hypothetical accident scenarios. The accident analysis helps ensure that no credible accident could lead to unacceptable radiological consequences to the PUR-1 staff, the public, or the environment. Additionally, the licensee analyzes the consequences of the maximum hypothetical accident (MHA), which is an event involving the cladding failure of an irradiated fuel element in air. The MHA is considered the worst-case fuel failure scenario for PUR-1 that would lead to the maximum potential radiation hazard to facility personnel and members of the public. The results of the MHA are used to evaluate the ability of the licensee to respond and mitigate the consequences of this postulated radioactive release.

None of the credible accidents postulated would lead to the failure of the cladding of any fuel plates or the uncontrolled release of fission products. However, the licensee postulated an enveloping, MHA event involving fuel cladding failure exposing fuel surface areas equivalent to stripping all the cladding from one face of one fuel plate. The postulated fuel plate cladding failure is consistent with the MHA failure criterion for MTR fuel elements as recommended in NUREG-1537. This event would lead to the maximum potential radiation hazard from fuel failure to facility personnel and members of the public. The licensee makes no assumptions as to the cause of the failure. The licensee evaluated only the potential consequences of this event, not the likelihood or mechanisms of the event's occurrence. The licensee and the NRC staff have evaluated other possible accident sequences that originate in the intact reactor core; none pose a significant risk of cladding failure or release of fission products.

The basic limit used to insure safe operation of the MTR-type fueled reactors is the fuel temperature. The NRC has accepted that no fuel damage or cladding failure is expected if the fuel temperature never exceeds 550 °C (1022 °F) in an MTR (uranium-silica) fuel plate (Ref. 35). The safety limit given in TS 2.1 is conservatively lower at 530 °C (986 °F). This is also the criterion used for a loss-of-coolant accident with air cooling and no fuel damage (Ref. 3).

The typical energy generation of the PUR-1 is less than 3 kW-hr/yr and routine operation will not result in a saturated fission product inventory (Ref. 7). Expected operations at 12 kWt power level would also result in fission products less than the saturation levels. However, because

there is no limitation on operating time in the license or TS, the accident analyses assume an inventory of saturated fission products.

The PUR-1 SAR considers the following accident scenarios:

- Maximum Hypothetical Accident
- Insertion of excess reactivity
- Loss-of-coolant accident (LOCA)
- Loss of coolant flow
- Mishandling or malfunction of fuel
- Experiment malfunction
- Loss of normal electrical power
- External events
- Mishandling or Malfunction of Equipment

13.2 Accident Scenarios

13.2.1 Maximum Hypothetical Accident

The licensee discusses the MHA in Section 13.2.1 of the SAR. The MHA is defined as a hypothetically conceived fission product release accident for which the risk to the public health and safety is greater than that from any other fission product release event that can be postulated mechanistically, and as such, it bounds the consequences of all other fission product release postulated accidents for the facility. The licensee assumes that the accident occurs, but does not attempt to describe or evaluate all of the mechanical details of the accident or the probability of its occurrence; only the consequences are considered. In defining the MHA, the licensee stated that a fuel plate would undergo severe mechanical damage resulting in fuel cladding failure exposing fuel surface areas equivalent to stripping all cladding from one face of one fuel plate. It is assumed that 100 percent of the gaseous activity including iodine produced within the recoil range of the exposed fuel surface area (2.07 percent of the total volatile activity) instantaneously escapes into the reactor room air (Ref. 43).

In response to RAI 11 (Ref. 20) the licensee demonstrated that the consequences of the failure of the cladding of one face of one fuel plate in air bounds potential accidents from fueled experiments performed within the limits of proposed TS 3.5. The NRC staff finds that the postulated fuel plate cladding failure is consistent with the MHA failure criteria for MTR fuel elements as recommended in NUREG-1537. The MHA scenario also assumes the instantaneous release of the noble gases and halogen fission products directly into the reactor room air without radioactive decay. Boundary conditions and assumptions include the use of conservative fuel element power density with saturated inventories and no credit for dilution or filtration removal by the ventilation system. The released nuclides would diffuse in the air of the reactor room and ultimately be released to the environment.

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

facilities. The licensee must follow the requirements of 10 CFR Part 20, as amended, for all aspects of facility operation.

Nuclide Inventory

The licensee calculated fission product inventory assuming that the reactor had been operating for an infinite period of time at 18 kWt, which is 50 percent higher than the maximum proposed normal operating power level of 12 kWt. All fission products had reached their saturated activity (a conservative assumption considering the typical operating history of the PUR-1). The analysis was done for a fuel plate with the maximum power density derived from the steady-state MCNP reactor core analysis, maximizing the fission product inventory. The MCNP code is the generally accepted code used by the nuclear industry for fission product generation.

The NRC staff compared the licensee's calculations of fission product inventory to estimates of radionuclide inventories in NUREG/CR-2079, "Analysis of Credible Accidents for Argonaut Reactors," Battelle Pacific Northwest Laboratory/NRC, April 1981 (Ref. 43), for select halogens and noble gases at shutdown. Based on the information provided above, the NRC staff concluded that the licensee's MHA radioactive source estimates are acceptable.

Release Fractions

The licensee calculated the fraction of fission products released from the fuel matrix based on data presented in NUREG/CR-2079. The report suggests that 100 percent of the gaseous activity and iodine produced within the recoil range of the exposed fuel surface area (2.07 percent of the total volatile activity) instantaneously escapes into the reactor room air (Ref. 43).

The licensee assumed that the fuel clad failure occurs in the air, and the noble gases and halogens in the recoil range release directly to the reactor room. This assumption is conservative for halogens (e.g., iodines) because they are chemically active and are not volatile below about 180 °C (356 °F). In addition, iodines typically become trapped by materials with which they come into contact.

The guidance typically used (Technical Information Document (TID) 14844, "The Calculations of Distance Factors for Power and Test Reactor Sites," March 1962; and US NRC RG 1.4 "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," June 1974) indicates that most iodines either will not become airborne, or will not remain airborne after they are released. This guidance provides that only 50 percent of the halogens are released to the outside environment. However, to be certain that the MHA failure scenario leads to upper limit dose estimates; the licensee assumed that 100 percent of the iodines become airborne and could be released to the environment.

The analysis assumed no radioactive decay during the time of the accident initiation and fission product release into the reactor room air. All halogen and noble gas activity released to the reactor room is instantly mixed uniformly with the air. There will be a time delay between the time of fuel plate failure and fission product release to the environment with uniform mixing. This assumption ignores radioactive decay during the finite mixing time. All of the reactor room air containing the radioactive gases is available for release to the environment.

Atmospheric Dispersion Factor χ/Q

The licensee used the Gaussian plume diffusion model with a building vent release to compute the nuclide concentrations and consequent doses external to the building. This methodology conservatively assumes that the release occurs at ground level. The licensee assumed a highly stable atmospheric class (Pasquill F) with a wind speed of 1 meter/second, and neglected deposition processes. The licensee considered horizontal meandering processes using Equations (1)-(3) of US NRC Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1, issued February 1983 (Ref. 44). In addition, the licensee estimated the dispersion factors in lateral (y-axis) and axial (z-axis) axis from Fig. 1 and 2 of RG 1.145. The licensee further assumed that the prevailing wind at the time of the release is always in the direction of the receptor.

The NRC staff used the RG 1.145, Revision 1, method to determine the atmospheric dispersion factors at the selected distance from the reactor building for comparisons. The NRC staff verified by comparison that the licensee's method and data used in the atmospheric dispersion factor calculations for the PUR-1 MHA dose calculations were reasonable, conservative, and acceptable.

Dose Calculations

The licensee calculated the occupational dose for an individual in the reactor room. Boundary conditions for these calculations included assuming the failure of the fuel element with the peak power density, incorporating conservative release fractions, and using the reactor room volume of 424 m^3 (14973 ft^3). Other parameters used in the dose calculations include a breathing rate of 333 cm^3/s (0.71 ft^3/min) consistent with the value given in Appendix B to 10 CFR Part 20, and a ventilation rate of 0.2 m^3/s (423 ft^3/min). In addition, the licensee used dose conversion factors (DCFs) for the inhalation and submersion external exposure pathways from the U.S. Environmental Protection Agency (EPA) Federal Guidance Report (FGR) No. 11, "Limiting Values of Radionuclide Intake and Air Concentration, and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," issued September 1988 (Ref. 45). The licensee used DCFs for submersion in the air to the halogen isotopes from FGR No. 12, "External Exposure to Radionuclides in Air, Water, and Soil," issued September 1993 (Ref. 46).

