Safety Evaluation Report

Renewal of the Facility Operating License for the Rhode Island Nuclear Science Center Reactor

License No. R-95
Docket No. 50-193

Rhode Island Atomic Energy Commission

United States Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

January 2017

ABSTRACT

This safety evaluation report summarizes the findings of a safety review conducted by 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, as supplemented, filed by the Rhode Island Atomic Energy Commission (the licensee) for a 20-year renewal of Facility Operating License R-95 to continue to operate the Rhode Island Nuclear Science Center Reactor (RINSC reactor or the facility). The facility is located on the Narragansett Bay Campus of the University of Rhode Island in Narragansett, Rhode Island. In its safety review, the NRC staff considered information submitted by the licensee, including past operating history recorded in the licensee's annual reports to the NRC, interactions with the NRC staff, inspection reports (IRs) prepared by NRC personnel, and first-hand observations.

On the basis of this review, the NRC staff concludes that the licensee can continue to operate the RINSC reactor, in accordance with the application and for the period of the renewed license, without undue risk to the health and safety of the members of the public.

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ABBREVIATIONS, SYMBOLS, AND ACRONYMS

AC	alternating current
ADAMS	Agencywide Documents Access and Management System
AEA	Atomic Energy Act
ALARA	as low as reasonably achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
AWSS	auxiliary water supply system
CACS	Center for Atmospheric Chemistry Studies
CDE	committed dose equivalent
CEDE	committed effective dose equivalent
CFR	Code of *Federal Regulations*
CHF	critical heat flux
CPS	counts per second
CVS	confinement ventilation system
DAC	derived air concentration
DC	direct current
DIF	Dry Irradiation Facility
DOE	Department of Energy
ECP	estimated critical position
EES	emergency evacuation system
EP	emergency plan
EPRI	Electric Power Research Institute
EPS	emergency power source
EPZ	emergency planning zone
ESF	engineered safety feature
FC	forced convection
FR	*Federal Register*
FY	fiscal year
HEPA	high-efficiency particulate air
HEU	high enriched uranium
HP	high power
HV	high voltage
I&C	instrumentation and control
IR	inspection report
ISG	Interim Staff Guidance
LCC	limiting core configuration
LCO	limiting condition for operation
LEU	Low-enriched uranium
LOCA	loss-of-coolant accident
LOFA	loss-of-flow accident
LP	low power
LRA	license renewal application
LSSS	limiting safety system setting
MHA	maximum hypothetical accident
MMI	modified Mercalli intensity
MTR	materials testing reactor
MWt	mega-Watts
NC	natural convection

NFM	neutron flux monitoring
NRC	Nuclear Regulatory Commission
NRSC	Nuclear and Radiation Safety Committee
OCC	operational core configuration
ONB	onset of nucleate boiling
PCI	process control and instrumentation system
PCS	primary coolant system
PSP	physical security plan
RAI	request for additional information
RCS	reactor control system
RG	regulatory guides
RI	Rhode Island
RIAEC	Rhode Island Atomic Energy Commission
RINSC	Rhode Island Nuclear Science Center Reactor
RMS	radiation monitoring system
RO	reactor operator
RPP	Radiation Protection Program
RPS	reactor protection system
RSO	Radiation Safety Officer
RTR	research and test reactor
SAFSTOR	safe storage
SAR	Safety Analysis Report
SCFM	standard cubic feet per minute
SCS	secondary coolant system
SDM	shutdown margin
SER	Safety Evaluation Report
SL	safety limit
SNM	special nuclear material
SR	surveillance requirement
SRM	staff requirements memorandum
SRO	senior reactor operator
SSC	structures, systems and components
T&H	thermal and hydraulic
TEDE	total effective dose equivalents
TS	Technical Specifications
USAEC	United States Atomic Energy Commission
VAC	volts AC
VDC	volts DC
WC	water column
WR	wide range

Technical Parameters and Units

C	Celsius
cm	centimeters
cps	counts per second
F	Fahrenheit
gpm	gallons per minute
k_{eff}	k-effective; the eigenvalue for a nuclear core
kW(t)	kilowatts thermal
kph	kilometer per hour
kW	kilowatt
m	meter
MW	Megawatt (10^6 watts)
mho	unit of conductivity (reciprocal of resistance unit of Ohm)
mph	miles per hour
pH	potential of hydrogen
S	Siemens (measure of conductivity, 1S/cm = 1mho/cm)
T	temperature
V	Volts
$\Delta k/k$	expression of reactivity, relative to critical

1. INTRODUCTION

1.1 Overview

By letter dated May 3, 2004, as supplemented on January 19, February 4, August 6, August 18, September 3, September 8, November 8, November 26, December 7, and December 14, 2010; January 24, February 24, and July 15, 2011; March 15, September 16, and December 19, 2013; February 24, April 28, and June 30, 2014; August 7 and August 11, 2015; and January 20, February 26, March 1, April 21, July 20, October 6, November 1, November 14, December 1, December 8, December 13, and December 15, 2016, the Rhode Island Atomic Energy Commission (RIAEC or the licensee), submitted to the U.S. Nuclear Regulatory Commission (NRC) a license renewal application (LRA) for a 20-year the renewal of the Class 104c Facility Operating License No. R-95 (NRC Docket No. 50-193) for the Rhode Island Nuclear Science Center Reactor (RINSC reactor or the facility). A Notice of Opportunity for Hearing was published in the *Federal Register* (FR) on October 24, 2016 (81 FR 73148-73153). No requests for a hearing were received.

Title 10 of the *Code of Federal Regulations* (10 CFR) (Ref. 9) Section 50.51(a) states, in part, 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 the date of issuance." The RIAEC was issued Construction Permit No. CPRR-73 on August 27, 1962, as amended by License Amendment No. 1 to the Construction Permit issued October 10, 1963, which authorized the construction of the reactor at the RINSC site. The original license was to expire midnight on August 27, 2002, but a construction time recapture amendment (License Amendment No. 27) issued July 28, 2000 (Ref. 10) extended the expiration date to July 21, 2004. Because of the timely renewal provision contained in 10 CFR 2.109(a), the licensee is permitted to continue operation of the reactor under the terms and conditions of the current license until the NRC staff completes action on the LRA. A renewal would authorize continued operation of the reactor for an additional 20 years.

The NRC staff based its review of the request to renew the RINSC reactor facility operating license on the information contained in the LRA, as well as supporting supplements and licensee responses to the NRC staff's requests for additional information (RAIs). Specifically, the LRA included the updated safety analysis report (SAR), as supplemented, an environmental report, financial qualifications, operator requalification program, and technical specifications (TSs). The LRA indicates that there were no requested changes to the RINSC physical security plan (PSP) and the emergency plan (EP) as a result of the renewal request. The NRC staff conducted site visits on April 15, 2014, and November 2 - 3, 2016, to observe facility conditions and to discuss RAIs and RAI responses. The NRC staff issued RAIs on November 24, 2009 (Ref 66); April 13, 2010 (Ref. 67); December 17, 2012 (Ref. 68); August 15 (Ref. 69) and October 21, 2013 (Ref. 70); January 9 (Ref. 71) and March 20, 2014 (Ref. 72); September 3, 2015 (Ref. 73); and August 3 (Ref. 74) and October 14, 2016 (Ref. 75). In addition, the NRC staff conducted telephone conference calls with the licensee on several occasions.

The licensee provided responses to the RAIs in letters dated January 19 (Ref. 46), February 4 (Ref. 43), August 6 (Ref. 3), August 18 (Ref. 3), September 3 (Ref. 26), September 8 (Ref. 3), November 8 (Ref. 25), November 26 (Ref. 3), December 7 (Ref. 3), and December 14, 2010 (Ref. 3); January 24 (Ref. 3), February 24 (Ref. 3), and July 15, 2011 (Ref. 3); March 15, (Ref. 3) September 16 (Ref. 3), and December 19, 2013 (Ref. 22); February 24 (Ref. 3),

April 28 (Ref. 3), and June 30, 2014 (Ref. 35); August 7 (Ref. 37) and August 11, 2015 (Ref. 61); and January 20 (Ref. 62), February 26 (Ref. 4.h), March 1 (Ref. 4), April 1 (Ref. 65); July 20 (Ref. 63), October 6 (Ref. 64), November 1 (Ref. 54), November 14 (Ref. 5), December 1 (Ref. 49), December 8 (Ref. 57), December 13 (Ref. 58), and December 15, 2016 (Refs. 56, 59, 60).

Although the LRA did not request changes to the PSP and the EP as part of license renewal, the NRC staff reviewed these plans to ensure they are consistent with current NRC regulations and guidance. As part of the review, the NRC staff also reviewed annual reports of facility operation submitted by the licensee and IRs prepared by NRC personnel. Information from RIAEC annual reports reviewed cover the period for the years 2007 through 2016 (Ref. 16) and the NRC IRs reviewed cover the period for the years 2011 through 2016 (Ref. 28).

With the exception of the PSP, EP, and portions of the SAR and RAI responses that contain security related information, material pertaining to this review may be examined or copied for a fee at the NRC's Public Document Room, Room 01 F 21, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852. The NRC staff maintains an Agencywide Documents Access and Management System (ADAMS), which provides text and image files of NRC's public documents. Publicly available documents related to this license renewal may be accessed online through the NRC's Public Library, ADAMS Public Documents collection at http://www.nrc.gov/reading-rm/adams.html. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC's Public Document Room staff by telephone at 1-800-397-4209 or 301-415-4737, or send an e mail to PDR at Resources@nrc.gov. The PSP is protected from public disclosure under 10 CFR 73.21, "Protection of Safeguards Information: Performance Requirements." The EP and material containing security-related information is protected under 10 CFR 2.390, "Public inspections, exemptions, requests for withholding." Since portions of the SAR and RAI responses contain security related information and are protected from public disclosure, redacted versions are provided to the public in ADAMS.

Chapter 18, "References," of this safety evaluation report (SER) contains the dates and associated ADAMS accession numbers of the licensee's renewal application, related supplements, annual reports, and IRs.

In conducting its safety review, the NRC staff evaluated the facility against the requirements of the regulations, including 10 CFR Part 20, "Standards for Protection against Radiation"; 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material"; 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities"; 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions"; and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material." The NRC staff also considered the recommendations of applicable regulatory guides (RG) and relevant accepted industry standards, such as those of the American National Standards Institute/American Nuclear Society (ANSI/ANS)-15 series. The NRC staff also considered the recommendations contained in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," issued February 1996 (Ref. 11). The NRC staff compared calculated dose values for accidents against the requirements in 10 CFR Part 20.

In SECY-08-0161, "Review of Research and Test Reactor License Renewal Applications," dated October 24, 2008 (Ref. 12), the NRC staff provided the Commission with information on plans to streamline the review of LRAs for research and test reactors (RTRs). The Commission

issued its staff requirements memorandum (SRM) for SECY-08-0161, dated March 26, 2009 (Ref. 13). The SRM directed the staff to streamline the renewal process for such reactors, using some combination of the options presented in SECY-08-0161. The SRM also directed 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, "Detailed Description of Research and Test Reactor License Renewal Streamlining Options the Staff Has Considered," of SECY-08-0161. 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," dated October 15, 2009 (Ref. 14), to assist in the review of LRAs. The streamlined review process is a graded approach based on licensed power level. Under this process, the facilities are divided into two tiers. Facilities with a licensed thermal power level of 2 megawatt (MWt) and greater, or requesting a power level increase, undergo a full review using NUREG-1537. Facilities with a licensed power level less than 2 MWt may 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 conducted the RIAEC LRA review using the guidance in the final RTR ISG-2009-001 (Ref. 14), and because RINSC reactor's licensed power level is 2 MWt, the NRC staff performed a full review on the licensee's application in accordance with the guidance in in RTR-ISG-2009-001 using NUREG-1537.

The NRC staff separately evaluated the environmental impacts of the renewal of the license for the RINSC reactor in accordance with 10 CFR Part 51. The NRC staff published an Environmental Assessment and Finding of No Significant Impact in the *Federal Register* on January 5, 2017 (82 FR 1364), which concluded that renewal of the RINSC reactor license will not have a significant effect on the quality of the human environment.

The purpose this SER is to summarize the findings of the NRC staff safety review and to delineate the technical details considered in evaluating the radiological safety aspects for continued operation of the RINSC reactor. This SER provides the technical basis for renewing the RIAEC license for the operation of the RINSC reactor at power levels up to 2 MWt.

This SER was prepared by Patrick G. Boyle, Alexander Adams, Jr., John T. Adams, Cindy K. Montgomery, Edward M. Helvenston, Eben S. Allen, Geoffrey A. Wertz, Duane A. Hardesty, Steven T. Lynch, and Michael F. Balazik, in the NRC's Office of Nuclear Reactor Regulation (NRR), Division of Policy and Rulemaking, Research and Test Reactors Licensing Branch; and Michael Purdie in the NRR, Division of Inspection and Regional Support, Financial International Projects Branch. URS Corporation and Energy Research Inc., acting as NRC contractors, also provided input to the SER.

1.2 Summary and Conclusions Regarding the Principal Safety Considerations

In its evaluation, the NRC staff considered 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 NRC staff and firsthand onsite observations. On the basis of this evaluation and resolution of the principal issues reviewed for the RINSC reactor, the NRC staff concludes the following:

- The design, use, testing, and performance of the RINSC reactor's structures, systems, and components important to safety during normal operation discussed in Chapter 4 of the SAR, as supplemented, in accordance with the TSs, are safe, and safe operation of the facility can reasonably be expected to continue.

- The licensee's activities will continue to be useful in the conduct of research and development activities, as described in SAR Section 1.1.2.

- The licensee considered the expected consequences of a broad spectrum of postulated credible accidents and a maximum hypothetical accident (MHA). The licensee performed analyses, using conservative assumptions, of the most serious credible accidents and the MHA and determined that the calculated potential radiation doses to the facility staff and members of the public, would not exceed 10 CFR Part 20 dose limits.

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

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

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

- The licensee maintains a PSP for the facility and its special nuclear material (SNM) in accordance with the requirements of 10 CFR Part 73, "Physical Protection of Plants and Materials," which provides reasonably assurance that the licensee will continue to provide the physical protection of the facility and its SNM.

- The licensee maintains an EP in compliance with 10 CFR 50.54(q) and Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," which

provides reasonable assurance that the licensee will continue to be prepared to assess and respond to emergency events.

- The licensee's procedures for training its licensed reactor operators (ROs) and the operator requalification program provide reasonable assurance that the licensee will continue to have qualified staff who can safely operate the reactor.

- Operation of the facility and the handling of radioactive material under the control of the RIAEC Radiation Protection Program are not expected to result in doses to personnel in excess of 10 CFR Part 20 dose limits and are expected to be consistent with ALARA principles.

On the basis of these findings, the NRC staff concludes that the RIAEC can continue to operate the RINSC reactor, in accordance with the Atomic Energy Act (AEA) of 1954, as amended; NRC regulations; and Renewed Facility Operating License No. R-95 without endangering the health and safety of the public, facility personnel, or the environment. The NRC staff further concludes that the issuance of the renewed license will not be inimical to the common defense and security.

1.3 General Facility Description

As described in SAR Section 1.2, the reactor is housed in a building on the Narragansett Bay Campus of the University of Rhode Island. The RINSC facility contains the reactor and associated equipment in its own building, and attached laboratories, classrooms, and offices. Access to the entire facility is controlled.

The lower levels of the facility are of thick brick construction, as the facility was built on the foundation of the gun batteries of a decommissioned coastal defense site.

As described in SAR Section 4.3, the reactor is suspended in an open pool of light water. The pool is 9.8 meters (m) (32 feet (ft)) deep, and the bottom plate of the reactor is 8 m (26.3 ft) below the surface of the pool. The concrete biological shield surrounding the pool provides structural support for the reactor pool, and also provides axial radiation shielding protecting personnel on the lower levels of the reactor building from radiation. The shield is several feet of poured concrete. The pool is lined with aluminum to prevent seepage of primary coolant into the concrete. The pool has different regions and can be subdivided by installing pool gates. Using the gates, portions of the pool can be drained, if needed.

As described in SAR Section 4.1, the reactor fuel elements contain low enriched (less than 20 percent) uranium plate-type Uranium Silicide Aluminum (U_3Si_2-Al) fuel in aluminum cladding. Each fuel element contains two non-fueled end plates and 22 fueled plates, forms a 7.62 centimeters (cm) by 7.62 cm (3-inch (in) by 3 in) square in cross-section, and is approximately 1 m (40 in) long. The core is a 7 by 9 grid with the four corners occupied by the suspension pillars. Fuel occupies the central section of the grid and may be surrounded by graphite or beryllium reflector elements. Special irradiation baskets which are placed in grid positions at the edge of the reactor core are available for sample irradiation.

As described in SAR Section 4.2.2, reactivity control for the reactor is provided by four shim safety blades. Their drive mechanisms are positioned on the reactor bridge outside of the fuel region of the core. These shim safety blades use boron carbide as the neutron poison and are clad in aluminum. These blades are located within guide shrouds and are lifted by

electro-magnets. Upon interruption of power to the magnets, the shim safety blades will drop into the core, shutting down the reactor. The reactor also uses a regulating rod for fine adjustments to reactivity. The regulating rod is a stainless-steel box securely attached to its drive shaft. The regulating rod reactivity worth is not included for safe shutdown of the reactor, since it is not designed to scram. In various licensee documents, the shim safety control blades are described as the “shim safety,” “safety,” or “shim blades,” and the “regulating blade” and “regulating rod” are used interchangeably to describe the regulating control blade.

As described in SAR Section 4.1, the entire core and control mechanisms are suspended from a steel bridge that sits on rails atop the reactor support structure. The bridge can be moved by means of a hand crank to position the reactor at different locations in the pool. Power and instrument cables for the core are routed through a standpipe that extends above the reactor top deck with sufficient cable length to reach all bridge positions.

The reactor is licensed for steady-state operation at 2 MWt. The reactor can be operated using either forced flow or natural convection (NC) of coolant. However, the reactor power level is limited to 0.1 MWt with NC coolant flow. The reactor does not have pulse capability. Heat of the reactor pool is removed through the primary cooling system to heat exchangers and, ultimately, conveyed to the secondary cooling system and then to the environment using cooling towers adjacent to the reactor building.

As described in SAR Chapter 7, the control room consoles can monitor and control reactor function and coolant flow, and monitor effluents. The analog-based instrumentation displays flow, temperature, pH, and conductivity on digital meters and feeds the information to a monitoring computer touch screen with trend capabilities. The pumps and fans can be turned on and off utilizing the touch screen features, or with installed switches. A two-position switch allows selection of the control input.

The shim safety blade and regulating rod control system has computer based controls, but the previously installed analog control system is available as a backup to the digital system. Computer displays are available for indication of reactor power, nuclear instrumentation, core configuration, and control rod position. The required safety features of the system (e.g., automatic scrams) are not impacted by the digital upgrades that were performed. The new digital display system utilizes existing engineered outputs (isolated spare analog current output device) and performs the analog to digital conversion within the display device preventing any impact to the safety portion of the system. A more detailed evaluation of the system modifications and the relationship to the reactor protection system (RPS) is included and found acceptable in SER Chapter 7.

Area radiation monitors and air monitors provide radiological monitoring. These monitors provide readouts in the control room and have local alarms to alert personnel to radiation or airborne concentration levels above preset points. Installed smoke and fire alarm and emergency lighting systems provide for personnel safety. An emergency generator serves select electrical loads in the event of a power failure. The TSs do not require any safety-related equipment to be operable when the reactor is secured.

As described in SAR Chapter 5, the reactor pool water is filtered and purified using mixed bed demineralizers. A portion of the water in the pool is extracted through a pipe, purified, and returned to the pool through the makeup return line. Water conductivity and pH are monitored and controlled to provide acceptable conditions for the fuel and reactor components. This water

acts as a biological shield and heat sink. The reactor pool water also acts as an effective neutron reflector.

As described in SAR Chapter 10, the major experimental facilities include a flux trap, a thermal column, irradiation baskets, a pneumatic transfer system, and neutron beam port facilities. A modified beryllium reflector segment is used as a flux trap in the center of the core to hold samples. The pneumatic transfer system allows samples to be sent from a sending-receiving station to a position adjacent to the reactor core. Six beam ports are located around the reactor, as well as a through port that passes beneath the reactor.

1.4 Shared Facilities and Equipment

The RINSC shares a heating system with the adjacent Center for Atmospheric Chemistry Studies Building (CACS) of the University of Rhode Island. Water is heated in boilers in the CACS building and supplied to the RINSC facility. Demineralized water generated at the RINSC is shared with laboratories in the CACS building. The demineralized water supply to the CACS can be isolated, if necessary.

1.5 Comparison with Similar Facilities

The RINSC reactor uses plate-type low-enriched uranium (LEU) fuel, which is similar to other plate-type RTRs in the U.S. The control blades and drives are similar to the reactor at the University of Massachusetts at Lowell.

1.6 Summary of Operation

The licensee has operated the RINSC reactor in accordance with Facility Operating License No. R-95 and established procedures to facilitate experiments and research. The RINSC reactor is used for teaching, performing nuclear research, and providing a range of irradiation services. The RIAEC annual reports for the years 2007 through 2016 (Ref. 16) indicated that the reactor is critical for an average of 300 hours per year. This value represents the expected annual facility operation during the period of the renewed license. The annual reports (Ref. 16) did not indicate any significant degradation of fuel element integrity, control rod operability issues, or radiological exposure concerns. The review of the NRC IRs from years 2011 through 2016, find two non-cited violations: a violation of TS 6.5.4 in 2013, and a violation of TS 6.2.2 in 2012 (Ref. 28). These violations and corrective actions are discussed in SER Section 11.1.2, “Radiation Protection Program.”

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 a facility operating license for a research reactor, that the licensee shall have reached an agreement with the U.S. Department of Energy (DOE) for the disposal of high-level radioactive wastes and spent nuclear fuel. In a letter dated May 3, 1983, R.L. Morgan of DOE informed H. Denton of the NRC that DOE has determined that universities and other government agencies operating non-power reactors have entered into contracts with DOE providing that DOE retains title to the fuel and is obligated to take the spent fuel and high-level waste for storage or reprocessing.

An e-mail, dated January 15, 2014, (Ref. 55), sent from K. Osborne (DOE) to D. Hardesty (NRC) confirms this contractual obligation with respect to the fuel at the RINSC reactor

(DOE Contract No. 7743) continues and is valid from August 15, 2008, to December 31, 2017. Additionally, DOE states that it renews these contracts before their expiration to ensure that they remain valid. By entering into such an agreement with DOE, the licensee has satisfied the requirements of the Nuclear Waste Policy Act of 1982.

1.8 Facility Modifications and History

As described in SAR Section 1.1, on July 21, 1964, the U.S. Atomic Energy Commission issued Facility License No. R-95 to Rhode Island and Providence Plantation Atomic Energy Commission for the operation of the RINSC reactor at power level of up to 1 MWt. License Amendment No. 1, issued on September 10, 1968, authorized the operation of the RINSC reactor up to a maximum power level of 2 MWt. The reactor used high-enriched uranium (HEU) fuel until it was converted to use LEU fuel. The order to convert to LEU fuel was part of License Amendment No. 17 issued by the NRC on March 17, 1993.

As described in SAR Section 1.7.2, the licensee is continuing to modernize its equipment, replacing obsolete components and introducing computer-based displays and control systems. The nuclear instrumentation and associated reactor protection RPS was upgraded to a Gamma-Metrics system in 2006. The current round of upgrades utilizes spare outputs on the existing analog components to provide a modern computer-based display. Physical security upgrades of the facility are also continuing to be performed. All modifications to the facility are reviewed in accordance with the requirements in 10 CFR 50.59, “Changes, Tests and Experiments,” and approved by the Nuclear and Radiation Safety Committee, prior to installation.

The following table (reproduced from SAR Section 1.8) list changes to the facility.

Table 1-1 Facility Modifications

Facility Modernizations and History	Date
Reactor Room Exhaust Isolation Valve - Installation	1966
Fire Sprinkler - South Basement Area	1968
Reactor Room Intake Isolation Valve	1969 & 1973
New Laboratory Building Addition	1971
Cleanup Demineralizer Tank-Replaced	1977 & 1978
Primary System Heat Exchanger-Replaced	1973
Fire Sprinklers - North Basement Area	1984
Reactor Room Crane Upgrade - (Analysis in 1987)	1988
Reactor Building Roof Surface-Replaced	1991
Reactor Room Air Intake Isolation Valve & Duct-Replaced	1992
Facility Modernizations and History	**Date**
Regulating Rod-Replaced	1993
Cooling Tower #1-Replaced	1993
Shared Heating System Installation	1993
LEU (Fuel & Beryllium Reflectors) Upgrade	1993
Emergency Core Cooling System	1995
Primary Pump Replace & Piping Upgrade	1996
Heat Exchanger #2-Installation	1996
Primary System Diaphragm Valves-Replaced with Butterfly valves	1997

Makeup Demineralized Water System-Replaced	1997
Facility Modernizations and History	**Date**
Reactor Console Equipment	1997
Area Radiation Monitors-Replaced	1998
Fire detection and alarm system replaced	1999
Vehicle Barriers installed @ front & rear of confinement building 2003	2003

2. SITE CHARACTERISTICS

2.1 Geography and Demography

2.1.1 Geography

Chapter 2 of the SAR (Ref. 2), as supplemented by the licensee's responses to RAIs 2.1 and 2.2 (Ref. 3), and RINSC TS 5.1.1, indicate that the reactor is located on the Narragansett Bay Campus of the University of Rhode Island in Narragansett, Rhode Island (RI).

Narragansett, RI, is located in Washington County in the southern portion of the State. The campus sits at an elevated position overlooking the West Channel of Narragansett Bay. The reactor site was formerly one of the batteries of Ft. Philip Kearney, a coastal defense reservation decommissioned after World War II. The land surrounding the campus is a mixture of residential and forested areas. The RINSC reactor is on land leased to the RIAEC from the Rhode Island Department of Higher Education. The land adjacent to the RINSC reactor is owned and controlled by the University of Rhode Island.

The three areas concerning the normal operation, safety, and emergency actions associated with the reactor facility are: (1) the area within the operations boundary, (2) the area within the site boundary, and (3) the emergency planning zone (EPZ).

The operations boundary is the reactor confinement building. The area within this boundary is a "restricted access" area for which the RINSC Director has direct authority and control over all activities, normal and emergency. There are pre-established evacuation routes and procedures known to personnel who frequent this area. The operations boundary is within the site boundary.

The site boundary is presented in SAR Figure 2-5. The minimum distance from the operations boundary to the site boundary is 140 ft (42.7 m) to the southeast of the facility.

In addition, an EPZ has been established for which emergency plans have been developed to ensure that prompt and effective actions can be taken to protect the public in the event of an accident. The RINSC EPZ boundary is the reactor confinement building and lies completely within the site boundary.

Important aspects of the site description are included in the design features section of the RINSC TSs.

TS 5.1.1 states:

> 5.1 Site and Facility Specifications
>
> 5.1.1 The reactor facility is located in Narragansett, Rhode Island on a 3 acre section of the University of Rhode Island (URI), Narragansett Bay Campus (NBC).

TS 5.1.1 describes the licensee's site on the University of Rhode Island Narragansett Bay Campus. The NRC staff during site visits confirmed that the site description is accurate. The

NRC staff finds that the site description is consistent with its use in the licensee's SAR. Based on its review and site observations, the NRC staff concludes that TS 5.1.1 is acceptable.

An evaluation of TS 5.1.2 within TS 5.1, is provided and found acceptable in SER Section 6.2.1, "Confinement System."

2.1.2 Demography

As described in SAR Section 2.1.3, the population of the Town of Narragansett, RI, was 16,361, according to the 2000 U.S. Census. The NRC staff reviewed the U.S. Census Bureau data for the 2010 U.S. Census and finds that the population was 15,868. The population of Narragansett and the area surrounding the facility is considered well-developed and no great changes in the population are expected.

According the RINSC SAR, the University of Rhode Island Narragansett Bay Campus services between 200 and 300 students and faculty. The Narragansett Industrial Park is located less than 0.25 mi (0.40 km) west of the facility along South Ferry Road and houses several light industrial buildings and businesses. To the north of South Ferry Road is a residential area containing single family homes of about one per 0.5 acre (2,023 m^2). Within one mile (1.6 km) of the facility are scattered residential units and a small business strip along Route 1A to the west.

The NRC staff reviewed local maps and satellite photos to confirm that the area surrounding the immediate vicinity of the RINSC is mostly developed, and significant increases in the local population are not expected during the 20-year license renewal period. The NRC staff concludes that the demographic information provided by the licensee is sufficient to allow accurate assessments of the potential radiological impact on the public resulting from operation of the facility.

2.2 Nearby Industrial, Transportation, and Military Facilities

As described in SAR Section 2.2.1, the area surrounding the RINSC does not contain heavy industry. Other than the university, the local economy is primarily based on tourism.

As described in SAR Section 2.2.2, the closest major transportation route is Interstate Highway 95, approximately 12 mi (19.3 km) from the RINSC in an arc from the west to the north of the facility. The largest primary highway serving the vicinity of the facility is U.S. Route 1, which is approximately 2 mi (3.2 km) west of the facility. The nearest rail line to the facility passes through West Kingston, RI, approximately 6 mi (9.6 km) to the west.

The largest major airport is the T.F. Green State Airport in Warwick, RI, which is more than 15 mi (24 km) north of the RINSC. Closer to the RINSC is the Quonset State Airport in North Kingston, RI, approximately 6 mi (9.6 km) to the north. This airport serves general aviation traffic and the Rhode Island Air National Guard, in addition to serving as the home of the 143rd Airlift Wing utilizing C-130 transport aircraft. The main runway is not directly oriented towards the RINSC. Shipping traffic to and from the Port of Providence passes through the West Passage of the Narragansett Bay. This passage is approximately 4 mi (6 km) west of the facility and is blocked from a direct view of the RINSC by Conanicut Island.

As described in SAR Section 2.2.3, the nearest military facility is the Naval Station Newport in Newport, RI, home of the Naval War College. The War College is an educational institution located in the city of Newport.

In SAR Section 2.2.4, the licensee states that there are no nearby industrial, transportation, or military facilities with the potential of causing a credible accident (which could prevent a safe reactor shutdown or result in a release of radioactive material from the reactor facility) that would exceed the members of the public exposure limits of 10 CFR Part 20. The basic design and structure of the facility provides significant protection for the reactor. For example, the core is located near the bottom of a 32-ft (9.75-m) deep, aluminum-lined concrete pool. Additionally, the front and rear of the confinement building is protected at a minimum of 50 ft (15.2 m) with vehicle barriers of the bollard type at the front and the jersey type at the rear.

The NRC staff confirmed the locations and orientations of local transportation facilities through a review of local maps and satellite photos. A review of local maps also confirmed the characterization of nearby military facilities. The NRC staff finds that the military bases, transportation routes, and airports do not possess any significant consequence to the operational safety of the RINSC facility due to the distance of these facilities to the RINSC reactor. Based on its review of the character and distances of local industry, transportation, and military 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 RINSC reactor.

2.3 Meteorology

SAR Section 2.3 describes the meteorology near the RINSC. The RINSC is located adjacent to Narragansett Bay, which is connected to the Atlantic Ocean. According to information supplied by the licensee and confirmed by the NRC staff through a review of National Weather Service data, monthly average temperatures range from a low of -7 degrees Celsius (°C) (19 degrees Fahrenheit [°F]) in January to a high of 28 °C (82 °F) in July. Temperature extremes are a low of -22 °C (-8 °F) and a high of 34 °C (93 °F). There is precipitation throughout the year with an annual average of 123 cm (48 in).

As described in response to RAI 2.1 (Ref. 3), Rhode Island has a humid climate, with cold winters and short summers. The humidity varies depending on wind direction and ocean temperature. According to the Newport, RI weather station (Ref. 18), the average wind speed varies from 16 to 21 kilometers per hour (kph) (10 to 13 miles per hour [mph]), primarily from the west northwest.

According to SAR Section 2.3.2.5, and confirmed by reports from the Newport, RI weather station, the facility has experienced hurricanes during its license period (Refs. 2, 18). The only damage reported was some roof damage as a result of Hurricane “Bob” in August 1991, which recorded a high gust in Newport County of 159 kph (99 mph) (Ref. 19). According to the supplemental information provided on November 14, 2016 (Ref. 5), there was no damage to the facility from the most recent hurricanes in the area: Hurricane Irene (2011) and Hurricane Sandy (2012). Tornados are rare in RI and tend to only do limited damage to lightly constructed buildings.

Based on the meteorological information supplied by the licensee and identified by the NRC staff’s independent review, the NRC staff finds the meteorological information provided for the region around the RINSC reactor to be sufficient. Based on the information provided above, the

NRC staff concludes that meteorological-related events of credible frequency or consequences do not pose any significant risk for continued operation of the RINSC reactor.

2.4 Hydrology

SAR Section 2.4 describes the hydrology in the vicinity of the RINSC facility. The RINSC reactor is located on elevated ground above the West Passage of the Narragansett Bay. The NRC staff reviewed the relevant U.S. Geological Survey topographic map (Ref. 20) and the supplemental information provided (Ref. 5), and finds that the facility sits at approximately 97 ft (30 m) above mean sea level and 550 ft (168 m) from the bay, which is also far above extreme storm surge heights. As described in SAR Section 2.4.2, surface flooding is not a factor since the Bay Campus has a storm drainage system that intercepts local runoff and discharges it away from the site. There are no bodies of water above the elevation of the facility and no noted streams in the vicinity. Storm water piping is below the lowest RINSC ground level and drainage of the area is into the Narragansett Bay. According the maps noted in SAR Figure 2-4 and described in SAR Section 2.4.5, the RINSC is above areas that flood and, therefore, no credible source of flooding exists.

Combining a review of facility drawings with facility tours, the NRC staff observed that the shielding of the reactor pool and supporting structure does not allow neutrons from the reactor core to reach soil structures and groundwater. For this reason, the NRC staff concludes that the activation or contamination of the groundwater from reactor operations is not credible.

According to the response to RAI 2.2 (Ref. 3) and observations by NRC staff during a site visit, the level of the reactor pool is monitored and alarmed so that significant leakage would be detected and mitigated. Leakage from contaminated systems would be detected through the normal facility radiation surveys that are part of the facility's RPP described in SER Section 11.1.2. Leakage from the pool would accumulate in the bottom level of the reactor building and would discharge into the sanitary sewer system described in response to RAI 2.1 (Ref. 3). The concentration of radioisotopes in the pool water is consistently below the effluent release limits stipulated in 10 CFR Part 20, Appendix B, Table 3, "Releases to Sewers."

The NRC staff verified the information from the licensee in this section by reviewing local maps and making first-hand observations during facility tours. Based on its review, the NRC staff finds that the design of the facility and licensee's radiation protection practices minimize the potential for groundwater contamination. Any releases would likely be detected and found to be within the regulatory limits. Therefore, based on the information above, the NRC staff concludes that the local hydrology does not pose a significant risk to the continued safe operation of the RINSC.

2.5 Geology, Seismology, and Geotechnical Engineering

SAR Section 2.5 describes the geology in the vicinity of the facility. The RINSC is built upon the Narragansett Till Plains remaining from the glaciation of the region. The Till Plains are soil and rock formations deposited when a stationary glacier melted.

The State of Rhode Island, in general, has low seismic activity with few events greater than a modified Mercalli intensity level of V in the recorded history of the past 300 years. SAR Section 2.5.2 references a study indicating that the strongest seismic event for the area would be a modified Mercalli intensity of VI.

SAR Figure 2-8 illustrates seismic hazards for the United States based on new seismological, geophysical, and geological information (Ref. 2). The figure shows a relatively low ground-motion risk for a large area surrounding the RINSC site. The 2008 National Seismic Hazard Map shows only a 2-percent probability that in 50 years the peak lateral ground acceleration will exceed 0.07 times the acceleration due to gravity (Ref. 21).

The NRC staff finds that the licensee has provided sufficient information about geological features and potential seismic activity at the RINSC site. Based on the above information, the NRC staff concludes that the geology of the RINSC site is suitable for supporting the reactor building, structure, and systems, and that potentially damaging seismic events are unlikely to occur during the period of the renewed license. The NRC staff also reviewed the accident scenarios described in SER Chapter 13 and concludes that it is highly unlikely that a seismic event would cause damage to the facility that would result in the release of fission products greater than the MHA.

2.6 Conclusions

Based on the information above, the NRC staff concludes that the RINSC site has experienced no significant geographical, meteorological, or geological change since the issuance of the initial facility operating license. The site, therefore, remains suitable for the continued operation of the reactor. The demographics of the area surrounding the reactor have not significantly changed, nor is any change projected at this time, that could increase the risk to public health and safety from continued operation of RINSC for the 20-year period of the license renewal. Hazards related to industrial, transportation, and military facilities will not pose a significant risk to the continued safe operation of the facility. Infrequency of the occurrence of hurricanes, tornadoes, and earthquakes and the robustness of the facility continue to make the site suitable for operation of the reactor.

3. DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.1 Design Criteria

Section 3.1 of the SAR (Ref. 2) describes the design criteria for structures, systems, and components (SSCs) at the RINSC reactor to ensure that they are able to perform their intended functions. Some of the principal SSCs include the fuel, core support structure, reactor safety system, reactor pool, and reactor building.