The licensee performed MHA dose analyses for a number of scenarios in responses to RAI12 (Ref. 20), RAI 11 (Ref. 21), and RAI 73(Ref. 22) providing bounding dose estimates for an MHA event. The ventilation system is required to be operating during reactor operation in accordance with PUR-1 TS 3.4, Specification a, item 1. The licensee stated in its response (Ref. 22) that if a significant radiological release occurs, reactor operators maintain the ventilation system operating, because this would reduce the radionuclide concentrations in the reactor room and the dose to operating staff. As demonstrated by the calculations below, leaving the ventilation system on would also result in doses that are well below 10 CFR Part 20 dose limits for members of the public outside the reactor building. The scenarios considered are as follows:

Scenario 1 – Reactor Room, occupational dose to operating staff

The licensee considers this area a radiological restricted area and assumes the occupants are radiation workers (occupational exposure limits apply) who will be exposed to the airborne gaseous fission products with no credit for radionuclide decay. Although, as stated above, the ventilation system would remain on following a significant

radiological accident at the facility, the calculation conservatively assumes that the ventilation system is off. The exhaust fan is shut down and the dampers isolate the reactor building.

For estimating occupational doses the licensee assumed that the facility would evacuate personnel from the reactor pool room within a 1-minute timeframe. In the response to RAI 96 (Ref. 16), the licensee estimated the maximum amount of time required for a staff member to evacuate the reactor room is less than 15 seconds. Based on firsthand observation of the reactor room, access to exit, and the licensee's emergency procedures, a conservative evacuation time of one minute is used. The licensee periodically conducts emergency exercises and past performance indicates that the evacuation times are below the time assumed in the scenarios. The licensee stated that there is an alarm in the reactor room and in the hallway of the nuclear engineering laboratories that can be triggered from inside the reactor room (Ref. 14). After a fuel cladding failure the increasing radiation dose would cause a visual and audible radiation alarm.

<u>Scenario 2</u> – Class Room, areas adjacent or above the reactor room

The licensee considers this a radiological unrestricted area and assumes that occupants may not be radiation workers (i.e., 10 CFR Part 20 public dose limits apply). Although, as stated above, the ventilation system would remain on following a significant radiological accident at the facility, the calculation conservatively assumes that the ventilation system is off. The exhaust system is shut down with dampers closed resulting in no fission product release. A member of the public is assumed to be in a class room above the reactor room subjected to gamma exposure from the fission product cloud inside the reactor room through the reactor room ceiling.

A member of the public located in the closest, unrestricted location (near a sealed doorway) is not subjected to inhalation and submersion radiation effects due to fission products in air leakage from the restricted area, since air preferentially leaks into the reactor room from the unrestricted area. (However the purpose of this scenario is to consider the impact of radiation fields and not inhalation and submersion).

The licensee provided an analysis of gamma ray exposure due to the postulated MHA in a response to RAI 12 (Ref. 20). The calculations assumed a semi-infinite cloud of radioactive isotopes evenly distributed in the reactor room emitting gamma rays at specific energies. The potential shielding effect of the concrete floors and walls are neglected due to the very low dose rate as shown below. Even though the dose rate is extremely low, the building would be evacuated under the University Police and Fire Department procedures, further limiting the potential exposure to members of the public.

<u>Scenario 3</u> – Outside the PUR-1 facility at the nearest residence or maximally exposed location:

The licensee calculated the dose outside the facility at the nearest residence (70 m (230 feet) from the facility), which is a radiological unrestricted area with occupants who are not radiation workers (i.e., 10 CFR Part 20 public dose limits apply). Dose calculations were performed with the ventilation system on, releasing the radioactive gases to the outside atmosphere at a rate of .2 m^3/s (423 ft^3/min). The exhaust system

keeps operating with the dampers remaining open and the fission products are released and dispersed into the atmosphere. Although the fission products would be released from the facility stack, the licensee calculated the dispersion by assuming the fission products are released at ground level. The licensee assumes the entire MHA radioactive source to be exhausted to the outside with one complete room air exchange, which would occur in less than 35 minutes. The occupant would be exposed (submersion and inhalation) to the fission product plume for the entire period of time it would take the plume to pass.

In responses to RAI 12 (Ref. 20) and RAI 73(Ref. 22), the licensee considered the effect of dilution of the radioactive isotope concentrations in the reactor room air as the ventilation system keeps operating. The exhaust fan expels the reactor room air, but also draws in clean air diluting the reactor room air concentration.

The NRC staff reviewed the analytical modeling, input data, scenarios, and the methodology described above for estimating the doses within and beyond the confines of the reactor facility in case of an MHA fission product release. The NRC staff concludes that, in general, the scenarios and methodology are conservative, consistent with the guidance provided in NUREG_1537, and adequate to calculate occupational and public radiation doses. Assumptions and boundary conditions were generally consistent with accepted nuclear industry practices. However, the NRC staff noted that the licensee's analysis of public doses outside of the facility (Scenario 3) calculated the dose at the nearest residence, which may not necessarily be the location of the maximally exposed member of the public. Although the nearest residence is the nearest continually-occupied location, members of the public are also present at other locations closer to the facility, and these members of the public could also potentially be exposed to airborne fission product releases from an accident. Therefore, the NRC staff's confirmatory calculation for Scenario 3, which is discussed below, included an analysis to determine the location of the maximally-exposed member of the public, and calculated the dose at that location.

<u>MHA Dose - Confirmatory Analysis</u>

The NRC staff performed confirmatory calculations of the MHA total effective dose equivalent (TEDE), in order to demonstrate the adequacy of the licensee's MHA results. These calculations were performed using assumptions, geometry, and radiological source terms that were verified to be consistent with those used by the licensee, except as discussed below. For the evaluations of the dose results, the NRC staff used the saturated isotope inventories. The occupational and public TEDE doses were calculated for Scenarios 1 and 3 above using the licensee's provided isotope inventory and the DCFs from FGR No. 11 and FGR No. 12 (Ref. 45 and Ref. 46).

For Scenario 3, dose calculations were performed for an elevated 15 m (50 feet) release (crediting the facility stack) analyzed by HotSpot (Ref. 47), an atmospheric dispersion code that provides a first-order approximation of the radiation effects associated with the atmospheric release of radioactive materials. The methodology utilized in the code conforms to NRC Regulatory Guide 1.145 (Ref. 44). The code uses the Gaussian-plume dispersion model and the total effective dose equivalents are calculated using FGR-11 and FGR-12 dose coefficients along the plume sector center lines. As discussed above, the licensee's calculation conservatively assumed a ground release and highly stable (Pasquill F) atmospheric conditions. The NRC staff noted that highly stable atmospheric conditions are most conservative for a

ground release; however, for an elevated release, the most conservative atmospheric condition varies. Therefore, since the NRC staff's confirmatory calculation assumed an elevated release, the NRC staff performed calculations to determine the maximum dose for varying atmospheric conditions. The NRC staff determined that the maximum dose, given the elevated release assumed in the NRC staff's calculation, is 3.6 mrem and occurs 34 m (112 ft) from the point of release during highly unstable (Pasquill A) conditions and 1 m/sec (2.2 mi/hr) wind speed.

The confirmatory analyses determined the adequacy of the results presented by the licensee. Based on the information provided above, the NRC staff concludes, that the licensee's estimated radionuclide inventories and MHA analysis are acceptable.