The NRC staff reviewed the information in SAR Section 3.1 on design criteria for SSCs. Some of the more important considerations evaluated during normal operation and credible accident scenarios include the following:

- The fuel must prevent the release of fission products.
- The core support structure must maintain its orientation, geometry, and structural integrity.
- The reactor safety system must be able to shut down the reactor.
- The reactor pool must provide adequate shielding of radiation emitted from the reactor core and provide for heat removal from reactor components.
- The reactor building must provide a controllable environment for the movement of air and protects the reactor from external environmental conditions.

The fuel specifications for the U_3Si_2-Al plate fuel used by the RINSC were developed by the Idaho National Laboratory and evaluated by the NRC during the fuel conversion of the RINSC to LEU fuel. The aluminum matrix with dispersed fuel is designed to minimize the release of fission products. SER Chapter 4 evaluates and finds the fuel design acceptable.

As described in the response to RAI 4.8 (Ref. 3), the core support structure forms a rigid box holding the fuel assemblies. It is suspended from the reactor bridge by four corner posts that are bolted to the lower support plate and cross-braced. The core support system is designed to support the weight of the core as well as the control and cooling elements. After 40 years of supporting the weight of the structure, there was no observed appreciable deterioration recorded during inspections. The radiation and corrosion exposure to the structure are low enough, such that the strength of the materials will not be impacted by the continued operation of the facility (see discussion in the response to RAI 4.8 [Ref. 3]). SER Chapter 4 contains an additional discussion of the core support structure.

The reactor safety system consists of four safety blades that are raised by blade drive mechanisms. The blade drive mechanisms are attached to the blades by electromagnets during reactor operation. Upon detection of a condition requiring a reactor shutdown, the electromagnets are de-energized and the safety blades fall by gravity back into the reactor. The reactor safety system contains diverse and redundant instrumentation and scram functions and is designed to fail in a safe condition in the event of a loss of power. SER Chapter 7 evaluates and finds the reactor safety system acceptable.

The reactor pool absorbs much of the radiation emitted from the core to protect facility personnel who may be standing at the reactor top. The reactor pool and primary coolant system (PCS) also remove the heat from the operation of the core. Heat is transferred from the reactor pool and primary coolant to a secondary coolant system (SCS), and ultimately into the environment via a cooling tower. SER Chapter 5 evaluates and finds the reactor coolant system acceptable.

The reactor building is a confinement structure designed to maintain a negative pressure relative to the outside environment. The ventilation system keeps the interior of the reactor building at a lower than ambient pressure. As such, leakage into the confinement is from the outside environment. The exhaust from the facility is monitored and certain ventilation pathways are filtered. SER Chapter 6 evaluates and finds acceptable the reactor confinement.

The design and construction of the RINSC is in accordance with the license issued by the U.S. Atomic Energy Commission in 1964.

The RINSC reactor SSCs have been changed through license amendments or by licensee review without prior NRC approval under 10 CFR 50.59, “Changes, Tests, and Experiments.” Maintenance of SSCs was conducted using procedures developed and updated in accordance with TSs and 10 CFR 50.59. The NRC staff previously evaluated all amendments to the facility license, and the NRC inspection program verified that the licensee has conducted proper reviews under 10 CFR 50.59. The application for license renewal under review includes changes made to the facility since initial licensing. SER Chapter 16 evaluates and finds acceptable age-related issues.

Based on the discussion above, the NRC staff concludes that the design and construction of the RINSC reactor provides reasonable assurance that the reactor components and systems will continue to meet the design criteria throughout the license renewal period. The design criteria applied to the RINSC reactor are based on appropriate standards, codes, and criteria and provide reasonable assurance that the facility SSCs have been built and will function as designed and required by the analyses in the SAR. The licensee has implemented acceptable TSs to control important aspects of the facility design. Additionally, the design criteria provide reasonable assurance that the public will be protected from radiological risks resulting from operation of the facility.

3.2 Meteorological Damage

SER Section 2.3 and SAR Section 3.2 discuss the meteorology in the RINSC vicinity. The meteorological data documents the rare occurrences of extreme weather conditions that could affect the structure of the RINSC facility.

The NRC staff independently reviewed the meteorological data and determined that the highest recorded wind speed in the area was a peak wind gust of 159 kph (99 mph) in 1991, associated with Hurricane Bob. According to SAR Section 3.2, the reactor building sustained minor damage to the roof in that event. The reactor building is built to the appropriate building codes used at the time of construction. The reactor building has survived more than 40 years of weather phenomena at the site while sustaining only minor damage. The basic design and structure of the facility provides significant protection for the reactor. For example, the core is located near the bottom of a 32-ft (9.8-m) deep, aluminum-lined concrete pool. Additionally, the biological shield and construction atop a military concrete bunker provide protection against

natural phenomena that could result in damage to the core. As such, given the meteorological data for the site, the NRC staff finds that significant meteorological damage is unlikely.

Based on the information above, the NRC staff concludes that the design of the reactor building to protect against meteorological damages provides reasonable assurance that the facility SSCs will perform safety functions to effect and maintain safe reactor shutdown conditions, and to protect the health and safety of the members of the public from radioactive materials and radiation exposure.

3.3 Water Damage

SER Section 2.4 and SAR Section 2.4 discuss the hydrology in the vicinity of the RINSC. There are no bodies of water in the immediate vicinity of the RINSC site that are at an elevation higher than the reactor building. According to the supplemental information provided (Ref. 5), the lowest elevation at the facility is 97 ft (30 m) above sea level and the facility is located 550 ft (168 m) from the Narragansett Bay. The elevation is sufficient to protect against even extreme storm surge. Historic high levels of precipitation would not raise the water table to the point of inundating the reactor building structure.

Based on the information above, the NRC staff concludes that the likelihood that the RINSC could be damaged by water to the extent that would interfere with the safe operation or shutdown of the reactor is insignificant.

3.4 Seismic Damage

SER Section 2.5 and SAR Section 3.4 discuss the seismicity in the vicinity of the RINSC. The Rhode Island area is classified as being in Seismic Zone 2, as defined in the Uniform Building Code. The RINSC building, reactor foundation, shielding structure, reactor tank, and core support structure have been designed and constructed in accordance with this code. Since the reactor core is suspended in the pool by the reactor bridge, an earthquake of a sufficient magnitude would cause the reactor pool structure to shake, and a beyond design basis earthquake could cause dislocations between the reactor and immediate surroundings. However, in response to RAI 4.6 (Ref. 3), and described in SER Section 4.2.5, the licensee indicates that the core, pool, and supporting structure are designed to move as a unit in response to seismic forces. During a seismic event, the seismic scram detector de-energizes the shim safety blade electromagnets, releases the control blades, and shuts the reactor down. Sufficient tolerances between the shim safety blade and the shroud preclude the binding of a free-falling shim safety blade. If seismic activity did not de-energize the shim safety blade electromagnets and shut down the reactor due to power loss, the RO could manually shut down the reactor. In addition, as described more fully in SER Section 4.2.2, TS 3.2 provides a scram on seismic activity that can safely shut down the reactor.

The reactor is contained in a pool with an aluminum liner. SAR Chapters 4 and 5 describe the reactor core and the pool. The licensee analyzed a loss-of-coolant accident (LOCA) in SAR Chapter 13 and concluded that, under conservative assumptions for the core operating history, the maximum fuel temperature would remain below the temperature for melting the fuel elements. Additionally, SAR Section 3.4 states that any break in the primary coolant pipes would allow the pool to drain no lower than 12 ft (3.65 m) above the core due to pipe location in the pool concrete and anti-siphon provisions. SAR Section 3.4 further states that any resultant radiological doses from a seismic event would be bounded by the analysis in SAR Chapter 13.

Additional discussion on the RINSC accident analysis, including dose consequences from events such as a LOCA, is included in SER Chapter 13.

Based on the information above, the NRC staff concludes that the design of the facility is sufficient to protect the public in the event of seismic activity that can reasonably be expected to occur during the period of license renewal.

3.5 Systems and Components

Section 3.5 of the SAR describes the design bases for the systems and components required to function for safe reactor operation and shutdown. The licensee has identified the instrumentation and control (I&C) system, missile protection, reactor design, electric power systems, fluid systems, reactor confinement, and radioactivity control as important for safe operation of the facility.

SER Chapter 7 evaluates and finds acceptable the design of the I&C systems, including the reactor control and reactor safety systems such as the RPS. I&C systems also include shim safety blade control. Details on the design and function of the shim safety control blades is provided in SAR Section 4.2.2 and evaluated and found acceptable in SER Section 4.2.2.

As described in SAR Section 3.5.2, the reactor core is protected from external missiles by being surrounded by a large block of reinforced concrete. Additionally, the piping systems are anchored and imbedded in the concrete biological shield walls. As evaluated and found acceptable in SER Section 13.8, the thick monolithic structure housing the core provides sufficient protection for credible external events, such as missiles.

The reactor design is evaluated and found acceptable in SER Chapter 4. An evaluation of accident scenarios associated with this design is provided and found acceptable in SER Chapter 13. These chapters show that the most rapid possible reactivity insertion rates are adequately compensated for by period alarm and trip provisions.

SER Chapter 8 evaluates and finds acceptable electric power systems, including the 15-kW emergency backup system provided in case of power failure. SER Chapter 5 evaluates and finds acceptable the coolant system utilized to cool the reactor pool water during normal operation. SER Section 6.2.1 evaluates and finds acceptable the reactor confinement as part of the facility's engineered safety features (ESF). Radioactivity control, including the management of liquid radioactive sources and fuel storage and handling practices are evaluated and found acceptable in SER Sections 11.1.1 and 9.2, respectively.

Based on the information above and evaluated and found acceptable in other SER chapters, the NRC staff concludes that these discussions show that the reactor safety system design bases and the related TSs provide reasonable assurance that the reactor safety systems will function as designed to ensure the safe operation and safe shutdown of the reactor.

3.6 Conclusions

Based on the above findings, the NRC staff concludes that the design bases and operation since the issuance of the original operating license for this facility provide reasonable assurance that the RINSC SSCs will function as designed to ensure the continued safe operation and safe shutdown of the reactor. The NRC staff also concludes that the RINSC facility is adequately

designed and built to withstand any credible and probable wind, water, and seismic events associated with the site.

4. REACTOR DESCRIPTION

4.1 Summary Description

SAR Section 4.1 states that the RINSC reactor is a 2 MWt open-pool design with both natural and forced convection (FC) modes using water to cool the fuel.

The reactor core was fueled with HEU, until converted to LEU in 1993. Graphite and beryllium reflectors were added during the conversion to LEU fuel (Ref. 23).

The reactor core assembly is located near the bottom of a 9.8 m (32 ft) deep open pool of water. The reactor core assembly can be moved to one of three sections - the high power (HP) section, dual storage section, or low power (LP) section. The reactor is repositioned using a mechanical rail system located on the top ledge of the pool wall. The core may operate in the LP section under NC up to 0.1 MWt. At power levels greater than 0.1 MWt, the reactor is required to be operated in forced circulation (see Figure 4-1 below). The reactor core assembly must be in the HP section to engage the forced flow connections with the cooling system.

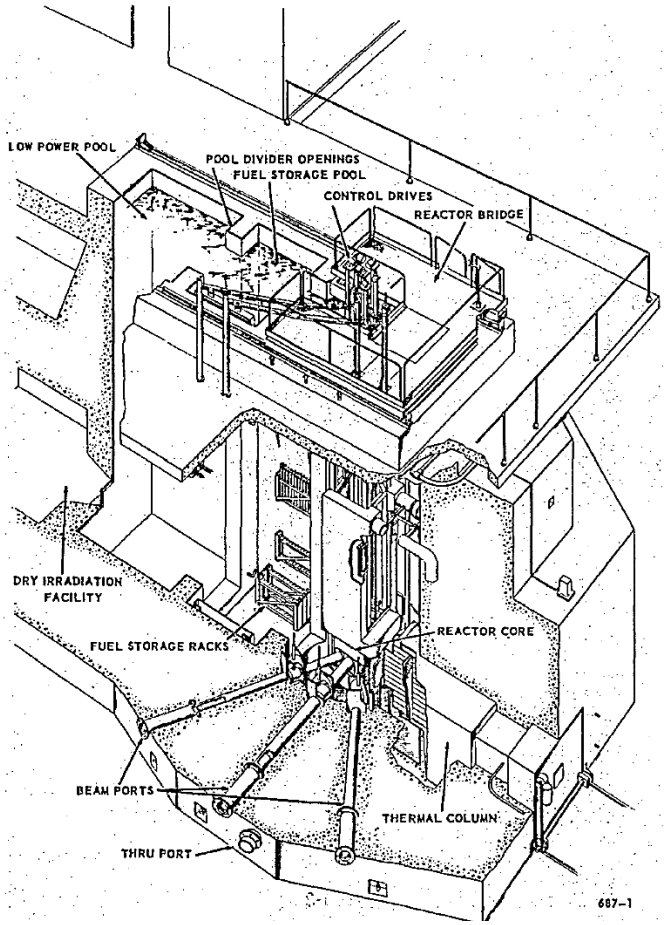

Figure 4-1 RINSC Reactor

Under forced cooling flow, the reactor grid box assembly is cooled by water circulated at approximately 7,410 liters per minute (1,950 gallons [gal] per minute [gpm]) by one or two primary pumps. The heat from reactor operation in the PCS is transferred to the SCS by means of two heat exchangers. Two cooling towers then dissipate the heat into the environment.

As described in SAR Section 4.1, the reactor accommodates several experimental locations. The beryllium (Be) reflector element in the center of the core has a flux trap for experiments. A Be plug is inserted when the flux trap is not being used. Experiments can be placed in irradiation baskets along the edge of the core opposite the thermal column. These baskets are designed for large samples and/or long duration irradiations. The reactor also provides irradiation locations for experiments by utilizing two pneumatic tubes for small targets and six horizontal beam ports for long-term irradiations and neutronic beam extraction experiments such as neutron scattering and neutron spectroscopy. A thermal column containing graphite is used for neutron radiography. A dry irradiation room located adjacent to the LP section of the pool supports gamma irradiation activities.

The reactor core is an arrangement of fuel elements (also referred to as fuel assemblies) in a rectangular array surrounded by graphite and Be reflectors. Four shim safety blades and a servo-actuated regulating rod provide reactivity control. Core elements are contained in a grid box that forces flow into the grid locations when the reactor is operating in forced cooling flow mode. The grid box assembly and the blade and rod drive mechanisms are supported by the suspension frame. The fuel and reflector elements that make up the core sit on a 7 by 9 position grid plate with the four corner positions occupied by the suspension frame comer posts. These corner posts connect the grid plate to the reactor bridge that spans the open pool. The neutron detectors are suspended within the water-filled corner posts. The grid plate is suspended about 8 m (26 ft) below the pool water surface. The core suspension system includes the reactor bridge, the suspension frame, the locating plate, and the blade and rod drive mechanisms.

4.2 Reactor Core

SAR Section 4.2 states that the core assembly consists of the reactor fuel, neutron absorbing shim safety blades, a neutron absorbing regulating control rod, the neutron moderator and reflector, the neutron source (as needed), and the core support structure.

The core consists of a 7 by 9 position array of 7.62-cm (3-in) square elements, with the center of the array filled with fuel and reflector elements and the four corners occupied by the suspension posts. Fuel elements consist of 22 fuel plates contained in a rectangular fuel box. Graphite and Be reflector elements surround the fuel section. A Be reflector with a flux trap occupies core position D-5 in the center of the core. Figure 4-2, below, graphically illustrates the core configuration.

SAR Chapter 4 describes two different operating core configurations. The first configuration uses 14 fuel elements in a 7 by 9 position array. The second configuration uses 17 fuel elements in the same 7 by 9 position array, with the additional three fuel elements replacing graphite reflectors near the thermal column. Figure 4-2 shows the two core arrangements, the first one containing 14 fuel elements, and the second arrangement containing 17 fuel elements. In response to RAI 4.30 (Ref. 3), the licensee further clarifies that the 17-element configuration provides a faster flux to the thermal column experimental facility.

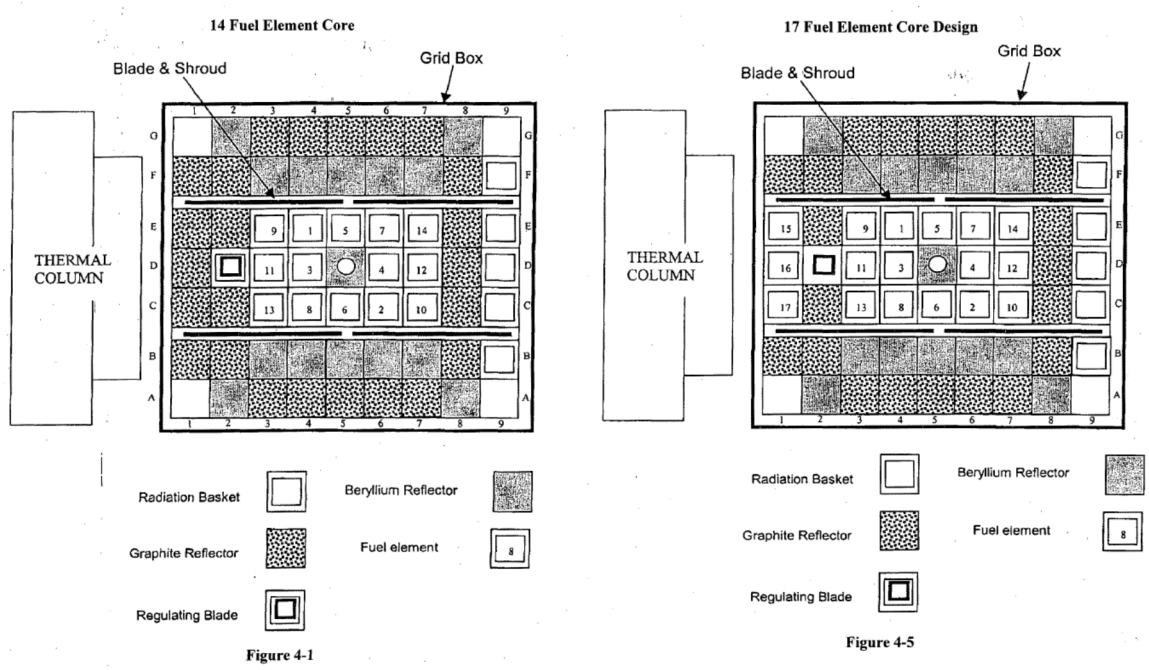

Figure 4-2 RINSC Reactor Core Layout

TS 5.4 states:

5.4 Reactor Core

5.4.1 The reactor core box consists of a grid plate with a 9x7 array of 3 inch square modules designed to receive various components (ex. fuel elements, reflectors, experimental baskets, detectors) and four aluminum side walls. The four corner positions also serve as structural support posts.

5.4.2 The standard core consists of 14 fuel assemblies arranged symmetrically between the shim safety blades in the center of the core box. An alternate core with an additional 3 fuel assemblies installed at the thermal column end of the core is also approved for use.

5.4.3 All core designs shall insure that the temperature coefficient is negative.

TS 5.4.1 requires the reactor core be assembled using a grid box of the stated dimensions. The NRC staff finds that this specification is consistent with the construction of the facility as shown in SER Figure 4-2 and as observed during site visits. The NRC staff also finds that this specification helps ensure that core configuration is controlled in a manner as to be consistent with the analysis provided. On the basis of this information, the NRC staff concludes that TS 5.4.1 is acceptable.

TS 5.4.2 identifies that the standard core is a 14-assembly combination of fuel assemblies. The NRC staff finds that this specification is consistent with the normal operating core as evaluated and found acceptable in SER Section 4.5.3. The NRC staff also finds this specification helps

ensure that core configuration is controlled in a manner as to be consistent with the analysis provided. On the basis of this information, the NRC staff concludes that TS 5.4.2 is acceptable.

TS 5.4.3 requires that all RINSC core designs have a negative temperature coefficient. This means that the combined temperature coefficient (fuel and moderator) needs to be negative over all allowed operating temperature ranges. The NRC staff finds this specification helps ensure that all core configurations will have a negative reactivity response to increases in temperature and will thus contribute to maintaining acceptable control over changes in reactor power. On the basis of this information, the NRC staff concludes that TS 5.4.3 is acceptable.

4.2.1 Reactor Fuel

SAR Section 4.2.1 and the response to RAI 4.2 (Ref. 3) describe the reactor fuel. The fuel is a dispersion-type fuel composed of Uranium Silicide-Aluminum (U_3Si_2-Al), which has an enrichment of less than 20 percent of uranium 235 (U-235) (LEU). Each fuel element consists of two aluminum side plates and 22 equally spaced flat fuel plates. The fuel itself has an active length of 61 cm (24 in) and is surrounded with aluminum cladding. The plates are separated by the side plates of the fuel element box allowing water to flow between the plates. Two end boxes of a similar size and shape allow the fuel to be positioned in any core location and fuel assemblies can be rotated in the x-y plane or flipped vertically for efficient fuel utilization.

TS 5.2 states:

> 5.2 Reactor Fuel
>
> 5.2.1 Each fuel element shall contain 22 plates containing uranium silicide fuel enriched to less than 20% in the isotope U-235 clad with aluminum.
>
> 5.2.2 Each fuel element shall contain no more than 283 grams of U-235.

TS 5.2.1 requires that each fuel element be fabricated with 22 fuel plates. TS 5.2.2 requires that each fuel element have a U-235 loading not to exceed 283 grams. The NRC staff finds these specifications help ensure that the fuel loading used is consistent with assumptions employed in the neutronics, thermal-hydraulic (T&H), and fuel failure analyses. Based on the information above, the NRC staff concludes that TS 5.2.1 and TS 5.2.2 are acceptable.

TS 3.9.2 states:

> 3.9.2 Low Enriched Uranium Fuel
>
> 3.9.2.1 The reactor shall not be operated with known fuel defects unless it is to facilitate the determination of which fuel element is damaged.

TS 3.9.2.1 requires the licensee to inspect the fuel elements for defects. The NRC staff notes that the fuel inspection procedures include, as a minimum, visual inspections for indications of defects, blisters, deformation, and oxide buildup or other problems that could restrict coolant flow between the fuel plates. The NRC staff finds that this specification helps ensure the early detection of an incipient fuel failure by examining the physical attributes, which will also help to minimize the potential release of radionuclides from the fuel. TS 3.9.2.1 also allows reactor operation with damaged fuel in order to determine which fuel element is damaged. Reactor operation is needed in cases where the fission products only leak from the fuel during reactor

operation resulting in elevated fuel temperatures. Fuel development and the qualification of the RINSC fuel are described in NUREG-1313, "Safety Evaluation Report Related to the Evaluation of Low-Enriched Uranium Silicide-Aluminum Dispersion Fuel for Use in Non-Power Reactors" (Ref. 27). TS 3.9.2.1 requires the fuel to be without physical defects when the reactor is operated unless that operation is needed to determine fuel damage. Based on the information above, the NRC staff concludes that TS 3.9.2.1 is acceptable.

TS 4.9.2 states:

> 4.9.2 Fuel Elements
>
> 4.9.2.1 The fuel elements shall be visually inspected and functionally fit into the core grid box on a rotating basis not to exceed five years such that:
>
> 4.9.2.1.1 The annual surveillance shall include at least one fifth of the fuel elements that are in the core,
>
> 4.9.2.1.2 The annual surveillance shall include fuel elements that represent a cross section with respect to burn-up,
>
> 4.9.2.1.3 If a fuel element is removed from use and the time since its last surveillance exceeds five years, it shall be visually inspected and functionally fit into the core grid box prior to being placed in use, and
>
> 4.9.2.1.4 If damage is visually determined or detected by Technical Specification 4.3.1.2 or otherwise discovered, then the surveillance shall be expanded to include all of the fuel elements prior to use, and annually thereafter.

TS 4.9.2.1 requires the licensee to periodically inspect fuel elements and cites the conditions that need to be met in order to satisfy the specification. These conditions include: (1) one fifth of the fuel elements be inspected annually, (2) a mix of fuel elements by burnup be inspected annually, (3) fuel elements out-of-core and not inspected for 5 years be inspected prior to use, and (4) any damage detected shall be a basis for more frequent inspections. The NRC staff finds that periodic fuel inspection helps ensure that the fuel continues to operate with effective barriers to prevent the inadvertent release of fission products. The NRC staff also finds that the surveillance frequencies are consistent with the guidance in NUREG-1537, ANSI/ANS-15.1-2007, and other facilities with materials testing reactor (MTR)-type fuel. The NRC staff finds that inspections of the fuel elements required by TS 4.9.2.1 provide adequate oversight of the physical condition of the fuel. The NRC staff reviewed the surveillance intervals in TS 4.9.2.1 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. Based on the information above, the NRC staff concludes that TS 4.9.2.1 is acceptable.

The NRC staff finds that SAR 4.2.1 and RAI responses, as discussed above, accurately characterize the RINSC reactor fuel elements. These discussions include the design limits of the fuel elements and provides the technological and safety bases for these limits. The application refers to the fuel development program that determined all fuel characteristics and parameters important to the safe operation of the reactor. Information on the design and

development program for this fuel offers reasonable assurance that the fabricated fuel can function safely in the reactor without adversely affecting the health and safety of the public. Based on the information above, the NRC staff concludes that the RINSC descriptions of the fuel in the SAR, the responses to the RAIs, and the associated TSs are acceptable.

4.2.2 Control Blades

SAR Section 4.2.2 states that the RINSC reactor has five independent control blades, four scrammable shim safety blades, and one non-scrammable regulating rod. In response to RAI 14.27 (Ref. 3), the licensee states that the shim safety blades are fabricated from a neutron absorbing material that compensates for fuel burnup, temperature, and poison effects. SAR Section 4.2.2 states that the absorbing material used is boron carbide, and it is sandwiched between aluminum side plates.

The regulating rod is used to control power either manually or automatically through the Servo-Controlled Regulating Blade Drive System described in SAR Section 7.2.7, and the Automatic Power Level Channel evaluated and found acceptable in SER Sections 7.3 and 7.4. SAR Section 4.2.2 states that the regulating rod is a 63.5 cm (25 in) long by 5.04 cm (1.98 in) square stainless steel channel.

According to the supplemental information provided (Ref. 49), the facility has an analog rod drive system that has a digital indication of rod position. The digital system can be used to drive the analog system. The digital system is in series with the analog system. The control system is designed to ensure that the control blades are all independent, so a malfunction in one drive system would not affect the insertion or withdrawal of any other. The shim safety blades are positioned with a stepper motor and reducers connected to the shim safety blade through an electromagnet. When a reactor scram occurs, the electromagnet in the control rod drive mechanism is de-energized allowing the shim safety blades to insert into the core by gravity.

In response to RAI 4.6 (Ref. 3), the licensee explains the design characteristics of the core that provide the assurance of proper shim safety blade insertion. The response states that the core is suspended from a bridge that is mounted over the top of the reactor pool and rests upon a military gun pad. The pool is constructed of a large mass of reinforced concrete and consequently, in the event of an earthquake, the pool, the bridge, and the core are expected to move as a unit. The shim safety blades fit inside a shroud, which is part of the core grid box. When fully withdrawn, the ends of the shim safety blades remain inside the shroud, which prevents misalignment when the shim safety blades are scrammed. A significant earthquake would likely shake the shim safety blades free from the magnets. However, the reactor is fitted with a seismic scram sensor that scrams the reactor upon the detection of an earth tremor. This is evaluated and found acceptable in SER Section 7.4.

The licensee states in response to RAI 4.4 (Ref. 3) that RIAEC staff performs an annual inspection of the shim safety blades by raising each blade to its full upper-most position and visually inspecting each shim safety blade. As discussed below, TS 4.2.2 requires measurement of the shim safety blade reactivity insertion rate and TS 4.2.1 requires measurement of the shim safety blade drop times. These measurements provide indication that the shim safety blade motion is not hindered. The measurement of the reactivity insertion rate allows a comparison of the differential control rod worth as it changes over time. Unexpected changes in the reactivity worth of the shim safety blade may provide an indication of the degradation of the shim safety blade. In response to RAI 4.4 (Ref. 3), the licensee confirms that

inspections of shim safety blades show no signs of a reduction in their function during their operational history.

According to SAR Section 3.5.1, shim safety blade ejection is not a credible event due to the fact that the PCS operates at the same pressure as the atmosphere. The NRC staff reviewed this assumption and considers it to be conservative as the water above the shim safety blades will provide downward pressure during NC mode, and the water discharged into the core in FC mode is also downward. Furthermore, the licensee states that it is not credible for the shim safety blades and the regulating rod to be forced out of the bottom of the core because in the full down position, the blades are approximately 2.54 cm (1 in) above the safety plate located near the bottom of the tank. The dashpot assembly slows the rate of shim safety blade insertion near the bottom of the stroke to limit deceleration forces. The NRC staff finds these design considerations acceptable.

TS 3.2.1, 3.2.2 and 3.2.3 state:

> 3.2 Reactor Control and Safety System
>
> The reactor shall not be operated unless:
>
> 3.2.1 All four shim safety blades and the regulating rod are operable.
>
> 3.2.2 All four shim safety blades are capable of being fully inserted into the reactor core within 1 second from the time that a scram condition is initiated.
>
> 3.2.3 The total reactivity insertion rate of any one shim safety blade and the regulating rod simultaneously does not exceed 0.02%Δk/k per second.
>
> 3.2.4 See SER Section 7.4

TS 3.2.1 requires that all four of the shim safety blades and the regulating rod are operable during reactor operation. The bases for TS 3.2.1 state, in part, that "[t]his ensures that all control rods are being controlled by the reactor control system (RCS) and the licensed operator." The NRC staff considers withdrawal and insertion of the control rods while observing positive indication of motion on the console indicators to be an acceptable method to satisfy the requirements in TS 3.2.1. This specification helps to ensure that the blades are responding to operator commands and are thus capable of performing their manual and automatic functions. Based on the information above, the NRC staff concludes that TS 3.2.1 is acceptable.

TS 3.2.2 requires that the scrammable shim safety blades be fully inserted within 1 second after a scram signal is initiated. The NRC staff finds this helps ensure that the scram times assumed in the safety analysis are satisfied by the actual shim safety blade performance. The NRC staff reviewed this specification and compared it to the assumptions in the analyses in SAR Chapter 13. A variety of power transients are analyzed using the assumed scram time (see SER Section 13.2). These analyses show that if the reactor is operated in accordance with the TS, this time delay will not cause an over power condition that exceeds the safety limit (SL). The NRC staff finds that TS 3.2.2 provides key performance criteria for ensuring that the shim safety blades can perform their intended scram function. The NRC staff also finds that this scram time is typical of other facilities and is consistent with the guidance in NUREG-1537,

Appendix 14.1. Based on the information above, the NRC staff concludes that TS 3.2.2 is acceptable.

The NRC staff reviewed the TS 3.2.3 limit on the rate of reactivity insertion in conjunction with the assumptions used in the reactivity addition accidents presented by the licensee. The NRC staff finds that the total reactivity inserted is significantly below the allowable step insertion limit of 0.02 Δk/k (TS 3.1.1.3.2 and TS 3.1.1.2.1) evaluated and found acceptable in SER Section 13.2.2. Based on the information above, the NRC staff concludes that TS 3.2.3 is acceptable.

TS 4.2.1 and TS 4.2.2 state:

4.2 Reactor Control and Safety System

4.2.1 Shim safety drop times shall be measured:

- 4.2.1.1 Annually
- 4.2.1.2 Whenever maintenance is performed which could affect the drop time of the blade
- 4.2.1.3 When a new core is configured
- 4.2.1.4 Following control blade changes

4.2.2 Shall measure each shim safety blade and regulating rod reactivity insertion rates:

- 4.2.2.1 Annually
- 4.2.2.2 Whenever maintenance is performed which could affect the reactivity insertion rate of the blade
- 4.2.2.3 When a new core is configured
- 4.2.2.4 Following control blade changes

4.2.3 See SER Section 7.4

- 4.2.3.1 See SER Section 7.4
- 4.2.3.2 See SER Section 7.4
- 4.2.3.3 See SER Section 7.4
- 4.2.3.4 See SER Section 7.4

TS 4.2.1 requires the licensee to periodically measure safety blade scram drop time and cites conditions that must be met in order to satisfy the specification. A measurement must take place: (1) on at least on an annual basis, (2) whenever maintenance potentially affecting drop time takes place, (3) whenever the core configuration is changed, and (4) whenever the blades are changed. The NRC staff finds that these measurement criteria are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007, the surveillance intervals are reasonable, and they help ensure that the shim safety blades are operable. The NRC staff finds that this surveillance requirement (SR) supports TS 3.2.1 and TS 3.2.2. Based on the information above, the NRC staff concludes TS 4.2.1 is acceptable.

TS 4.2.2 requires the licensee to periodically measure safety blade and regulating rod insertion rates and cites conditions that shall be met in order to satisfy the specification. A measurement must take place: (1) on at least on an annual basis, (2) whenever maintenance potentially affecting reactivity insertion rate takes place, (3) whenever the core configuration is changed, and (4) whenever the blades are changed. The NRC staff finds that these measurement criteria are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007, the surveillance intervals are reasonable, and they help ensure that the safety blades and regulating rod are operable. The NRC staff finds that this SR supports TS 3.2.3. Based on the information above, the NRC staff concludes that TS 4.2.2 is acceptable.

The NRC staff concludes that the continued operation, as limited by the above TSs, offers reasonable assurance that the RCS can meet the design objectives of reactor operability and shutdown capability, which are necessary to protect fuel integrity and the health and safety of the public and is therefore acceptable.

4.2.3 Neutron Moderator and Reflector

According to SAR Section 4.2.3, the RINSC core utilizes light water as a moderator. Reactor-grade graphite and Be are used as reflectors. Graphite was the reflector in the original design, and Be was added after the conversion to LEU fuel for efficiency and operational purposes. The core configuration, as discussed above, determines the number of graphite reflectors in the core. The design of the core takes into account thermal expansion, irradiative growth, and gas evolution of the graphite. Beryllium reflectors are replaced when reaching their life expectancy.

TS 3.9.1 states:

> 3.9.1 Beryllium Reflectors
>
> The maximum accumulated neutron fluence shall be $1x10^{22}$ neutrons/cm^2.

TS 3.9.1 is based on an analysis performed at another facility utilizing the same reflector material. The University of Missouri Research Reactor staff, after a failure of a Be reflector element in that reactor, identified a limiting fluence which is higher than the TS limit proposed by the RIAEC. That analysis also references previous work performed at the high flux integral reactor at the DOE, where the presence of small cracks at fast fluence of 1.8×10^{22} neutrons per cm^2 (nvt) were noticed and suggest that “a value of 1×10^{22} nvt (>1MeV) could be used as a conservative lower limit for determining when replacement of a beryllium reflector should be considered.” The RINSC limit of 1×10^{22} nvt is even more conservative because it credits flux from all neutron energies (not just fast neutrons i.e. energies >1 MeV). Therefore, the NRC staff finds that this limit is acceptable. Based on the information above, the NRC staff concludes that TS 3.9.1 is acceptable.

TS 4.9.1 states:

4.9.1 Beryllium Reflector Elements

4.9.1.1 The maximum neutron fluence of any beryllium reflector shall be determined and verified to be less than $1x10^{22}$ neutrons/cm^2 annually.

4.9.1.2 The beryllium reflectors shall be visually inspected and functionally fit into the core grid box on a rotating basis not to exceed five years such that:

4.9.1.2.1 The annual surveillance shall include at least one fifth of the beryllium reflectors that are in the core,

4.9.1.2.2 If a beryllium reflector is removed from use and the time since its last surveillance exceeds five years, it shall be visually inspected and functionally fit into the core grid box prior to being placed in use, and

4.9.1.2.3 If damage is discovered, the damaged reflector shall be removed from service and the surveillance shall be expanded to include all of the beryllium reflectors prior to use, and annually thereafter.

TS 4.9.1.1 requires the licensee to evaluate the effects of neutron fluence on the Be reflectors annually. The NRC staff finds this specification helps ensure that the stated fluence exposure limit is not exceeded. Based on the information above, the NRC staff concludes that TS 4.9.1.1 is acceptable.

TS 4.9.1.2 requires the licensee to inspect the Be reflectors and cite the conditions that shall be met in order to satisfy the specification. These conditions include: (1) one fifth of the Be reflectors shall be inspected annually, (2) those out-of-core and not inspected for 5 years shall be inspected prior to use, and (3) any damage detected shall be a basis for increased inspection frequency for all Be reflectors and any damaged Be reflector will be removed from service. The NRC staff finds that these conditions are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007, and the surveillance intervals are reasonable. The NRC staff also finds the SRs support TS 3.9.1 and are therefore acceptable. Based on the information above, the NRC staff concludes that TS 4.9.1.2 is acceptable.

The NRC staff reviewed the design of the reflectors and compared the design with similar reflectors at other RTRs. The NRC staff also reviewed the calculation regarding lifetime neutron fluence and the Be reflector analysis. The NRC staff finds that the graphite reflector is able to withstand the lifetime neutron flux over the licensed period, and the TSs imposed are sufficient to preclude damage to the Be reflector. Based on the information above, the NRC staff concludes that continued operation within the requirements of the TSs provides reasonable assurance that the reflector systems designed for this reactor will perform as necessary and will not adversely affect safe reactor operation or shutdown, or cause an uncontrolled release of radioactive material into the unrestricted environment.

4.2.4 Neutron Startup Source

The licensee states, in response to RAI 4.7 (Ref. 3), that there are three neutron sources available for use as a start-up source. The first is a pair of plutonium Be (PuBe) sources that are stored together in a common container, the second is an Antimony-Be source, and the third consists of the Be reflectors in the reactor core. The reactor start-up channel has a neutron count interlock of 3 counts per second (cps), which is the minimum neutron count rate that must be present in the core in order to perform a startup. Any one of these three sources may be used as a start-up source.

However, in response to RAI 4.7 (Ref. 3), the licensee indicates that a discrete neutron startup source is not needed or utilized for the reactor. The reason given is because the gamma decay from the fission products in the fuel interacts with the Be reflector to produce a sufficient level of photo neutrons to have a neutron count rate of at least 3 cps. Therefore, external sources are generally not needed to have a neutron count rate of at least 3 cps in the core. The 3-cps count rate satisfies the minimum count rate for the startup interlock in TS 3.2.

The NRC staff evaluated the use of the described neutron sources and finds that they are comparable to those used in other licensed RTRs with a Be reflected core. The NRC staff reviewed the operational history and the design and finds it adequate for source range indication and subcritical measurements. Based on its review, the NRC staff concludes that the continued use of any of the three neutron start-up sources in accordance with the applicable TSs and procedures provides reasonable assurance that the sources can perform the required functions safely and reliably.

4.2.5 Core Support Structures

According to SAR Section 4.3, the reactor core support structure consists of a suspension frame bolted to a movable bridge. The core sits on a 7 by 9 position grid plate with the four corner grid positions occupied by the frame support posts. The grid box is immediately above the grid plate with enclosed sides. The entire suspended frame is capable of movement from one end of the pool to the other by using a hand crank. The bridge moves along rails embedded in the top of the concrete biological shield. In response to RAI 4.8 (Ref. 3), the licensee states that the core support structure has the ability to support the weight of the core, the control blades, and cooling structure. The support structure is constructed of 6061-T6 alloy aluminum, which is resistant to corrosive environments. The licensee states that the primary coolant chemistry monitoring program, which has preserved the core support structural integrity during its 40-year lifetime, will continue to perform this function.

Cross braces and stiffeners provide the strength for and align the upper half of the frame. The coolant flow channels align the lower half of the frame and provide the flow paths when the reactor is positioned in the HP region of the pool. Three sides of the frame have stiffeners, while the fourth side is open to provide access to the core. In response to RAI 4.6 (Ref. 3), the licensee indicates that the core, pool, and supporting structure are designed to move as a unit in response to seismic forces. Control blades move within shrouds and the grid box that provide their alignment during movement. A portion of each safety blade remains within the shroud at the fully withdrawn position. The seismic scram detector de-energizes the shim safety blade electromagnets and shuts the reactor down during a seismic event. Sufficient tolerances between the safety blade and the shroud preclude the binding of a free-falling safety blade.

The NRC staff reviewed the SAR and supplemental materials regarding the design of the core support structure and its ability to perform its function during the license renewal period in accordance with the guidance in NUREG-1537. The NRC staff finds that the RINSC structure is consistent with corresponding structures at other RTRs, and the binding of a free-falling shim safety blade is not likely to occur because of tolerances inherent in the design. The NRC staff reviewed the RINSC LRA and RAI responses and finds that they adequately describe the design for the structural support of the core that ensures a stable and reproducible core configuration for all anticipated conditions throughout the renewal period. The NRC staff also finds that the core support structure is conducive to a sufficient coolant flow that is compatible with the coolant and radiation environment. Based on the information above, the NRC staff concludes that the core support structure is acceptable for the continued safe operation of the RINSC during the license renewal period.

4.3 Reactor Pool

According to SAR Section 4.3, the reactor core is on a 7 by 9 position grid plate near the bottom of a 9.8-m (32-ft) deep pool. The pool is made of aluminum-lined concrete walls that holds approximately 152,000 l (40,000 gal) of light water, if the primary coolant piping system is included. The pool level is automatically controlled by the primary coolant water makeup system evaluated and found acceptable in SER Section 5.5.1. The referenced height for the pool is 7.22 m (23.8 ft), which is the depth of water above the top of the active fuel sitting in the reactor grid box 40.6 cm (16 in) below the suspension frame base plate elevation. There are six horizontal beam ports and a through port located within the pool wall for long-term irradiations and neutron beam extraction experiments, such as neutron scattering and neutron spectroscopy. Reactor penetrations and piping are designed to prevent siphoning from uncovering the core. The LOCA analysis in SER Section 13.3 describes the consequences for a loss of coolant through the beam ports.

In response to RAI 4.9 (Ref. 3), the licensee describes the long-term ability of the pool liner to resist radiation, chemical, and thermal degradation and to continue to perform its function during the relicensing period. The pool liner is constructed of 6061-T6 alloy aluminum. Corrosion of this material is not expected because of the water monitoring and inspections. Radiation effects on the biological shield are minimal due to the shielding effect of the pool water. The relatively low temperature of the water will not lead to thermal damage of either the pool liner or the biological shield.

According to the Section 4.3 of the licensee's supplemental information (Ref. 5), each of the beam ports and the through tube have a 1.27 cm (0.50 in) drain line associated with them for the detection of leaks. These lines come together and are collected in the basement near the make-up system and ion exchanger, which is checked daily for signs of leakage from these experimental facilities. As part of securing the facility each day, the change in the running total volume of make-up water that has been added to the pool is recorded on RINSC Form NSC-15, "RINSC Checklist for Securing Reactor Facility." If the volume has changed by more than 100 gal (379 l) over a 3-day period, the Radiation Safety Officer (RSO) is notified. The RSO and Health Physicist then investigate and determine the cause. See Section 2.4 for additional evaluation and finding of acceptability of primary coolant leakage.

TS 5.6 states:

5.6 Reactor Pool

5.6.1 The reactor pool is made of concrete with an aluminum liner.

TS 5.6.1 describes the important design features of the reactor pool. The NRC staff finds that this specification helps ensure that analysis in the SAR are consistent with the reactor pool design. Based on its review, the NRC staff concludes that TS 5.6.1 is acceptable.

The NRC staff reviewed the SAR, as supplemented, and finds that it adequately describes the reactor pool design features. In addition, acceptable detection measures and preventive maintenance procedures provide reasonable assurance that the associated components are capable of withstanding the corrosion and radiation environment for the extended period of the license. The reactor system and experiment facility penetrations and piping are designed to prevent siphoning to minimize the potential for a pool boundary integrity failure that could lead to a loss of coolant or other types of malfunction. Based on the information above, the NRC staff concludes that the reactor pool is acceptable for the continued safe operation of the RINSC.

4.4 Biological Shield

SAR Section 4.4 discusses the RINSC biological shield. The stated purpose of the biological shield is to provide radiation shielding to workers on or around the reactor facility. The shield is designed to keep radiation levels below 1.0 millirem per hour (mrem/hr) at any point above or outside the pool. This shield consists of 7.3 m (24 ft) of water above the core, with water and concrete shielding in the LP section of the pool. The beam ports are shielded by lead plugs, and the thermal column can be covered by a steel door. The nitrogen-16 delay tank holds up the water containing nitrogen-16 for a period of time before returning it to the pool. This allows the dose from the nitrogen-16 to be reduced to less than 1 mrem/hr before returning into the pool. SER Chapter 10 evaluates and finds acceptable the Dry Gamma Room located inside the biological shield at the LP end of the pool.

The NRC staff reviewed the RINSC biological shield design and compared it to shields at similar RTRs. The staff finds that the pool liner is constructed of corrosion-resistant materials, and the degradation of the biological shield from neutron irradiation is shown to be negligible over the relicensing term. Based on the information above, the NRC staff concludes that with the required surveillance of the reactor water chemistry, the biological shield, and the pool liner will continue to perform their design functions during the relicensing period.

4.5 Nuclear Design

In the neutronics analysis, the licensee provided reactor core configuration information indicative of a typical RINSC operational core configuration (OCC), for use in the reactor core analyses as discussed in SER Section 4.5.1. The OCC is an as-built core that provides benchmarking information for reactor neutronic and T&H calculations. The results of the OCC analyses are compared to measurements which help to validate that the codes and methods used are accurate. Using the same codes and methods to analyze the limiting core configuration (LCC) helps to provide confidence in the predicted results of the LCC analysis.

The licensee follows the guidance provided in NUREG-1537, Section 4.5.1, to establish a LCC. The LCC is defined in NUREG-1537 as the core configuration that would yield the highest

power density using the fuel authorized for use in the reactor. The LCC establishes limiting operating conditions and represents a core that typically has not been configured by the licensee, but could be under the approved TSs. The configuration of the RINSC LCC is defined in the licensee's supplemental information (Ref. 4) as being a 14-fuel assembly core as discussed in SER Section 4.6.

The information discussed in this section establishes the design basis for the content of other chapters, specifically the safety analysis and portions of the TSs. The analysis presented in this section includes both neutronic and T&H results. Neutronic results utilize the Argonne National Laboratory (ANL) Code System Using Variational Nodal Methods and Finite Difference Methods to Solve Neutron Diffusion and Transport Theory Problems (DIF3D/VARI3D/ANL), Code System for Analysis of Fast Reactor Fuel Cycles (REBUS) code package, ANL Deterministic Code System for Reactor Lattice Calculation (WIMS/ANL), and Monte-Carlo Neutron Particle Transport Code System (MCNP) codes. T&H results utilize the ANL FORTRAN based code for plate reactor T&H analysis (PLTEMP) and ANL natural convection T&H analysis (NATCON) computer codes. These are evaluated in more detail and found acceptable in subsequent sections of this SER.

4.5.1 Normal Operating Conditions

In a paper presented to the RTR community (Ref. 24), the RINSC reactor major operating factors have been characterized using comparisons of DIF3D calculations and facility measurements. This paper describes the depletion of the fuel thus incorporating into the model the accumulation of fission products and transmuted elements. The NRC staff finds that these comparisons provide a basis for determining the suitability for the using the DIF3D model for predictive calculations.

Figure 4-3, below, shows the first LEU core 12-fuel assembly configuration and the estimated critical position (ECP) for this configuration. Note that the regulating rod is not indicated in the graphic, but is located to the left of assembly 11. The central assembly location is a beryllium reflector.

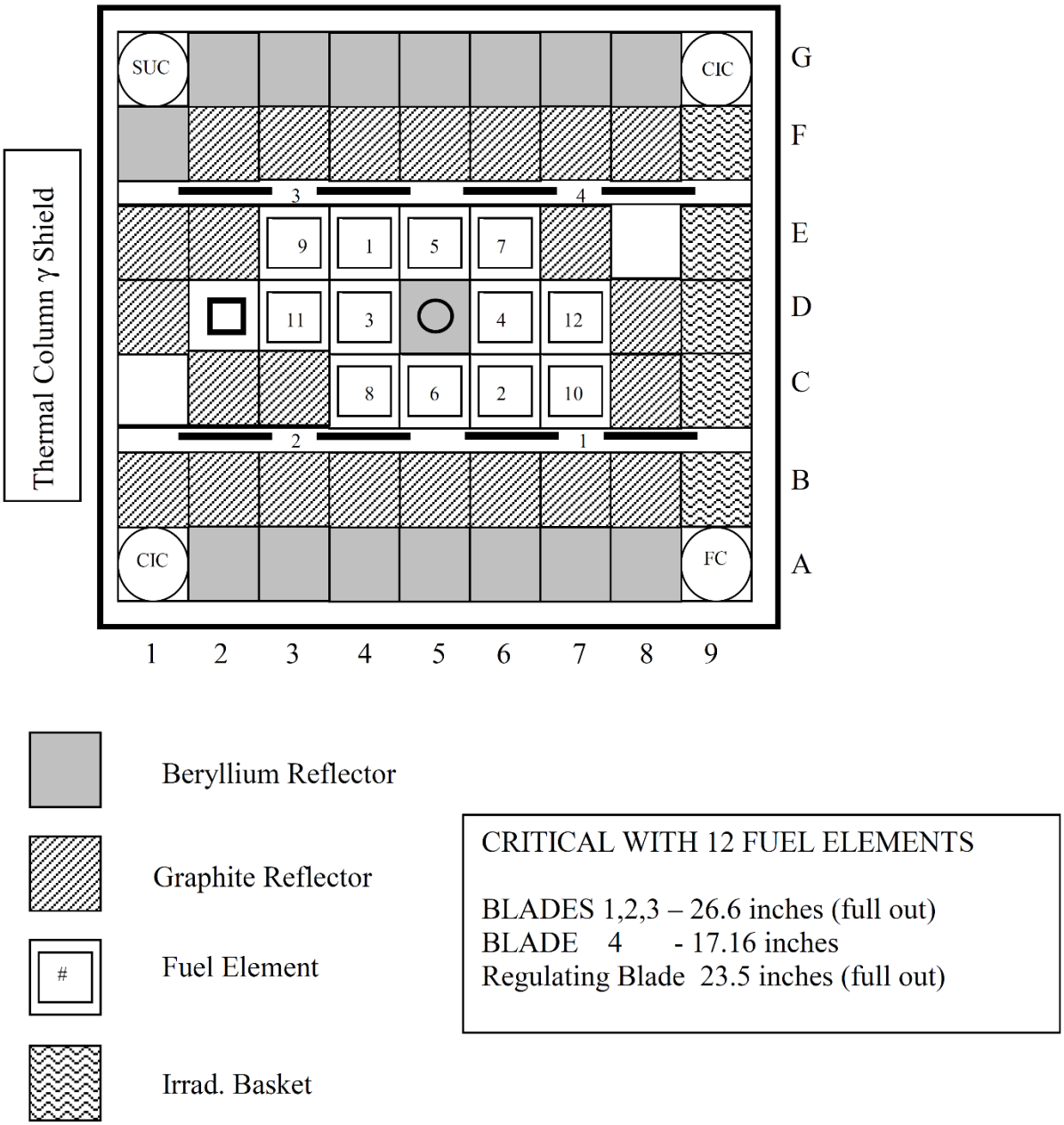

Figure 4-3 LEU First Critical Configuration

This configuration was used for a number of neutronic comparisons that are discussed further below.

Excess Reactivity

Excess reactivity was measured for the LEU startup core at 2,700 percent milli-rho (pcm), as compared to a design calculation of 3,000 pcm (Refs. 25, 26). The traditional method for comparing excess reactivity is a simple difference, which in this case is 300 pcm. The design and measured values for both cores are within the TS limit of 4,700 pcm. The calculated values

were made using the DIF3D code, which was supplied cross-sections from the Electric Power Research Institute (EPRI)-cell code.

The NRC staff reviewed the licensee's documentation of the ECP calculations for the RINSC reactor. The degree of agreement between the MCNP predictions of the ECP and measurements is 300 pcm, which the NRC staff finds acceptable. Because the licensee uses appropriate codes, validates them against measured data, and achieves agreement that is acceptable, the NRC staff concludes that the licensee's calculation methodology for criticality is suitably predictive and generally acceptable.

Control Blade Worth

The LEU first core was then utilized to determine the control rod worths (Refs. 24, 25, 26). In the smaller LEU core, the blades surround the active fuel region and are symmetric with respect to the flux. While the control blade measurements were consistent with the predictions, the regulating rod measured value was significantly less than the predicted value. The traditional means for comparison is to use a simple ratio of calculated/measured worths. The NRC staff notes that the LEU core uses a new stainless steel regulating rod, which has less of a reactivity effect than the previous boral regulating rod. It is also located on the core periphery where gradients, and hence expected deviations, are expected.

Table 4-1 Control Blade Worths

Blade	Total Calculated Reactivity Worth (pcm)	Total Measured Reactivity Worth (pcm)	Ratio
1	-2390	-2270	1.05
2	-2390	-2100	1.14
3	-2390	-2300	1.04
4	-2390	-2160	1.11
Regulating	-410	-269	1.52

The NRC staff reviewed the licensee's methodology and the calculated and measured control blade worth data for the RINSC reactor. On the basis of this information, the NRC staff finds the methodology appropriate and consistent with the methodology used at other RTRs. As the calculated values are acceptably in agreement with the measured values, the NRC staff concludes that the values for blade worth calculations using the licensee's methodology are suitably predictive and generally acceptable.

Fuel Burnup

In the response to RAI 14.55 (Ref. 3), the licensee provides the basis for having no fuel burnup limit based upon statements in NUREG-1313 (Ref. 27). The licensee states that no TS is required for the burn-up limit, because the fuel qualification limit of a 98 percent burn-up is not achievable at the RINSC as long as the current operating and refueling schedule produces an average discharge burn-up of 21 percent. Based on the information above, the NRC staff concludes that no burnup limit is required for RINSC fuel.

Core Configuration

According to SAR Section 4.5, the RINSC reactor is normally operated one shift per day. SAR Table 4-1 describes multiple historical core configurations and their operational characteristics. The core configurations analyzed are compliant with the TS requirement for a symmetrical core of 14 fuel elements in a 7 by 9 position grid, with the four corner positions filled with structural supports as required by TS 5.4. The second configuration utilized in normal operations is the 17-element core, which replaces three graphite reflector elements with fuel. Both arrangements of the core, the 14-element and 17-element, are compact and utilize all available grid positions with either fuel or reflectors (see Figure 4-2).

In response to RAI 14.142 (Ref. 3), the licensee states that they achieved an equilibrium core in October 2008.

The RINSC core can be positioned within the pool. According to supplemental information provided by the licensee (Ref. 5), the reactor pool is separated into three different sections:

- High Power (HP) Section
- Middle Section
- Low Power (LP) Section

When the reactor is located in the HP section, it is coupled with the reactor FC cooling system. When the reactor is placed in the LP section of the pool for operation adjacent to the Dry Irradiation Facility (DIF), the reactor is limited to using NC cooling. The reactor may not be operated in the middle section. The reactor may not be operated with the dam in place in any location, only stored. This ensures that the entire pool volume is available for shielding and cooling during reactor operation at any power level or position within the pool.

The dam is normally stored on the north side of the middle section. The dam can be placed on either the east or west sides of the middle section, facing the LP or HP sections, respectively. The dam may be used to isolate the HP or LP sections for maintenance, repairs, or leak detection or mitigation. Once the dam is in place, the isolated section can be drained.

The dam is made of aluminum plate and is 132 cm (52 in) wide and 9.6 m (32 ft) tall with a 11.4 cm (4.5 in) thick frame having a rubber gasket on one side. Prior to installation, the aluminum cat walk is removed from the pool top. A guide rope is then installed on each side of the bridge and the crane hook is attached to the cable located on the top of the dam. The dam may be positioned so that the side with the gasket is facing the side to be drained. The dam is then lowered onto the damming hooks. Figure 4-4, below is an image of the dam installed in the LP section of the pool.

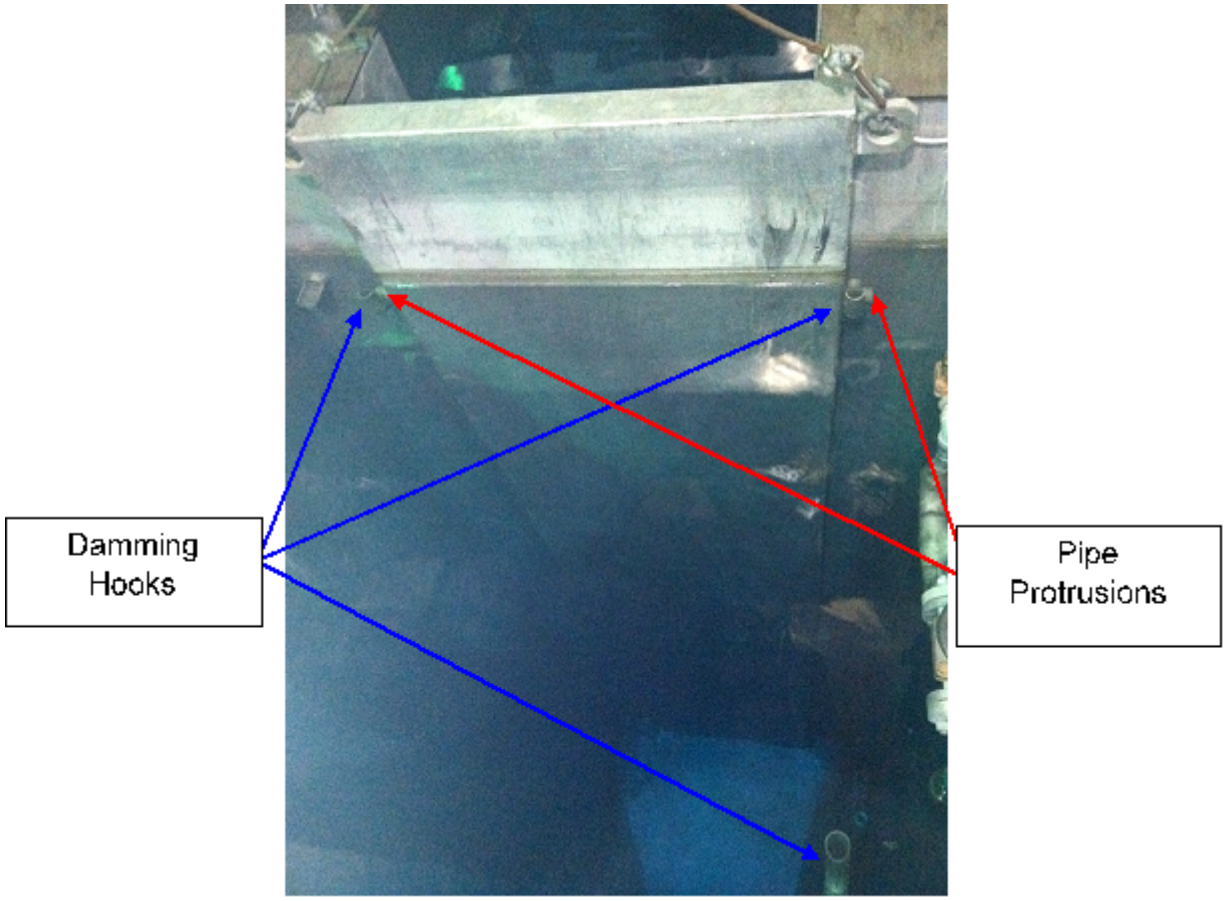

Figure 4-4 Reactor Pool Dam

TS 3.1.2 states:

3.1.2 Core Configuration Limits

3.1.2.1 All core grid positions shall contain fuel elements, baskets, reflector elements, or experimental facilities during reactor operations.

3.1.2.2 The pool dam shall be in its storage location during reactor operations.

TS 3.1.2.1 requires that all grid positions be utilized for operation. The NRC staff finds this specification helps prevent the reduction of coolant flow through the fuel channels resulting from flow bypassing the actively fueled region of the core through unoccupied grid locations. The NRC staff also finds that this specification also helps ensure that core configurations loaded in the RINSC reactor are bounded by the accident analyses in SAR Chapter 13. Based on the information above, the NRC staff concludes that TS 3.1.2.1 is acceptable.

TS 3.1.2.2 requires that the pool gate that is used to separate the sections of the pool be in its storage location when the reactor is in operation. The NRC staff finds that this specification helps ensure that there will be a sufficient heat sink for reactor operations, and the full volume of the pool water will be available in the event of a LOCA. The NRC staff reviewed the LOCA analysis in Section 13.3 and finds that TS 3.1.2.2 is consistent with the initial conditions of the

analysis. Based on the information above, the NRC staff concludes that TS 3.1.2.2 is acceptable.

TS 4.1.2 states:

> 4.1.2 Core Configuration Limit
>
> 4.1.2.1 Prior to the first reactor start-up of the day, inspect the core to confirm that all grid positions contain fuel elements, baskets, reflector elements, or experimental facilities.
>
> 4.1.2.2 Prior to the first reactor start-up of the day, inspect to ensure that the pool dam is in its storage location.

TS 4.1.2.1 requires a surveillance to verify that all grid positions are properly occupied before operating the reactor. The NRC staff finds this specification helps ensure that the core flow is apportioned properly thus maintaining the flow conditions assumed in the T&H analysis. The NRC staff finds that this specification is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007, and the surveillance interval is reasonable. Based on the information above, the NRC staff concludes that TS 4.1.2.1 is acceptable.

TS 4.1.2.2 requires a surveillance to verify that the pool dam is in the storage location before operating the reactor. This specification helps to ensure that the pool volume assumed in the T&H and LOCA analysis is available to provide a heat sink for the fuel as well as maintaining the expected level of shielding. The NRC staff finds that this specification is consistent with the guidance stipulated in NUREG-1537 and ANSI/ANS-15.1-2007, and the surveillance interval is reasonable. Based on the information above, the NRC staff concludes that TS 4.1.2.2 is acceptable.

<u>Conclusions</u>

The licensee has described their typical core configuration that envelopes all planned configurations for this fuel design. The NRC staff concludes that:

- The licensee's assumptions and methods are justified and their demonstrated validity is acceptable. These comparisons of measured and calculated ECPs and blades worth demonstrate acceptable agreement that indicate that the models are suitably predictive of RINSC reactor behavior.

- The analyses include changes resulting from burnup, plutonium buildup, and the accumulation of fission products.

- The criticality analyses establish the ability of the licensee to predict core excess reactivity and control blade worth.

- The analyses address the steady power operation and kinetic behavior of the reactor and show that the dynamic response of the control blades and instrumentation is designed to prevent uncontrolled reactor transients.

- The analyses include consideration of those parameters that ensure the provision of a limiting core analysis. Since this core configuration has the highest power density, the licensee uses it to determine the limiting T&H characteristics for the reactor.

- The analyses and information in this section describe a reactor core system that could be designed, built, and operated without unacceptable risks to the health and safety of the public.

- The licensee justifies the appropriate TSs controlling the core configuration.

4.5.2 Reactor Core Physics Parameters

In responses to RAIs 4.10, 4.12, and 4.13 (Ref. 3), the licensee provided calculated equilibrium core values for reactivity, temperature, void, and power (Doppler) coefficients. Reactivity coefficients were calculated using the VARI3D code within the operating temperature envelope of the reactor fuel and primary coolant. Calculations extend to a fuel temperature of 600 °C (1112 °F) and a coolant temperature of 100 °C (212 °F), which are well beyond anticipated fuel and coolant conditions.

Kinetics Parameters

In response to RAI 4.10 (Ref. 3), the licensee provided a revised analysis of the RINSC kinetic parameters using the VARI3D code, in addition to the prompt neutron lifetime and the 6-group delayed neutron parameters with and without xenon present. These parameters tend to be sensitive to fuel content, burnup, and core leakage - parameters that have remained unchanged for many years - and they are relatively insensitive to the analytical methods employed, as long as the methods have acceptably demonstrated the ability to model the RINSC reactor behavior. The NRC staff finds that such information is commonly presented in this manner, as it is convenient for code input and the values are typical of similar RTRs.

Table 4-2 RINSC Equilibrium Core Kinetics Parameters

	2004 Report	2010 Calculation	
		Equilibrium Xe-135	No Xe-135
Delayed Neutron Fraction, β-eff	0.00764	0.00755	0.00756
Neutron Generation Time, μsec	68.3	69.4	68.6
Delayed Neutron Parameters	group	fraction	fraction
	1	2.6580×10^{-4}	2.6580×10^{-4}
	2	1.3707×10^{-3}	1.3707×10^{-3}
	3	1.3188×10^{-3}	1.3188×10^{-3}
	4	2.8985×10^{-3}	2.8985×10^{-3}
	5	1.1990×10^{-3}	1.1990×10^{-3}
	6	5.0074×10^{-4}	5.0074×10^{-4}
		λ	λ
	1	1.3337×10^{-2}	1.3337×10^{-2}
	2	3.2712×10^{-2}	3.2712×10^{-2}
	3	1.2075×10^{-1}	1.2075×10^{-1}
	4	3.0279×10^{-1}	3.0279×10^{-1}

	5	8.4966×10^{-1}	8.4966×10^{-1}
	6	2.8538	2.8538

Coefficients of Reactivity

In response to RAIs 4.11 and 4.12 (Ref. 3), the licensee provided the prompt neutron lifetime and the 6-group delayed neutron parameters with and without xenon present. These coefficients remain negative for all temperatures and anticipated conditions above the reference temperature of 20 °C (68 °F). In response to RAI 4.13 (Ref. 3), the licensee also provided the Doppler coefficient of reactivity. These coefficients remain negative for a fuel temperature range between 20 and 600 °C (68 and 1112 °F). The NRC staff finds the values are typical of similar RTRs. Based on the information above, the NRC staff concludes that the reactivity coefficients are acceptable.

Table 4-3 Equilibrium Core Reactivity Coefficients

Moderator Temperature Coefficient (tabular)	Temp. °C		Δk/k
	20		0.00000
	30		-0.00116
	40		-0.00230
	50		-0.00345
	60		-0.00459
	70		-0.00572
	80		-0.00684
	90		-0.00796
	100		-0.00908
Moderator Density Coefficient (tabular)	Temp. °C	Density (mg/ml)	Δk/k
	20	0.99811	0.00000
	30	0.99564	-0.00061
	40	0.99227	-0.00137
	50	0.98810	-0.00226
	60	0.98323	-0.00330
	70	0.97773	-0.00448
	80	0.97171	-0.00580
	90	0.96525	-0.00727
	100	0.95845	-0.00887
Fuel Temperature Coefficient (tabular)	Temp. °C		Δk/k
	20		0.00000
	30		-0.00020
	40		-0.00040
	50		-0.00059
	60		-0.00079
	70		-0.00098
	80		-0.00117
	90		-0.00136
	100		-0.00155
	150		-0.00248

	200		-0.00337
	300		-0.00507
	400		-0.00663
	500		-0.00806
	600		-0.00936

Based on its review of the information, the NRC staff concludes that:

- The analyses of the neutron lifetime, effective delayed neutron fraction, and coefficients of reactivity use methods that are appropriate.

- The numerical values for the reactor core physics parameters depend on features of the reactor design that are included in applicable models along with information that is acceptable for use in the analyses of the RINSC reactor operation.

4.5.3 Operating Limits

The regulations in 10 CFR 50.36(c)(1)(i)(A) require TSs to include SLs and limiting safety system settings (LSSSs). SLs are defined as "limits upon important process variables that are found to be necessary to reasonably protect the integrity of the physical barriers that guard against the uncontrolled release of radioactivity."

The regulations in 10 CFR 50.36(c)(1)(ii)(A) define LSSSs, in part, as "settings for automatic protective devices related to those variables having significant safety functions. Where a LSSS is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded."

The principal physical barrier to the release of radionuclides for a MTR-type reactor is the fuel plate cladding, and the most important parameter to maintain the fuel plate cladding integrity is the fuel and cladding temperature. A loss in the integrity of the fuel plate cladding may occur if the temperatures reach the point of allowing cladding blisters to form.

In its analysis, the licensee uses the LCC to demonstrate the acceptability of operating the RINSC reactor within the bounds established by the TSs.

<u>Safety Limit and Limiting Safety System Settings</u>

TS 2.1 states:

> 2.1 Safety Limit
>
> Specification:
>
> The temperature of the reactor fuel cladding shall be less than or equal to 530° C.

TS 2.1 requires that the SL of the RINSC fuel cladding shall be temperature based with the limiting value of 530 °C (986 °F) to ensure that fuel integrity is maintained. The licensee stated, in response to RAI 14.32 (Ref. 3), that the primary design objective of the RINSC fuel is the

maintenance of fuel integrity under any operating and credible abnormal conditions. However, this is inconsistent with the guidance in NUREG-1537, Appendix 14.1, which states that for the LEU uranium-silicide fuel the NRC staff finds 530 °C (986 °F) to be an acceptable value for the cladding and the fuel, not just the cladding. The NRC staff has calculated the expected fuel temperature assuming a calculated cladding temperature of 530 °C (986 °F) and finds that there is an increase of about 3.5 °C (6.3 °F) from cladding temperature to fuel temperature. This slightly higher temperature resulting in the fuel portion of the fuel element is not expected to challenge the fuel element integrity since the melting point of the U_3Si_2 is significantly higher than the Al cladding.

The NRC staff finds that the RINSC SL on the fuel and cladding temperature is supported by research and testing documented in NUREG-1313, "Safety Evaluation Report Related to the Evaluation of Low-Enriched Uranium Silicide-Aluminum Dispersion Fuel for Use in Non-power Reactors" (Ref. 27). According to NUREG-1313, the fuel design utilized by the RINSC reactor retains most mixed fission products and reduces the leakage of halogens and noble gases. Swelling of the fuel in tests conducted to a high burn-up was found to be negligible. Blister resistance for U_3Si_2-Al fuel is such that a fission product release is not expected until fuel cladding temperatures reach above 530 °C (986 °F).

The staff finds that TS 2.1 is otherwise consistent with the guidance of NUREG-1537 and ANSI/ANS-15.1-2007 and meets the requirements of 10 CFR 50.36, "Technical Specifications." Based on the information above, the NRC staff concludes that TS 2.1 is acceptable.

The regulations in 10 CFR 50.36 require TSs to include LSSSs to initiate the automatic safety system to ensure that SLs are not exceeded. The licensee proposes TS 2.2.1 and TS 2.2.2 to ensure the SL of TS 2.1 is not exceeded. These specifications are the subject of RAIs 4.22, 14.32, and 14.36 (Ref. 3) and the supplemental information (Ref. 5).

TS 2.2.1 states:

> 2.2.1 Limiting Safety System Settings for Natural Convection Mode Operation
>
> 2.2.1.1 The limiting safety system setting for reactor thermal power shall be 115 kW.
>
> 2.2.1.2 The limiting safety system setting for the height of coolant above the top of the uranium silicide fuel shall be 23 feet 7 inches.
>
> 2.2.1.3 The limiting safety system setting for the bulk pool temperature shall be 127° F.

TS 2.2.1.1 requires a maximum thermal power level when in natural-convection cooling mode.

TS 2.2.1.2 requires a minimum height of coolant above the fuel when in natural-convection cooling mode.

TS 2.2.1.3 requires a maximum bulk pool water temperature when in natural-convection cooling mode.

The analysis for operating when in natural-convection cooling mode supporting these specifications is provided as a supplement to the response to RAI 4.28 (Ref 26). Multiple

conservative assumptions were made when developing the input conditions for the analysis. The simultaneous application of these conservative assumptions provides an additional margin of safety in the analysis. The resulting analysis shows that under these conditions, peak cooling channel power would have to reach 1.7812 kWt in order for the onset of nucleate boiling (ONB) to occur, which corresponds to a fuel cladding temperature that is below the 530 °C (986 °F) SL value at which damage to the fuel cladding could occur. The hottest cooling channel reaches a peak power of 1.7812 kWt when core power is 369 kWt; this is far above the allowed power of 100 kWt. The peak fuel temperature reached is 78.9 °C (26.0 °F), which is less than the SL. The NRC staff finds that this analysis demonstrates that the TSs 2.2.1.1, 2.2.1.2, and 2.2.1.3 are conservative LSSS set points for the RINSC reactor when operated in NC cooling mode. Further analysis is provided in SER Section 4.6. Based on the information above, TSs 2.2.1.1, 2.2.1.2 and 2.2.1.3 are acceptable.

TS 2.2.2 states:

> 2.2.2 Limiting Safety System Settings for Forced Convection Mode of Operation
>
> 2.2.2.1 The limiting safety system setting for reactor thermal power shall be 2.3 MW.
>
> 2.2.2.2 The limiting safety system setting for the height of coolant above the top of the uranium silicide fuel shall be 23 feet 7 inches.
>
> 2.2.2.3 The limiting safety system setting for the primary coolant inlet temperature shall be 122° F.
>
> 2.2.2.4 The limiting safety system setting for the primary coolant flow rate shall be 1560 gpm.

TS 2.2.2.1 requires a maximum limit on thermal power level when in forced cooling mode.

TS 2.2.2.2 requires a minimum coolant height above the top-of-fuel-meat when in forced cooling mode.

TS 2.2.2.3 requires a maximum limit on the bulk pool temperature when in in forced cooling mode.

TS 2.2.2.4 requires a minimum primary coolant flow rate when in forced cooling mode.

The analysis for the operation under FC cooling supporting these specifications is provided in the response to RAI 4.28 (Refs. 3, 26). This analysis uses power of 2.4 MWt, coolant height of 7.18 m (23 ft 6.5 in), bulk pool temperature of 51.7 °C (125 °F), and coolant flow of 1,580 gpm as the conditions for steady state operation; and power of 2.2 MWt, coolant height of 23 ft 9.1 in (7.24 m), bulk pool temperature of 123 °F (50.6 °C), and coolant flow of 5,981 lpm (1,740 gpm) as the operating conditions for an over power transient. Each of these conditions will individually make the analysis more conservative than actual operating conditions; their simultaneous use makes the analysis even more conservative. The peak fuel temperature reached is 87.9 °C (190.2 °F), which is far less than the SL. The NRC staff finds that this analysis conservatively demonstrates that TSs 2.2.2.1, 2.2.2.2, 2.2.2.3, and 2.2.2.4 are conservative LSSS setpoints for the RINSC reactor. Based on the information above, the NRC staff concludes that TSs 2.2.2.1, 2.2.2.2, 2.2.2.3, and 2.2.2.4 are acceptable.

The NRC staff reviewed TS 2.2.1 and TS 2.2.2. The NRC staff finds that the RINSC LSSSs provide a safety margin between the operational limit and the SL to allow for measurement and analytical uncertainties, as well as anticipated operational transients. The licensee provided further details supporting the selection of LSSS values in the responses to RAI 4.20 and 14.36 (Ref. 3). The staff finds that the LSSS values provide reasonable assurance that the SL (TS 2.1) will not be exceeded. The staff also finds that TS 2.2.1 and TS 2.2.2 are consistent with the guidance of NUREG-1537 and ANSI/ANS-15.1-2007 and meet the requirements of 10 CFR 50.36. Based on the information above, the NRC staff concludes that TS 2.2.1 and TS 2.2.2 are acceptable.