Table 13-1 summarizes the MHA dose results the licensee provided in a number of responses to RAIs (Refs. 20, 21, and 22) along with the results from the NRC staff's confirmatory calculations. The NRC staff noted that the Scenario 3 dose calculated by the licensee is significantly greater than the Scenario 3 dose calculated by the NRC staff, even though the NRC staff's calculation calculated the dose for the maximally-exposed location, and the licensee calculated it for the nearest residence. However, as noted above, the licensee's calculation did not take credit for the fact that the release would occur from the facility stack, and used the extremely conservative assumption that the release occurs at ground level. This explains the large discrepancy between the licensee- and NRC-calculated doses for Scenario 3. Nonetheless, both the licensee's and the NRC staff's results indicate that the occupational and public doses remain below the regulatory limits in 10 CFR 20.1201, "Occupational dose limits for adults," and 10 CFR 20.1301, "Dose limits for individual members of the public".

Table 13-1: MHA Dose Estimates

Scenario	Description	Total Effective Dose Equivalent (mrem)		
		PUR-1 Results TEDE	NRC Confirmatory Analysis, TEDE	Dose Limit
1	Maximum dose to operating personnel (1 min. exposure) **Occupational Dose**	317	294	5000
2	Maximum dose to personnel in class room (1 hr exposure) **Public Dose**	9.5×10^{-8}	N/A	100
3	Maximum dose to persons at nearest residence (licensee calculation) or maximally exposed location (NRC calculation) **Public Dose**	47.1	3.6	100

The NRC staff finds that the potential MHA dose to workers and the public was conservatively calculated and the result was within the regulatory limit for each scenario evaluated. The NRC staff concludes, based on the licensee's dose calculations, and the results of the NRC staff's confirmatory calculation, that the MHA dose results demonstrate that the maximum TEDEs are below the occupational limit in 10 CFR 20.1201 and the public dose limit in 10 CFR 20.1301.

MHA Dose Calculation Conclusions

The licensee states, in the SAR, that the calculated exposures are below the occupational and public exposure limits given in 10 CFR 20.1201, "Occupational dose limits for adults," and 10 CFR 20.1301, "Dose limits for individual members of the public."

The NRC staff reviewed the analysis performed by the licensee 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. The independent confirmatory dose calculations that the NRC staff performed demonstrate that the licensee conservatively evaluated the postulated doses from the MHA scenarios. The results of the NRC staff's confirmatory dose calculations are consistent with the dose results that the licensee provided, except as discussed above. In addition, the doses from the postulated scenarios provided above demonstrate that the maximum TEDEs are below the occupational limits in 10 CFR 20.1201, and the public exposure limits in 10 CFR 20.1301.

Based on the information provided above, the NRC staff concludes that the potential MHA calculated doses are within acceptable limits.

13.2.2 Insertion of Excess Reactivity

In Sections 13.1.2 and 13.2.2 of the SAR (Ref. 3) the licensee provides analyses of reactivity insertion accident scenarios. The initiating event is an insertion of excess reactivity in the reactor core that is analyzed with the assumptions that the control rods operate as required (scram) and that they fail to insert (no scram). These calculations were performed using the MCNP5, NATCON and PARET/ANL computer codes. These computer codes are accepted tools for analysis of research reactor physics, thermal-hydraulics and reactor transients and are acceptable for use in the licensee's analyses.

Reactivity Insertion with Scram

The maximum allowable excess reactivity, based upon TS 3.1, Specification d, was specified as the limiting case for the analysis. The event is initiated by the rapid insertion of excess reactivity of 0.6% $\Delta k/k$ into the reactor core operating at a steady state power level of 10 kWt. It was conservatively assumed that the period trip failed, and that reactor scram would be initiated at a nominal power trip of 12 kWt (120 percent of steady state power level). The licensee assumed a conservative uncertainty of 50 percent for bounding analysis, and, as a result, the actual scram signal is initiated when reactor core power reaches 18 kWt. Control rod motion is assumed to begin after a delay of 0.1 second following signal initiation. It is assumed that the least reactive control rod, SS-2 is dropped to scram the reactor.

The licensee considered two cases of sudden insertion of reactivity:

(1) Step insertion of 0.6% $\Delta k/k$ with scram
(2) Ramp insertion of 0.6% $\Delta k/k$ over 10 seconds with scram

The calculation results show that the peak clad temperature for the two cases is less than 58 °C (136 °F), less than the fuel safety limit of 530 °C (986 °F) and below the incipient boiling temperature:

Table 13-2: Reactivity Insertion, Peak Power

Case	Peak Power kWt	Peak Fuel Cladding Temperature
Step insertion of 0.6% Δk/k	46.4	57.40 °C
Ramp insertion of 0.6% Δk/k (10 s)	18.4	57.45 °C

The accidental removal of a secured experiment that would result in a ramp insertion of 0.6% Δk/k reactivity over 10 seconds (0.06% per 1 second) requires a failure to follow written procedures as the reactor operator unfastens the experiment and pulls it rapidly out of the core. The licensee states that this accident is less severe than the accidental step insertion accident primarily because the removal of the experiment occurs over a much longer period than the step insertion. The slower reactivity addition time will result in activating the 120 percent power scram at a lower portion of the reactivity insertion curve, resulting in lower peak power. However, the ramp reactivity insertion event is still more severe than withdrawing a control rod that is limited to a maximum withdrawal rate of 0.04% Δk/k per second. The results demonstrate that fuel integrity would be maintained in the event of the limiting reactivity insertion accident.

The licensee provided a sensitivity analysis on initial power level in a response to RAI 6 (Ref. 21). The excess reactivity analysis above was performed assuming that the initial power level corresponds to the steady state power level of 10 kWt. In the sensitivity analysis, the step insertion of excess reactivity is initiated at the maximum licensed power level of 12 kWt with the same assumptions on the control rod motion delay of 0.1 second. It is assumed that the reactor period is also the same, 1 second, which is conservative since the reactor period also increases with initial power. The licensee calculated that from the initial power level of 12 kWt the peak power would increase to 48.9 kWt as compared to 46.4 kWt from 10 kWt, which is well below the onset of nucleate boiling power level of 98.6 kWt.

The licensee states that both of these scenarios are extremely unlikely because they require either the failure of an operator to follow written procedures or deliberate violation of written procedures with rapid removal of an experiment. Even in these unlikely scenarios with conservative assumptions, the analytical results indicate that the TS 2.1 safety limit of 530 °C (986 °F) would not be exceeded.

<u>Reactivity insertion without scram</u>

The licensee calculated the reactor response to the reactivity insertion of 0.6% Δk/k for the cases of a step and a ramp insertion of 10 seconds assuming the unlikely failure of all reactor protective systems (without scram). The reactor temperature and moderator density feedback mechanisms provide the negative reactivity feedback that serve to control the reactor excursion and bring the reactor under control. The reactor is assumed to be operating at 10 kWt at the time of the reactivity insertion.

The calculation results for these cases demonstrate that while the reactor power rises to 2.38 MWt and that the reactor remains at elevated power levels for considerable time (more than 600-700 seconds), the cladding temperature never exceeds 133 °C (271 °F), well below the fuel temperature limit of 530 °C (986 °F). The results demonstrate that fuel integrity would be maintained even in the event of the limiting reactivity insertion accident with no reactor scram.

Conclusions

The NRC staff concludes that fuel damage events from accidental reactivity insertions are unlikely. The license renewal SAR, as amended, demonstrates that the combination of technical specifications and the physical characteristics of the fuel provide reasonable assurance that fuel integrity would be maintained and that no fission products would be released from the fuel as a result of accidental reactivity insertions.

The NRC staff reviewed the licensee's evaluation of a postulated reactivity insertion event, as described in the SAR, and finds that the licensee's analysis is comprehensive, and the results demonstrate that the maximum fuel temperature reached during the positive reactivity addition event would remain well below the safety limit specified in TS 2.1. The NRC staff finds that the licensee's postulated excess reactivity insertion scenarios that could result in a potential positive reactivity insertion event to be conservative examples for a MTR plate-type research reactor. The NRC staff also finds that the licensee's stated design features and administrative controls would render the postulated positive reactivity insertion event to be unlikely. They are consistent with the controls and design features found at other MTR type reactors; and, can be maintained, tested and/or inspected by the licensee to help ensure operability and TS compliance. Based on the information provided above, the NRC staff concludes that the licensee has sufficient design features and administrative restrictions in place that would make insertion of excess reactivity during reactor operation unlikely, and that the safety limit of TS 2.1 would not be exceeded.