Based on these findings, the NRC staff concludes that the continued operation, as limited by TS 2.1, TS 2.2.1, and TS 2.2.2, offers reasonable assurance that the fabricated fuel can meet the design objective of maintaining fuel integrity; and it will thereby function safely in the reactor without adversely affecting the public health and safety.

<u>Excess Reactivity and Shutdown Margin</u>

TS 3.1.1.1 states:

> 3.1.1.1 Core
>
> 3.1.1.1.1 The core shutdown margin shall be at least 1.0 %Δk/k.
>
> 3.1.1.1.2 The core excess reactivity shall not exceed 4.7 %Δk/k.
>
> 3.1.1.1.3 The reactor shall be subcritical by at least 3.0 %Δk/k during fuel loading changes.

TS 3.1.1.1.1 requires the minimum shutdown margin (SDM) under any operational circumstances. The specified SDM helps to ensure that the reactor will be suitably subcritical subsequent to a scram from any operating condition. The definition of SDM states that the reactor will remain subcritical after cool down, xenon decay, and experiment removal, even if the most reactive scrammable shim safety blade fails in the most reactive position. No credit is taken for the negative reactivity worth of the regulating rod because it is not scrammable. An example of the application of this specification is demonstrated in Table 4-4 below. On the basis of this information and this demonstration, the NRC staff concludes that TS 3.1.1.1.1 is acceptable.

TS 3.1.1.1.2 requires a value on the upper limit for allowed excess reactivity. This means that the excess reactivity evaluation shall include the effect of all experiments that have a positive worth to the core upon insertion. The NRC staff specifically disallows including the contribution of negative worth experiments in this evaluation since upon removal they have the effect of increasing core reactivity. This specification helps to ensure that the SDM requirement can be met under all circumstances. The definition of excess reactivity states it is determined when the core is in the reference core condition. These conditions clarify the acceptable core temperature and xenon conditions. An example of the application of this specification is demonstrated in Table 4-4 below. On the basis of this information and this demonstration, the NRC staff concludes that TS 3.1.1.1.2 is acceptable.

TS 3.1.1.1.3 requires that core reactivity be subcritical by 3.0 %Δk/k during fuel loading activities. This specification helps to ensure that changes to the core configuration are

conducted in a suitably conservative manner and that reactivity during such changes is substantially subcritical, even as fuel assemblies are inserted, removed, or relocated. Under such conditions, inadvertent criticality accidents are prevented. Based on the information above, the NRC staff concludes TS 3.1.1.1.3 is acceptable.

TS 3.1.1.2 states:

3.1.1.2 Control Rods

3.1.1.2.1 The reactivity worth of the regulating rod shall not exceed 0.6 %Δk/k.

TS 3.1.1.2 requires a limit for the regulating rod worth. This specification helps to ensure that the reactivity value for the regulating rod, which is not scrammable and can be used with the automatic servo system, is less than the delayed neutron fraction of the reactor core. This helps ensure that the reactor could not become prompt critical if the automatic servo system were to fail. Based on the information above, the NRC staff concludes that TS 3.1.1.2 is acceptable.

TS 4.1.1.2 states:

4.1.1.2 Control Rod Reactivity Limit

4.1.1.2.1 The reactivity worth of the shim safety blades and the regulating rod shall be determined:

4.1.1.2.1.1 Annually

4.1.1.2.1.2 Whenever the core reflection is changed

4.1.1.2.1.3 Whenever the core fuel loading is changed

4.1.1.2.1.4 Whenever maintenance is performed that could have an effect on the reactivity worth of the control rod

TS 4.1.1.2 requires a surveillance to verify the reactivity worth of shim safety blades and the regulating rod. This specification helps to ensure that the control rods worth used for the evaluation of SDM and other reactivity dependent measurements are appropriate to changing core conditions. This specification is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.1.1.2 is acceptable.

Table 4-4 Excess Reactivity-SDM Evaluation

Condition	Source	Δk/k
Maximum allowed excess reactivity (ρ_{excess})	TS 3.1.1.1.2	+0.04700
Worth of neglecting the regulating rod using measured worth from Table 4-1 (ρ_{reg})	TS 3.1.1.2	+0.00600
Worth of safety blades except the highest worth blade using measured worths from Table 4-1 ($\rho_{blade-1}$)	(Ref. 24)	-0.06530
Net reactivity ($\rho_{excess} + \rho_{reg} + \rho_{blade-1}$)		-0.01230
ρ_{SDM}	TS 3.1.1.1	-0.01000

After a scram, the net core maximum reactivity must be at least 1.0 percent Δk/k subcritical (k_{eff} = 0.99) with the most reactive shim safety blade and regulating rod withdrawn; consideration of excess reactivity must include the worth of experiments having positive net value to the core when inserted. Using the blade worths from Table 4-1 indicates that the SDM at representative conditions is -1.230 percent Δk/k or -1230 pcm, and is within the TS 3.1.1.1 limit.

The NRC staff verified that the 1.0 percent Δk/k SDM is sufficient to keep the reactor subcritical after shutdown with the most reactive blade and the regulating rod withdrawn. The NRC staff compared these limits to those of other RTRs and finds the licensee's SDM and limit on excess reactivity are consistent with the corresponding limits used at other non-power plate-type reactors. Based on the information above, the NRC staff concludes that TS 3.1.1.1.1 and TS 3.1.1.1.2 are acceptable.

TS 4.1.1.1 states:

4.1.1.1 Core Reactivity Limit

4.1.1.1.1 The core shutdown margin shall be determined:

4.1.1.1.1.1 Annually

4.1.1.1.1.2 Whenever the core reflection is changed

4.1.1.1.1.3 Whenever the core fuel loading is changed

4.1.1.1.1.4 Following control blade changes.

4.1.1.1.2 The core excess reactivity shall be determined:

4.1.1.1.2.1 Annually

4.1.1.1.2.2 Whenever the core reflection is changed

4.1.1.1.2.3 Whenever the core fuel loading is changed

4.1.1.1.2.4 Following control blade changes.

4.1.1.1.3 The core shutdown reactivity shall be determined to remain greater than 3 %ΔK/K prior to and during fuel loading changes.

TS 4.1.1.1.1 requires that the core SDM shall be determined annually and whenever there is a change in core loading, core reflection or control blade changes. Measurements made whenever the core loading or reflection is changed provide the assurance that core reactivity limits are not being exceeded as a result of changes in the core configuration. Determining

SDM after a change in control blades will help ensure that the SDM is determined using the changed blades. This specification helps to ensure that the SDM is maintained. SDM is important because it demonstrates the ability to make the reactor subcritical by the amount defined, even if a scrammable safety blade fails to insert. The regulating rod of the RINSC is not scrammable. SDM is determined by considering all combinations of the three safety blades by inserting and selecting the combination that provides the minimum negative reactivity. That shim safety blade worth is then added to the excess reactivity. The resulting value must be negative (reactor subcritical), and the magnitude must be more negative than the SDM requirement. The surveillance interval is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Therefore, based on the information above, the NRC staff concludes that TS 4.1.1.1.1 is acceptable.

TS 4.1.1.1.2 requires that the core excess reactivity shall be determined annually and whenever there is a change in core loading, core reflection or control blade changes. Measurements made whenever the core loading or reflection is changed provide the assurance that core reactivity limits are not being exceeded as a result of changes in the core configuration. Determining excess reactivity after a change in control blades will help ensure that the excess reactivity is determined using the changed blades. This specification helps to ensure that changes in excess reactivity are monitored and controlled. Excess reactivity is a core parameter that is important to determining the SDM and is also adjusted in accordance when experiments are inserted. It is also used as an input parameter to some elements of the safety analysis. Monitoring this parameter also serves the purpose of detecting core reactivity anomalies such as misaligned blades, disconnected blades, fuel misloading, and fuel failures. It also serves to detect model inaccuracies. The surveillance interval is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.1.1.1.2 is acceptable.

TS 4.1.1.1.3 requires that the core shutdown reactivity shall be determined to be more negative that the cited value prior to and during fuel loading changes. This specification helps to ensure that changes in core reactivity are monitored and controlled during refueling operations. Based on the information above, the NRC staff concludes that TS 4.1.1.1.3 is acceptable.

Conclusions

The NRC staff has reviewed key parameters of the RINSC operating limits including the SL, LSSSs, excess reactivity, and the SDM in this subsection of the SER, and as provided in the SAR and the RAI responses referenced above. The NRC staff concludes the following:

- The licensee has derived the SL and LSSSs from the T&H analysis. The values proposed for the limits are consistent with the analyses performed. The T&H analyses demonstrated that no overheating of the fuel would occur during any operation or credible event and fuel integrity will be maintained. Further analysis of T&H characteristics are provided in SER Section 4.6.

- The licensee has discussed and justified all excess reactivity factors needed to ensure a complete and operable reactor core. The licensee has also considered the design features of the control systems to ensure that this amount of excess reactivity is fully controlled under normal operating conditions.

- The definition of the SDM is negative reactivity obtainable by control rods to ensure a reactor shutdown from any reactor condition. With the assumption that the most reactive

shim safety blade is inadvertently stuck in its fully withdrawn position, and the non-scrammable regulating rod is in the position of maximum reactivity addition, the analysis derives the minimum negative reactivity necessary to ensure a safe reactor shutdown. The licensee conservatively proposes a SDM of 0.01 $\Delta k/k$ in the TSs. This value is readily measurable and is thus acceptable.

Based on the information described above, the NRC staff concludes that the nuclear design is adequate for the continued safe operation of the RINSC reactor.

4.6 Thermal-Hydraulic Design

SAR Section 4.6 states that the RINSC reactor has a steady-state operating power of 2 MWt and open-pool water cooling with the ability to have forced flow directed into the core region. A separate secondary cooling loop removes heat from the pool through heat exchangers and transfers it into the atmosphere by means of a cooling tower. The reactor has two different operating modes: LP with natural convective cooling below 100 kWt and 2 MWt with FC cooling. SER Chapter 13 evaluates and finds acceptable the anticipated transient conditions.

Peaking Factors

Peaking factors used in T&H analysis are derived from the neutronic analysis. In response to RAI 4.28 (Refs. 3, 25, 26), the licensee presented two new thermal hydraulic analyses using PLTEMP/ANL. These analyses were provided by ANL and they provide comprehensive results that characterize the behavior of the RINSC reactor system under conditions of natural circulation and forced flow.

The TS that are currently in force at the RINSC identify the SLs in terms of reactor power, coolant flow rate through the core, coolant outlet temperature, and height of water above the core. The first draft of the proposed TSs for the license renewal identified the same SLs. In support of the numerical values for these parameters ANL performed an analysis for the RINSC reactor for natural and forced (convection (Refs. 25 and 26). Subsequent to the application for the license renewal, a decision was made to redefine the SL in terms of the cladding temperature, since that is the limit related to the primary fission product barrier for safe operation of the reactor. The parameters related to reactor power, coolant flow through the core, coolant outlet temperature, and height of water above the top of the core are now identified more appropriately as the LSSSs for the reactor.

The ANL staff performed their analysis to determine the limiting reactor power in MWt for the onset of nucleate boiling (ONB), as evidenced in the manner in which ANL reported their results. When the decision was made to change the SL to peak cladding temperature, the analysis was not re-performed to focus on the cladding temperature, as adequate margin to the peak cladding temperature was demonstrated when calculating the ONB. As a result, it should be understood that the limiting entry into ONB will likewise limit the cladding temperature to a value less than the SL.

Models Employed

The PLTEMP/ANL model for forced flow is shown in Figure 4-5. In this down-flow model, a common source pressure exists at the top of the assemblies and a common sink pressure exists in the outlet plenum. Differences in the velocity of the coolant between the source and sink regions are assumed to be small and are ignored. The thermal column gamma shield

hydraulics have the same inlet and outlet pressures as is assumed for the fuel assemblies. The inlet to the gamma shield is connected to the core inlet pipe. The outlet to the gamma shield is connected to the core outlet pipe. However, the flow through the gamma shield is not part of the flow rate cited in the results.

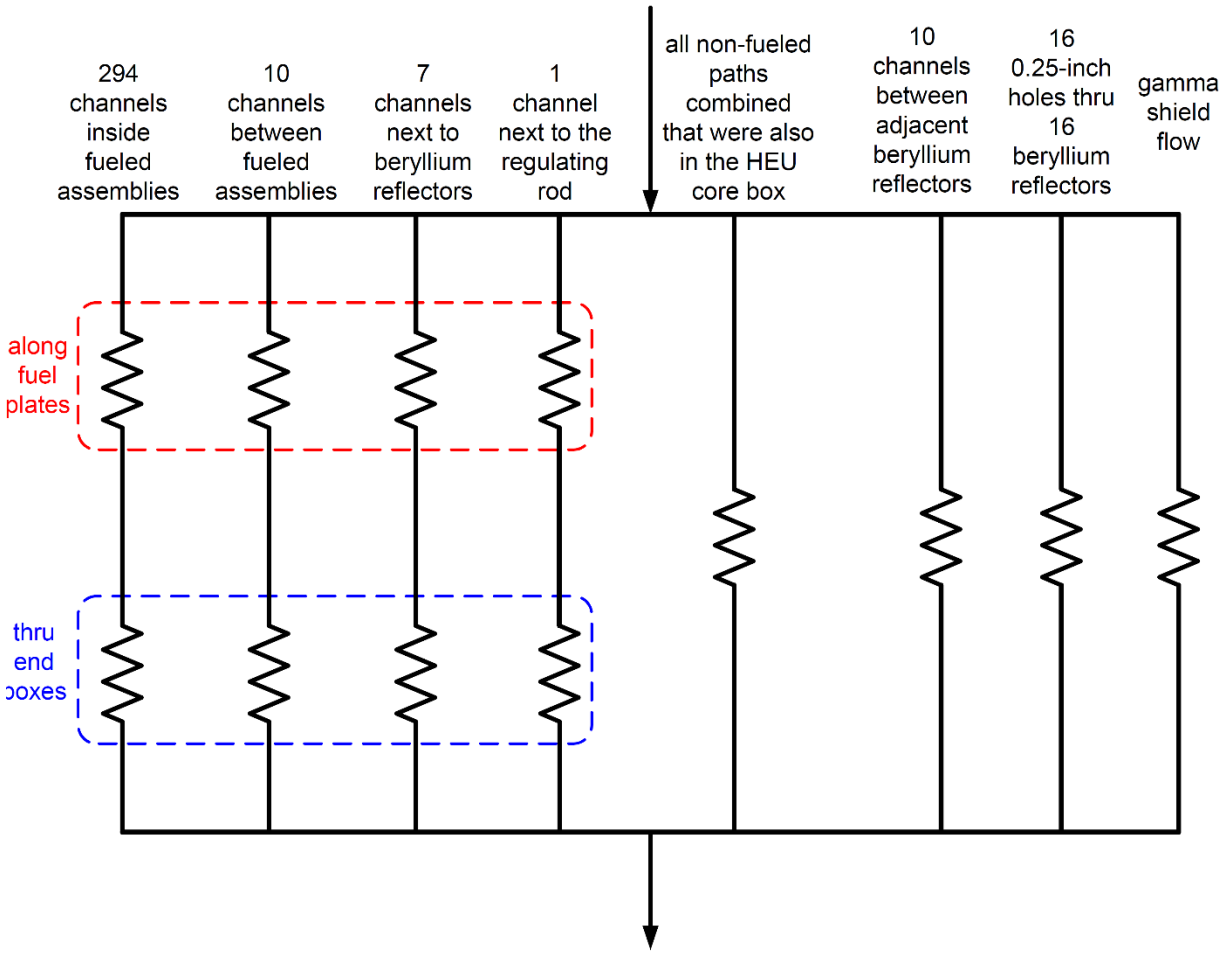

Figure 4-5 PLTEMP/ANL Forced Flow Model

For natural circulation, a simpler model is used since the only mechanism moving flow upward through the core is buoyancy. This buoyancy is due to the higher coolant temperatures in the core relative to that of the open pool. There is an end box at the top and bottom of each assembly which could serve as a chimney, but is not included in the model for conservatism.

PLTEMP/ANL determines the flow rate by balancing the buoyancy-induced pressure rise in the core with the friction and K-loss pressure drops through the core combined with the sum of the K-loss pressure drops through the ductwork from the coolant header gate to the core inlet. The NRC staff finds that the models employed in the T&H analyses are typical of plate reactor analyses and they use a code that has been demonstrated previously to be acceptable for such use.

Peaking Factors

Both of the ANL documents (Refs. 25 and 26) provide a detailed analysis of the peaking factors used. The individual factors vary slightly because of the flow conditions that are applicable to the two flow regimes. The peaking factors include allowances for manufacturing variations, power density, flow distribution, measurements, and coefficients utilized. The NRC staff reviewed the material submitted regarding the factors used and on the basis of the information supplied finds that they are justified and appropriate. Additionally, based on its review of the means for combining the uncertainties, the NRC staff also finds that the method is acceptable.

Forced Flow Analysis Results

There are 14 fuel assemblies in the analyzed core. Each has 22 fuel plates, for a total of 308 plates. The highest power fuel plate is the one immediately adjacent to the beryllium reflector in assembly D6. PLTEMP/ANL was used to determine the flow rate in the limiting channel as a function of flow rate through the reactor flow meter. The limiting channel for the analysis is one that is between two fuel plates of the same fuel assembly. The PLTEMP/ANL analysis shows that ONB occurs when the power in the plate between the two half channels is at 22.80 kWt. The analysis of the power distribution in the equilibrium core shows that when the reactor is operating at 2.0 MWt, the power in the limiting plate is 9.653 kWt. Thus, 22.80 kWt corresponds to reactor operation at 4.72 MWt. Figure 4-6 identifies the power and flow conditions in relation to ONB. The LSSS proposed for the RINSC are well within the power/flow region that has significant margin to ONB, which is well below the departure from nucleate boiling conditions. The NRC staff has also reviewed the supplied analysis which includes a calculation of the estimated cladding temperature. Using geometry and properties consistent with the fuel design, the NRC staff estimates that the increase in fuel temperature above the cladding temperature is approximately 3.5 °C (6.3 °F). The NRC staff finds that the forced flow analysis is at temperature and coolant elevations that are conservative with respect to the actual LSSS setpoints, and thus are acceptable assumptions.

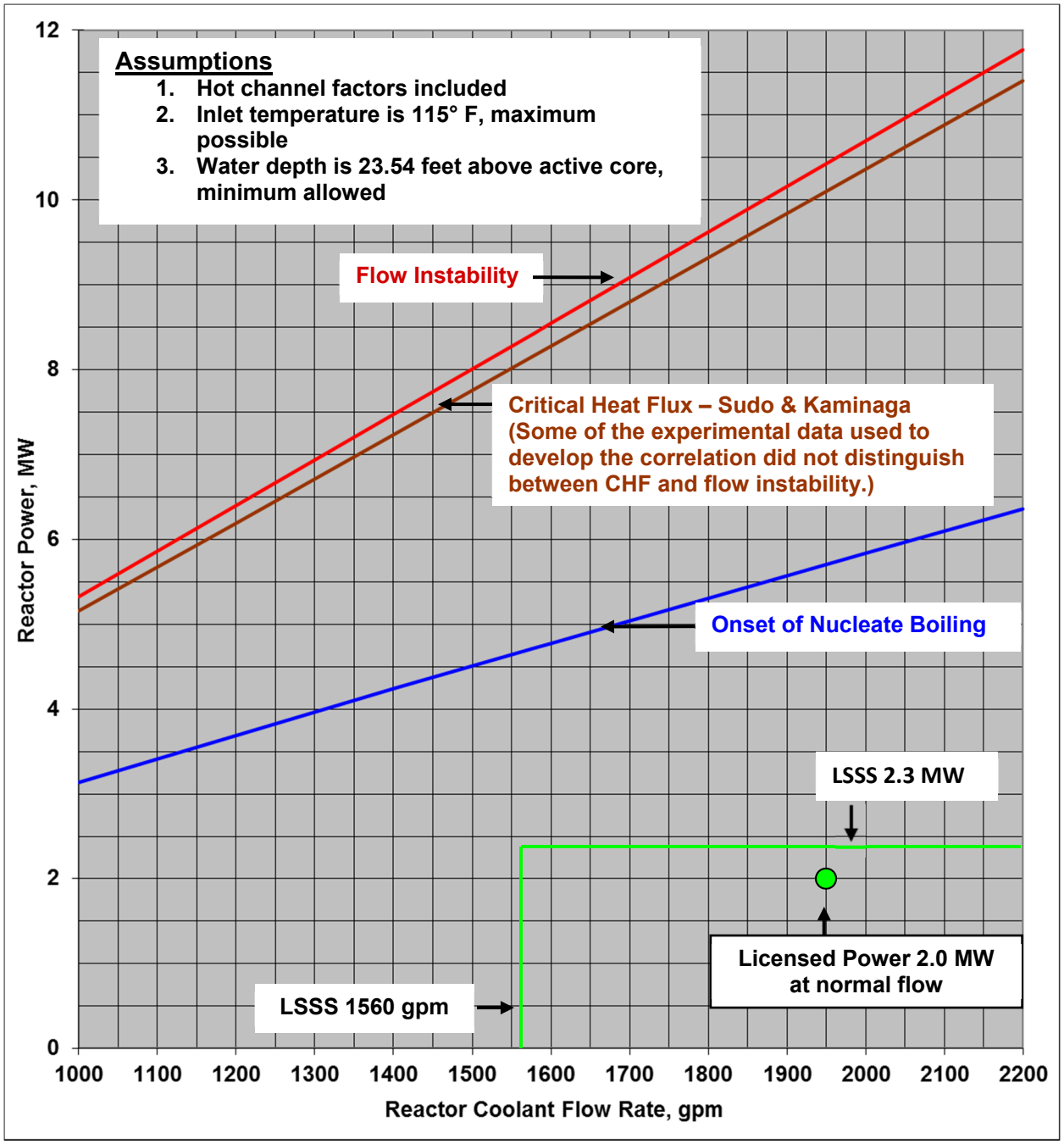

Figure 4-6 Forced Flow ONB Results

Natural Circulation Analysis Results

For the analysis of the RINSC reactor under conditions of natural circulations, the inlet temperature of 130 °C (266 °F) and water depth of 7.17 m (23.54 ft) above the active core were used. The hydraulic resistance along the inlet flow path from the coolant header gate to the lower plenum was considered in the analysis and is represented by a K-loss value of 7 at the inlet to the limiting coolant channel. The PLTEMP/ANL analysis shows that ONB occurs at a

power of 369 kWt with all uncertainties included, and the maximum fuel temperature attained is 78.9 °C (175.8 °F). The NRC staff finds that the natural circulation flow analysis is at coolant elevations that are conservative with respect to the actual LSSS setpoints, and it is an acceptable assumption.

Conclusions

The licensee's results demonstrate that the natural circulation steady-state maximum power allowed by TS 2.2.1, "115 kW," is at least 50 percent below the power that would result in the ONB (369 kWt). The licensee demonstrated that the sensitivity of the results to various input assumptions on pressure drop and power density does not drastically alter the results. The licensee states that as long as the ONB is prevented from occurring within the fueled channel, fuel temperatures will remain below the SL.

The licensee also presented results from calculations demonstrating that the forced flow steady-state maximum power allowed is at least 50 percent below what would result in the ONB (4.72 MWt). The licensee demonstrated that the sensitivity of the results to various input assumptions on pressure drop and power density does not drastically alter the results. The licensee also demonstrated that flow instability is not reached until 8.17 MWt. The licensee stated that as long as the ONB and flow instabilities are prevented from occurring within the fueled channel, fuel temperatures will remain below the SL.

The NRC staff has reviewed the licensee's T&H data and analyses and finds the RINSC reactor T&H characteristics are acceptable and sufficient to ensure that fuel integrity will be maintained under all analyzed conditions. Limits provided by the TSs provide reasonable assurance that the critical heat flux (CHF) will not be exceeded, thereby maintaining fuel plate temperatures within the SL. The NRC staff concludes that the thermal-hydraulic design, as limited by the TSs, is adequate to demonstrate that it establishes conditions that are appropriate to allow the continued safe operation of the RINSC reactor.

4.7 Conclusions

Based on the above findings and conclusions, the NRC staff concludes that the licensee has adequately described the bases and functions of the reactor design to demonstrate that the reactor can be safely operated and shut down from any operating condition or accident assumed in the safety analysis. The systems provide an adequate control of reactivity, the containment of coolant, barriers to the release of radioactive material, and sufficient radiation shielding for the protection of facility personnel. Nuclear and thermal-hydraulic design and operating limits, as established by the TSs, will adequately provide for the protection of fuel integrity. For this reason, no cladding breach will occur when the reactor is operated in accordance with the TSs. The NRC staff concludes that continued operation of the RINSC within the limits of the TSs and facility license will not result in undue risk to the health and safety of facility personnel, the public or the environment.

5. REACTOR COOLANT SYSTEMS

5.1 Summary Description

SAR Chapter 5 describes the reactor coolant system, which consists of the primary and secondary coolant systems. The primary system includes the reactor pool, N-16 hold up tank, two heat exchangers, and associated pumps and piping. The primary system contains demineralized water that performs the function of radiation shielding and removing the heat from the primary system to the secondary coolant system via the heat exchangers. According to SAR Section 5.1, only one cooling loop is needed at any time to provide enough heat removal for 2 MWt operation. The SCS includes two cooling towers adjacent to the reactor building to dissipate the heat from the SCS into the environment.

The analysis evaluated and found acceptable in SER Section 4.6 demonstrates that in the NC mode, when reactor power is limited to 100 kWt, the secondary coolant system is not required.

5.2 Primary Coolant System

SAR Section 1.8 and 5.2 describe the PCS. The PCS removes the fission and decay heat from the fuel during reactor operation in NC mode at 100 kWt, in FC mode at full power (2 MWt), and decay heat during reactor shutdown, while maintaining the pool water within an acceptable temperature range. The PCS is a closed loop system that consists of the pool, the coolant flow channels in the core, the N-16 delay tank, two primary system cooling loops, two primary cooling pumps, and two heat exchangers. The original construction provided for 2 cooling loops, but only had 1 such loop fully installed. The installation of the components for cooling loop 2 were completed in 1996. PCS piping location is installed so that if there was a break in the piping, the siphoning action would not drain the pool lower than 12 ft (3.66 m) above the core.

The reactor pool is evaluated and found acceptable in SER Section 4.3.

The NRC staff conducted a site visit in November 2016. During the walk down of the PCS in the facility, the NRC staff observed the changes as described in SAR Section 1.8, “Facility Modernizations and History,” and compared first hand observations with SAR Figure 5.4 (reproduced as SER Figure 5-1 below), and historical documents. The NRC staff observed that the outlet from the pool to the N-16 delay tank is a common pipe. Coolant loop 1 was fully configured in the original construction. No heat exchanger, connecting pipes, or pump were provided for cooling loop 2 when the facility was originally constructed. Piping for coolant loop 2 exits the holdup tank and penetrates the shielding wall that encloses the delay tank room. Originally, it terminated with a shutoff valve and a flange plate. The corresponding coolant loop 2 connection to the return line union also penetrated the shield wall and terminated with a shutoff valve and flange plate.

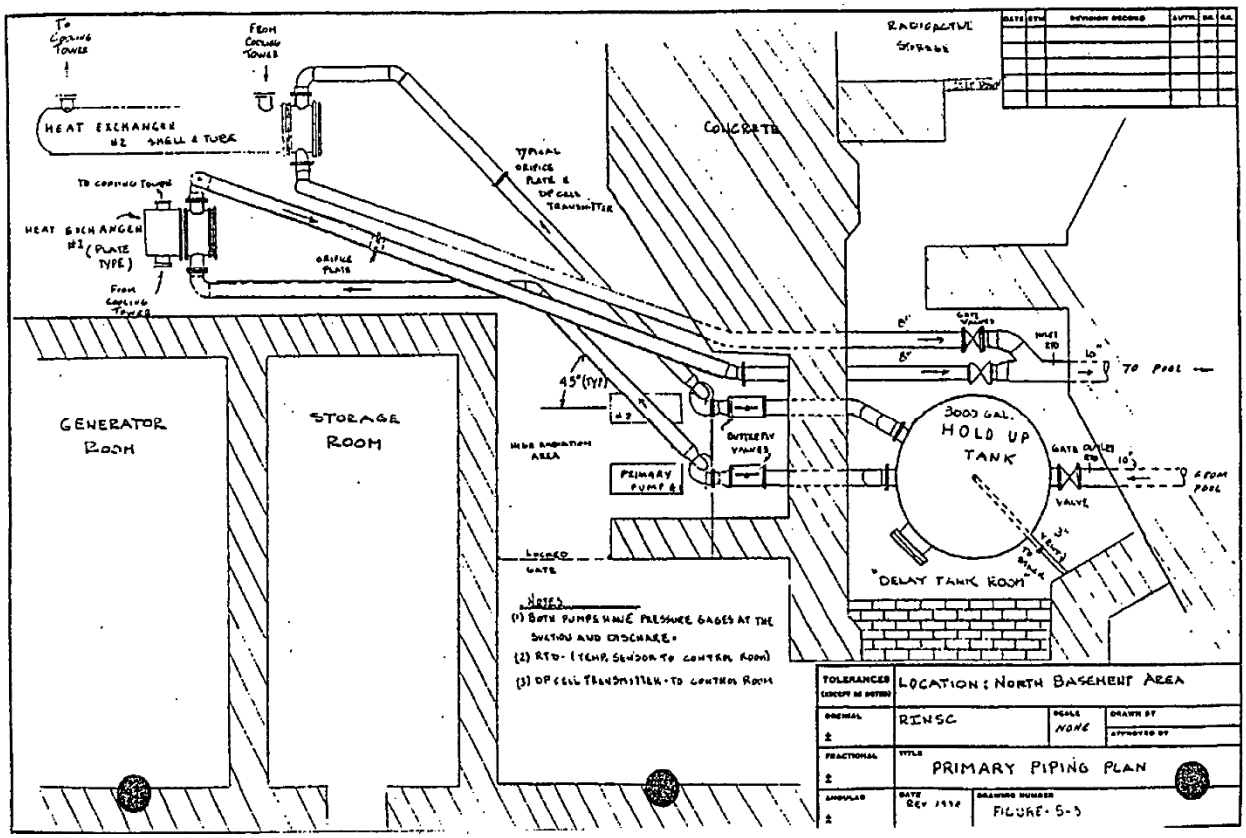

Figure 5-1 Primary Coolant System (from SAR Figure 5.4)

The LSSS for the height of coolant above the top of the fuel, inlet temperature, and coolant flow in FC mode are evaluated and found acceptable in SER Section 4.5.2.

A float switch system continuously monitors the pool level. This system is tied into the facility alarm system, which is monitored by an offsite alarm company. In the event that the pool level drops to within 1 in (2.54 cm) below the LSSS, the automatic pool fill is started. If the pool level drops to the LSSS, a scram occurs, the operator receives an alarm, and the alarm company notifies the RINSC staff member that is on call.

TS 4.3.1.3 states:

> 4.3.1.3 Primary Coolant Level Inspection
>
> The primary coolant level shall be verified to be greater than or equal to the Limiting Safety System Setting value prior to the initial start-up each day that the reactor is started up from the shutdown condition.

TS 4.3.1.3 specifies the requirement to periodically verify that the PCS level is at an acceptable level above the reactor core. This specification helps to ensure that the primary coolant level is inspected and acceptable prior to the first reactor start-up of each day. The NRC staff reviewed TS 4.3.1.3 and finds that a daily verification of the pool level prior to starting up the reactor provides adequate assurance that the automatic pool fill system is working to maintain the pool

level. The surveillance interval is consistent with guidance in NUREG-1537, Section 4.3. Based on the information above, the NRC staff concludes that TS 4.3.1.3 is acceptable.

The NRC staff reviewed the RINSC coolant systems TSs and concludes the following:

- The PCS is designed in accordance with the design bases and supports the T&H and accident analysis in the SAR.

- Design features of the PCS and components give reasonable assurance of fuel integrity under all possible reactor conditions. The system is designed to remove sufficient fission heat from the fuel under all possible reactor conditions without exceeding the established LSSSs that are included in the TSs.

- The PCS is designed to convert into a passive or fail-safe method to a NC flow that is sufficient to avoid a loss of fuel integrity (see SER Section 13.4).

- The size and shape of the pool will provide: (1) sufficient radiation shielding to maintain personnel exposures below the limits in 10 CFR Part 20 (see SER Sections 4.3 and 4.4), and (2) a heat reservoir sufficient for anticipated reactor operations.

- Designs and locations of PCS components have been specifically selected to avoid coolant loss that could lead to fuel failure, and uncontrolled release of excessive radioactivity (see SER Section 13.3).

- The TS surveillance provides reasonable assurance of necessary PCS operability for reactor operations as analyzed in the SAR.

5.3 Secondary Coolant System

SAR Section 5.3 describes the SCS, which consists of two separate pumps, loops, heat exchangers, and cooling towers. Either loop is capable of transferring 2 MWt heat from the PCS to its respective cooling tower under the most limiting anticipated metrological conditions. The system minimizes the potential for leakage of pool water into the environment, and radioactivity monitoring allows sufficient time for corrective action to mitigate any leakage.

TS 3.3.2 states:

> 3.3.2 Secondary Coolant System
>
> Sodium-24 activity in the secondary coolant shall be maintained at levels that are indistinguishable from background.

TS 3.3.2 requires a limit on the Sodium-24 radioisotope in the SCS. Sodium-24 is produced by the activation of the aluminum structural materials in the primary pool, and a small concentration of it is present in the primary coolant during and immediately following the operation of the reactor. The NRC staff reviewed this specification and finds that if the Sodium-24 isotope is found in the secondary coolant, it may indicate a primary to secondary system leak and heat exchanger failure. This specification helps to ensure that a primary-to-secondary leakage will

be detected. Based on the information above, the NRC staff concludes that TS 3.3.2 is acceptable.

TS 4.3.2.1 states:

> 4.3.2.1 Secondary Coolant Activity
>
> Sodium-24 activity in the secondary coolant shall be measured monthly.

TS 4.3.2.1 requires a surveillance to verify that SCS coolant activity is indistinguishable from background. The NRC staff finds that this specification helps to ensure the detection of Sodium-24 activity in the SCS, and identify heat exchanger failure. The surveillance interval is consistent with the schedule recommended by NUREG-1537, Section 4.3. Based on the information above, the NRC staff concludes that TS 4.3.2.1 is acceptable.

Regarding the SCS, the NRC staff concludes that:

- Design features of the SCS and components will allow the transfer of the reactor heat from the PCS under all allowed reactor and meteorological conditions.

- The TSs provide reasonable assurance of necessary SCS operability for normal reactor operations.

5.4 Primary Coolant Cleanup System

SAR Sections 2.4.6 and 5.4 describe the primary coolant cleanup system. The primary coolant cleanup system maintains the water purity in the PCS to reduce the potential for corrosion to reactor components. It circulates water from the clean-up pump through a mixed bed demineralizer, and back into the pool through the make-up/clean-up return line.

The demineralizer resin must be replaced periodically after it is spent, which is indicated by an increase in water conductivity. Bulk pool temperature is limited by procedure to less than 140 °F (60 °C) to avoid damaging the resin. In addition to the demineralizer, there is a surface filtration unit in the pool that skims debris from the surface of the water to prevent contamination of the pool from any dust or debris. The cleanup pumps, skimmer, filter, resin tanks, valves, and piping, are located within the RINSC reactor and equipment rooms. Ion-exchange resins and any contaminated water leakage from this equipment will be wiped up and disposed of as dry solid, low-level radioactive waste.

TS 3.3.1.1 states:

> 3.3.1.1 Primary Coolant Conductivity
>
> The reactor shall not be operated unless primary coolant conductivity is ≤ 2 μmhos / centimeter.

TS 3.3.1.1 requires an upper limit on PCS water conductivity. The required primary coolant resistivity shall be maintained at a value less than or equal to 2 μmhos/cm. The NRC staff reviewed this specification and finds that operation within these limits helps to ensure control of the corrosion of the aluminum components in the PCS and the fuel element cladding. A requirement to maintain pH in an acceptable band is not necessary because conductivity is kept

less than 5 micromhos per cm. Controlling these limits also minimizes the activation of primary coolant water impurities. The NRC staff reviewed this specification and finds that it is consistent with the guidance of ANSI/ANS-15.1-2007 and the safety evaluation on RTR pool water electrolytic conductivity (Ref. 53). Based on the information above, the NRC staff concludes that TS 3.3.1.1 is acceptable.

TS 4.3.1.1 states:

> 4.3.1.1 Primary Coolant Conductivity
>
> The conductivity of the primary coolant shall be tested monthly.

TS 4.3.1.1 requires a surveillance to verify that PCS conductivity is within the band established in TS 3.3.1.1. The NRC staff reviewed this specification and finds that it helps to ensure that the conductivity of the PCS is within acceptable limits and monitors pool water quality and resistivity changes that could accelerate the corrosion of the primary system components. The NRC staff finds this specification is consistent with guidance in ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.3.1.1 is acceptable.

TS 3.3.1.2 states:

> 3.3.1.2 Primary Coolant Activity
>
> The reactor shall not be operated unless Cesium - 137 and Iodine - 131 activity in the primary coolant is indistinguishable from background. An exception can be made if the reactor operation is solely for the purpose of identifying which fuel assembly is damaged.

TS 3.3.1.2 requires fission product activity detection in the primary coolant. These isotopes are prominent fission products. Using this methodology, if either of these isotopes are detected in the primary coolant, it may indicate fission products escaping from the fuel cladding. The NRC staff reviewed this specification and finds that it is consistent with the guidance of ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 3.3.1.2 is acceptable.

TS 4.3.1.2 states:

> 4.3.1.2 Primary Coolant Activity
>
> Cesium-137 and Iodine-131 activity in the primary coolant shall be measured monthly.

TS 4.3.1.2 requires a surveillance to verify radioactivity in the PCS. The NRC staff reviewed this specification and finds it helps to ensure that the cesium - 137 and the iodine - 131 activity in the primary coolant are detected, which are indicators of fuel failure. The surveillance interval is consistent with the schedule recommended by ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.3.1.2 is acceptable.

The NRC staff reviewed the primary coolant cleanup system and TSs 3.3.1.1, 4.3.1.1, 3.3.1.2, and 4.3.1.2 and concludes that:

- The design helps ensure that corrosion of and oxide buildup on fuel cladding and other reactor components in the PCS are minimized.

- Conductivity of the primary coolant is acceptably controlled. This also allows pH of the primary coolant to be acceptably controlled.

- The primary coolant cleanup system and its components have been designed so that any malfunction or leaks would be confined to the reactor and equipment rooms.

5.5 Water Coolant Makeup System

5.5.1 Primary Coolant Makeup Water System

SAR Section 5.5.1 describes the primary coolant makeup water system, which is used to replace water in the PCS. City water is processed through two independent systems so that one can be used until it exceeds the desired purity, and then the switch is made to the other system while a replacement is made. A check valve (backflow preventer) is installed to prevent flow back to the city water system. The water in each system goes through a five-micron filter, an activated charcoal filter, two mixed-bed demineralizers, a one-micron filter, and a conductivity indicator before merging into the makeup/cleanup return line.

The pool level is controlled by manual operation or a solenoid valve that senses if the water level drops 1 in (2.54 cm) below the normal pool level. After the water level raises the float to the full level, the valve closes to shut off the water supply. The response to RAI 5.1 (Ref. 3), describes the operation of the secondary overflow solenoid switch that terminates the make-up water addition, if the water level exceeded the setpoint. According to additional supplemental information provided by the licensee (Ref. 56), the standard makeup rate for this system is 5 gpm.

The NRC staff reviewed the design and operation of the primary coolant makeup water system against the guidance in NUREG-1537, Section 5.5. The NRC staff finds that the system design is consistent with the guidance as the system prevents backflow, contains purification equipment, and has sufficient capacity to compensate for minor leaks and evaporation in order to maintain an acceptable pool level.

The NRC staff concludes that:

- The design bases, functional descriptions, and procedures for the primary coolant makeup water system give reasonable assurance that the quantity and quality of water required will be provided.

- The system design or procedures will prevent overfilling of the PCS or a malfunction of the makeup water system and will prevent the loss or release of contaminated primary coolant.

- The system design or procedures will prevent contaminated primary coolant from entering the potable water system through the makeup water system.

5.5.2 Secondary Coolant Makeup Water System

SAR Section 5.5.2 describes the secondary coolant makeup water system. The secondary coolant water makeup system is supplied by city water. The system is activated by a resistive level sensor in each cooling tower basin. Normal discharge of the water is to the storm drain. A 3,000-gal holding tank is available for re-use of the water. The NRC staff compared this system to comparable systems at other RTRs and finds them to be adequate to maintain an acceptable water level. Based on the information in the SAR, the NRC staff concludes that the design bases, functional descriptions, and procedures for the secondary coolant makeup water system give reasonable assurance that the quantity and quality of required water will be provided.

5.6 Nitrogen-16 Control System

SAR Section 5.6 describes the nitrogen-16 control system. Cooling water from the pool's primary outlet pipe flows to a 3,000-gal (11,356.24 L) delay tank, which holds the coolant water for a sufficient time for the nitrogen-16 radioactivity in the primary water to decay. The mean residence time of the nitrogen-16 in the tank is about 90 seconds, which allows time for most of the nitrogen to decay before exiting the tank. The inlet and outlets to the tank have a baffle plate that reduces the mixing of the incoming water with the water that is next to the exit of the tank. The licensee measured dose rates for the system to be 5 to 6 rem/hr (50 to 60 mSv/hr) at the delay tank, 1 to 2 rem/hr (10 to 20 mSv/hr) at the heat exchanger, and less than 1 rem/hr (10 mSv/hr) at the secondary pumps. Since elevated radiation levels are expected, the area is controlled and posted as a high radiation area. Continuous air monitoring devices and alarms described in Table 3.2 of the TS are used to alert facility personnel to radiation hazards in the area. The NRC staff compared this system to comparable systems at other RTRs and finds it to be adequate to control nitrogen-16.

The NRC staff reviewed the design and operation of the nitrogen-16 control system and finds it consistent with the guidance in NUREG-1537, Section 5.6. The NRC staff concludes that design bases and design features give reasonable assurance that the nitrogen-16 control system can function, as proposed, and can reduce potential doses to personnel, so that doses do not exceed the requirements of 10 CFR Part 20, and are consistent with the facility ALARA program.

5.7 Auxiliary Systems Using Primary Coolant

SAR Section 5.7, as supplemented (Ref. 5), describes the auxiliary water supply system (AWSS). The AWSS provides an independent source of water for suppling water to the pool. In the SAR supplement (Ref. 5), the licensee states that this system is not covered under the TSs. AWSS water is supplied from the fire sprinkler system supply, through a series of manual valves, up to the top of the pool. Since the water from this system does not go through a clean-up system, it is for emergency use only, and can only be activated manually. The flow is about 60 gpm.

The NRC staff conducted a site visit in November 2016. During the walkdown of the facility, the NRC staff viewed the components of the AWSS. Use of this system requires that a RINSC staff member obtain a key from the control room and use a 12-ft (3.66 m) ladder to unlock and open two valves. Opening of the valves supplies city water to the pool.

The NRC staff reviewed SAR Section 5.7 and supplemental information, and finds that this alternate source of water is not necessary to be credited as performing a safety function, to protect the integrity of the fuel.

5.8 Conclusions

Based on the information above, the NRC staff concludes that the design of the RINSC cooling systems, as described in the SAR, are adequate for the removal of heat generated during continuous full power reactor operation, and for the removal of decay heat after a shutdown from an extended full-power operation. The systems contain sufficient features to protect personnel from excessive radiation hazards, minimize corrosion of system components and fuel, prevent loss of coolant, and provide one of the barriers to prevent a fission product release into the environment. The NRC staff concludes the following:

- The licensee described and analyzed the RINSC coolant systems, has derived the design bases from other chapters of the SAR, and provided acceptable methods to remove sufficient heat to ensure the integrity of the components.

- TSs, including testing and SRs, provide reasonable assurance of necessary cooling system operability for all modes of operation.

Based on the information above, the NRC staff concludes that the RINSC coolant systems are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007, and sufficient for continued safe reactor operation during the renewal period.

6. ENGINEERED SAFETY FEATURES

6.1 Summary Description

Chapter 6 of the RINSC SAR (Ref. 2) describes the ESFs that are capable of mitigating the consequences of an accident and with helping maintain any potential radiological dose below the limits allowed in 10 CFR Part 20, "Standards for Protection Against Radiation." The confinement system, including the confinement ventilation system (CVS) and the emergency evacuation system (EES) work in unison when manually initiated to minimize the consequences of any radiological release in the confinement. The CVS and EES can be powered from the emergency electrical power system and as such, will be available in the event of a loss of electrical power. These systems do not actuate automatically, nor are they required to actuate to mitigate any accidents or abnormal operating conditions. The design of these safety features are based on the assumed radiological release that is postulated to result from the MHA. In the event of an accident, the confinement and emergency power systems are designed to minimize any radiological release through the maintenance of negative pressure in the reactor building relative to the outside atmosphere, and filtering exhaust air prior to discharge through the elevated stack.

6.2 Detailed Descriptions

6.2.1 Confinement System

The confinement is described in TS 5.1.2 and SAR Section 6.2.1, as supplemented in RAI 6.1 (Ref. 3).

The CVS establishes and maintains the required negative pressure in the reactor building, as required and monitored by TS 3.2 "Reactor Control and Safety System, Table 3.1.3, line no. 7, which is evaluated and found acceptable in SER Section 7.4, "Reactor Protection System." There are twelve confinement penetrations, five of which are associated with the ventilation system. The remaining penetrations are either sealed or normally closed. Personnel access into the confinement volume is through the portal entrance from the laboratory wing of the facility.

When there is indication of a release of radioactive material into the reactor building atmosphere, the confinement is used to control the release of radioactive material to the outside environment. Air monitors are located in the ventilation system, as evaluated and found acceptable in SER Section 7.7, and can monitor the radiation dose rates in the confinement building air.

This CVS creates a flow of air into and out of the reactor building with a negative differential pressure between the building and the outside atmosphere. By throttling the dampers on the dilution, normal, and emergency blower subsystems, a negative differential pressure is maintained dynamically with air flow into the reactor building. The CVS changes modes from normal to emergency when a building evacuation alarm is manually initiated. In supplemental information provided by the licensee (Ref. 5), the graphic below is provided to illustrate the major aspects of this system.

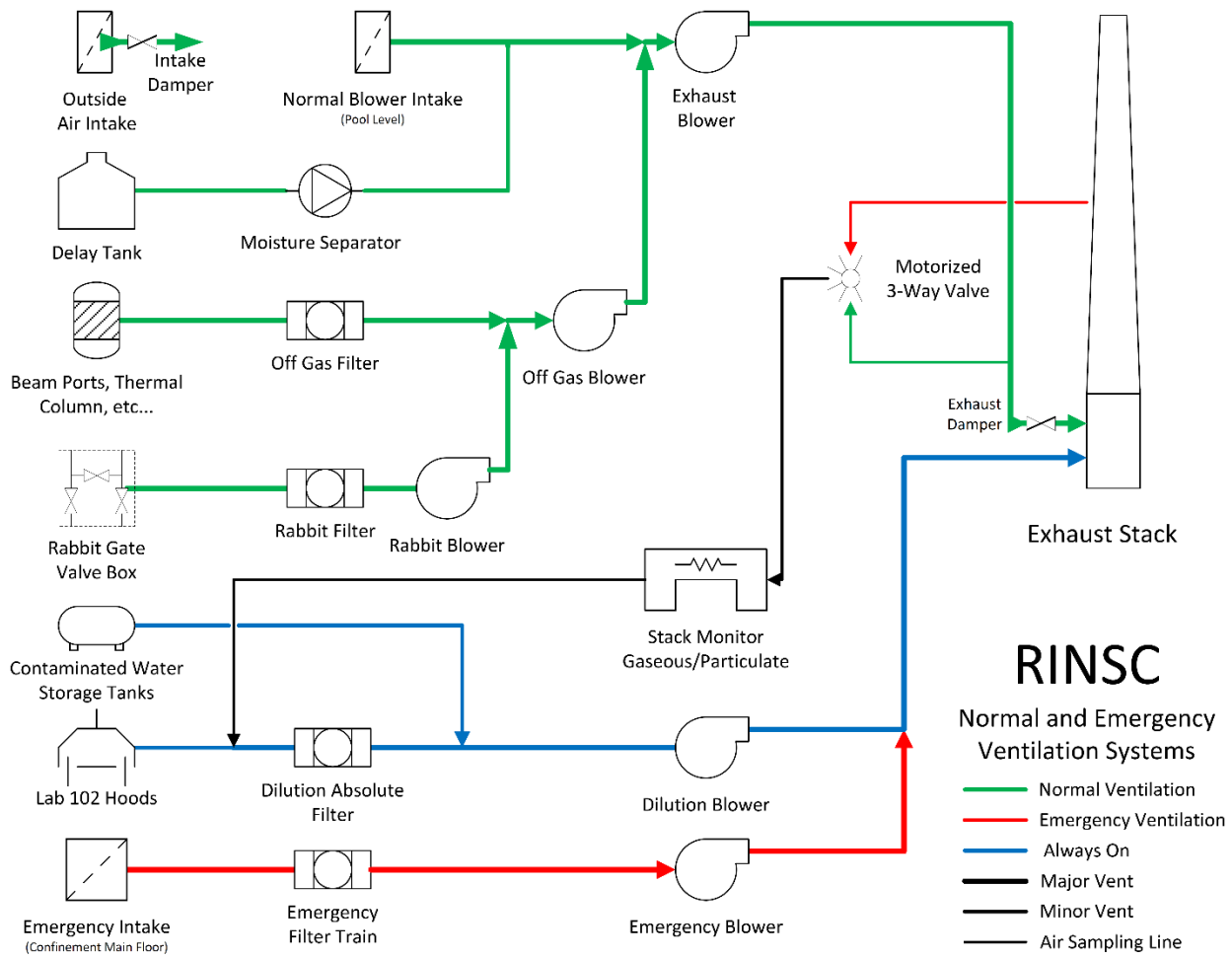

Figure 6-1 Heating, Ventilation and Air Conditioning System

As described in SAR Section 9.1.1 and depicted by the blue path in Figure 6-1, air from the dilution blower portion of the CVS is drawn from the contaminated water storage tanks, laboratory hoods, and the output of the stack monitor and is exhausted into the stack. The dilution air blower is required to operate whenever the CVS is in operation regardless of the mode (normal or emergency), since it dilutes the radioactive exhausts from other sources and it also prevents the backflow of radioactive exhaust through the dilution portion of the CVS.

As depicted by the green path on Figure 6-1, exhausted air during normal operation of the CVS is from the normal intake at pool level, the delay tank, the off gas system (beam ports, thermal column, etc.) and the rabbit system. The normal exhaust line includes a sample connection to the stack monitor which, after sampling, is returned to the stack via the dilution blower portion of the CVS. The bulk of the exhaust air during normal operation empties into the stack where it is diluted with the exhaust from the dilution blower. The stack is 35 m (115 ft) in height and is of steel construction. As operation is changed from normal to emergency mode, as depicted by the red path on Figure 6-1, the normal flow rate drops causing pneumatically operated switches to trigger the normal exhaust and intake dampers to close, as discussed in the response to RAI 6.1 (Ref. 3).

As described in SAR Sections 9.1.1 and 6.2.2, the function of the emergency mode of CVS operation is to assure that, in the event of an accident which could involve the release of radioactive material, the reactor building air is exhausted through a system of filters, is sampled allowing quantification of the amount of material released, and is released out of the elevated stack with the maximum opportunity for dilution. The emergency mode may only be entered by manually depressing one of the five evacuation alarm buttons. When this happens, the actions are to: (1) isolate all normal ventilation blowers in the reactor building, (2) close the dampers on the normal intake and exhaust lines, and (3) activate the emergency blowers.

There are no dampers on the emergency exhaust system. The exhaust from the emergency intake passes through the emergency filter system. The emergency filter system directs air from the reactor building through a roughing filter, an absolute particulate filter, a charcoal filter for removing radioiodine, and an absolute filter for removing charcoal dust that may become contaminated with radioiodine. Each absolute filter cartridge is individually tested and certified by the manufacturer to have an efficiency of not less than 99.97 percent when tested with 0.3-micron diameter dioctylphthalate smoke. The minimum removal efficiency of the charcoal filters for iodine is 99 percent based on Oak Ridge National Laboratory data and measurements performed locally. After exhausting the air from the emergency blower into the stack, a tap from the stack sends a sample to the stack monitor. This sample includes the effect of dilution from the dilution blower. The emergency electrical power system is capable of providing power to the exhaust system, in the event of a loss-of-offsite-power.

As shown above in Figure 6-1, the 3-way valve changes functions between normal and emergency modes. In the normal mode, it allows normal exhaust samples to be diverted from the exhaust line to the stack monitor before the exhaust enters the stack. In the emergency mode, it allows exhaust from the stack to be diverted to the stack monitor. In either mode, the sample is returned to the stack via the dilution blower.

The following TSs apply to the operation and maintenance of the confinement system:

The confinement at the RINSC is described in TS 5.1 and TS 5.5.

TS 5.1, “Site and Facility Specifications,” Specification 5.1.2 states:

> 5.1.2 The facility consists of a Confinement Building (also referred to as the reactor building), including the basement area and an office wing and lab building. The Confinement Building and Confinement Building basement serve as the restricted area.

TS 5.1.2 establishes the design features for the confinement building considered important to safety for the design and safety analysis parameters.

TS 5.5 states:

> 5.5 Confinement (Reactor) Building
>
> 5.5.1 The nominal free volume of the Confinement Building (volume of the building minus volume of the pool structure, including the water in the pool) shall be 181,955 cu ft.

TS 5.5.1 requires the RINSC confinement building to have a minimum nominal free volume. This specification helps to ensure that the assumptions used in effluent calculations using the CVS are suitably conservative. Based on the information above, the NRC staff concludes that TS 5.5 is acceptable.

TS 3.4 states:

> 3.4 Confinement System
>
> Whenever the following operations are in progress:
>
> - The reactor is operating.
> - Irradiated fuel handling is in progress.
> - Experiment handling is in progress for an experiment that has a significant fission product, or gaseous effluent activation product inventory.
> - Any work on the core or control rods that could cause a reactivity change of more than 0.60 %Δk/k is in progress.
> - Any experiment movement that could cause a reactivity change of more than 0.60 %Δk/k is in progress.
>
> 3.4.1 The confinement system shall be operable.

TS 3.4 and TS 3.4.1 establish the requirement that the confinement system be operable whenever the stated operations are in progress. These specifications help to ensure that the CVS is operating during operations that could result in the release of radionuclides. The NRC staff finds that these specifications are consistent with the guidance of NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 3.4 and TS 3.4.1 are acceptable.

TS 4.4 states:

> 4.4 Confinement System
>
> 4.4.1 It shall be verified each day that the Confinement System is operable and working in conjunction with the Confinement Ventilation System, ref TS 4.5, maintaining a minimum of -0.5”WC differential pressure across the Confinement System boundary prior to any of the following conditions:
>
> 4.4.1.1 Reactor operations.
>
> 4.4.1.2 Handling of irradiated fuel.
>
> 4.4.1.3 Experiment handling for an experiment that has a significant fission product, or gaseous effluent activation product inventory.

4.4.1.4 Performing any work on the core or control rods that could cause a reactivity change of more than 0.60 %Δk/k is in progress.

4.4.1.5 Performing any experiment movement that could cause a reactivity change of more than 0.60 %Δk/k is in progress.

4.4.2 It shall be verified that the Confinement System remains operable during an initiation of a facility evacuation.

4.4.2.1 Monthly

4.4.2.2 Following any maintenance that could affect the operability of the system

4.4.3 It shall be verified that the Confinement System remains operable during an initiation of a facility evacuation alarm concurrent with a loss of normal AC power to the facility.

4.4.3.1 Quarterly

4.4.3.2 Following any maintenance that could affect the operability of the system.

TS 4.4.1 establishes the surveillance performance criteria for the CVS minimum differential pressure for system operability. The ability of the system to maintain minimum differential pressure indicates acceptable system operation. The NRC staff finds that this specification, and the stated conditions, are generally consistent with the guidance of NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.4.1 is acceptable.

TS 4.4.2 requires a surveillance to verify that the CVS retains the ability to maintain -0.5 water column inches of differential pressure upon initiation of the manual mode change from normal to emergency CVS operation. This specification helps to ensure that during normal and emergency conditions that there is net in-leakage of the reactor building. This helps ensure that after initiation of the emergency CVS operating mode, all vented air and effluent is mitigated through filtration and dilution and measured to determine the released inventory of radionuclides. The NRC staff finds that this specification, and the stated conditions, are consistent with the guidance of NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.4.2 is acceptable.

TS 4.4.3 requires a surveillance to verify that the CVS retains the ability to maintain -0.5 water column inches of differential pressure upon initiation of the manual mode change from normal to emergency CVS operation using emergency power. This specification helps to ensure that during normal and emergency conditions, there is net in-leakage of the reactor building. This helps ensure that after initiation of the emergency CVS operating mode all vented air and effluent is mitigated through filtration and dilution and measured to determine the released inventory of radionuclides, even if the building's normal electrical system is unavailable. The NRC staff finds that this specification, and the stated conditions, are consistent with the guidance of NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.4.3 is acceptable.

TS 5.7 states:

5.7 Confinement Ventilation System

5.7.1 The confinement building ventilation system emergency filtration train absolute filters shall be certified by the manufacturer to have an efficiency of not less than 99.97% when tested with 0.3 micron diameter dioctylphthalate smoke.

5.7.2 The containment exhaust stack terminates at a minimum height equal to or greater than the confinement building.

TS 5.7.1 requires the confinement building ventilation system emergency filtration train absolute filters minimum performance. This specification helps to ensure that the assumptions used in effluent calculations using the CVS are suitably conservative. Based on the information above, the NRC staff concludes that TS 5.7.1 is acceptable.

TS 5.7.2 requires a minimum height of exhaust stack terminus. This specification helps to ensure that the assumptions used in effluent calculations using the CVS are suitably conservative. Based on the information above, the NRC staff concludes that TS 5.7.2 is acceptable.

TS 3.5 states:

3.5 Confinement Ventilation System

Whenever the following operations are in progress:

- The reactor is operating.
- Irradiated fuel handling is in progress.
- Experiment handling is in progress for an experiment that has a significant fission product, or gaseous effluent activation product inventory.
- Any work on the core or control rods that could cause a reactivity change of more than 0.60 %Δk/k is in progress.
- Any experiment movement that could cause a reactivity change of more than 0.60 %Δk/k is in progress.

3.5.1 The Confinement Ventilation System shall be operable and maintaining a minimum differential pressure of -0.5” WC across the Confinement System boundary.

TS 3.5 and TS 3.5.1 require that the ventilation system be operable for the operations cited and that it provides a differential pressure across the CVS boundary of -0.5 water column inches equivalent when the reactor is operating. These specifications help to ensure that during normal and emergency conditions that there is net in-leakage of air into the reactor building. After initiation of emergency CVS operating mode, all vented air and effluent is mitigated

through filtration and dilution and measured to determine the released inventory of radionuclides. The NRC staff finds that these specifications are consistent with assumptions in the safety analysis evaluated and found acceptable in SER Sections 13.1 and 13.6, and with the guidance in ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 3.5 and 3.5.1 are acceptable.

TS 4.5 states:

4.5 Confinement Ventilation System

4.5.1 It shall be verified each day that the Confinement Ventilation System is operable and working in conjunction with the Confinement System, ref TS 4.4, maintaining a minimum of -0.5"WC differential pressure across the Confinement System boundary prior to any of the following conditions:

4.5.1.1 Reactor operations.

4.5.1.2 Handling of irradiated fuel.

4.5.1.3 Experiment handling for an experiment that has a significant fission product, or gaseous effluent activation product inventory.

4.5.1.4 Performing any work on the core or control rods that could cause a reactivity change of more than 0.60 %Δk/k is in progress.

4.5.1.5 Performing any experiment movement that could cause a reactivity change of more than 0.60 %Δk/k is in progress.

4.5.2 It shall be verified that the Confinement Ventilation System Emergency Mode activates and maintains greater than a differential pressure of -0.5" WC during an initiation of a facility evacuation alarm.

4.5.2.1 Monthly

4.5.2.2 Following any maintenance that could affect the operability of the system

4.5.3 It shall be verified that the Confinement Ventilation System Emergency Mode activates and maintains greater than a differential pressure of -0.5" WC during an initiation of a facility evacuation alarm concurrent with a loss of normal AC power to the facility.

4.5.3.1 Quarterly

4.5.3.2 Following any maintenance that could affect the operability of the system.

4.5.4 The Emergency Filter Bank shall be verified to be at least 99% efficient for removing iodine:

4.5.4.1 Biennially

4.5.4.2 Following any maintenance that could affect the operability of the system.

4.5.5 The ventilation flow through the Emergency Filter Bank shall be verified to be less than or equal to 1500 SCFM:

4.5.5.1 Biennially

4.5.5.2 Following any maintenance that could affect the operability of the system.

TS 4.5.1 requires a surveillance to verify the CVS minimum differential pressure for system operability. Meeting this specification helps to ensure that during normal and emergency conditions that there is in-leakage to the reactor building. After initiation of the emergency CVS operating mode, all vented air and effluent is mitigated through filtration and dilution and is measured enabling the quantification of the released inventory of radionuclides. The NRC staff finds that this specification, and the stated conditions, are generally consistent with the guidance of NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.5.1 is acceptable.

TS 4.5.2 requires a surveillance to verify the CVS retains the ability to maintain a minimum -0.5 water column inches of differential pressure upon transition from the normal to emergency mode of CVS operation. This specification helps to ensure that during normal and emergency conditions, there is in-leakage to the reactor building thus ensuring that after initiation of the emergency CVS operating mode all vented air and effluent is mitigated through filtration and dilution and is measured enabling the quantification of the released inventory of radionuclides. The NRC staff finds that this specification, and the stated conditions, are consistent with the guidance of NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.5.2 is acceptable.

TS 4.5.3 requires a surveillance to verify the CVS retains the ability to maintain -0.5 water column inches of differential pressure upon initiation of the manual mode change from normal to emergency CVS operation using emergency power. This specification helps to ensure that during normal and emergency conditions that there is in-leakage to the reactor building, thus ensuring that after initiation of the emergency CVS operating mode, all vented air and effluent is mitigated through filtration and dilution and is measured enabling the quantification of the released inventory of radionuclides even if the building normal electrical system is unavailable. The NRC staff finds that this specification, and the stated conditions, are consistent with the guidance of NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.5.3 is acceptable.

TS 4.5.4 requires a surveillance to verify that the emergency filter for iodine is capable of having the cited efficiency to sorb iodine. This specification helps to ensure that the iodine filter will satisfy accident analysis requirements. The NRC staff finds that this specification is consistent with assumptions in the safety analysis evaluated and found acceptable in SER Sections 13.1 and 13.6, and the surveillance frequency is consistent with the guidance in ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.5.4 is acceptable.

TS 4.5.5 requires a surveillance to verify the emergency blower ventilation rate limit is satisfied. This specification helps to ensure that the iodine filters can perform at the expected efficiency by

controlling air contact time with the charcoal. The NRC staff notes that the TS has no minimum flow rate. However, if the flow rate were to decrease to a low level, the system would not be able to maintain the TS required negative pressure between the reactor building and the outside environment. The NRC staff finds that this specification is consistent with assumptions in the safety analysis evaluated and found acceptable in SER Sections 13.1 and 13.6, and the surveillance frequency is consistent with the guidance in ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.5.5 is acceptable.

During a site visit, the NRC staff observed the layout and operation of the Confinement and Emergency Exhaust System. The NRC staff reviewed the design against the guidance of ANSI/ANS-15.1-2007 and the accident analysis contained in SAR Chapter 13 and finds that:

- The CVS design addresses appropriate sources of airborne radioactive material and ensures that these sources are diluted, diverted, or filtered so that occupational doses do not exceed the requirements of 10 CFR Part 20 and are consistent with the facility ALARA program.
- The design and operating features of the CVS ensure that no uncontrolled release of airborne radioactive material to the unrestricted environment will occur.
- The analyses of operations of the CVS show that planned releases of airborne radioactive material to the unrestricted environment will not expose the members of the public to doses that exceed the limits of 10 CFR Part 20 and the facility ALARA program guidelines.
- TSs ensure CVS operability.

Based on information above, the NRC staff concludes that the CVS at the RINSC is sufficient to control and mitigate the release of radioactive material.

6.2.2 Containment

According to SAR Section 3.5.6, the RINSC has a confinement and does not have a containment.

6.2.3 Emergency Core Cooling System

According to supplemental information provided by the licensee (Ref. 5), the RINSC does not have an emergency core cooling system. See the discussion in SER Section 5.7.

6.3 Conclusions

The NRC staff reviewed the RINSC SAR Chapter 6, as supplemented, and related TSs and concludes that the confinement, as described, provides reasonable assurance of limiting the consequences of a radioisotope release to the RINSC staff and members of the public of less than that allowed in 10 CFR Part 20.

7. INSTRUMENTATION AND CONTROL SYSTEMS

7.1 Summary Description

The revised SAR Chapter 7 (Ref. 49), describes the Instrumentation and Control (I&C) systems, including the design criteria and support design bases, and the functional and safety analyses of the I&C systems. Unless otherwise stated, all information in this chapter is from that source.

The I&C systems provide the RO with the required information to keep the reactor within its operational safety envelope. The I&C systems automatically scrams 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 control console and display instruments continuously monitor and display the neutron flux from the subcritical source multiplication range, through the critical range, and through the intermediate flux range to full power, while also providing reactor period information. In addition, the control console and display instruments provide input signals to the reactor control and reactor protection systems.

The reactor control system (RCS) enables manual control of reactor power from source to the power range levels and automatic control after a minimum power level has been attained. Only the regulating rod can be used to automatically control power. Automatic position control is available for the four shim safety blades. The shim safety blades can only be moved one at a time in manual positioning mode or automatic mode, with the exception that all four shim safety blades can be simultaneously inserted during a runback.

The RPS is designed to prevent operation of the reactor in regions in which fuel damage may occur. This is accomplished through promptly placing the reactor in a subcritical, safe shutdown condition by a reactor scram, which initiates the instantaneous drop of the shim safety blades by interrupting power to their electromagnets should a monitored parameter exceed a predetermined value. Inputs which govern the RPS output are supplied from the neutron flux monitors, process transducers, and safety interlocks. A reactor scram may also be initiated manually by the RO.

As stated in the revised SAR Section 7.5, ESFs are not required at this facility to mitigate identified accidents to keep radiological exposures to the operating staff or members of the public within the limits of 10 CFR Part 20.

The radiation monitoring system (RMS) detects and quantifies radiation and activity levels at various locations within the facility, within various reactor systems, and within the exhaust gases released to the uncontrolled environment.

In revised SAR Section 7.1, the licensee stated that I&C systems consists of sensors, electronic circuitry, display devices and actuating components that give the operators the information and capability to safely operate the reactor. Furthermore, I&C systems are in place to monitor and display the following reactor and support systems parameters:

- Neutron flux in the core
- Reactor power level
- Shim safety blade and regulating rod position
- Primary and secondary cooling system flows and temperatures
- Reactor pool temperature
- Confinement system status
- Radiological conditions within the facility and effluent paths

Additionally, the I&C systems provide input for interlocks, automatic control, alarm actuation and automatic scram initiation. The licensee explained that since initial licensing of the RINSC reactor in 1964, many of the original components of the systems discussed were replaced or upgraded. The licensee stated that such changes were reviewed in accordance with the requirements of 10 CFR 50.59. Subsequent sections describe the RCS, the RPS, the control console, and the RMS.

The I&C systems employed at the RINSC reactor are similar to those used by other RTRs operating in the United States. Control of the nuclear fission process is achieved using five control rods: four shim safety control blades 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 blade magnet power is removed allowing the blades to fall, by gravity, into the reactor. The regulating rod does not have scram capability.

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.

7.2 Design of Instrumentation and Control Systems

Revised SAR Section 7.2.1 states that I&C systems for the RINSC reactor are expected to continue to perform under adverse circumstances; however, they are not required to function to ensure reactor safety. I&C systems utilized for safe operation of the reactor are all located within the confines of the confinement building and confinement building basement. The control room, which encloses the control console and the instrument racks, is located within the confinement building and provides the ability to observe activities around the pool top.

The following design criteria are employed in the design of RINSC I&C systems utilized for reactor protection:

Single failure

- Use of this criterion ensures that no single component failure, single maintenance action, or single human error will disable the control systems causing loss of the capability to safely shutdown the reactor and maintain it in a shutdown condition for all postulated operating and or accident conditions.

Diversity

- Functional diversity uses multiple parameters to monitor reactor status
- Equipment diversity uses alternate devices for monitoring the same parameter
- Simple diversity uses duplication of devices for monitoring the same parameter

Independence

- Independence provides for physical separation such that failure of one channel has no effect on other channels.

Reliability

- Reliability is obtained by the use of technology that is qualified and/or proven through experience, testing, or both.

Testability

- Testability provides for the capability to perform periodic checks, calibrations, and function tests.

Manual Initiation

- Manual initiation provides a simple and direct means for the operator to take control and initiate a reactor shutdown under any circumstances.

Access Control

- Access control provides for the use of physical barriers/procedures to prevent unauthorized use of the reactor control system.

The I&C systems are designed to ensure the reactor can be operated safely and will automatically shut down in the event that an operating limit is approached or a single reactor protection system component failure occurs.

The I&C systems perform the following functions:

- Provide the operator with information on the status of the reactor systems and facility.
- Provide the means for controlling the reactivity of the reactor.
- Provide the means for detecting radiation levels within the facility and in the ventilation exhaust pathway.

Reactor Control System

Several safety features are designed into the RCS, they include:

- The RCS cannot be energized unless the keyed master switch is unlocked and placed in the on position
- All of the scram relays must be reset in order to energize the electromagnets that couple the shim safety blades to the drive mechanism.
- The RO can scram the reactor at any time by depressing the scram button on the control console.
- The RO can take manual control of the RPS at any time by simply selecting the manual mode button on the control console.
- Only one shim safety blade can be withdrawn at a time.
- All four shim safety blades can be scrammed from any position in the core.
- All four shim safety blades can be lowered simultaneously in the event of a runback signal.

There are also several interlocks designed into the RCS:

Safety Withdrawal:

- Shim safety blades cannot be withdrawn unless the neutron flux monitor start-up channel reads greater than 3 cps; this ensures that the nuclear instrumentation is on scale and there is adequate indication of neutron population for monitoring a reactor start up.
- The neutron flux monitor test selector switch must be in the OFF position in order to raise the shim safety blades; this ensures that actual detector output is being monitored while raising the blades versus a test signal.

Automatic Regulating Rod Control:

- The reactor period must be greater than 30 seconds to place the regulating rod in automatic. With a period less than 30 seconds, there is no assurance that the regulating rod could insert reactivity fast enough to compensate for the rate of change in power.
- The regulating rod must be full out to place it in automatic to ensure that the first movement the regulating rod makes in the automatic mode will be inward. If a runback is initiated, either manually or automatically, the regulating rod will shift back to the manual mode and these conditions will have to be re-established in order to return the regulating rod to the automatic mode of operation.

Runback:

- An automatic runback occurs if reactor period is less than or equal to 7 seconds. A runback inserts all four shim safety blades simultaneously. The runback will terminate once the reactor period is greater than 7 seconds. If the period continues to decrease and reaches 4 seconds, a short period scram will occur.

- A manual runback can be initiated by the RO by selecting the runback button on the rod control display. A manual runback will continue until stopped by the operator.

Reactor Protection System

The revised SAR Section 7.2.2.2 states that the RINSC RPS ensures the LSSS are not exceeded as the result of any abnormalities or transients of the type and magnitude evaluated and found acceptable in SER Chapter 13. The RPS addresses those parameters required to provide reactor protection: reactor power level, primary coolant flow, primary coolant temperature and reactor pool level. These parameters are all integral in the analysis for design bases events that demonstrate that the integrity of the fuel cladding is maintained under all circumstances and therefore, there are no uncontrolled releases of radioactivity that would result in radiation doses to members of the public or staff personnel that exceed 10 CFR Part 20 limits.

The RPS, as discussed in more detail in later sections, has both redundancy and diversity built into its design, and inputs that are received will independently initiate a reactor scram. TS 3.2 provides a listing of all RPS reactor scrams and set points.

The design bases analysis considers that an automatic protective action, referred to as a scram, occurs at the point where any or all of the parameters mentioned above reach their LSSS set point as defined in RINSC TSs.

Reactor Console

The revised SAR Section 7.2.2.4 states that the control console and instrument racks are located in the control room. The displays are designed to provide the RO with a suitable representation of the essential parameters required to safely monitor and operate the reactor. The intent is to provide adequate and reliable information from which the operator can ascertain the condition of the reactor and its support systems and take appropriate actions as necessary. The displays show not only the parameters discussed in the RPS section above, but also parameters associated with other support systems and facility conditions. Annunciators are also provided to alert the operator to abnormal conditions. User interfaces in the control room allow the operator to start and stop various equipment throughout the facility to support reactor operation

Radiation Monitoring System

The revised SAR Section 7.2.2.5 states that the RMS system is designed to provide the RO with adequate and reliable information, as it pertains to radiological conditions throughout the facility, as well as the confinement building exhaust effluent. The RMS provides for both local and remote indication of radiological conditions with detector failure, alert and high alarms. The RMS provides for redundant and diverse detection for both gamma and neutron area radiation monitoring, particulate and noble gas monitoring of confinement atmosphere, and exhaust effluent.

The NRC staff finds that the design criteria and functional capabilities of the I&C system are typical of other RTRs, are consistent with the licensing basis for RINSC, and on the basis of the information supplied are acceptable.

7.3 Reactor Control System

The revised SAR Section 7.3 describes the RCS. Reactivity is controlled in the RINSC reactor by means of a regulating rod and four shim safety blades. The regulating rod is mechanically coupled to the operating mechanism and therefore, has no scram capability. The four shim safety blades are coupled to the operating mechanisms via an electromagnet. All four shim safety blades can be scrammed manually or automatically by the RPS system.

The RCS has two modes of operation: manual and automatic. The operation mode refers primarily to operation of the regulating rod. Manual mode allows the operator to control the height of the regulating rod, as well as the shim safety blades, whereas automatic mode controls the regulating rod height based on a preset reactor power set point. Manual mode is normally used for performing reactor start-up, shut-down and major changes in power level. Automatic mode is used for steady state operation.