13.2.3 Loss of Coolant

The licensee provides an analysis of a loss-of-coolant (LOCA) event in Section 13.1.3 of the SAR, as supplemented and additional LOCA analysis in a response to RAI14 (Ref. 20), RAI 15 (Ref. 21), and RAI 74 (Ref. 22). The outer wall of the reactor pool is made of stainless steel, individual rings, which are welded together for the full height. The outer pool wall is surrounded by another corrugated metal wall with sand filling in between. Because it is an in-ground pool, the reactor pool contains no drains or horizontal beam tubes and a sudden loss of the coolant inventory is not considered credible. Piping does not extend deeply into the pool making siphoning of significant pool water not credible. The licensee considered two possible scenarios for a significant LOCA event: 1) evaporation of the reactor tank coolant; and 2) a breach of the reactor tank resulting in a leakage to the surrounding soil. Evaporation of the reactor tank coolant would occur over a long period, allowing sufficient time for cooling replenishment by operations staff.

Tank failure due to corrosion would lead to a slow loss of coolant. A slow leak would be detected early as a result of water level checks that take place during daily routine operations and actions would be taken to mitigate the consequences by adding water to the reactor pool.

The instantaneous, total loss of coolant is not credible given the double-walled reactor tank design. Should such scenario be postulated, the reactor would shut down due to loss of moderation and the core would be cooled by natural convection of ambient air. In response to RAI 77 (Ref. 19) the licensee states that decay heating from an assumed power level of 18 kWt would produce approximately 5.2 watts of average fuel plate power and at such a heat rate the fuel plate would not be damaged.

Previous studies for a HEU MTR-type fuel plate with a 420 W steady-state operating power have shown that after a total loss-of-coolant with reactor scram the fuel plate cladding temperature would rise about 17 °C (30 °F) due to decay power and air cooling would be sufficient (Ref. 48). Reference 48 further states that the LEU fuel element has a larger coolant volume fraction and a lower power per fuel plate as compared with the analyzed HEU fuel element. Based on these two factors, it is expected that in a LOCA scenario the LEU fuel plate temperature increase should be approximately equal or less than in the HEU fuel plate. In the PUR-1 core the fuel plate with the peak power of 80 W and a cladding temperature of 43 °C (109 °F) would have at least a factor of five lower decay power than the analyzed HEU fuel plate (Ref. 48), and correspondingly an even lower cladding temperature rise. Even conservatively assuming the same temperature rise as for the HEU fuel plate in Reference 48, the increase would result in a cladding temperature of 60 °C (140 °F), which is far below the safety limit of 530 °C (986 °F). The NRC staff concludes that the hypothetical total loss-of-coolant scenario would not result in damage to the reactor fuel plates.

In the unlikely event of a leak from the reactor pool, the impact on public safety and the environment would not be significant. If the tank failure occurs above ground, the top 3 feet (0.9 m) portion of the pool water would collect on the reactor room floor resulting in 3 inches (7.6 cm) of standing water, without any damage to reactor components and the core would remain covered. Any water would be collected and analyzed before disposal. If the failure occurs below ground, the pool water would be released to the sand filling surrounding the reactor tank.

The impact on public safety and the environment would not be significant, since the radioactivity level would be low. The periodic monitoring of the pool water ensures that the radioactivity level of the water is low and within the limits found in Appendix B, “Annual Limits on Intake (ALIs) and Derived Air Concentrations (DAC) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage,” of 10 CFR Part 20 for release to the sewer. Based on the information provided above, the NRC staff concludes that the potential total loss of primary coolant is an extremely unlikely event and will not have significant impact on the health and safety of the public or environment.

The licensee provided an analysis of a slow leak event in a response to RAI 14 (Ref. 20). In the analysis the licensee stated that the most probable event initiating a LOCA would be a crack developing between one of the welds on the pool wall. The potential crack in the tank wall welding would lead to water seeping out and filling the small voids between the sand particles. The potential consequence of a slow leak LOCA is the reduction of the 13-foot high water column above the reactor core and an increase in gamma ray dose to the surroundings from the core.

The licensee postulates that a weld crack develops at the lowest level of the tank as the most conservative assumption resulting in the fastest drainage lasting the longest time. Since the pool would drain into the space between the stainless steel pool wall and the outer corrugated metal wall, the drainage is restricted to the space available in the sand filling.

The licensee states, that even though the water level would decrease, it would reach an equilibrium level and would still be 2.7 meter (5.1 feet) above the reactor core. The licensee further estimates that the time to reach this level would be 210 seconds. This is a conservative estimate because it is based on the assumption that the flow rate out of the crack is from a free standing tank with unrestricted flow into air. However, the flow rate would be greatly reduced,

since the tank is not draining into empty space, but rather to and around the sand in-fill. The licensee's analysis also indicates that following the initiation of a LOCA event, the reactor would scram in 80 seconds when the radiation level would reach 50 mrem/hr (0.5 mSv/hr), which is the setpoint for the pool top radiation monitor. The operator would then follow up with a manual scram and evacuate the reactor room in 30 seconds. The licensee states that this time frame is sufficient for an operator to become aware of the situation, initiate a scram and begin an evacuation.

However, even if the scram is not initiated, the loss of water results in a loss of moderation and reactor subcriticality with only decay power. Therefore, the licensee's dose analysis considers only the reactor decay power, which is acceptable to the NRC staff.

In the first 80 seconds the gamma ray intensity corresponds to the full operating level that is reduced to decay power level after the scram is initiated. The licensee's shielding analysis considered the time dependent change in the decreasing water level and conservatively assumed that the operator is at the edge of the pool surface, while a member of the public is located at 5 meters distance from the pool shielding wall. These are very conservative assumptions, since the operator is not directly exposed to the gamma rays, but only through scattered dose from the building walls. Any member of the public is either located in a class room above the reactor room shielded by a concrete floor or outside the reactor room door that provides additional shielding

The licensee calculates that the total dose to the operating staff is 0.49 mrem (0.0049 mSv) and to a member of the public is 0.15 mrem (0.0015 mSv) with the following assumptions:

- Reactor pool water level at 2.7 meter (5.1 feet) above the core
- Reactor scram 80 seconds after LOCA initiated
- Reactor area evacuated in 30 seconds
- Operator located at the edge of the reactor pool, while a member of the public is 5 meter from the pool wall or in the classroom above the reactor room.

These dose levels are bounded by those analyzed for the MHA.

The NRC staff reviewed the information provided in the SAR, as supplemented and additional LOCA analysis provided in responses to RAIs (Refs. 20, 21, and 22) with respect to a potential leak from the reactor pool and finds that the licensee has the ability to monitor the pool water level, and respond to low-level alarms. The temperature rise to be expected upon loss of coolant from an operating power of 12 kWt is small and the maximum fuel temperature would be far below the fuel temperature limit. The LOCA does not result in damage to the reactor fuel. In addition, doses within the reactor facility are within the limits that would allow timely recovery operations to proceed.

The licensee's analysis of a LOCA event indicates that radiation doses to the workers, 0.49 mrem (0.0049 mSv) and members of the public, 0.15 mrem (0.0015 mSv) would remain within the limits of 10 CFR Part 20. Even in a non-credible, instantaneous, total loss of coolant, the fuel element integrity would remain intact. Based on the information provided above, the NRC staff concludes that a potential loss of primary coolant is extremely unlikely and, if it occurs, the impact on the health and safety of the public or the PUR-1 staff will be minimal.

13.2.4 Loss of Coolant Flow

Section 13.1.4 of the licensee's SAR discusses loss of coolant flow. The PUR-1 reactor core is cooled by natural convection of pool water up between the fuel plates and the pool water is cooled in a cooling system by forced circulation to a heat exchanger. Heat is removed from the shell side of a heat exchanger by a Freon chiller compressor-condenser unit. Loss of forced circulation of pool water through the heat exchanger or the failure of the heat exchanger-chiller system would be observed by operators using various indicators at the console and the reactor could be shut down if needed. If shutdown, the reactor would continue to be safely cooled by natural circulation while the reactor pool would slowly heat at decay heat level at a maximum rate of 0.465 °C/hr (0.84 °F/hr) dropping to undetectable after operation at 12 kWt. 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. If the reactor remained in operation, operations would be terminated before exceeding the pool temperature limit of 30 °C (86 °F) in TS 3.3, Specification c.