The major components of the RCS are briefly discussed below.

Master Switch

The master switch for RCS is located in instrument rack #1, and is powered directly from the control power circuit breaker. The key is kept in a security locker with limited access. When unlocked and turned to the “ON” position, the master switch energizes the 24 volt direct current (DC) power supply, control blade drive motors and control circuits, power level interlocks, electromagnet magnet power supplies, trip actuation circuits, “Reactor On” light, and the control panel annunciator.

Since the switch is powered directly from the control power breaker, if the facility power supply is lost then the switch is no longer powered. This will de-energize trip actuator amplifiers, and thus the safety blade electromagnets, causing a reactor scram.

Shim Safety Blade Electromagnet Power Supply

Each of the four shim safety blades is coupled to the drive mechanisms by an electromagnet when the reactor is in operation. These magnets are positioned above the reactor pool normal water level, directly above the control blade armature.

The magnet power supplies supply 24-volt DC power to the magnets at less than 1 amp each. The power supplies receive a 12-volt DC signal from the nuclear instrumentation, by means of the logic element. The power supplies also receive a 120-volt alternating current (AC) signal from the reactor control computer, through the annunciator panel. A loss in the DC or AC signal signifies an electronic or mechanical scram, respectively. After the initiation of a scram signal, the magnet power supplies de-energize, allowing the shim safety blades to be dropped into the core. Power can be restored to the magnets after clearing all scram signals by restoring the DC and AC signals and pressing the scram reset button on the annunciator panel.

The shim safety blade magnets and magnet power supplies are independent from any of the operator controls or shim safety blade drive systems. The initiation of a scram cannot be overridden by an operator. This means that no other system will interfere with the initiation of a reactor scram.

Shim Safety Blade Control and Drive System

The RCS changes are performed using the reactor control computer. This computer is the interface and controls the output to the rod control display, through which the operator can access all features of the RCS. The system can be operated in manual mode using the selector switches and toggle switches on the control console, which are part of the original RCS, or in automatic mode by using the reactor control computer and rod control display interface. In either mode, the rod control display is required to be functional to display shim safety blade and regulating rod position.

For operation using the reactor control computer and rod control display, there is a select button for each shim safety blade which activates a pair of relays operating in a binary logic system to designate which blade is manipulated. This mechanism replicates the original mechanical selector switch in preventing multiple blades being withdrawn simultaneously. The select button for the selected shim safety blade changes state to indicate it is the shim safety blade that is selected to be manipulated. The selected shim safety blade can then be manipulated with manual withdraw and insert buttons, mimicking the toggle switch on the control console. Additionally, the selected shim safety blade can be manipulated using the auto blade position feature. This allows the operator to enter a specific withdrawal position between 0 and 26 in. The start button begins moving the shim safety blade toward the specified position; the stop button stops the shim safety blade immediately. The shim safety blade will automatically stop moving once it reaches the specified position. Blade movements are still subject to the various alarms, scrams, and reactivity insertion rates during the use of the auto blade position feature. All movements use separate blade up and blade down relays. Should a malfunction occur and both relays become energized, the blade down function would override and cause the selected shim safety blade to insert.

The manual rundown button functions similarly to the original manual rundown switch on the console by inserting all four shim safety blades simultaneously. Unlike the original switch on the console, the manual rundown button will reset itself once all shim safety blades are fully inserted.

Each shim safety blade has a separate controller which powers the associated stepper motor that is connected to the original shim safety blade drive gear through a gear reducer. A digital encoder located underneath the gear reducer measures the angular movement of the drive mechanism. The angular movement of the drive gear correlates to the vertical movement of the shim safety blade and is displayed for each shim safety blade on the rod control display.

Each drive mechanism assembly has a limit switch located at the top and bottom of the travel range to indicate full-in and full-out position. The activation of these switches overrides the shim safety blade movement commands and prevents shim safety blades from moving beyond the desired travel range.

The original controls for the drive mechanisms located on the control console remain as a redundant system in parallel with the reactor control computer. A four-position selector switch allows the operator to choose which shim safety blade is manipulated, allowing for only one shim safety blade to be withdrawn at a time. The selected shim safety blade is inserted or withdrawn using a momentary toggle switch that defaults to the off position. The manual rundown switch is a maintained toggle switch that will insert all control blades to their full-in position. A two-position selector switch allows the operator to select manual mode to use the control console to operate the drive mechanisms, or auto mode to use the reactor control

computer and rod control display. The rod control display is the sole shim safety blade and regulating rod position display.

The drive mechanisms use a DC motor to raise and lower the drive shaft. The drive mechanisms are directly coupled to electromagnets which when energized, will couple the drive mechanisms to the shim safety blades. Coupling the electromagnets to the shim safety blades is accomplished via power received from the RPS system, which requires all of the scram conditions and scram relays to be reset.

Shim Safety Blade Withdrawal Interlocks

Withdrawing the shim safety blades requires that the electromagnets be energized to couple the drives to the shim safety blades. Power to energize the electromagnets comes from the RPS through the master switch and the electromagnet amplifiers, and cannot be provided unless the following conditions are met:

- Master switch is unlocked and in the “ON” position.
- All scram input signals are clear and scram relays reset.

With the drive mechanisms in the full down position and the electromagnets energized, a “Blade Engaged” button will be displayed on the rod control display just below each of the shim safety blades position indicators. Once engaged, RCS control logic prevents the operator from withdrawing more than one shim safety blade at any one time. In addition, there are two interlocks which prevent shim safety blade withdrawal:

- Neutron flux monitor test select switch must be in the “OFF” position
- Source range start-up counts must be great than 3 cps

Regulating Rod Drive System

As stated in the revised SAR Section 7.3.6, the regulating rod is also controlled by the reactor control computer and rod control display. The controls allow for manual insertion and withdrawal, similar to the original control console controls. The reactor control computer also allows for the regulating rod to move to the full-in or full-out position through a single command. A stop button allows the operator to stop the regulating rod immediately.

The commands for manipulation of the regulating rod are sent to a pair of relays in the instrument rack with one relay for each direction of movement. The regulating rod also has a separate controller which receives the signal from the instrument rack for manipulation of the regulating rod. Signals for regulating rod position and position limits are sent back to the instrument rack and reactor control computer.

The reactor control computer also allows for the automatic manipulation of the regulating rod for minor adjustments to reactor power. When interlocks are satisfied (regulating rod is full-out and reactor period is greater than 30 sec), the operator may place the regulating rod in automatic mode. In this mode, the operator will have set the power schedule on the rod control display to the desired reactor power level as a function of percent of full power. The reactor control computer compares the desired power level to the current power level from wide range channel #1. The computer will generate an error signal output to a servo-system controller to

insert or withdraw the regulating rod to adjust the current power level until it agrees with the power level set point. The regulating rod motion is powered by a stepper motor control.

The original regulating rod console controls remain as a redundant system in parallel to the reactor control computer. The regulating rod can be manipulated using a momentary toggle switch to insert or withdraw. The reactor control computer is the sole regulating rod display and is the only means of engaging the regulating rod automatic mode or adjusting the desired power level setting.

Conclusion

The NRC staff reviewed the RINSC RCS operating characteristics in SAR Chapters 4 and 13, as well as the description by the licensee in supplemental information, and concludes that:

- The licensee has analyzed the normal operating characteristics of the reactor facility, including thermal steady-state power levels and the planned reactor uses. The licensee has also analyzed the functions of the RCS and components designed to permit and support normal reactor operations, and confirms that the RCS and its subsystems and components will give all necessary information to the operator or to automatic devices to maintain planned control for the full range of normal reactor operations.
- The RCS is designed to sense all parameters necessary for facility operation with acceptable accuracy and reliability, to transmit the information with high accuracy in a timely fashion, and control devices are designed for compatibility with the analyzed dynamic characteristics of the reactor.
- The RCS has sufficient interlocks to limit hazards to personnel and to ensure compatibility among operating subsystems and components in the event of single isolated malfunctions of equipment.
- The RCS is designed so that any single malfunction in its components will not prevent the RPS from performing necessary functions and will not prevent safe shutdown of the reactor.

7.4 Reactor Protection System

The supplemental responses for SAR Chapter 7 (Ref. 49), describes the RPS as having two modes of operation: the 2 MWt mode, also referred to as the high-power or FC cooling mode, and the 0.1 MWt mode, also referred to as the LP or NC cooling mode. In the 0.1 MWt mode of operation, the scrams associated with primary flow are bypassed.

Nuclear Instrumentation System

Revised SAR Section 7.4.1 states that three individual channels of nuclear instruments (NIs) monitor and provide display, alarm and scram input for reactor power. Two channels, referred to as the wide range linear power (WR) channels 1 and 2, measure the neutron flux utilizing a compensated ion chamber detector and display their output in the control room over 14 ranges of power starting with a 600-MWt range, and ending with a 2 MWt range. The third channel, referred to as the neutron flux monitoring (NFM) channel, measures the neutron flux utilizing a fission chamber providing indication from the start-up range up through and including the power range. The NFM has multiple displays including cps, reactor period, wide range logarithmic power, and wide range linear power. The function of the two WR channels is to monitor reactor

power level over several ranges of output. These channels overlap the NFM channel and provide redundant indication and trip capability for reactor power.

A trip signal from any one of the three NIs will initiate a reactor scram. WR channel 1 is also used to provide input to the servo controller, which controls the regulating rod when in the automatic mode of operation. The NFM channel provides input for the regulating rod interlocks and the shim safety blade withdrawal interlocks.

WR Channel Trip Circuit

The trip circuit for the WR channels contains identical bi-stable circuits to generate trip signals for loss-of-high voltage (HV), high-power and annunciator alarm initiation. Once a trip signal has been generated, it remains locked-in and must be reset even if the output signal causing the trip returns to nominal value.

WR Channel Output Isolator

The output signal from the WR channel modules is used for remote indications and controls.

WR Channel 1 and 2 Display and Output

Each module has a 0 to 125 percent linear power indicator, range indicator, range select for local operation, HV indication, compensating voltage indication and potentiometer, a gain adjust potentiometer, a test/calibration ramp select and potentiometer and four LED lights for bi-stable status indication.

Neutron Flux Monitor Channel

The function of the NFM channel is to monitor the power level starting from extremely low-power levels, start-up range, and up through the power range. The NFM overlaps the WR linear power channels and provides a more accurate assessment of changing neutron fluence when very low in power. The NFM channel consists of a fission chamber detector, pre-amplifier and a wide range logarithmic power module to provide both reactor power and reactor period indication.

NFM Channel Trip Circuit

The NFM trip circuit is similar in design to the WR channels and contains identical bi-stable circuits to generate trip signals for short period, high-power and annunciator alarm initiation. A comparator monitors the incoming signal and compares it to a reference voltage.

NFM Channel Isolation Circuit

The NFM isolation circuit is similar in design to the WR channels and provides isolation for both the power output and the period output.

NFM Channel Display and Output

The NFM utilizes a fission chamber to detect neutron flux and rate of change in neutron flux in the core. The NFM has 0 to 125 percent linear power indication, $10\times^{-8}$ to 100 percent wide range logarithmic indicator, a $10\times^{-1}$ to 105 cps indicator, a source range and wide period indicator.

The bi-stable outputs are used for:

- Linear power 115 percent Hi Flux scram
- Wide Range period less than 7-second rundown
- Wide range period less than 30-second regulating rod interlock
- Wide range period less than 4-second scram
- Start-up channel less than 3 cps rod withdrawal interlock

The NFM module is completely independent of the WR modules, and a bi-stable status change requiring a reactor trip on this module will result in a scram actuation regardless of the condition or status of the WR modules.

Process Control and Instrumentation System

Revised SAR Section 7.4.2 states that the process control and instrumentation system (PCI) is composed of the following channels: (1) primary coolant system flow, (2) primary coolant inlet temperature, (3) primary coolant outlet temperature, (4) bulk pool water temperature, and (5) pool water height. The PCI includes a display screen on the instrument rack, which has a mimic one line of the process systems and redundant display of the temperatures and flows. This display screen is also the user interface for operation of the various components in the cooling water and ventilation systems supporting reactor operation.

Primary Coolant Flow Measurement

The primary flow is measured in each cooling loop by use of an orifice installed in a straight run of pipe in the heat exchanger room. A differential pressure transmitter provides an analog output signal that is proportional to the differential pressure across the orifice plate caused by the flow through the pipe. The signal is displayed in gallons per minute (gpm) on one of the PCI channels. Each cooling loop has independent indication, alarm, and trip capability. A trip signal is generated by the primary flow device when primary flow decreases below the scram set point.

Primary Coolant Temperature Measurement

The primary cooling water has two temperature measuring channels: one measures inlet temperature to the core; the second measures outlet temperature from the core. Both channels are the same design and function independently. The inlet temperature detector is mounted in the system as it passes through the delay tank room just after the two cooling loops merge, near the pool wall penetration. The outlet temperature detector is located in the system as it passes through the delay tank room in the coolant piping, just before the primary coolant enters the delay tank. Both channels display temperature in degrees Fahrenheit, and are set up to provide alarm on temperatures exceeding a set point. The inlet temperature also provides a trip signal upon exceeding set point.

Bulk Pool Temperature Measurement

There is one temperature measuring channel associated with reactor pool water which measures bulk pool temperature. This channel is similar to the primary coolant temperature channels, but completely independent. It provides indication and trip capability. The bulk pool temperature detector is mounted on the west wall of the reactor pool. Bulk pool temperature is displayed in degrees Fahrenheit and provides an alarm when the temperature is above the set point and the reactor is in NC mode.

Reactor Pool Height

The height of water above the top of the fuel in the core is monitored by two independent float switches. Both switches actuate on low pool water level. One switch provides make-up water control (see SER Section 5.5.1), and the second switch generates a scram signal. The pool water height is not actually measured, and the float provides no indication to the control room. In the event that the water level falls below the scram set point, a signal is also sent to the alarm system which is monitored continuously. The company that is contracted to monitor the system will call a staff member to investigate the alarm.

Scram Circuit

The RCS for the reactor utilizes four electromagnets to couple the shim safety blades to the control rod drive mechanisms, allowing the blades to be pulled out of the core and increase power level. In the event of the initiation of a scram, the power to these magnets is de-energized, allowing the shim safety blades to fall back into the core due to gravity. Power for these magnets is supplied by two trip actuator amplifiers. The amplifiers draw their power from a 120 volt AC supply from the reactor console, and a 12 volt DC input from a logic element/trip reset. If either of these power supplies is lost, the amplifiers lose power and de-energize the electromagnets. The automatic or manual scram of the reactor can be initiated by a number of relays, limit switches, and electronic input signals that comprise the scram circuit. Any of these devices has the capability to de-energize either the 120 volt AC or the 12 volt DC power supply when scram conditions are present.

Three independent power level channels can be used for both natural and FC cooling modes of operation, each of which is independently capable of scramming the reactor before reactor power exceeds 115 percent of licensed power. This provides defense in depth.

One rate of change of power channel is utilized for both cooling modes of operation. A 4-second period limit serves as additional protection to assure that the reactor fuel would not be damaged in the event that there was a power transient.

Each power channel has a high-voltage failure scram. These channels rely on detectors that require HV in order to be operable. These scrams assure that the reactor will not be operated when one of these detectors does not have proper HV.

One low pool level channel is available for both forced and NC cooling modes of operation. This channel ensures that the reactor will not be in operation if the pool level is below the minimum level required.

One manual scram button, that is located in the control room, is available to the operator to initiate a scram at any time conditions dictate that a scram is warranted.

One rod control communication scram is available for both modes of operation. The RCS has a communication link between the digital display in the control room, and the stepper motor controllers out at the pool top. There is a watchdog feature that verifies that this communication link is not broken. In the event that the link is broken, a scram will occur. This communication link is only for the RCS and does not affect any other signals or magnet power coming from or going to the reactor bridge.

One seismic disturbance scram is available for both modes of operation. In the event a seismic disturbance is detected, the seismic scram relay will de-energize, initiating a scram.

One bridge movement scram is available for both modes of operation. This scram ensures that the reactor will be shut down in the event that the bridge moves while the reactor is in operation.

One bulk pool temperature channel is utilized for the NC cooling mode of operation. This channel is capable of scramming the reactor when the temperature increases to above the set point. This channel provides the over temperature protection when the reactor is operated in the NC cooling mode.

One primary inlet temperature channel is utilized for FC cooling mode operation. This channel will initiate a scram signal if inlet temperature increases above the set point.

One primary coolant flow rate channel is utilized for FC cooling mode operation. This channel assures that the reactor will not be operated at power levels above 100 kWt with a primary coolant flow rate that is less than the minimum required.

One coolant gate open scram input is used on each coolant duct. Either of the gates being open while the reactor is being operated in the FC cooling mode will cause a reactor scram. This scram ensures that coolant flow through the inlet and outlet ducts is not bypassed during FC cooling.

One no flow thermal column scram is utilized during FC cooling mode operation. This scram ensures that there is coolant flow through the thermal column gamma shield during operations above 100 kWt.

One bridge LP position scram is utilized for FC cooling mode operation. In order for the FC cooling system to work, the reactor must be seated against the HP section pool wall. This scram ensures that the reactor is properly positioned in the pool so that the coolant ducts are properly coupled with the cooling system piping.

Annunciator System

The annunciator system provides alarm and indication for the operator indicating that some condition or parameter is abnormal. When an alarm condition is reached, an alarm horn will sound and the appropriate annunciator window will illuminate. The operator can acknowledge and silence the alarm horn, but cannot reset the annunciator window until the condition clears. The following conditions will activate an annunciator window:

- Hi Temp Primary Coolant - This is initiated on inlet or outlet temperature increasing to above set point.
- Lo Flow Pri Coolant - This is initiated on primary flow decreasing to below set point.

- Bridge Lo Power Pos - This is initiated when the bridge is moved off of the stops at the HP end of the reactor pool.
- Coolant Gate Open - This is initiated when either of the coolant duct gates are open, in the forced convection (FC) mode of operation; this will also initiate a scram.
- Lo Lev Cooling Tower - This annunciator is no longer used.
- Bridge Movement - This is initiated when the bridge is moved while the reactor is in operation regardless of mode; this will also initiate a scram.
- High Rad Area - This is initiated when the locked gate at the heat exchanger room or the double doors at the north entrance to the heat exchanger room are opened while the reactor is being operated. Warns the operator that someone is entering a potentially high radiation area.
- Inst Trouble - This is initiated when a loss of HV occurs on any of the nuclear instrumentation channels.
- Reg Blade Limit - This is initiated when the regulating rod is at the limit of travel, either high or low.
- Cont Blade Disengd - This is initiated when any one shim safety blade is uncoupled from its drive mechanism.
- High Cond - This annunciator is no longer used.
- Low Pool Level - This is initiated when pool level decreases to below set point and initiates a scram.
- High Neutron Flux - This is initiated when either WR 1 or 2 linear power channels exceeds 110 percent of full power.
- Short Period - This is initiated when the NFM period exceeds the runback set point or the scram set point.
- Seismic Scram - This is initiated when a gross seismic event is detected and initiates a scram.
- Annun Reset - This allows the operator to reset the annunciator window associated annunciator relay once the condition has cleared.
- Annun Ackn - This allows the operator to acknowledge the alarm and silence the audible alarm horn.
- Scram Reset - This allows the operator to reset the scram relays if all scram conditions are clear.

Natural Convection Mode Operation

The reactor can be operated up to a steady state power level of 0.1 MWt with no primary coolant pumps running. This is referred to as the NC cooling mode of operation, also referred to as the LP mode or the .1 MW mode of operation. When the power level select switch is in the .1 MW position, a set of contacts is aligned such that primary coolant parameters, reactor power and bridge position scram inputs are changed to support reactor operation in this mode of operation. Specifically, the high temperature scram switches from inlet temperature to bulk pool temperature, low primary coolant flow and coolant gates open no longer initiate a scram signal, the over power scram is now activated at 57 percent on the 200-kW scale on WR channels 1 and 2, and the bridge LP position no longer initiates a scram signal.

TS 3.2.4 of TS 3.2 states:

3.2 Reactor Control and Safety System

The reactor shall not be operated unless:

3.2.4 The instrumentation shown in Table 3.1, Required Safety Channels, is operable and capable of performing its intended function:

Table 3.1 Required Safety Channels

Table 3.1.1 Required Safety Channel Scrams					
Line #	**Protection**	**Op Mode**	**Channels Required**	**Function**	**Set Point**
1.	Over Power	Both	2	Scram before power is greater than	115% Power for selected mode
2.	Rate of Change of Power	Both	1	Scram before period is less than	4 seconds
3.	Detector HV Failure for Lines 1 & 2 above	Both	1 per operable channel	Scram on a loss of HV power	50 V below suggested operating voltage
4.	Low Pool Level	Both	1	Scram before pool level is less than	23 feet 7 inches above the top of the fuel
5.	Manual Scram	Both	1	Scram when	Control Room Scram Button Depressed
6.	Control Rod Drive Communication (Watchdog)	Both	1	Scram if loss of communication for greater than	10 seconds
7.	Seismic Disturbance	Both	1	Scram when	Seismic Disturbance Detected
8.	Bridge Movement	Both	1	Scram when	Bridge Movement Detected
9.	Pool Temperature	NC	1	Scram before temperature is greater than	127° F
10.	Primary Coolant Inlet Temperature	FC	1	Scram before temperature is greater than	122° F
11.	Primary Coolant Flow Rate	FC	1	Scram before flow rate is less than	1560 gpm
12.	Coolant Gates Open	FC	1	Scram when	Inlet or outlet gate open
13.	No Flow Thermal Column	FC	1	Scram when	No Flow Detected
14.	Bridge Low Power Position	FC	1	Scram when	Bridge Not Seated at HP End

Table 3.1.2 Required Safety Channel Interlocks					
Line #	**Protection**	**Op Mode**	**Channels Required**	**Function**	**Set Point**
1.1	Servo Control Interlock	Both	1	Regulating rod cannot be placed in automatic servo mode if	Regulating rod not full out
1.2	Servo Control Interlock	Both	1	Regulating rod cannot be placed in automatic servo mode if reactor period is less than	30 seconds
2.	Shim Safety Blade Withdrawal Interlock	Both	1	No shim safety blade withdrawal if start up channel count rate less than	3 counts per second
2.2	Shim Safety Blade Withdrawal Interlock	Both	1	No shim safety blade withdrawal if Neutron Flux Monitor Test / Select switch is	Not in the Off position
2.3	One Shim Safety Blade Withdrawal Interlock	Both	1	Only one SSB can be withdrawn at any one time	Select switch if in manual mode, binary logic must be satisfied if in auto mode

Table 3.1.3 Required Safety Channel Indications					
Line #	**Description**	**Op Mode**	**Channels Required**	**Function**	**Set Point**
1.	Wide Range Linear Power	Both	1	Provide indication of reactor power	N/A
2.1	Log Power	Both	1	Provide indication of reactor power	N/A
2.2	Log Power Start-up Counts	Both	1	Provide indication of start-up channel counts	N/A
2.3	Log Period	Both	1	Provide indication of rate of change in reactor power	N/A
3.	Pool Temperature	NC	1	Provide indication of bulk pool temperature	N/A
4.	Primary Coolant Inlet Temperature	FC	1	Provide indication of primary coolant inlet temperature	N/A
5.	Primary Coolant Outlet Temperature	FC	1	Provide indication of primary coolant outlet temperature	N/A
6.	Primary Coolant Flow Rate	FC	1	Provide indication of primary coolant flow	N/A
7.	Confinement Building Pressure	Both	1	Provide indication of Confinement Building Pressure	N/A

TS 3.2.4 establishes several requirements for the I&C system. These requirements help to ensure that the required complement of channels are specified, their functions are explained, and their setpoints are established as a function of reactor mode (NC or forced flow). This information is provided in the supplied TS Table 3.1. SER Section 7.4 evaluates and finds these requirements acceptable. The NRC staff's review of this table finds that it is consistent with the LSSS setpoints as evaluated and found acceptable in this SER. The NRC staff also finds that this information is consistent with the supplied T&H analysis, and that the scram functions are consistent with the assumptions used in the SAR Chapter 13 safety analysis as evaluated and found acceptable in this SER. The table provides the key requirements and parameters important to each channel. Based on the information above, the NRC staff concludes that TS 3.2.4 and Table 3.1 are acceptable.

TSs 4.2.3, 4.2.4, 4.2.5 and 4.2.6 of TS 4.2 state:

4.2 Reactor Control and Safety System

4.2.3 The following reactor safety and safety related instrumentation shall be verified to be operable by performing a channel test prior to the initial start-up each day that the reactor is started up from the shutdown condition, and after the channel has been repaired:

- 4.2.3.1 Control room manual scram button
- 4.2.3.2 Power level channels
- 4.2.3.3 Period channel
- 4.2.3.4 Rod control communication watchdog scram

4.2.4 The following reactor safety and safety related instrumentation shall be verified to be operable by performing a channel test prior to the initial start-up each day that the reactor is started up from the shutdown condition, and for which reactor power level will be greater than 100 kW, and after the channel has been repaired:

- 4.2.4.1 All of the reactor safety and safety related instrumentation listed in 4.2.3.
- 4.2.4.2 Primary coolant flow scram

4.2.5 The following reactor safety and safety related instrumentation scrams, and interlocks shall be channel tested annually:

- 4.2.5.1 The following detector HV failure scrams:
  - 4.2.5.1.1 Power level channels
  - 4.2.5.1.2 Period channel
- 4.2.5.2 The following shim safety withdrawal interlocks:
  - 4.2.5.2.1 Start-up count rate
  - 4.2.5.2.2 Test / Select switch position
  - 4.2.5.2.3 Shall verify that only one shim safety blade can be withdrawn at a time
- 4.2.5.3 The following servo control interlocks:

4.2.5.3.1 Regulating blade not full out

4.2.5.3.2 Period less than 30 seconds

4.2.5.4 The following coolant system channel temperature scrams:

4.2.5.4.1 Primary inlet temperature

4.2.5.4.2 Pool temperature

4.2.5.5 The following coolant system channel flow scrams:

4.2.5.5.1 Primary flow and flow rate

4.2.5.5.2 Coolant gates open

4.2.5.5.3 No flow thermal column

4.2.5.6 Low pool level scram

4.2.5.7 The following bridge scrams:

4.2.5.7.1 Bridge movement

4.2.5.7.2 Bridge low power position

4.2.5.8 Seismic scram

4.2.6 The following reactor safety and safety related instrumentation shall have a channel calibration performed annually:

4.2.6.1 Power level channels

4.2.6.2 Primary flow channel

4.2.6.3 Primary inlet and outlet temperature channels

4.2.6.4 Pool temperature channel

TS 4.2.3 requires a surveillance to verify certain I&C components and these conditions are applicable at all power levels. This specification indicates the reactor safety and safety related instrumentation that must be verified to be operable prior to the initial reactor start-up of each day, and helps to ensure that they are capable of performing their intended functions at all power levels. This specification is consistent with the facility design requirements. Based on the information above, the NRC staff concludes that TS 4.2.3 is acceptable.

TS 4.2.4 requires a surveillance to verify the operability of certain I&C components, and these conditions are applicable at all power levels greater than 100 kWt. This specification indicates the reactor safety and safety related instrumentation that must be verified to be operable prior to the initial reactor start-up of each day and helps to ensure that they are capable of performing their intended functions whenever power is greater than 100 kWt. For operations where power is above 100 kWt, this specification requires that the primary coolant flow rate scram be verified to be operable prior to the initial start-up of the reactor. This specification is consistent with the facility design requirements. Based on the information above, the NRC staff concludes that TS 4.2.4 is acceptable.

TS 4.2.5 requires a surveillance to verify the cited alarms, scrams, and interlocks are functioning, as required, by performing channel tests. For all of the scrams listed in these sections, the annual requirement is consistent with the existing facility surveillance frequency. Based on the information above, the NRC staff concludes that TS 4.2.5 is acceptable.

TS 4.2.6 requires a surveillance to calibrate listed I&C channels. This specification helps to ensure that the indicated channel value is within accepted tolerances of the real value. This then helps to ensure that the response to channel indications are consistent with the safety analysis evaluated and found acceptable in SER Chapter 13. These requirements are consistent with the facility surveillance frequency, and is within the range recommended by ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.2.6 is acceptable.

The NRC staff reviewed the reactor safety system described in the SAR, as supplemented with responses to RAIs, and conducted a site visit to observe the placement of I&C equipment. The staff reviewed the key parameters relevant to analyzed accidents in SAR Chapter 13 and concludes that the I&C components governed by TS 3.2 meet the guidance of ANSI/ANS-15.1-2007, and that TS 3.2 is acceptable.

7.5 Engineered Safety Features Actuation Systems

The revised SAR Section 7.5 states that ESFs are not required for the RINSC. The confinement system, ventilation system and the evacuation system all work in unison when manually initiated to minimize the consequences of any radiological release; however, they do not actuate automatically, nor are they required to actuate to mitigate any accidents or conditions. Consistent with the accidents analyzed in SAR Chapter 13 and evaluated and found acceptable in SER Section 13, the NRC staff concludes that no ESF systems are required for the RINSC reactor.

7.6 Control Console and Display Instruments

The revised SAR Section 7.6 describes the I&C systems as having a series of cabinets all located in the control room. They are the control console and four instrument racks. The control room is an enclosed, environmentally controlled room on the pool level of the confinement. It has glass on the north and west walls, which provide the operator visual access to all of the pool level and approximately a third of the main experimental floor. The control console serves as a central point for location of operating controls and instrumentation. The operator is provided with a vantage point from which to conveniently observe reactor performance and the pool area. The operator can adjust operations to varying requirements when needed for tests, experiments and power level requirements.

The control console consists of a desk-type cabinet. Located on the right of the console control panel is the RCS computer. This computer operates the software that is the primary means of manipulating the reactor controls.

The central portion of the console is occupied by the annunciator panel as previously discussed. Mounted below the annunciator panel is the manual scram switch. Below the scram switch in the center portion of the control panel are the original reactor controls, including the blade select switch, control blade manual control switch, manual rundown switch, and regulating rod manual control switch. Also located in this section is the auto/manual select switch. This switch designates which system selects the control blade to be manipulated, the computer RCS if in the AUTO position, or the original switches if in the MANUAL mode. The original switches are also capable of overriding the computer-controlled RCS, if needed.

The start-up count rate and period, logarithmic power level and period, and the linear power level indicators are located on the left side of the control panel, giving the operator three

independent and redundant indications of reactor power. Mounted below the linear indicators are the linear range switches and the reactor on indicator switch. Based on the information above, the NRC staff concludes that the console is typical of other RTRs of this type, and is acceptable.

7.7 Radiation Monitoring Systems

The revised SAR Section 7.7 (Ref. 49), describes the area RMS including the remote area monitors indicating gamma and neutron radiation levels at several locations throughout the confinement building and basement areas, noble gas monitors and exhaust stack particulate and gaseous monitors. The individual monitors indicate and alarm locally as well as in the control room. None of the radiation monitors initiate any actuation or control of equipment. Table 7-1 reproduces TS Table 3.2, which lists all of the radiation monitors and their locations. The maximum set point for every item in Table 7-1 is 2 times normal, with the exception of TS Table 3.2.1, Line # 1, for which the maximum set point is 2.5 times normal. Based on the information above, the NRC staff finds that the indicated complement of radiation monitors is typical and appropriate for reactors of this type, and concludes that TS Table 3.2 is acceptable.

Table 7-1 Radiation Monitoring Equipment

3.2.1 Required Radiation Monitors				
Line#	**Description**	**Minimum Required**	**Function**	**Operating Mode**
1.1	Confinement Building Exhaust Stack Gaseous	1	Indication and alarm both locally and in control room	As per TS 3.7.1.1.1
1.2	Confinement Building Exhaust Stack Particulate	1	Indication and alarm both locally and in control room	As per TS 3.7.1.1.1
2.	Reactor Bridge Area Monitor	1	Indication and alarm both locally and in control room	As per TS 3.7.1.1.3
3.	Main Floor of Confinement Building (At least one of 3.2.2, lines 3, 6 or 7)	1	Indication and alarm both locally and in control room	As per TS 3.7.1.1.3

3.2.2 Other Available Radiation Monitors (NO MINIMUM REQUIRED)				
Line #	**Description**	**Detector Type**	**Function**	**Operating Mode**
1.	Main Floor Particulate Monitor	Alpha Beta Gamma	Indication and alarm both locally and in control room	N/A, Can be used as temporary alternate for stack particulate monitor
2.	Fuel Safe Area Monitor	Gamma Neutron	Indication and alarm both locally and in control room	N/A
3.	Thermal Column Area Monitor	Gamma Neutron	Indication and alarm both locally and in control room	N/A

4.	Heat Exchanger Area Monitor	Gamma Neutron	Indication and alarm both locally and in control room	N/A
5.	Primary Clean-Up Demineralizer Area Monitor	Gamma Neutron	Indication and alarm both locally and in control room	N/A
6.	Beam Port Area Monitors (4 total)	Gamma Neutron	Indication and alarm both locally and in control room	N/A
7.	Dry Irradiation Facility Area Monitor	Gamma Neutron	Indication and alarm both locally and in control room	N/A
8.	Rabbit room Area Monitor	Gamma Neutron	Indication and alarm both locally and in control room	N/A
9.	Rabbit Room Noble Gas Monitor	Noble Gas	Indication and alarm both locally and in control room	N/A
10.	Pool Level Noble Gas Monitor	Noble Gas	Indication and alarm both locally and in control room	N/A

TS 3.7.1.1 states:

3.7.1.1 Required Radiation Monitoring Systems

Whenever the following operations are in progress:

- The reactor is operating.
- Irradiated fuel handling is in progress.
- Experiment handling is in progress for an experiment that has a significant fission product, or gaseous effluent activation product inventory.
- Any work on the core or control rods that could cause a reactivity change of more than 0.60 %Δk/k is in progress.
- Any experiment movement that could cause a reactivity change of more than 0.60 %Δk/k is in progress.

3.7.1.1.1 A minimum of one radiation monitor that is capable of warning personnel of high radiation levels in the confinement gaseous and particulate effluent (Table 3.2, Required Radiation Monitors, lines 1.1 and 1.2) shall be operating.

3.7.1.1.2 If the detector described in specification 3.7.1.1.1 fails during operation, within one hour, place in service a suitable alternative air monitor or begin an hourly grab sample analysis (grab sample analysis applies to particulate only) in lieu of having a functioning monitor.

3.7.1.1.3 A minimum of one gamma sensitive radiation monitor that is capable of warning personnel of high radiation levels shall be on the main floor of the Confinement Building and over the pool.

3.7.1.1.4 If the detector described in specification 3.7.1.1.3 fails, within one hour, place a suitable gamma sensitive alternative meter with alarming capability meeting all of the requirements as the detector originally used to satisfy 3.7.1.1.3 in service.

TS 3.7.1.1.1 requires a minimum of one confinement gaseous and one confinement particulate effluent radiation monitor that is capable of warning personnel of high airborne radiation levels in the confinement. These monitors are required to be in operation whenever the reactor is operating, irradiated fuel handling is in progress, experiment handling is in progress for an experiment that has a significant fission product or gaseous effluent product inventory, any work on the core or control rods is in progress that could cause a reactivity change in excess of 0.60percent $\Delta k/k$, or any experiment movement that could cause a reactivity change of more than 0.60 percent $\Delta k/k$ is in progress. This specification helps to ensure that releases of radioisotopes into containment will be detected, and the RO will be provided the information needed for initiating the containment mode change from normal to emergency. The NRC staff finds that this specification supports the facility monitoring system, the facility ALARA program, and is consistent with the monitoring commitments in the SAR. Based on the information above, the NRC staff concludes that TS 3.7.1.1.1 is acceptable.

TS 3.7.1.1.2 establishes that if the TS 3.7.1.1.1 detector fails, then it is permissible to replace it. If any of the detectors described in Specification 3.7.1.1.1 fail during operation, a suitable alternative gaseous or particulate air monitor may be used, or an hourly grab sample analysis may be made in lieu of having a functioning particulate monitor. This specification allows operation to continue as long as the functional requirements continue to be met. The NRC staff finds that this specification supports the facility monitoring system and related commitments, including ALARA, and is consistent with the monitoring commitments in the SAR. Based on the information above, the NRC staff concludes that TS 3.7.1.1.2 is acceptable.

TS 3.7.1.1.3 requires that a minimum of one gamma sensitive radiation monitor that is capable of warning personnel of high radiation levels be over the pool and on the main floor of the confinement building. These monitors are required to be in operation whenever the reactor is operating, irradiated fuel handling is in progress, experiment handling is in progress for an experiment that has a significant fission product or gaseous effluent product inventory, any work on the core or control rods is in progress that could cause a reactivity change in excess of 0.60 percent $\Delta k/k$, or any experiment movement that could cause a reactivity change of more than 0.60 percent $\Delta k/k$ is in progress. This specification helps to ensure that abnormal radiation fields will be detected providing the RO with information needed for initiating the containment mode change from normal to emergency. The NRC staff finds that requiring operation of these monitors supports the facility ALARA program, and is consistent with the monitoring commitments in the SAR. Based on the information above, the NRC staff concludes that TS 3.7.1.1.3 is acceptable.

TS 3.7.1.1.4 establishes that if a detector required by TS 3.7.1.1.3 fails, it is permissible to replace it within one hour of failure with a suitable gamma sensitive alternative meter with alarming capability. This specification allows operation to continue as long as the functional requirements of TS 3.7.1.1.3 continue to be met. The NRC staff finds that this substitution is

consistent with the monitoring commitments in the SAR. Based on the information above, the NRC staff concludes that TS 3.7.1.1.4 is acceptable.

TS 3.7.1.2 states:

3.7.1.2 Radiation Monitoring System Alarm Set Points

Radiation monitor alarm set points shall be established as follows:

3.7.1.2.1 The stack gaseous monitor shall alarm when radiation levels of the stack gas are 2.5 times normal levels, or greater.

3.7.1.2.2 The stack particulate monitor shall alarm when radiation levels of the stack particulates are 2 times normal levels, or greater.

3.7.1.2.3 The area radiation monitors shall alarm when radiation levels are 2 times normal levels, or greater.

TS 3.7.1.2.1 requires that the stack gaseous monitor alarm setpoint may be no higher than 2.5 times the normal gaseous radiation level such that the monitor will alarm when the radiation level is 2.5 times normal or greater.

TS 3.7.1.2.2 requires the stack particulate monitor alarm setpoint may be no higher than 2 times the normal gaseous radiation level such that the monitor will alarm when the radiation level is 2 times normal or greater.

TS 3.7.1.2.3 requires the area radiation monitor alarm setpoint may be no higher than 2 times the normal gaseous radiation level such that the monitor will alarm when the radiation level is 2 times normal or greater.

In the bases for these specifications, the licensee explains that the purpose of defining set points in terms of “normal” radiation levels is to account for the fact that the levels vary in the confinement room depending on the kind of experiments that are being performed. The NRC staff accepts the position of the licensee that ability to adjust the setpoints to conform to known conditions is necessary so alarm conditions serve the purpose of warning staff of abnormal conditions. The process used will be described in procedures that can be reviewed during NRC inspections. Based on the information above, the NRC staff concludes that TS 3.7.1.2 is acceptable.

TS 4.7.1 states:

4.7.1 Required Radiation Monitoring Systems

4.7.1.1 The following radiation monitors shall be operable each day prior to the reactor being started up from the shutdown condition, and after the channel has been repaired:

4.7.1.1.1 At least one experimental level area radiation monitor

4.7.1.1.2 At least one pool top area radiation monitor

4.7.1.1.3 The gaseous effluent air monitor

4.7.1.1.4 The particulate air monitor

4.7.1.2 The following radiation monitors shall be channel calibrated and channel tested annually:

4.7.1.2.1 The experimental level area radiation monitor

4.7.1.2.2 The pool top area radiation monitor

4.7.1.2.3 The gaseous effluent air monitor

4.7.1.2.4 The particulate air monitor

TS 4.7.1.1 requires a surveillance to verify that the radiation monitors are operable each day prior to the reactor being started up from the shutdown condition, and after the channel has been repaired. This specification helps to ensure that the monitors required to detect radiation are suitable for use. The NRC staff finds that this specification is consistent with assumptions in SER Sections 7.7, 11.1.4, and 13.1, and the surveillance frequency is consistent with the guidance in ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.7.1.1 is acceptable.

TS 4.7.1.2 requires a surveillance to verify that the radiation monitor instrumentation be channel tested and calibrated. This specification helps to ensure that the monitors are providing indications of radiation levels that are properly related to true values. The NRC staff finds that this specification is consistent with the guidance in ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.7.1.2 is acceptable.

The NRC staff reviewed the type and locations of required radiation monitoring instrumentation and finds them consistent with the guidance of ANSI/ANS-15.1-2007 and appropriate for alerting facility personnel of changes in radiation levels that could indicate problems with reactor operation or use. The NRC staff finds the designs and operating principles of the I&C of the radiation detectors and monitors have been described, and are applicable to the anticipated sources of radiation. Based on the information above, the NRC staff concludes that the RMS described in the SAR give reasonable assurance that dose rates and effluents at the facility will be acceptably detected, and that the health and safety of the facility staff and the members of the public will be acceptably protected.

7.8 Conclusions

The NRC staff finds that the reactor control systems are adequately designed and implemented to provide safe and reliable startup, operation, and shutdown of the reactor during normal operation. Further, the NRC staff finds that the RPS is adequate to protect the SL on fuel temperature and maintain the reactor in a state as analyzed in the accident analysis. The NRC staff also finds that the RMS is adequately designed and detectors appropriately located to assure that ROs will be appropriately warned when abnormal radiation levels are detected.

Based on the information above, the NRC staff concludes that the I&C systems of RINSC reactor are consistent with the guidance of NUREG-1537 and are sufficient for continued safe reactor operation within the related limits of the facility license and TSs.

8. ELECTRICAL POWER SYSTEMS

8.1 Normal Electrical Power Systems

The supply of normal electrical power to the RINSC is described in SAR Section 8.2 (Ref 2). Normal electrical power is supplied by the local electric utility to a transformer onsite. The power is then stepped down for local needs. The larger loads, such as coolant pumps, use 480 volt alternating current (VAC). Smaller loads use either 220 VAC or 110 VAC. The RCS uses 24 volt direct current (VDC) power, which is converted from the onsite 110 VAC.

Power cables to the shim safety blade drive mechanisms and other equipment and instruments on the reactor bridge are routed through a fixed standpipe at the top of reactor shield assembly. The standpipe and power cabling observed during a facility walkdown were noted to be of sufficient cable length to easily reach all potential locations of the bridge structure.

As described in SER Chapter 4, reactor shutdown is passive and fail-safe in that if normal power is lost, the shim safety blades automatically fall into the core due to gravity, shutting down the reactor. Loss of normal electrical power initiates start-up of the emergency electrical power system.

The NRC staff reviewed the normal power system and finds it adequate to support normal operations at the RINSC. The RINSC reactor will shut down with loss of normal power and does not rely on normal power for safe shut down. TSs are not required for normal electrical power, since normal electrical power provides no safety-related functions. Based on the information above, the NRC staff concludes that the design of the normal electrical power system is acceptable.

8.2 Emergency Electrical Power Systems

Supplemental information from the licensee (Ref. 5) clarifies and updates the information supplied in the SAR regarding the emergency power source (EPS). The EPS is available to power the emergency exhaust system, should a power failure occur at the same time a radiological release is occurring in confinement. After about two hours sufficient air exchange has occurred in confinement that the emergency exhaust is no longer required to function. The emergency exhaust system is discussed in more detail in chapter 13 of this SER. The EPS is not utilized or required for cooling of the reactor fuel. On a loss of power to the site, the reactor automatically scrams and sufficient decay heat is removed, by natural convection, from the open pool to prevent fuel damage. Chapter 5 of this SER discusses the reactor core cooling in more detail.

The EPS is depicted in Figure 8-1 below. The EPS uses commercial quality equipment to supply 15 kW of power from a propane gas source. The loads presented to the EPS are listed in Table 8-1. The starting load is less than the rating of the generator. According to SAR Section 8.3 (Ref. 2), there is a 10-second time delay associated with the automatic start and switchover from normal to emergency power.

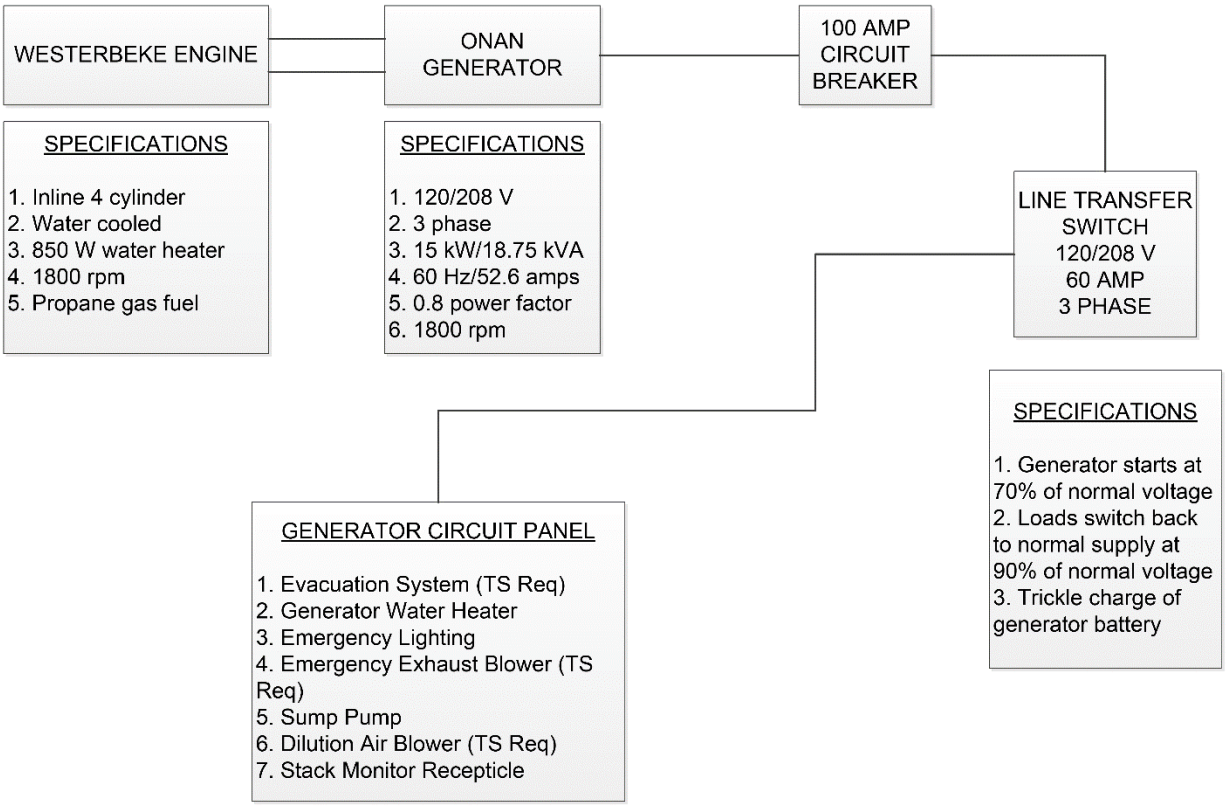

Figure 8-1 Emergency Power Source

Table 8-1 Emergency Generator Loads

Emergency Generator Loads		
Load	Run Amps	Full Load Starting Amps
Evacuation System	0.4	0.4
Generator Water Heater	0.2	0.2
Emergency Lighting	4.2	4.2
Emergency Exhaust Blower	3.5	8.9
Sump Pump	8.0	8.0
Dilution Air Blower	16.0	18.4
Stack Monitor Receptacle	5.6	5.6
Total Amps	37.9	45.7

The manufacturer's fuel consumption table shows that the RINSC generator uses approximately 110 ft^3 of propane vapor per hour of operation at full load. The licensee determined that the generator uses about 3 gal of liquid propane per hour. There are two propane tanks that each have the capacity for 100 gal of liquid propane. The maximum amount of time that the generator could operate under full load was determined by the licensee to be about 67 hours. If the tanks are kept at least half full, there will be enough fuel to run approximately 30 hours, which is longer than the time required should an airborne radiological release occur during a power failure.

TS 3.6, “Emergency Power System,” requires that the emergency electrical power system be operable for reactor operation and whenever the confinement system is required to be operable.

TS 3.6 and 3.6.1 state:

> 3.6 Required Emergency Power System
>
> Whenever the following operations are in progress:
>
> - The reactor is operating.
> - Irradiated fuel handling is in progress.
> - Experiment handling is in progress for an experiment that has a significant fission product, or gaseous effluent activation product inventory.
> - Any work on the core or control rods that could cause a reactivity change of more than 0.60 %Δk/k is in progress.
> - Any experiment movement that could cause a reactivity change of more than 0.60 %Δk/k is in progress.
>
> 3.6.1 The Emergency Power System shall be operable.

TS 3.6.1 requires an emergency electrical power source to be available when the stated operations are underway. This specification helps to ensure that emergency power is available to support the operation of the confinement system so that it can perform all required functions in the event of a power outage. According to SAR Section 8.3, a 15-kW diesel generator will start up after a 10-second delay upon loss of normal electric power. The generator will supply the equipment listed in SER Section 8.2 with electricity so that they will continue to operate so that the CVS will continue to provide negative pressure on the reactor building. SAR Section 8.3 states that the emergency generator will automatically start and assume the loads upon a loss of normal electrical power. The automatic bus transfer switch will transfer the load to the generator and back to the normal circuits again as normal electric power is restored. The NRC staff finds that the inter-related design features of the EPS and the reactor containment helps to ensure that in the event of a power outage the intended reactor building conditions will be maintained. Based on the information above, the NRC staff concludes that TS 3.6 and TS 3.6.1 are acceptable.

TS 4.6 states:

> 4.6 Emergency Power System
>
> 4.6.1 It shall be verified that the Emergency Power System is operable at least daily prior to any of the following conditions:
>
> 4.6.1.1 The reactor is operating.
>
> 4.6.1.2 Irradiated fuel handling is in progress.
>
> 4.6.1.3 Experiment handling is in progress for an experiment that has a significant fission product, or gaseous effluent activation product inventory.
>
> 4.6.1.4 Any work on the core or control rods that could cause a reactivity change of more than 0.60 %Δk/k is in progress.

4.6.1.5 Any experiment movement that could cause a reactivity change of more than 0.60 %Δk/k is in progress.

4.6.2 Perform an operability test to verify that the Emergency Power System starts and loads (see TS 4.5.3) in the event of a facility power outage.

4.6.2.1 Quarterly

4.6.2.2 Following emergency system load changes

4.6.3 It shall be verified that the available fuel for the emergency generator is at least 50% of full capacity.

4.6.3.1 Monthly

TS 4.6.1 requires a surveillance to verify the EPS is operable, in the event of a power outage. This specification helps to ensure that the "start" portion of the requirement for the EPS to start and take load is accomplished. This requirement is consistent with the guidance recommended by ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.6.3 is acceptable.

TS 4.6.2 requires a surveillance to verify the EPS to takes on electrical load in the event of a power outage. This specification helps to ensure that the "take load" portion of the requirement for the EPS to start and take load is accomplished. This requirement is consistent with the guidance recommended by ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.6.2 is acceptable.

TS 4.6.3 requires a surveillance to verify the fuel tank levels for the EPS. As explained in SER Section 8.2, 50 percent of a full fuel capacity allows about 30 hours of operation under full load. The NRC staff finds that this is more than sufficient to satisfy operability requirements for the ventilation system, should it be required to operate using this power source, and is consistent with the accident analysis in SER Chapter 13. Based on the information above, the NRC staff concludes that TS 4.6.3 is acceptable.

8.3 Conclusions

The NRC staff has reviewed the design bases, functional characteristics, and safety analysis relating to the normal and emergency power supplies and concluded that the systems will provide necessary service. This NRC staff finds that:

- The design bases and functional characteristics of the EPS have been reviewed, and the system is capable of providing the necessary replacement power.
- The design and operating characteristics of the RINSC emergency electrical power source is typical of those used in similar applications and have been demonstrated as reliable.
- The TSs, including surveillance and testing, provide reasonable assurance of necessary system operability and availability.

Based on the information above, the staff concludes that emergency electrical power source is supports the safe operation of the facility.

9. AUXILIARY SYSTEMS

9.1 Heating, Ventilation, and Air-Conditioning Systems

As described in SAR Section 9.1.1, as supplemented (Ref. 5), and as observed during the facility walkdown, air from the dilution blower portion of the CVS takes a suction from the office portion of the facility, the retention tanks, and the stack monitor return and is exhausted into the stack. The dilution blower is required to operate whenever the CVS is in operation regardless of the mode (normal or emergency) since it dilutes the potentially radioactive exhausts from other sources and it also prevents backflow of radioactive exhaust.

As described in SAR Section 9.1.1, during normal operation the CVS allows fresh air to enter the building through the intake damper and a pair of balancing dampers. The balancing damper positions are fixed and are used to adjust intake flow to maintain a minimum of -0.5 water column inches across the confinement building envelope. The intake air is not temperature controlled. The control room is equipped with a small air conditioning unit which recirculates cooled air for operator comfort and to help maintain control room humidity levels. The confinement building main floor has several local heating elements that utilize forced hot water with a temperature-controlled fan for heating. The confinement exhaust blower takes air from the pool level of confinement during normal operation; it also takes a suction on the delay tank vent, the off-gas blower exhaust and the rabbit blower exhaust. It exhausts through the exhaust damper to the stack where it mixes with the dilution blower exhaust before being discharged. There is a sample tap on the normal exhaust line upstream of the exhaust damper. This sends a sample of the effluent to the stack monitor where it is monitored for the release of both particulate and gaseous activity, and is then returned to the stack via the dilution blower portion of the ventilation system. The steel constructed exhaust stack is 35 m (115 ft) in height and provides an elevated release point. A detailed discussion CVS operation is provided in SER Section 6.2.1.

The emergency electrical power system is capable of providing power to all of the equipment required to initiate the evacuation system and the emergency mode of operation for the CVS in the event of a loss of offsite power.

Based on the information above, the NRC staff concludes that the heating, ventilation, and air conditioning system is sufficient to maintain acceptable conditions for personnel and equipment.

9.2 Handling and Storage of Reactor Fuel

Upon receipt, new fuel is physically examined in accordance with RINSC procedures. Following acceptance, the new fuel is stored in a new fuel safe or underwater in the pool storage racks. All reactor fuel is stored or used in a controlled access area in accordance with the RINSC Security Plan. Fuel handling activities are conducted by trained qualified operators under the direct supervision of a Senior Reactor Operator. Specially-designed tools are used to move fuel elements. These tools have locking features for gripping fuel elements and minimizing the potential for dropping elements. ALARA practices are used to receive, inspect, and store fuel.

TS 5.3 states:

5.3 Reactor Fuel Storage

5.3.1 All dry new fuel storage facilities shall have a configuration where k_{eff} is less than 0.8 under water flooded conditions.

5.3.2 A maximum of four fuel elements shall be stored in the fuel safe with no two elements in adjacent positions in the storage rack or in adjacent rows.

5.3.3 All irradiated fuel and experimental fissionable material not installed in the reactor core shall be stored in the reactor pool in storage racks in a configuration that ensures adequate cooling and is designed to maintain k_{eff} less than 0.9 under all conditions of moderation and reflection.

TS 5.3.1 requires that all new dry fuel storage locations have a configuration requiring that fuel elements stored there will have a flooded k_{eff} that is less than 0.8. This specification helps to ensure that stored fuel will be maintained sufficiently subcritical. The NRC staff finds that these design features are consistent with the guidance of NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 5.3.1 is acceptable.

TS 5.3.2 establishes the maximum storage limit of 4 new fuel elements in the new fuel safe and prohibits the storage any two fuel elements in storage rack positions adjacent to each other. This specification helps to ensure that stored fuel in the new fuel safe will be maintained sufficiently subcritical. Further technical demonstration of the specification effectiveness at preventing inadvertent criticality is provided in Rhode Island Nuclear Science Center Conversion from HEU to LEU Fuel (Refs. 23, 24) which discusses the approach to critical with LEU fuel. During the initial fuel loading and criticality it was demonstrated that a k_{eff} equal to 1.0 required 12 LEU fuel elements, symmetrically loaded around the flux trap, with optimal conditions of moderation and reflection. The NRC staff finds that these design features are consistent with the guidance of NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 5.3.2 is acceptable.

TS 5.3.3 establishes the configuration requirements for in-pool fuel storage. SAR Section 9.2.3.2 states that the fuel storage racks are made of sandwiched aluminum-cadmium plates. All materials of construction are chemically compatible with reactor in-core components and can also accommodate the graphite and beryllium reflectors and aluminum radiation baskets. The fuel racks have a k_{eff} less than 0.8 and are safe when used with the LEU fuel. These specifications help to ensure that stored fuel will be sufficiently subcritical. The NRC staff finds that these design features are consistent with the guidance of NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 5.3.3 is acceptable.

Based on its review, the NRC staff finds that the licensee's methods, analyses, and systems for storage of new and irradiated fuel are sufficient to prevent criticality (k_{eff} not to exceed 0.90) under all conditions of moderation and reflection and that the TSs define controls on fuel storage that are appropriate.

Based on the information above, the NRC staff concludes that the controls to handle and store fuel are adequate to prevent doses from exceeding 10 CFR Part 20 requirements and maintain

facility personnel doses ALARA. Further, the TSs related to the handing of reactor fuel meet the guidelines of ANSI/ANS-15.1-2007.

9.3 Fire Protection Systems and Programs

The fire protection system for the RINSC is described in SAR Section 9.3. The reactor building is of concrete construction with structural steel. These materials are fireproof in nature. The fire protection program consists of both detection and mitigation equipment and includes portable fire extinguishers, smoke alarms, pull stations, smoke detectors, and fuse activated sprinklers, which are located throughout the facility as noted on the facility walkdown by the NRC staff. A fire hydrant was observed within 50 ft from the facility. Additional detection capability is provided by an Aerotherm system, which consists of a small-diameter copper tube that is placed around the reactor biological shield. Local increases in temperature, indicative of a fire, will result in a rise in pressure in the copper tube, which trips local and remote alarms.

As stated in SAR Section 9.3, offsite agreements are in place with local fire departments, police departments, and other agencies as part of the facility EP. TS 6.4, “Procedures,” requires the facility to have a procedure to implement the EP.

Based on its review, the NRC staff finds that the plans for preventing fires helps ensure that the facility meets fire and building codes, and that systems that are designed to detect and combat fires at the facility can function, as described, and limit damage and consequences at any time.

Based on these observations and the information above, the NRC staff concludes that systems designed to detect and combat fires at the facility can function as described in the SAR and limit damage and consequences, and the potential radiological consequences of a fire will not prevent safe reactor shutdown. The NRC staff reviewed the RINSC EP and finds that there are procedures to support immediate response and notification of a fire. The appropriate sections of the facility EP adequately address any fire-related release of radioactive material from the facility to the unrestricted environment. Additionally, an agreement is in place acknowledging the commitment from the local fire department and other emergency response organizations. Based on the information above, the NRC staff concludes that the Fire Protection Program is acceptable.

9.4 Communication Systems

Communication systems at the RINSC are described in SAR Section 9.4. Telephones are in place throughout the facility, as observed by the NRC staff during facility visits. In addition, the facility has a general paging system. Communication over the paging system is possible using any phone. In addition, as stated in SAR Section 9.4, the facility has walkie-talkie radios. These units have a complete range within the facility, as well as the EPZ. The general paging system is used for notification of personnel during weekly tests of the EES.

Based on its review, the NRC staff finds that the facility communication systems are designed to provide two-way communication between the reactor control room and all other locations necessary for safe reactor operation; the communication systems allow the RO on duty to communicate with the supervisor on duty and with health physics personnel; and the communication systems allow a facility-wide announcement of an emergency. Based on the information above and observation of use of the system during facility visits, the NRC staff concludes that the communications systems are adequate to provide communications between

the control room and all other RINSC locations, communications between operators and health protection staff, and communications to summon emergency assistance.

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

The SAR did not identify any other license for the possession and use of byproduct and SNM. All materials possessed under the R-95 license are controlled under the Radiation Protection Program, as evaluated and found acceptable in SER Chapter 11. The operating license conditions, pursuant to the Atomic Energy Act, as amended, 10 CFR Part 30 and 10 CFR Part 70, include the following authorizations:

- to receive, possess, and use, but not separate, in connection with the operation of the facility, up to 10 kilograms of contained uranium-235 enriched to less than 20 percent in the form of MTR-type reactor fuel;
- to receive, possess, and use, but not separate, in connection with the operation of the facility, up to 32 grams of plutonium encapsulated in two plutonium-beryllium neutron sources for reactor startup;
- to receive, possess, and use, but not separate, in connection with the operation of the facility, up to 40 grams total of special nuclear material, of any enrichment, in the form of detectors, fission plates, foils, and solutions;
- to receive, possess, and use, but not separate, in connection with the operation of the facility, such special nuclear material as may be produced by the operation of the facility;
- to receive, possess, and use, in connection with the operation of the facility, a sealed antimony-beryllium neutron startup source; and,
- to receive, possess, and use, in connection with operation of the facility, such byproduct material as may be produced by operation of the reactor, which cannot be separated except for byproduct material produced in non-fueled reactor experiments.

As all materials are covered by a single license, the NRC staff concludes that the facility and procedure program adequately address management of byproduct materials. Further, the NRC staff concludes that design features and RPPs prevent and minimize exposure to workers and members of the public in the unrestricted environment. Based on the information above, the NRC staff concludes that the above license conditions are acceptable.

9.6 Cover Gas Control in Closed Primary Coolant Systems

The RINSC does not use a reactor cover gas. The relatively low reactor power and open pool results in negligible disassociated hydrogen accumulation.

9.7 Other Auxiliary Systems

9.7.1 Building Water System

The water supply system for the RINSC is described in SAR Section 9.5. Water is supplied to the facility from a 300,000-gallon storage tank that serves the Narragansett Bay Campus of the University of Rhode Island. Water is supplied through a 20-cm (8-in) line, which supplies fire water and also other facility needs. Domestic water pumps used to maintain the water supply at the campus all have emergency backup power.

The NRC staff reviewed the design and operation of the building water system and finds it adequate to supply water needs at the RINSC.

9.7.2 Reactor Building Overhead Crane

SAR Section 9.2.2.6 describes the design and construction of the reactor building overhead crane. The crane has a capacity of 15 tons and is supported by the building structure. The major function of the crane is to move fuel casks. Facility procedures limit the travel path of the cask with relation to the reactor core when moving heavy shipping casks. This action minimizes the risk of dropping an object or the failing of the crane structure with the reactor under the crane.

The NRC staff reviewed the design and operation of the reactor building overhead crane and compared it to similar cranes at other RTRs. The design and use restrictions are consistent with other facilities and the NRC staff concludes that the design and operation of the crane is adequate to perform lift functions at the RINSC. Furthermore, procedures for use of the crane have identified the conditions under which it may have an effect on the reactor and provide appropriate limitations to provide reasonable assurance that the reactor is protected from operation of the crane.

9.8 Conclusions

The NRC staff reviewed the auxiliary systems, as described in SAR Chapter 9 and licensee's responses to RAIs, and finds that the systems are designed to perform the functions required by the design bases. The NRC staff also finds the design of the systems considers functions and potential malfunctions that could affect reactor operations and no analyzed functions or malfunctions could initiate a reactor accident, prevent safe reactor shutdown, or initiate uncontrolled release of radioactive material. Additionally, the TSs provide assurance that fuel elements are appropriately handled and that there is no significant risk to the health and safety of the public from the storage and movement of fuel. Based on the information above, the NRC staff concludes that auxiliary systems at the RINSC support safe operation of the facility.

10. EXPERIMENTAL FACILITIES AND UTILIZATIONS

10.1 Summary Description

Chapter 10 of the SAR describes the RINSC experimental facilities that are used to provide irradiation services to researchers and commercial entities, as well as for education. The main purpose of the RINSC is to provide neutrons to the experimental facilities. Other experimental facilities are designed to provide mainly gamma-rays. Examples of current experimental applications at the RINSC include neutron activation analysis and radiography. Various TSs provide limitations on experiment reactivity and materials and means for technical and safety review of experiments.

10.2 Experimental Facilities

SAR Section 10.2 describes, in general terms, the following experimental facilities:

- six beam ports
- the through-port
- the thermal column
- the dry gamma room
- radiation baskets
- the flux trap
- the pneumatic transfer system
- non-reactor irradiation facilities

10.2.1 Beam Ports

SAR Section 10.2.1 describes the design and construction of the beam ports. There are a total of six beam ports. The RINSC utilizes one 20.3-cm (8-in) and two 15.24-cm (6-in) beam ports on the north side of the reactor and a similar set on the south side. The tubes are constructed of aluminum and are secured with a lead-filled aluminum shutter when not in use. Unused beam ports are sealed with stepped concrete plugs with a bolted outer cover. Each port has both a drain connected to the waste water retention facility and a gas vent connected to the off-gas removal system.

The beam ports are used to channel neutrons and gamma radiation from the reactor core to target areas outside the reactor and biological shield. These target areas are shielded to minimize dose to facility personnel. The beam ports are shielded by dense concrete plugs and steel doors to provide radiation protection for facility personnel. Experiments using the beam ports are subject to the review process described in SER Section 10.3.

Pool leakage through beam ports is analyzed as a potential accident in SAR Section 13.2.3 and is evaluated and found acceptable in SER Section 13.3. Following shutdown from full power operation for an infinite period, the fuel has sufficient decay heat for about 4.5 hours to potentially damage the cladding in the event of being uncovered. Applying actual reactor operating history, as documented in annual reports (Ref. 16), results in a significant decay heat

retention time period shorter than 4.5 hours. TS 3.9.3.1 provides administrative controls limiting access to beam ports for 4.5 hours after shutdown.

10.2.2 Through Port

SAR Section 10.2.2 describes the design and construction of the through-port. The through-port is a 6-in diameter horizontal aluminum tube that passes beneath the reactor and through the pool and biological shield. As the through-port is lower than the reactor, procedural controls are in place to ensure that the flanges are secure prior to operation. TS 3.9.3.1.3 requires gate valves to be installed whenever the through-port is in use.

Irradiations using the through-port are similar to those using the beam ports and are subject to the experiment review process described in SER Section 10.3.

10.2.3 Pneumatic System

SAR Section 10.2.3 describes the design and construction of the pneumatic tube system (also called the “rabbit” system). Supplemental information in response to RAI 10.5 (Ref. 3) described the relocation of the pneumatic tube system terminus to a new location adjacent to the reactor building, but outside the confinement boundary. This pneumatic transfer system is available to insert samples adjacent to the reactor core. The sample containers are commonly known as “rabbits,” and are transferred using air from a blower system. Two separate tubes are available from a single sending/receiving station.

As the “rabbit” irradiation location is near the core, irradiated materials are subject to the TS reactivity value limits and the material limits and must be reviewed before insertion as described in SER Section 10.3.

10.2.4 Thermal Column

SAR Section 10.2.4 describes the design and construction of the thermal column. This structure contains up to 2.44 m (8 ft) of graphite in a 1.42 m by 1.42 m (56 in by 56 in) cross-section. The column includes a lead shield adjacent to the core to reduce gamma radiation. The structure is connected to the off-gas system and the ventilation stack that allow the removal of radioactive gases generated in the thermal column.

Access to the face of the thermal column requires movement of the thermal column door. This heavy concrete door has stepped edges to minimize radiation streaming. Since the thermal column face is located away from the core, material to be irradiated does not have a reactivity effect on the core. However, experiments are required to be evaluated as described in SER Section 10.3.

10.2.5 Dry Gamma Room

SAR Section 10.2.5 describes the design and construction of the dry gamma room. This room is located within the biological shield and provides gamma irradiation for larger samples than is available from other facilities at the RINSC. The room is adjacent to the low-power end of the reactor pool. Gamma radiation from the core passes through an aluminum window for irradiating samples. As stated in SAR Section 10.2.5, this facility is currently not in use and both personnel access and the vent line have been sealed shut. However, efforts are ongoing to refurbish the dry gamma room and return it to service as an irradiation facility. Since the dry

gamma room is located away from the core, material to be irradiated is not subject to the reactivity limitations of TS 3.1.1.3, but is required to go through the review process as described in SER Section 10.3.

10.2.6 Dry Gamma Tube

SAR Section 10.2.6 describes the design and construction of the gamma tube. This is a 7.62-cm (3-in) aluminum tube that extends from the reactor pool surface to a point adjacent to the fuel storage rack in the pool. As material is not placed adjacent to the core, it is not subject to the reactivity limitations of TS 3.1.1.3, but is required to go through the review process as described in SER Section 10.3.

10.2.7 Radiation Baskets

SAR Section 10.2.7 describes the design and construction of the radiation baskets. These are aluminum boxes that can be inserted in fuel and reflector element locations at the reactor core edge. Radiation baskets allow for in-core irradiations. Samples in the baskets are cooled by the reactor pool water. All TS and review requirements apply to use of the radiation baskets.

10.2.8 Flux Trap

SAR Section 10.2.8 describes the design and construction of the flux trap. The beryllium reflector at the center of the core has a plugged opening of 3.8 cm (1.5 in) in diameter and 74 cm (29 in) in length. The plug can be removed with fuel handling tools. Material to be irradiated in the flux trap is subject the TSs on experiments and must be reviewed as described in SER Section 10.3.

The following TSs and SRs apply to experimental facilities and the conduct of experiments:

TS 3.9.3.1 states:

> 3.9.3.1 Experimental Facility Configuration during Reactor Operation, Including a 4.5 hour period after shutdown
>
> Prior to reactor operation and for a period 4.5 hours after shutdown, the following experiment facility configurations will be established and maintained:
>
> 3.9.3.1.1 Each beam port shall have no more than a 1.25 inch diameter opening to confinement,
>
> 3.9.3.1.2 The drain valve from the through port shall be closed when the through port is in use.
>
> 3.9.3.1.3 When the through port is in use, gate valves shall be installed on the end(s) of the port that will be used for access.
>
> 3.9.3.1.4 When the through port is not physically manned and monitored, the ends of the through port shall be closed.

TS 3.9.3.1.1 requires a limit on beam port open areas to 1.25 in (3.18 cm) in diameter. This specification supports the LOCA analysis. This analysis demonstrates that as long as the pool level does not drain through an area greater than 1.48 square in (1.000645 m^2), which is

equivalent to a 1.37 in (0.000884 m^2) diameter opening, there will be sufficient time for the reduction in decay heat to prevent fuel temperature from exceeding the SL of TS 2.1. That analysis also shows that if any single beam port has a catastrophic failure, the remaining beam ports do not become pool drain pathways. Consequently, limiting the area of each experimental port that is open to confinement to 1.25 in (3.18 cm) in diameter is conservative. The NRC staff finds that this limitation is consistent with the LOCA analysis as reviewed in SER Section 13.3. Based on the information above, the NRC staff concludes that this specification is acceptable.

TS 3.9.3.1.2 requires that during reactor operation the through port drain valve shall be closed. The through port has a potential pool leak pathway by means of the drain valve. By keeping this drain valve closed during operation, that potential leak pathway is no longer a credible failure mode, and the potential for an unnoticed pool leak through this experimental facility is prevented. Based on the information above, the NRC staff concludes that this specification is acceptable.

TS 3.9.3.1.3 requires that when the through port is in use, gate valves shall be installed on the through port. The LOCA analysis has shown that the amount of time available for performing mitigating actions in the event of a non-catastrophic pool leak is on the order of hours. The consequence of a leak to the though port can be mitigated quickly by closing the gate valves. The NRC staff finds that this mitigating action is reasonable and achievable. Based on the information above, the NRC staff concludes that this specification is acceptable.

TS 3.9.3.1.4 requires that the gate vales to be closed when the through port is not being monitored for leakage. The NRC staff finds that this mitigating action is reasonable and achievable. Based on the information above, the NRC staff concludes that this specification is acceptable.

TS 4.9.3.1 states:

> 4.9.3.1 Experimental Facility Configuration during Reactor Operation, including a 4.5 hour period after shutdown.
>
> Prior to operating the reactor the following conditions shall be verified, these conditions shall be maintained for a period of 4.5 hours after shutdown:
>
> 4.9.3.1.1 Each beam port shall have no more than a 1.25 inch diameter opening to confinement.
>
> 4.9.3.1.2 The drain valve from the through port shall be closed when the through port is in use.
>
> 4.9.3.1.3 When the through port is in use, gate valves shall be installed on the end(s) of the port that will be used for access.
>
> 4.9.3.1.4 When the through port is not physically manned and monitored, the ends of the through port shall be closed.

TS 4.9.3.1.1 through 4.9.3.1.4 are facility-specific SRs for experimental facilities to verify the requirements of TS 3.9.3.1 are met prior to reactor operation. The NRC staff finds that this surveillance interval is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.9.3.1 is acceptable.

TS 3.9.3.2 states:

3.9.3.2 Experimental Facility Configuration Within the 4.5 Hour Period After Shutdown

If the experimental facility configuration specified in 3.9.3.1 cannot be maintained for 4.5 hours after the reactor is shutdown, the following actions shall be taken prior to changing the configuration required by 3.9.3.1:

3.9.3.2.1 The reactor shall be moved to the low power section of the pool where it is at the opposite end of the pool from the beam port extensions.

3.9.3.2.2 The pool dam shall be positioned so that the high power section of the pool is isolated in such a way that if a beam port extension were sheared off, the pool level in the low power section would not be affected.

TS 3.9.3.2.1 and TS 3.9.3.2.2 establish the requirements to move the reactor to the LP section of the pool and isolate it from the beam tubes using the pool dam under certain timing conditions. Moving the reactor to the low power end with the pool dam in place will ensure the coolant loss rate is less than that assumed in the LOCA which is a 3.18 cm (1.25 in) line shear. With the reactor in the low power end of the pool with the dam in place, a failure of a beam port line that is greater than the 1.38 cm (1.25 in) assumed in the LOCA would not drain the section of the pool containing the reactor in less than 4.5 hours. Based on the information above, the NRC staff concludes that TS 3.9.3.2.1 and TS 3.9.3.2.2 are acceptable.

TS 4.9.3.2 states:

4.9.3.2 Accessing an Experimental Facility Configuration Within the 4.5 Hour Period After Shutdown

4.9.3.2.1 Prior to changing the configuration required by 4.9.3.1, shall verify that the reactor has not operated in the previous 4.5 hours.