Adequate time would be available for corrective action, including providing a supply of makeup water from alternate sources. The NRC staff concludes that even if the pool water should evaporate leaving the core uncovered the fuel temperature rise would be far below the safety limit, as described in Section 13.2.3 of this SER.

The licensee stated that loss of coolant flow from blocked fuel cooling channels is not a credible scenario for the PUR-1 reactor. The reactor is cooled by natural circulation and, as such, no pumps are employed in forcing water through the core. As a result, heavier objects below the core cannot be drawn up into the core region.

In the event of a possible blockage of a coolant channel created by a foreign object lodged in the grid plate, the licensee states that at 18 kWt the individual plate power density is very low, and conduction of heat to adjacent channels would retain temperatures significantly below the ONB temperature of 112 °C (234 °F) and well below the fuel safety limit of 530 °C (986 °F) (Refs. 19, 20).

The NRC staff reviewed the licensee's description associated with a potential reactor core loss of coolant flow event, as described in the SAR, and finds that the loss of cooling would result in a slow temperature increase, which would be corrected by the PUR-1 reactor operators. The NRC staff also finds that a failure of the secondary coolant chiller systems has no impact on the ability to cool the reactor core because of the large inventory of primary coolant in the pool and the ability to add coolant to the pool, if needed. The NRC staff finds that in the event of a local blockage of a coolant channel by a foreign object, the fuel plate would still be maintained below the ONB temperature ensuring sufficient, continuing cooling. Based on the information provided above, the NRC staff concludes that should a loss-of-coolant flow condition occur, the reactor fuel plates would retain adequate cooling, and a loss-of-flow condition poses no adverse risk to the health and safety of the public or to the PUR-1 staff.

13.2.5 Mishandling or Malfunction of Fuel

Section 13.1.5 of the licensee's SAR discusses mishandling or malfunction of fuel. The licensee identifies two fuel handling scenarios which could lead to fuel damage: (1) fuel elements are sometimes moved, one at a time, from the core into the fuel storage space, and (2) a fuel element is removed from the pool annually for inspection. In the first instance a fuel element

may fall from the handling device and land on the pool floor with an overlying pool of water. In the second case, the fuel could land on the floor in the pool room.

The licensee has an established procedure for handling of fuel, has no recorded incidents of mishandling its fuel elements, and no incidents of cladding failure. TS 4.6 requires surveillance of representative fuel plates by visual inspection and annual measurement, in order to verify the continuing integrity of the fuel plate cladding. The removal of a fuel assembly from the reactor pool is controlled by a facility procedure.

In the event that fuel would be dropped while submerged, or if the fuel would suffer damage due to manufacturing defect or corrosion, the consequences of the incident would be bounded by the MHA since any damage would occur while the fuel is in the pool. If the fuel is submerged, then any fission product release would be significantly mitigated beyond the assumptions of the MHA analysis as a result of scrubbing by the pool water overlying the fuel element. Damage caused in air would also be bounded by the MHA.

The NRC staff reviewed the information provided in the SAR, as supplemented, and finds that the licensee has established procedures for handling fuel and has no recorded incidents of mishandling its fuel elements and no incidents of cladding failure. TS 4.6 requires surveillance of fuel plates by visual inspection to verify the continuing integrity of the fuel plate cladding. Based on the information provided above, the NRC staff concludes that any potential fission product release resulting from mishandling or malfunction of the fuel would be less than computed for the MHA analysis as discussed in Section 13.1.1 of this SER.

13.2.6 Experiment Malfunction

The licensee discusses experiment malfunction in Section 13.1.6 of the SAR. The primary use of the PUR-1 is training and performing associated experiments such as out of core dose measurements, and fueled and non-fueled irradiations. A malfunction of an experiment can result in over pressurizing an experiment or generating more radioisotopes than expected. The over pressurizing of an experiment can result in a sudden release of radioisotopes or damage to the core. The licensee's response to RAI 8 (Ref. 20) indicates that the experiment malfunction accident scenario can occur from two principal causes: 1) an unexpected reactivity insertion, or 2) a failure of an experimental capsule containing fissile material releasing fission products into the reactor room air.

The licensee controls and limits experiment activities through procedures and limitations set in TS 3.1, TS 3.5, TS 4.1, TS 4.5, TS 6.2 and TS 6.5 regarding the control and review of reactor experiments. Limitations on experiments are discussed in detail in Chapter 10 and 12 of this SER. The review process of a proposed experiment includes a safety analysis that assesses the complete range of safety issues such as the generation of radionuclides, the reactivity worth of the experiment, material properties such as chemical, physical, explosive, and corrosive characteristics of each experiment, and potential failures and malfunctions.

TS 3.1, Specifications e through g, limit the absolute value of the reactivity worth of any single experiment and the sum of all experiments to less than the maximum excess reactivity limit of TS 3.1, Specification d. Potential reactivity malfunctions are bounded by the insertion of reactivity accident discussed in Section 13.2.2 above.

Limits are placed on experiments installed in the reactor and associated experimental facilities through TS 3.1 and TS 3.5. TS 3.5, Specification a, requires that experimental capsules are made of corrosion resistant material. The NRC staff finds that the TSs help ensure that an anticipated failure of a capsule is highly unlikely and, therefore, are acceptable. TS 3.5, Specification c, prevents the introduction of explosive material in PUR-1.

TS 3.5, Specifications e and f, address experiment failures and malfunctions and limits the radioactive products produced in experimental materials that may release airborne radioactive material and provides conditions to be used in the safety analysis of the experiment. The purpose of TS 3.5, Specifications e and f, is to help ensure that potential releases of radioactive material from experiments are bounded by the exposure limits in 10 CFR Part 20 for PUR-1 staff and members of the public. This includes experiment failures under normal reactor operations, credible reactor accident conditions, and accident conditions in the experiment.

TS 3.5, Specification f, imposes limitations on the allowed inventories of iodine isotopes in a fueled experiment to ensure the health and safety of the facility workers in the event of an accident that would result in the release of these isotopes due to an accident in an experiment.

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 worth and other experiment materials to avoid accidental reactivity insertions, damage to reactor components and release of radioactivity. Explosive materials are excluded from the reactor pool. TS 6.2 and TS 6.5 require review and approval of all new experiments, during which the limits on experiments are analyzed and approved.

The licensee identified two experiment malfunctions: (1) failure of an irradiation capsule containing fissile material that has been irradiated in the reactor, and (2) flooding of the drop tube which results in a reactivity insertion event. The flooding event was calculated to lead to a reactivity insertion of 0.246% $\Delta k/k$, and therefore, the consequences are bounded by the maximum reactivity insertion of 0.6% $\Delta k/k$ analyzed in Section 13.2.2 of this SER.

The failure of an irradiation capsule containing fissile material is analyzed by the licensee in a response to RAI-8 in Ref. 20 and the consequences are shown to be bounded by the MHA. TS 3.1, Specification d, limits the maximum positive reactivity of the core and experiment to 0.6% $\Delta k/k$, and TS 3.5, Specification f, limits total iodine content produced in an experiment to 0.5 Ci, which are used to determine the maximum amount of fissile content. The licensee conservatively assumed that the capsule would fail in the air releasing all gaseous fission products. The dispersal of the fission products was analyzed using the same methodology as for the MHA (see Section 13.2.1 of the SER).

In analyzing the failure of an experimental capsule, the licensee made the following assumptions regarding the fission product releases that resulted in conservative consequence estimates:

- Experiment contains 0.5 Ci of radio-iodine as limited by TS 3.5, Specification f

- All iodine products escape the fuel and capsule into the reactor room air

This assumption does not account for any physical size or form of the material in the fueled experiment. Iodine can be trapped in the material matrices and the assumption does not provide any credit for trapping.

- In calculating the external dose rate outside the reactor building, it is assumed that the ventilation system keeps running exhausting all iodine radioisotopes out of the building in less than one hour with no radioactive decay and plating of the iodine isotopes.

  This assumption does not account for the removal of any iodine radioisotope due to radioactive decay and/or from plating out of the air. It also does not allow for manually shutting the ventilation fan.

The reactor room is assumed to be evacuated in about one minute, which is based on the licensee's experience. Based on firsthand observation of the reactor room, access to exit, and the licensee's emergency procedures, the NRC staff finds that an evacuation time of one minute is realistic and corresponds to the same evacuation time period used in the MHA analysis. Using the above assumptions and methods used to analyze the MHA (see Section 13.2.1), the licensee calculated occupational and public dose estimates presented in Table 13-6.

Table 13-3: Experiment Failure, Dose Estimates

Location	TEDE (mrem)	Dose Limit (mrem)
Reactor Room - Occupational	28.6	5000
Public -Unrestricted	1.6	100

The occupational TEDE of 28.6 mrem (0.286 mSv) is well below the limits in 10 CFR 20.1201. The dose to the member of the public in the unrestricted areas is also evaluated using the atmospheric dispersal methodology for the MHA. The calculated dose after one-hour exposure, assuming that the ventilation fan keeps running and the member of the public is continuously submerged in the radioactive plume is 1.6 mrem (0.016 mSv); below the limit set forth in 10 CFR 20.1301.

The NRC staff reviewed the information provided in the SAR and TSs, and finds that the licensee's limitations, controls, and procedures for experiments are in place, and are adequate to help to minimize the potential occurrence of an accidental experiment malfunction. The design of experimental irradiation facilities has also been reviewed in Chapter 10 of this SER and found acceptable. If an experiment were to malfunction, the TS controls, which limit the reactivity worth and experiment materials, would limit the accidental reactivity insertions, damage to reactor components, and the release of radioactivity.

Based on the PUR-1 TS limits for quantity and type of materials allowed in an experiment, the NRC staff concludes that the licensee's evaluation of the consequences of experiment malfunction leading to a radiological release are consistent with the guidance provided in NUREG-1537, and bounded by the MHA analysis for fission product releases evaluated in Section 13.2.1 of this SER.

Based on the information provided above, the NRC staff concludes that the licensee has proper controls established to minimize the potential occurrence of an accidental experiment malfunction, and to help ensure that the radiation dose consequences would not be more severe than the limits provided in 10 CFR Part 20.

13.2.7 Loss of Normal Electrical Power

The licensee discusses loss of normal electrical power in Section 13.1.7 of the SAR. The loss of normal electrical power would cause loss of power to the electromagnets that lift the shim safety rods. This would lead to insertion of the two shim safety rods and would result in reactor scram within one second of scram signal generation. The reactor is cooled by natural convection and no pumping power is required. Reactor decay heat would be dissipated through natural circulation in the primary coolant. The loss of forced circulation of the pool water through the heat exchanger would lead to a very slow heat-up of the reactor pool until power would be restored.

The electrical power system is not necessary to safely shut down the reactor and is not required to ensure public health and safety. 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 reviewed the information provided in the SAR, as supplemented, and finds that, upon a loss of electrical power, the reactor will shut down and 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 operation during the reactor shutdown. The NRC staff has reviewed the results of the licensee’s postulated loss of electrical power and finds the analysis to be acceptable. Based on the information provided above, the NRC staff concludes that loss of normal electrical power poses little risk to the health and safety of the public or to the PUR-1 staff.

13.2.8 External Events

The licensee discusses external events in Section 13.1.8 of the SAR. Sections 2 and 3 of the licensee SAR discusses meteorological events including tornado, hurricane, and flooding, and also discusses seismic events. The licensee states that the likelihood of external events such as hurricanes, floods and tornadoes is considered insignificant. It is judged that the structural integrity of the reactor building and the reactor pool are unlikely to be affected by hurricanes or tornadoes. The most severe result of a hurricane or tornado would be the loss of electrical power that would result in a controlled reactor shutdown as discussed in Section 13.2.7 of this SER.

The licensee stated that the probability of large earthquakes is extremely unlikely and not considered a credible hazard. The reactor building was constructed to seismic codes applicable at the time of construction. Reactor core damage is considered negligible due to the low probability of seismic activity and the expected limited damage to the building. An overhead crane may potentially fall onto the pool, but is generally parked outside of the pool and rarely used. In addition, the reactor core is located underground at the bottom of the pool and no damage would be expected even if the crane would fall onto the reactor room floor (Ref. 17). Calculations show that the fuel can be cooled in air. The potential consequences of external events would be bounded by the MHA.

Although the facility is close to an airport, and thus the possibility of an aircraft impact is not negligibly small, the probability of damaging the pool and thus the reactor is small. This is due to the design of the pool and reactor location, and the size of the possible aircraft involved, which are small aircraft due to the airport restrictions. The reactor is located below the ground level and the top portion of the reactor tank is protected by thick pool walls. The probability of an aircraft crashing into the reactor building is extremely low due to the runway trajectory. 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 finds that severe storms, floods, and extreme winds do not pose a threat to the PUR-1 building or structure. The seismic activity in the area is low, and the building was designed and built for the expected seismicity in the region. If an earthquake with significant severity could cause loss of shielding, cooling, or fuel element cladding failure, the consequences to the PUR-1 facility are not expected to result in events more severe than the events analyzed in the MHA. The consequence of a LOCA with the reactor core intact is not expected to result in clad failure, and should one occur, its consequences for the release of fission products would be bounded by the analysis for the MHA.

The NRC staff reviewed the licensees' analysis of an accident initiated by an external event, and finds that the probability of this type of potential accident to be extremely low. The NRC staff also finds that the fuel is protected by the facility design, structural barriers and its location below grade, from any accident initiated by an external event. The NRC staff also finds that the potential for an accident resulting from an aircraft collision to be unlikely to affect the fuel due to the thick, reinforced steel and concrete confinement building and pool walls, and location of the reactor below ground. Based on the information provided above, the NRC staff concludes that members of the public are not subject to undue radiological risk as a result of an external event, and that external events do not pose a risk to the health and safety of the public and to the PUR-1 staff that would exceed the fission product release results of the MHA.

13.2.9 Mishandling or Malfunction of Equipment

The licensee states that there are processes, procedures, or designs in place to help to ensure that any potential consequences of mishandling or malfunction of equipment is minimal and in no case exceeds the consequences of the MHA. Some of these potential accidents may occur as a result of human errors which could constitute a violation of the TSs, and which would be reported as required. The NRC staff reviewed the information provided and finds that the physical limitations of the PUR-1 design are such that mishandling or malfunction of equipment would lead to fission product release 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 PUR-1 staff.

13.3 Conclusions

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

- The licensee considered the expected consequences of a sufficiently broad spectrum of postulated credible accidents and an MHA, emphasizing those that could lead to a loss of integrity of fuel-element clad and a release of fission products.

- The licensee analyzed the most significant credible accidents and the MHA and determined that under conservative assumptions, the most significant credible accidents and the MHA will not result in occupational radiation exposure of the PUR-1 staff or radiation exposure to a member of the public in excess of the applicable 10 CFR Part 20 limits.

- The licensee has employed appropriate methods in performing the accident and consequence analysis.

- For accidents involving insertions of excess reactivity, the licensee has demonstrated that a reactivity insertion of 0.6% Δk/k will result in a peak fuel temperature well below the TS safety limit of 530 °C (986 °F). In the TSs the licensee has established an administrative limit of 0.6% Δk/k for the reactor excess reactivity.

- The review of the calculations, including assumptions, demonstrated that a LOCA would not result in unacceptable fuel element temperatures. The reactor can be safely cooled with all fuel elements in an air environment. Doses to individuals evacuating the reactor room and to the maximally exposed member of the public are calculated to be below the 10 CFR Part 20 limits.

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

- The accident analysis confirms the acceptability of the proposed licensed power of 12 kWt, including the response to anticipated transients and accidents.

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

The NRC staff reviewed the radiation source term and MHA calculations for the PUR-1. The NRC staff finds the calculations, including the assumptions, demonstrated that the source term assumed and other boundary conditions used in the analysis are acceptable. The radiological consequences to the public and occupational workers at the PUR-1 are in conformance with the requirements in 10 CFR Part 20. The licensee reviewed the postulated accident scenarios provided in NUREG-1537 and did not identify any other accidents with fission product consequences not bounded by the MHA. The PUR-1 design features and administrative restrictions found in the TSs help to prevent the initiation of accidents and mitigate associated consequences. Therefore, on the basis of its review, the NRC staff concludes that there is reasonable assurance that no credible accident would cause significant radiological risk and the continued operation of the PUR-1 poses no undue risk to the facility staff, the public or the environment.

14 TECHNICAL SPECIFICATIONS

The PUR-1 TS define specific features, characteristics, and conditions governing the operation of the facility. The NRC staff reviewed the format and content of the TSs for consistency with the guidance in NUREG-1537, Part 1, Chapter 14, and Appendix 14.1, and ANSI/ANS-15.1-2007 (Refs. 29, 34). Throughout this SER, the NRC staff specifically evaluated the content of the TSs to determine if the TSs meet the requirements in 10 CFR 50.36. The NRC staff also relied on the references provided in NUREG-1537 and the ISG (Ref. 32) to perform this review.

Based on information provided above, the NRC staff concludes that the PUR-1 TSs are acceptable for the following reasons:

- 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 include appropriate summary bases. Those summary bases are included in the TSs, but are not specifications required by the regulations.

- The PUR-1 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 TSs derived from analyses in the SAR, as supplemented by responses to RAIs.

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

- To satisfy the requirements of 10 CFR 50.36(c)(1), the licensee provided proposed TSs specifying a SL on the fuel temperature and an LSSS for the reactor protection system to preclude reaching the SL.

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

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

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

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

The NRC staff reviewed the proposed TSs and finds the proposed TSs acceptable and concludes that normal operation of the PUR-1 within the limits of the proposed TSs will not result in radiation exposures in excess of the limits specified in 10 CFR Part 20 for members of the general public or for the worker's occupational exposures. The NRC staff concludes that the proposed TSs provide reasonable assurance that the PUR-1 will be operated as analyzed in the

SAR as supplemented, and that adherence to the proposed TSs will limit the likelihood of malfunctions and the potential accident scenarios discussed in Chapter 13, “Accident Analysis,” of this SER.

15 **FINANCIAL QUALIFICATIONS**

15.1 Financial Ability to Operate the Facility

As stated in 10 CFR 50.33(f):

> 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.

Purdue University (Purdue) does not qualify as an "electric utility," as defined in 10 CFR 50.2, "Definitions." Furthermore, 10 CFR 50.33(f)(2) states the following:

> The application to renew or extend the term of any operating license for a nonpower reactor shall include the financial information that is required in an application for an initial license.

The U.S. Nuclear Regulatory Commission (NRC) staff has determined that Purdue must meet the financial qualifications requirements pursuant to 10 CFR 50.33(f), and is subject to a full financial qualifications review by the NRC. Specifically, Purdue must provide information to demonstrate that it possesses or has reasonable assurance of obtaining the funds necessary to cover estimated operating costs for the period of the license. Therefore, Purdue must submit estimates of the total annual operating costs for each of the first five years of facility operations from the expected license renewal date and indicate the source(s) of funds to cover those costs. This is consistent with the guidance described in NUREG-1537 "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," as it pertains to staff's review of the applicant's financial qualifications.

In the SAR, as supplemented by an RAI response (Ref. 59), Purdue provided information related to financial qualifications. Purdue submitted its projected operating costs for the Purdue University Research Reactor (PUR-1) for each of the fiscal years (FY) 2012 through FY 2016. The operating costs for PUR-1 were projected to be $180,189 in FY2012, $185,145 in FY2013, $190,249 in FY 2014, $195,507 in FY 2015 and $200,922 in FY2016.

According to Purdue, PUR-1 has limited operating expenditures, which includes salaries for three staff members and a modest Operations and Maintenance (O&M) budget. Based on the applicant's submittal, total salary costs for FY 2008 and FY 2009 for three staff members were $155,839, with an additional $75,363 in overhead, including benefits. The average O&M expenditures for FY 2006 through FY 2009 were $10,959. In an RAI response (Ref. 59), Purdue stated, in part, that the projected operating costs for PUR-1 are anticipated to remain consistent for FY 2016 to FY 2021. According to Purdue, its primary source of funding to cover its operating costs will be the funds associated with the school of Nuclear Engineering's recurring budget provided by the university, annually, or from grants. Purdue expects that these funding sources will continue for the aforementioned FYs. As part of its review, the NRC staff considered guidance in NUREG-1537, as well as the projected operating costs and associated funding for similar reactors to include the University of Florida and Ohio State University, and found these estimates and sources of funds to be reasonable.

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

Purdue is currently licensed as a facility that is useful in research and development under Section 104.c of the AEA, 42 U.S.C. §2234(c). The regulation in 10 CFR 50.21(c) provides for issuance of a license to a facility which is useful in the conduct of research development activities if no more than 50 percent of the annual cost of owning and operating the facility is devoted to production of materials, products, or the sale of services, other than research and development or education or training. The Purdue reactor was originally licensed by NRC as a non-commercial facility in 1962 and continues as an academic, non-commercial facility. In its response to RAI (Ref. 59), Purdue provided financial information that indicated no operating costs are devoted to commercial activities. Because 10 CFR 50.21(c) requires that the majority of reactor operating costs be funded by non-commercial uses and Purdue has no operating costs devoted to commercial activities, the NRC staff concludes that the licensee can be renewed as a Section 104.c license.

15.2 Financial Ability to Decommission the Facility

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

Under 10 CFR 50.75(d)(1), each non-power reactor applicant for or holder of an operating license for a production or utilization facility shall submit a decommissioning report as required by 10 CFR 50.33(k). Pursuant to 10 CFR 50.75(d)(2), the report must contain a cost estimate for decommissioning the facility, an indication of the funding method(s) to be used to provide funding assurance for decommissioning, and a description of the means of adjusting the cost estimate and associated funding level periodically over the life of the facility. The acceptable methods for providing financial assurance for decommissioning are described in 10 CFR 50.75(e)(1). The NRC staff applied guidance in NUREG-1537 to complete its review of the PUR-1 license renewal application as it pertains to financial assurance for decommissioning.

In the SAR, as supplemented by an RAI response (Ref. 59), Purdue provided a cost estimate for PUR-1 in the amount of $623,000 (2010 dollars), which considered the use of the DECON method for decommissioning of the facility. In calculating its estimate, Purdue applied a 25 percent contingency factor, then adjusted its costs considering a 5 percent rate of inflation; this resulted in a decommissioning cost estimate of $834,880 (2016 dollars). The estimate itemized costs by labor, radioactive waste disposal, and energy factors. Further, Purdue stated, in part, that the cost estimate considered a previously submitted cost estimate for the facility submitted to NRC. Purdue adjusted for changes in the scope of work, time, and resource requirements of the project, as its basis. The cost estimate adjustment also included updates to current values where available (e.g., transportation and disposal of HEU fuel from a prior conversion to LEU). As part of its review, staff independently verified the cost adjustments provided by Purdue.

Purdue 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. Therefore, the SOI must contain or reference a cost estimate for decommissioning and indicate that funds for

decommissioning will be obtained when necessary. Purdue provided an updated SOI (Ref. 60) stating that the signatory will "[r]equest that funds will be made available when necessary to decommission the PUR-1 reactor facility." Further, the signatory states that they will "request and obtain these funds over this period sufficiently in advance of required activities to assure timely funding of required activities." The updated SOI is signed by the Executive Vice President for Business and Finance, Treasurer of Purdue University.

As included in the updated SOI, Purdue stated that funding resources will be provided when needed by Purdue University or the State of Indiana, and the projected costs calculated using a conservative inflation rate of 5 percent, is still very small compared to the financial resources of the university and the State. Additionally, Purdue stated that a detailed adjustment of this funding level would not be required other than reasonable assurance that the inflation rate is staying near or below the 5 percent as discussed above.

To support the SOI and Purdue's qualifications to use an SOI, the application stated that Purdue is a non-profit educational institution and part of the state government of the State of Indiana, and included documentation which corroborates this statement. The application also provided information supporting Purdue's representation that the decommissioning funding obligations of the PUR-1 are backed by the full faith and credit of the State of Indiana. In an RAI response (Ref. 59), Purdue provided documentation signed by the Director of Radiation Laboratories, which stated that Purdue University is an Agency of the State of Indiana, established by the Indiana Code Title 21, IC 21-23-2. Purdue also provided documentation verifying that the Executive Vice President for Business and Finance, Treasurer of Purdue, the signatory of the SOI, is authorized to execute contracts on behalf of Purdue.

The NRC staff reviewed Purdue's information on decommissioning funding assurance as described above and finds that Purdue is a State government (non-power reactor) licensee under 10 CFR 50.75(e)(1)(iv), because the SOI is acceptable. The NRC staff also finds that the decommissioning cost estimate and the annual costs for decommissioning using the DECON method are reasonable; and Purdue's means of adjusting the cost estimate and associated funding level periodically over the life of the facility are also reasonable.

Therefore, the staff concludes that funds will be made available to decommission the facility and that the financial status of the applicant regarding decommissioning costs meets the requirements of 10 CFR 50.33(k) and 10 CFR 50.75.

15.3 Foreign Ownership, Control, or Domination

Section 104.d of the AEA of 1954, as amended, prohibits the NRC from issuing a license to "any corporation or other entity if the Commission knows or has reason to believe it is owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government." The regulations in 10 CFR 50.38, "Ineligibility of certain applicants," contain language to implement this prohibition.

In addressing foreign ownership, control and domination (FOCD), the NRC staff considered guidance in "Standard Review Plan on Foreign Ownership, Control and Domination" (Ref. 61), to determine whether Purdue is owned, controlled or dominated by an alien, a foreign corporation, or a foreign government.

According to the application, PUR-1 is owned and operated by Purdue, an entity (component unit) of the State of Indiana and is not owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government. Because the NRC staff does not know, or have reason to believe, otherwise, Purdue is eligible to receive a renewed license.

15.4 Nuclear Indemnity

The NRC staff notes that Purdue currently has an indemnity agreement with the Commission. It expires only when the Facility Operating License No. R-87 expires, provided all radioactive material has been removed from the location and transportation of radioactive material from the location has ended. Therefore, Purdue 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,” PUR-1, as a nonprofit educational institution, is not required to provide nuclear liability insurance. The Commission will indemnify PUR-1 for any claims that arise from a nuclear incident under the Price-Anderson Act and Section 170 of the AEA and in accordance with the provisions under its indemnity agreement in 10 CFR 140.95, “Appendix E—Form of indemnity agreement with nonprofit educational institutions,” up to $500 million. Therefore, applicable provisions of Part 140 have been satisfied. In addition, PUR-1 is not required to purchase property insurance under 10 CFR 50.54(w).

15.5 Conclusions

As described above, 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 PUR-1 and, when necessary, to shut down the facility and carry out the decommissioning activities. In addition, the NRC staff concludes that there are no foreign ownership, control or domination issues or indemnity issues that would preclude the issuance of a renewed license.

16 **PRIOR REACTOR UTILIZATION**

16.1 Prior Use of Components

As detailed in previous sections of this SER, the NRC staff concludes that continued operation of the PUR-1 will not pose a significant radiological risk. The bases for these conclusions include the assumption that the facility systems and components are in good working condition. However, reactor systems and components may experience chemical, mechanical, and radiation-induced degradation, especially over years of reactor operation. Systems and components that perform safety-related functions must be maintained or replaced to ensure that they continue to protect adequately against accidents. Such systems and components found at the PUR-1 include the fuel cladding and the reactor safety system. Section 4.2.1 of this document describes the reactor fuel. The original fuel was replaced during the HEU to LEU fuel conversion in 2007. Possible mechanisms of degradation of the fuel cladding over time include (1) thermal cycling and high fuel temperature, (2) radiation damage, (3) erosion, (4) mechanical impact or fuel handling, and (5) corrosion.

(1) Because the PUR-1 does not have any pulse capability, thermal cycling occurs only because of startup and shutdown of the reactor. During a cycle, the maximum cladding surface temperature change is approximately 23 °C (73 °F). The temperature change at the cladding-fuel interface is essentially the same. This temperature change does not have the potential to cause degradation of the fuel cladding. The licensee calculated a maximum cladding temperature of 43 °C (109 °F). This temperature is too low to cause degradation of the cladding.

(2) Aluminum-clad MTR-type fuel does not have a history of failure resulting from radiation damage. The Reduced Enrichment for Research and Test Reactors (RERTR) fuel development program tested the fuel type used at the PUR-1 at high burnup and observed no fuel failures. The staff evaluated those results in NUREG-1313. Exposure to radiation doses greater than those expected at the PUR-1 caused no significant degradation in similar fuel plates.

(3) Natural convection cooling does not generate the coolant velocities or pressures necessary to erode the cladding.

(4) The design of in-pool structures and components minimizes the chance for mechanical impact. The standard fuel assembly is constructed with an outside aluminum can, effectively shielding the cladding of the fueled plates. The design of the control rod fuel assembly places aluminum plates on either side of the center gap, thus effectively shielding the cladding of the outer fueled plates from impact with the control rod. The control rod assemblies are located in the center of the core and thus protected from external impacts. Fuel handling requires specially designed tools that do not come in contact with the fuel cladding. The core plenum shields the fuel plates from tools and small objects, should they fall into the reactor pool.

(5) TS 3.3 places requirements on the conductivity of the primary coolant. TS 4.3 specifies surveillance intervals for the chemical properties of the coolant. These TSs adequately ensure that no significant corrosion of the cladding has occurred or will occur.

The electrical design of the reactor safety system (e.g., safety channel circuitry, control rod magnets) precludes accidents as a result of failure of system components. As discussed in Chapter 7 of this SER, failure or removal for maintenance of required safety-related instrumentation and control (I&C) components causes a safe reactor shutdown. TS 4.2 specifies surveillance requirements for the reactor safety system. These requirements ensure that gradual degradation of system components will be detected and addressed. Additionally, the PUR-1 staff performs regular preventive and corrective maintenance and replaces system components as necessary. Nevertheless, some equipment malfunctions have occurred. The staff's review indicates that most of these malfunctions were one-of-a-kind and typical of even industrial-quality electrical and mechanical instrumentation and components. There is no indication of significant degradation of the instrumentation and components, and there is strong evidence that the PUR-1 staff will remedy any future degradation with prompt corrective action. The staff did not consider prior utilization of other systems and components because degradation would occur gradually, be readily detectable, or not affect the likelihood of accidents.

16.2 Conclusion

In addition to the considerations discussed above, the NRC staff reviewed licensee event reports and inspection reports. On the basis of this review and the preceding considerations, the NRC staff concludes that there has been no significant degradation of facility systems or components. The NRC staff further concludes that the surveillance requirements in the TSs provide reasonable assurance that the facility will continue to be adequately monitored for degradation of systems and components.

17 **CONCLUSIONS**

On the basis of its evaluation of the application for license renewal, at an increased power level of 12 kWt, as discussed in the previous chapters of this SER, the NRC staff concludes the following:

- The application for license renewal dated July 7, 2008, as supplemented by letters dated June 4, 2010; November 15, 2011; January 4, January 30, January 31, June 1, June 15, June 29, July 13, and August 11, 2012; April 10, 2013; July 24, 2015; and January 29, February 26, March 31, May 9, July 7, July 19, September 19, and September 29, 2016, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended, and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations*.

- The facility will operate in conformity with the application, as supplemented, as well as the provisions of the Atomic Energy Act of 1954, as amended, and the rules and regulations of the Commission.

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

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

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

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

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

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

- The receipt, possession and use of by product and special nuclear materials as authorized by this facility operating license will be in accordance with NRC regulations in 10 CFR Parts 30 and 70.

18 **REFERENCES**

1. Purdue University Research Reactor, *"*Application for Relicense of License Number R-87 with Power Uprate, Safety Analysis Report," dated July 07, 2008, ADAMS ML083040443.

2. U.S. Government Printing Office.Code of Federal Regulations, Title 10, Chapter 1, revised January 1, 2016.

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