4.9.3.2.2 If changing the configuration required by 4.9.3.1 within 4.5 hours after reactor shutdown is absolutely required, then it shall be verified that the following actions have been completed:

4.9.3.2.2.1 The reactor is in the low power section of the pool, opposite the end of the pool where the beam port extensions are located.

4.9.3.2.2.2 The pool dam is positioned so that the high power section of the pool is isolated in such a way that if a beam port extension were sheared off, the pool level in the low power section would not be affected.

TS 4.9.3.2.2 is a facility-specific SR to verify the 4.5 hour shutdown period prior to changing the configuration described in TS 4.9.3.1. Specification 4.9.3.2 is a facility-specific SR to verify that the actions of TS 3.9.3.2 are performed prior to changing the configuration in TS 4.9.3.1 within a 4.5 hour period after shutdown. The NRC staff finds that this surveillance interval is consistent

with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.9.3.2 is acceptable

Conclusions

The NRC staff reviewed the description of the experimental facilities, as described in the SAR and RAI responses, and finds that the licensee has proposed and justified acceptable TSs for the experimental facilities activities in accordance with the guidance in NUREG-1537. Additionally, the NRC staff finds that the design and functional information in the SAR gives reasonable assurance that the experimental facilities are capable of retaining necessary integrity during all anticipated operations and postulated accidents and are secured appropriately. The NRC staff also finds that these facilities are typical of other RTR facilities. Furthermore, the NRC staff finds that the consequences of the malfunction or failure of an experimental facility are considered in the analyses of reactor accidents in SAR Chapter 13. Based on the information above, the NRC staff concludes that the experimental facilities are acceptable, can be used without damaging the fuel, and do not pose a significant risk to the health and safety of the public or facility personnel.

10.3 Experiment Review

SAR Section 10.3 describes the process for review of experiments. The Nuclear and Radiation Safety Committee (NRSC) is responsible for evaluation of new experiments. Composition of the NRSC is defined in TS 6.1 and further evaluated and found acceptable in SER Chapter 12. The NRSC includes the Assistant Director for Radiation and Reactor Safety. Utilization of experimental facilities require the approval of the Director of the RIAEC.

TS 3.1.1.3 states:

3.1.1.3 Experiments

3.1.1.3.1 The total absolute reactivity worth of experiments shall not exceed the following limits:

3.1.1.3.1.1 Total Moveable and Fixed 0.6 %Δk/k

3.1.1.3.1.2 Total Moveable 0.08 %Δk/k

3.1.1.3.2 The maximum reactivity worth of any individual experiment shall not exceed the following limits:

3.1.1.3.2.1 Fixed 0.6 %Δk/k

3.1.1.3.2.2 Moveable 0.08 %Δk/k

TS 3.1.1.3.1 and TS 3.1.1.3.2 require a limit on both the total absolute and single experiment reactivity worth. The reactivity limit of 0.6 %Δk/k is evaluated and found acceptable in SER Section 13.2. The licensee provided an analysis in response to RAI 13.7 (Ref. 3) that presents the consequences of rapidly inserting 0.6 %Δk/k of reactivity while the reactor is at power resulting in negligible fuel temperature increases in either forced or natural circulation modes. The NRC staff finds that the TS 3.1.1.3.1 and TS 3.1.1.3.2 are consistent with the analysis supplied and that analysis shows a minimal impact on fuel temperature increase. Based on the information above, the NRC staff concludes that TS 3.1.1.3.1 and TS 3.1.1.3.2 are acceptable.

TS 4.1.1.3 states:

4.1.1.3 Experiment Reactivity Limit

4.1.1.3.1 The reactivity worth of new experiments shall be determined prior to the experiments initial use.

4.1.1.3.2 The reactivity worth of any on-going experiments shall be re-determined after the core configuration has been changed to a configuration for which the reactivity worth has not been determined previously.

TS 4.1.1.3.1 requires a surveillance to verify the reactivity worth of new experiments prior to their initial use. This specification helps to ensure that new experiments that have an unknown effect on core reactivity are evaluated and are within the requirements of TS 3.1.1. The NRC staff finds that this surveillance interval is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.1.1.3.1 is acceptable.

TS 4.1.1.3.2 requires the reactivity worth of ongoing experiments after changes to the core configuration that could alter the previously established reactivity worth. This specification helps to ensure that changes to core configuration do not result in on-going experiments having unacceptable reactivity worth. The NRC staff finds that this surveillance interval is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that this specification is acceptable.

TS 3.8.1 states:

3.8.1 Experiment Materials

3.8.1.1 Corrosives Materials

Corrosive materials shall be doubly contained in corrosion resistant containers. If failed container is suspected, all fuel assemblies and reactor structural components should be inspected.

3.8.1.2 Highly Water Reactive Materials

Highly water reactive materials shall not be placed inside the reactor, the reactor pool, or inside any reactor experimental facility where exposure to water is possible.

TS 3.8.1.3 states:

3.8.1.3 Explosive Materials

Explosive materials shall not be placed inside the reactor, the reactor pool, or inside the reactor experimental facilities.

3.8.1.4 Fissionable Materials

3.8.1.4.1 The quantity of fissionable materials used in experiments shall not cause the experiment reactivity worth limits to be exceeded.

3.8.1.4.2 The maximum quantity of fissionable materials used in an experiment shall be no greater than 87.5 milligrams of U-235 equivalent, and the maximum fission rate in a fissionable experiment shall be no greater than 2.1 X 10^{12} fissions per second.

3.8.1.4.3 Fissionable materials shall be doubly encapsulated.

3.8.1.4.4 Containers for experiments that have fissionable material shall be opened inside confinement.

TS 3.8.1.1 requires that corrosive materials be doubly encapsulated in a corrosion resistant container. This specification helps to ensure that the irradiation of corrosive materials cannot lead to a failure that is chemically adverse to core components or experimental facility materials. The possibility of experiment failure is minimized by requiring that corrosive materials be doubly contained. If an encapsulation fails, fuel assemblies and reactor structural components will be inspected to help ensure that the corrosive material did not cause damage. The NRC staff finds that the cited conditions are appropriate and consistent with the safety analysis reviewed in SER Section 13.6. Based on the information above, the NRC staff concludes that TS 3.8.1.1 is acceptable.

TS 3.8.1.2 requires that materials reactive with water not be placed in the reactor, the pool, or inside reactor experimental facilities where exposure to water is possible. This specification helps to ensure that damage does not arise as a result of highly water reactive materials reacting with the pool water. It makes this scenario impossible by limiting the use of highly water reactive materials in experiments. The NRC staff finds that the cited conditions are appropriate and consistent with the safety analysis reviewed in SER Section 13.6. Based on the information above, the NRC staff concludes that TS 3.8.1.2 is acceptable.

TS 3.8.1.3 requires that explosive materials are not be placed in the reactor, the pool, or inside reactor experimental facilities. This helps to ensure that damage does not arise as a result of explosive materials detonating inside an experimental facility. The NRC staff finds that the cited conditions are appropriate and consistent with the safety analysis reviewed in SER Section 13.6. Based on the information above, the NRC staff concludes that TS 3.8.1.3 is acceptable.

TS 3.8.1.4 requires an upper limit on the quantity of fissionable material used in experiments, in order to limit reactivity effects and to ensure that the failure of a fissionable experiment could not cause 10 CFR Part 20 dose limits to be exceeded. The TS also requires double encapsulation of experiments, and requires that experiment containers are opened inside the confinement building. Failures of experiments that contain fissionable materials have the potential to have an impact on reactor criticality, or to cause a radioactive material release. This specification helps to ensure that consequences of the failure an experiment containing fissionable material will not exceed the consequences of reactivity transients or radioactive material releases that have been analyzed (see SER Sections 13.1 and 13.2). The specification also helps ensure that the likelihood of encapsulation failures is minimized by requiring double encapsulation. Opening experiment containers only in the confinement building allows the emergency features of the ventilation system to be used if an unanticipated release of radioactive material were to occur. The NRC staff finds that the cited conditions are appropriate and consistent with the safety analysis reviewed in SER Sections 13.1, 13.2, and 13.6. Based on the information above, the NRC staff concludes that TS 3.8.1.4 is acceptable.

TS 3.8.2 states:

3.8.2 Experiment Failures or Malfunctions

3.8.2.1 Experiment shall be designed to ensure that credible failure of any experiment will not result in releases or exposures in excess of limits established in 10 CFR Part 20.

3.8.2.2 Experiment shall be designed to ensure that no reactor transient can cause the experiment to fail in such a way that it contributes to an accident.

3.8.2.3 Experiment shall be designed to ensure that credible failure of any experiment will not contribute to the failure of:

3.8.2.3.1 Other Experiments

3.8.2.3.2 Core Components

3.8.2.3.3 Principle physical barriers to uncontrolled release of radioactivity

TS 3.8.2.1, 3.8.2.2, and 3.8.2.3 establish requirements for the review of experiment design. Their purpose is to ensure that experiments comply with generally accepted practices in experiment review: (1) that experiment failure cannot challenge exposure limits; (2) that transients cannot lead to failure that then causes an accident; (3) that failure cannot lead to cascading failures; and (4) that experiment flooding cannot result in the reduction of SDM. The specifications include specific criteria for evaluation and approval of experiments at the RINSC. All experiments are evaluated by the RINSC NRSC to help ensure that they will not result in exceeding the excess reactivity limit and SDM of TS 3.1.1.1, or in the release of fission products under normal or accident conditions that could result in offsite concentrations of radioactive material in excess of 10 CFR Part 20 concentration limits. These specifications help to ensure that experiments do not undermine or challenge the safe operation of RINSC. The NRC staff finds that these specification are consistent with the guidance of provided in ANSI/ANS-15.1-2007 and NUREG-1537. Based on the information above, the NRC staff concludes that TS 3.8.2.1, 3.8.2.2 and 3.8.2.3 are acceptable.

TS 4.8 states:

4.8 Experiments

4.8.1 Experiments shall be reviewed to ensure that the design is within the limitations of the RINSC Technical Specifications and 10 CFR Part 50.59 prior to the experiments initial use.

TS 4.8.1 requires a surveillance to verify that all experiments are reviewed prior to initial use. The NRC staff finds that these requirements comply with 10 CFR 50.59 and are consistent with the guidance in NUREG-1537. Based on the information above, the NRC staff concludes that TS 4.8.1 is acceptable.

TS 6.5 states:

6.5 Experiment Review and Approval

6.5.1 All new experiments shall be reviewed by the NRSC, and approved by level 1 or level 2 management prior to bringing the reactor to power with the experiment loaded.

6.5.2 Substantive changes to previously approved experiments shall be reviewed by the NRSC, and approved by level 1 or level 2 management prior to bringing the reactor to power with the experiment loaded.

6.5.3 Minor changes that do not significantly alter the experiment may be approved by a Senior Reactor Operator or level 1, 2, or 3 management.

TS 6.5.1 requires an administrative review and approval of new experiments. The NRC staff finds that this specification is consistent with the guidance provided in ANSI/ANS-15.1-2007, Section 6.5, as accepted in NUREG-1537. Based on the information above, the NRC staff concludes that TS 6.5.1 is acceptable.

TS 6.5.2 requires an administrative review and approval of previously approved experiments with substantial changes. The NRC staff finds that this specification is consistent with the guidance provided in ANSI/ANS-15.1-2007, Section 6.5, as accepted in NUREG-1537, as well as the provisions of Section C.3 of RG 2.2 and 2.4 as cited in NUREG-1537, Section 6.5. Based on the information above, the NRC staff concludes that TS 6.5.2 is acceptable.

TS 6.5.3 requires an administrative review and approval of previously approved experiments with minor changes. The NRC staff finds that this specification is consistent with the guidance provided in ANSI/ANS-15.1-2007, Section 6.5 as accepted in NUREG-1537 as well as the provisions of Section C.3 of RGs 2.2, “Development of Technical Specifications for Experiments in Research Reactors,” and 2.4, “Review of Experiments for Research Reactors,” as cited in NUREG-1537, Section 6.5. Based on the information above, the NRC staff concludes that TS 6.5.3 is acceptable.

The NRC staff compared the review process and criteria with that of NUREG-1537 and ANSI/ANS-15.1-2007 and similar research reactors. The review criteria and the composition of the NRSC meets the guidance of NUREG-1537 and ANSI/ANS-15.1-2007, and is consistent with the experiment review process at other similar reactors. The NRC staff notes that for reviews under 10 CFR 50.59, changes to experiments are to be treated as new experiments. The NRC staff concludes that TS 6.5 is sufficient to ensure that experiments are reviewed and approved prior to irradiation. Based on the information above, the NRC staff concludes that TS 6.5 is acceptable.

10.4 Conclusions

Based on the information above, the NRC staff concludes that the licensee has the proper controls in place to continue to implement the experimental program safely. The NRC staff concludes that the review and approval process for experiments and the use of experimental facilities provides reasonable assurance that appropriate precautions are taken to minimize the risk to personnel from unintended radiation exposure. Furthermore, the NRC staff concludes that the review process provides reasonable assurance that the use of experiments or experiment facilities in accordance with the TSs will not damage the fuel and will not pose a significant risk to public health and safety, facility personnel, or the environment.

11. RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

11.1 Radiation Protection

The RIAEC radiation protection program (RPP) is described in SAR Chapter 11. Activities involving radiation at the RINSC are controlled through the RPP, which the licensee states meets the requirements of 10 CFR 20.1101, "Radiation Protection Programs," and is designed to minimize radiation exposure. The regulations in 10 CFR 20.1101 specify, in part, that each licensee shall develop, document, and implement a RPP and shall use, to the extent practical, procedures and engineering controls based on sound radiation protection principles to achieve occupational doses and doses to members of the public that are ALARA. As stated in SAR Section 11.1.5.4, all individuals who are granted unescorted access to the reactor facility are trained to the requirements of the RIAEC RPP. The health physics staff has the authority to interdict or terminate the use of radioactive materials or radiation sources. The basic aspects of the RPP include occupational and public exposure limits, training, surveys and monitoring, personnel dosimetry, and reporting.

The NRC inspection program routinely reviews radiation protection and radioactive waste management at the RINSC. The licensee's historical performance in these areas, as documented in NRC IRs (Ref. 28), in the annual operating reports (Ref. 16) of the RINSC facility, in the SAR, as supplemented, and as observed by the NRC staff during site visits, provide evidence that measures are in place to minimize radiation exposure to RINSC staff and the public, and to provide adequate protection against operational releases of radioactivity to the environment.

11.1.1 Radiation Sources

SAR Chapter 11 describes the radiation sources, including inventories, physical forms, and locations. The RINSC RPP and waste management program monitor and control the radiation sources. These sources are categorized as airborne, liquid, or solid.

Airborne Radiation Sources

SAR Section 11.1.1.1, and SAR Appendix A (Ref. 43) discuss airborne radiation sources at the RINSC. The licensee states that of the airborne radiation sources that are produced during reactor operation, the two that are of principal significance are argon-41 (Ar-41) and nitrogen-16 (N-16). Ar-41 is produced by neutron activation of natural, stable argon, a normal trace element in the atmospheric air. Ar-41 is mainly produced in the air in the reactor beam ports, experiment sample positions, pneumatic transfer system, as well as the air dissolved in the reactor coolant water. N-16 is generated by the fast neutron activation of oxygen-16 in the reactor coolant water as it passes through the reactor core. The reactor confinement building exhaust system is designed to provide a slightly negative pressure (relative to atmospheric pressure) within confinement to help ensure that all radioactive gases are released through the 115-ft (35-m) exhaust stack. The off-gas system is designed to remove the gases from the experiment facilities and maintain concentrations in the reactor confinement below the 10 CFR Part 20 dose limits for workers. As indicated in SAR Sections 9.1.1 and 10.2, the pneumatic transfer system and beam ports have vents that connect to the off-gas system and direct radioactive gases to the facility stack. Air in the reactor building is monitored for radiation, as required by TS 3.7.1 (see SER Section 7.7).

SAR Section 11.1.1.1 states that although N-16 produces high-energy photons as it decays, the N-16 generated during reactor operation is not a significant concern. At LP (below a thermal power level of 0.1 MWt), when the reactor is operating with NC cooling, N-16 production is limited. Additionally, N-16 produced in the core when the reactor is operating with NC cooling must diffuse through 23 ft (7 m) of water before reaching the pool surface, and the time for this diffusion to occur is long compared to the approximately 7-second half-life of N-16. At higher power, when the reactor is operating with FC cooling, the coolant is passed through a delay tank and heat exchanger before it re-enters the reactor pool and core. The time spent in the delay tank is at least 90 seconds, ensuring that the water re-entering the pool and core is essentially N-16 free. Because of the N-16 in the primary coolant in the decay tank, typical contact dose rates on the delay tank are approximately 5 to 6 rem per hour during extended reactor operation at full-power. However, these dose rates are expected, and areas near the delay tank and other primary system components are posted and access limited to help minimize any potential exposure to reactor staff. The NRC staff reviewed the information above, and finds that since little or no N-16 escapes from the primary coolant during low- or high-power operation, and RINSC staff access to PCSs components containing N-16 is controlled, any occupational or public dose from N-16 produced during reactor operation is not significant.

The licensee's SAR discussion of Ar-41 production relies on actual historical measurements of Ar-41 generated by reactor operation and released to the environment, and also provides calculated offsite public doses from Ar-41. The licensee stated that historic generation rates of Ar-41 show that approximately 0.14 ± 0.03 Curies (Ci) of Ar-41 are produced and released per megawatt-hour of reactor operation.

Because of the Ar-41 retained in the pool and experimental facilities, a limited amount of Ar-41 can be found in the reactor room (confinement) during operation. As discussed in SAR Section 11.1.5.2, the design of the reactor ventilation system and off-gas system help minimize occupational doses from Ar-41. The off-gas blower removes gases from the thermal column, beam tubes, and pneumatic system, and discharges them into the suction line of the reactor room exhaust blower. In addition, the reactor room exhaust blower constantly exchanges the air from the reactor confinement building. The exhaust blower inlet plenum is located near the pool platform to essentially sweep air across the pool surface, which helps remove airborne activity at the pool surface. The exhaust discharge is released through the stack. The licensee does not specify any measured Ar-41 concentrations for the confinement building in the SAR. However, the licensee provided (Refs. 57, 58) gamma dose rate measurements from the area monitor on the main floor of the confinement building, taken during an approximately 12 hour period when the reactor was operating at full power. These measurements showed that the dose rate from all gamma radiation sources, including Ar-41, fluctuated but remained below approximately 1 mrem (0.01 mSv) per hour during full-power reactor operation. The dose rate reached the maximum level early in the approximately 12-hour period, demonstrating that the dose rates are representative for equilibrium Ar-41 levels in the building. Conservatively assuming that all of this dose is from Ar-41 in the reactor building, that the reactor operated at full power for an entire year (this is allowed by the facility license, but the actual historical reactor utilization has been much lower, as indicated by the RINSC annual reports (Ref. 16)), and that a member of the RINSC staff occupied the reactor floor for 2,000 working hours in a year, the annual dose to that worker from Ar-41 would be about 2,000 mrem. This is below the 5,000 mrem occupational dose limit in 10 CFR 20.1201, "Occupational Dose Limits for Adults."

The licensee performed a calculation, dated August 24, 2016, using the COMPLY computer code to estimate the public dose from Ar-41 released to the environment during facility

operation, which it provided as a SAR supplement (Ref. 5). The COMPLY computer code is routinely used for this type of calculation at research reactors. For its COMPLY calculation, the licensee used an Ar-41 release rate of 54.91 Ci per year, which was the quantity of Ar-41 produced and released during the period from July 2015 through June 2016, as reported in the RINSC annual report for that period (Ref. 16). The licensee input wind rose information such that the calculation would consider the fraction of the year that the wind at the facility blows in each compass direction. For all wind directions, the licensee assumed an annual average wind speed of 2 m per second. The licensee considered receptors located 100 m from the base of the 35-m-high facility stack, since it assumed that the maximally-exposed members of the public in each direction from the facility would be at these locations. Based on the Ar-41 release rate of 54.91 Ci per year, the licensee calculated an annual public dose of approximately 1.2 mrem (0.012 mSv) per year. The licensee states in SAR Section 11.1.1.1 that its calculations of public Ar-41 dose using the COMPLY computer code show that the projected dose to a maximally-exposed member of the public is approximately 0.021 mrem per Ci of Ar-41 released. Given the 10 mrem (0.1 mSv) constraint on annual public dose from airborne emissions of radioactive material in 10 CFR 20.1101(d), the licensee states that its calculations show the facility can release up to approximately 476 Curies of Ar-41 per year, which would occur in about 3,400 MWt-hours of operation (equivalent to 1,700 hours of operation at the full licensed power of 2 MWt).

The NRC staff reviewed the licensee's calculation discussed above, and also performed a confirmatory calculation of the public dose from Ar-41 effluents from RINSC operation. The NRC staff's analysis used the Pasquill-Gifford methodology, and assumed that an annual average wind speed of 2 m per second and neutral (Pasquill D) atmospheric stability conditions occur for the entire year (these are similar to conservative default meteorological conditions used by the COMPLY code). Although the licensee's analysis assumed that the maximally-exposed member of the public would be located 100 m from the stack, the NRC staff analysis considered locations at varying distances from the RINSC stack. The NRC staff analysis assumed that all of the locations could potentially be occupied by members of the public 100 percent of the year. This is a conservative assumption, because it considers that any location around the facility could be the full-time residence of a member of the public, while in actuality, portions of the area around the facility are non-residential or undeveloped. The NRC staff's analysis assumed that the wind blows in the direction from the stack to the receptor for 11 percent of the year, which is also a conservative assumption because, as indicated by the wind rose used for the RINSC site, winds at the facility do not blow in any one direction more than approximately 10.5 percent of the year on average. The NRC staff's analysis conservatively ignored building wake effects and plume meander. Considering that there is no limit on reactor operation in the RINSC license, the NRC staff conservatively assumed that the facility would release, in 1 year, the quantity of Ar-41 that would be produced if the reactor operated continually for the entire year at full power. Based on the Ar-41 generation rate of approximately 0.14 ± 0.03 Ci per MWt-hour, approximately 2,454 Ci of Ar-41 would be produced in 1 year of continual operation. Using these inputs and assumptions, the NRC staff calculated a maximum public Ar-41 dose of 3.6 mrem (0.036 mSv), received by a person located approximately 500 m from the facility stack (although the NRC staff's calculation did not consider the specific direction from the facility in which this individual would be located, it can be assumed that the individual maximally-exposed to Ar-41 would be northeast of the facility, since the wind most often blows from the southwest). The NRC staff's calculated dose is above the 1.2 mrem (0.012 mSv) calculated by the licensee for a 54.91 Ci release. The difference is due to the NRC staff's use of a higher annual release quantity (2,454 Ci vs 55 Ci), as well as the other differences in the methodologies and assumptions used in the NRC staff and licensee calculations. The NRC staff finds that although the annual doses calculated by the licensee and

the NRC staff differ, they are both well below the 100 mrem (1 mSv) public dose limit in 10 CFR 20.1301, "Dose Limits for Individual Members of the Public," and they are also below the 10 mrem (0.1 mSv) constraint on public doses from airborne emissions of radioactive material in 10 CFR 20.1101(d).

The licensee- and NRC staff-performed calculations discussed above did not include a calculation of the annual public Ar-41 dose at the specific location of the nearest residence to the facility. The nearest residences to the facility are located approximately 500 m west-northwest and south of the facility, as noted in a SAR supplement (Ref. 5). As noted above, the NRC staff calculated that a maximally-exposed individual located 500 m from the facility (assumed to be northeast of the facility, because the wind blows most often from the southwest to the northeast), would receive an annual Ar-41 dose of approximately 3.6 mrem (0.036 mSv). Since full-time occupancy was assumed for the maximally-exposed individual, and since the wind blows less often toward the west-northwest and the south than it does toward the northwest, the doses at the nearest residence would be bounded by the 3.6 mrem (0.036 mSv) dose calculated for the maximally-exposed individual. However, to show how the dose at the nearest residences would compare to the dose to the maximally-exposed individual, the NRC staff performed a calculation of the public Ar-41 doses at the nearest residences, using the licensee's wind rose information in a SAR supplement (Ref. 5). The NRC staff calculated that the annual doses at the residences located approximately 500 m west-northwest and south of the facility would be approximately 0.4 mrem (0.004 mSv) and 2.0 mrem (0.02 mSv), respectively.

TS 3.7.2.1 requires that the annual total effective dose equivalent to the individual member of the public likely to receive the highest dose from air effluents will be calculated using a generally-accepted computer program and will not exceed 100 mrem (1 mSv) per year. TS 3.7.2.1 helps ensure that Ar-41 releases from the RINSC do not result in doses that exceed the 100 mrem (1 mSv) public dose limit in 10 CFR 20.1301. TS 3.7.2.1 is discussed and found to be acceptable in SER Section 11.2.3.

The NRC staff reviewed the recent RINSC annual operational reports for the years 2009 through 2016 (Ref. 16), which provide the measured Ar-41 releases for each reporting period, along with the calculated public doses during the reporting periods. The highest Ar-41 release of 129.4 Ci occurred during the period from July 2009 through June 2010, and the licensee calculated that this release resulted in a maximum public dose of 2.7 mrem (0.027 mSv). The average annual Ar-41 release from 2009 through 2016 was 77.90 Ci, and the average annual dose calculated by the licensee was 1.7 mrem (0.017 mSv). The NRC staff finds that the information in the annual reports shows that, with respect to Ar-41 emissions, historical operation of the facility has been in compliance with both the 100 mrem (1 mSv) public dose limit in 10 CFR 20.1301, and the 10 mrem (0.10 mSv) constraint in 10 CFR 20.1101(d).

The NRC staff reviewed the information above, including the licensee's calculations of doses from Ar-41. The NRC staff confirmed the adequacy of the licensee's results and methodologies for calculation of doses from Ar-41, and also performed confirmatory calculations of the public dose from Ar-41 at the location of the maximally-exposed individual and at the nearest residences, as discussed above. The NRC staff finds that there is reasonable assurance that the routine airborne radiation sources and gaseous effluent releases of Ar-41 and N-16 meet the occupational dose limits in 10 CFR 20.1201 and the public dose limits in 10 CFR 20.1301. The NRC staff also finds that there is reasonable assurance that the licensee will operate the facility in compliance with the 10 mrem (0.10 mSv) ALARA constraint in 10 CFR 20.1101(d). Therefore, based on the information above, the NRC staff concludes that the control of airborne

radiation sources at the RINSC is acceptable, and that the licensee has adequately described airborne radiation sources at the RINSC such that the information is sufficient to evaluate the facility's RPP and controls described in the remainder of SER Section 11.1.

Liquid Radioactive Sources

According to SAR Section 11.1.1.3, the primary liquid radiation sources at the RINSC is the reactor coolant. The level of impurities in the reactor coolant water is maintained very low. A filter and demineralizer is used to maintain the water purity. TS 3.3.1.1, which is discussed and found acceptable in SER Section 5.4, requires that the reactor not be operated unless primary coolant conductivity is less than or equal to 2 μmhos per centimeter. Although the reactor pool water is kept quite clean, occasionally there may be activation products from contaminants in the water, which are generated when neutrons interact with tank and structural components and the resulting radioactivity is transferred to the water. The radioactivity in the coolant consists primarily of these activation products, most of which is ultimately deposited in the mechanical filter and the demineralizer resins. Radionuclides such as sodium-24 and manganese-56 are common examples of waterborne radioactivity created in this manner. As discussed earlier in this section, the reactor coolant also contains Ar-41 and N-16. The entrained N-16 generated in the reactor coolant has a 7-second half-life and is only a radiation hazard during reactor operations or immediately after reactor shutdown. During full-power reactor operation, the decay tank, heat exchangers, and pumps have surface dose rates in excess of 1 rem/hr due to the N-16 radioactivity in the primary coolant, but the elevated dose rates are expected and are posted and RINSC staff access is controlled. Tritium is also present in the coolant due to activation of trace deuterium that is present in ordinary water.

According to SAR Section 11.1.1.3.1, the occupational exposure from liquid sources is limited because there are few operations that require contact with the primary coolant. In cases where there is potential worker contact with the primary coolant, such as in certain maintenance operations, the reactor coolant is allowed to decay for several days or more to significantly reduce radioactivity concentrations. Sodium-24, which is the predominant radionuclide in the reactor coolant, has a relatively short half-life of 14.9 hours, and after 48 hours of decay, the sodium-24 concentration would be reduced by about a factor of 10.

Liquid radiation sources at the RINSC also include liquid radioactive wastes. SAR Section 11.2 discusses radioactive waste at the RINSC, including liquid radioactive waste. Liquid radioactive waste sources are quite limited. The main source of liquid radioactive waste is primary coolant (i.e., primary coolant that is removed from the PCS for sampling or other purposes, and must be disposed of). Small quantities of liquid radioactive wastes can also be generated by other activities such as experiments. Low-activity aqueous liquid wastes, including primary coolant, are released into sanitary sewerage following sampling to ensure that the concentrations are below the concentration limits for sanitary sewer discharge in 10 CFR Part 20, Appendix B, Table 3 (TS 3.7.2.2, which is discussed and found acceptable in SER Section 11.2.3, requires that these discharges be within the limits in 10 CFR Part 20, Appendix B, Table 3.) Higher-activity aqueous liquid wastes, and organic liquid wastes such as liquid scintillation cocktail, are packaged and transported offsite for disposal in accordance with applicable NRC and Department of Transportation regulations. The high-activity aqueous liquid wastes are typically absorbed onto solid materials prior to disposal. Radioactive waste management, including liquid radioactive waste management, is also evaluated and found acceptable in SER Section 11.2.

Based on the information above, the NRC staff concludes that the description and characterization of the liquid radiation sources at the RINSC facility are reasonable for a RTR. The information is sufficient to evaluate the facility's RPP and controls described in the remainder of SER Section 11.1.

Solid Radioactive Sources

According to SAR Section 11.1.1.2, the principal solid radioactive sources at the RINSC are the fission products produced and retained within the fuel during normal reactor operation. These fission products are present in the fuel in the core, and in the spent fuel assemblies that are stored in fuel racks. The reactor core and fuel racks are surrounded by the concrete biological shield and are submerged in the reactor pool, which shields personnel from the radiation. Spent fuel movement and storage are evaluated and found acceptable in SER Section 9.2. Other solid radioactive sources include uranium in the reactor fuel, reactor fuel cladding, resins and filters, activated and/or contaminated reactor components, experiment components, fission chambers, the antimony-beryllium reactor startup source, various sealed instrument calibration sources, activated samples, and solid low-level radioactive waste.

SAR Section 11.2 discusses radioactive waste at the RINSC, including solid radioactive waste. Two main forms of solid low-level radioactive waste are generated at the RINSC. These are used ion exchange resins (which are dried before disposal), and laboratory waste materials (i.e., irradiated materials, and contaminated tools, toweling, etc.). These wastes are volume-reduced when practical, and are packaged and transported offsite (in accordance with applicable NRC and Department of Transportation regulations) to organizations authorized to receive the material for disposal. Radioactive waste management, including solid radioactive waste management, is also evaluated and found acceptable in SER Section 11.2.

Based on the information above, the NRC staff concludes that the description and characterization of the solid radiation sources at the RINSC facility are reasonable for a RTR. The information is sufficient to evaluate the facility's RPP and controls described in the remainder of SER Section 11.1.

Conclusions on Radiation Sources

The NRC staff reviewed the description of potential radiation sources and associated doses, including the inventories, chemical and physical forms, and locations of radioactive materials, and other facility radiation and operational parameters related to radiation safety presented in the SAR. This review included a comparison of the bases for identifying potential radiation safety hazards with the process and facility descriptions to verify that such hazards were accurately and comprehensively identified. This review and evaluation confirm that the SAR identifies the potential radiation safety hazards associated with the RINSC facility, and provides an acceptable basis for the development and independent review of the facility's RPP and controls.

11.1.2 Radiation Protection Program

The regulations in 10 CFR 20.1101(a) require each licensee to develop, document, and implement a RPP. The RINSC has a structured RPP with a health physics staff that has the equipment and capabilities to determine, control, and document occupational and public radiation exposures. The basic information in the SAR is supported by TSs that define the required details of the program, which are found in TS Section 6.0.

SAR Section 11.1.2 describes the RPP. The primary purpose of this program is to regulate the activities of, and protect the health and safety of, the RINSC staff, research associates, students, general public, and the environment in accordance with Federal and State regulations. The SAR establishes the commitment of the licensee to regulatory compliance and overall radiation safety.

According to SAR Sections 11.1.2.1 and 11.1.2.2, the radiological safety organization at the RINSC is comprised of the RIAEC, the Nuclear and Radiation Safety Committee (NRSC) and Subcommittee (NRSSC), the RSO, and other staff involved with radiation safety. The RIAEC, which operates the RINSC, is an agency of the Rhode Island State government. The RIAEC recommends the selection of a Director to the Governor, who is responsible for implementing and coordinating all decisions of the RIAEC staff. The RIAEC appoints the NRSC, which includes the Director, the Assistant Director for Operations, the Assistant Director for Radiation and Reactor Safety, and four representatives that are not RIAEC commissioners or staff. The function of the NRSC is to ensure compliance with all Federal and State regulations, including radiation safety. The NRSC has review and audit functions (including reviews of tests, experiments, modifications, and procedures) that are delineated in TS 6.2 (which is discussed and found acceptable in SER Section 12.2). TS 6.1.1 (which is discussed and found acceptable in SER Section 12.1) provides an organizational chart (TS Figure 6.1), which shows that line responsibility for radiation safety is derived from the RIAEC Director and resides with the Assistant Director for Radiation and Reactor Safety. The Assistant Director for Radiation and Reactor Safety also serves as the RSO, who is the chief administrative officer of the RPP. The organizational structure of the RINSC, including the radiation protection organization, is discussed further in SER Section 12.1.

TS 6.3 states:

> 6.3 Radiation Safety
>
> The facility shall have a qualified, designated individual that is responsible for implementing the Radiation Safety Program in accordance with 10 CFR Part 20. The Assistant Director for Radiation and Reactor Safety shall be the individual in the organization that fulfills this requirement. A qualified alternative may serve in this capacity if the Assistant Director is unavailable for an extended period of time.

TS 6.3 identifies the Assistant Director for Radiation and Reactor Safety as the responsible officer for implementation of the RPP. The requirements of the RPP are established in 10 CFR Part 20. The licensee provided clarification (Ref. 58) that the title of the individual responsible for the RPP is “Assistant Director for Radiation and Reactor Safety,” rather than “Assistant Director for Reactor and Radiation Safety,” and therefore, the TS is consistent with the actual title of the position. The NRC staff finds that this specification helps identify the responsible person for the implementation of the RPP. The NRC staff also finds this specification helps to ensure that the radiation safety aspects of the RINSC organization structure are properly delineated. Furthermore, the requirements of the position and the responsibility for the RPP are stated and appropriate. The NRC staff also finds that this specification is consistent with the guidance in NUREG-1537. Therefore, based on the information above, the NRC staff concludes that TS 6.3 is acceptable.

TS 6.1.4.1.4, which is discussed and found acceptable in SER Section 12.1, requires minimum levels of education and experience for the Assistant Director for Radiation and Reactor Safety (the RSO). The duties of the RSO include:

- Maintaining the RINSC Radiation Safety Guide;
- Administering the ALARA program (the ALARA program is discussed in SER Section 11.1.3);
- Administering the dosimetry program including record keeping and notifications;
- Establishing procedures for periodic radiation surveys;
- Developing and maintaining the survey instrument calibration program;
- Presenting briefings and training sessions for RINSC staff and others potentially exposed to radiation;
- Maintaining a call list providing 24-hour coverage in the event of a radiological accident;
- Reviewing and approving relevant radiation safety procedures;
- Encouraging compliance with all RINSC radiation safety procedures, and appropriate State and Federal regulations;
- Conducting audits, as appropriate; and,
- Managing the health physics staff, who report to the RSO.

SAR Section 11.1.2.5 indicates that the RSO provides training to all staff who work with or around radioactive materials. Non-radiation workers, such as custodial and security personnel, are retrained annually. Ancillary training is provided for specific job functions, as needed.

SAR Section 11.1.2.6 indicates that the RINSC facility has standard operating procedures for activities related to radiation safety and health physics. A RINSC administrative procedure describes the development, review, and approval of these standard operating procedures. TS 6.4, which is discussed and found acceptable in SER Section 12.3, requires the use of procedures for activities involving radiation safety.

The regulations in 10 CFR 20.1101(c) require that licensees shall periodically (at least annually) review the RPP content and implementation. TS 6.2.4.5, which is discussed and found acceptable in SER Section 12.2, requires that the RINSC radiation safety program shall be audited at least annually. SAR Section 11.1.2.7 states that these audits are performed to verify compliance with applicable Federal and State regulations and to determine the effectiveness of the RPP.

SAR Section 11.1.2.8 states that health physics records, including records related to personnel exposures and environmental releases, are maintained for the life of the facility. TS 6.8, which is discussed and found acceptable in SER Section 12.6, imposes requirements for the maintenance of records, including health physics records.

The NRC staff has reviewed the RINSC RPP, as described in the SAR, as supplemented, and the supporting TSs. The NRC staff finds that the licensee effectively describes:

(1) the roles, responsibilities, authorities, organization, and staffing of the radiation protection organization,
(2) the roles, responsibilities, authorities, staffing, and operation of committee responsible for the review and audit of the RPP,
(3) the effectiveness and comprehensiveness of the radiation protection training program,
(4) the radiation protection plans and information that form the bases of procedures, and the management systems employed to establish and maintain them,
(5) the effectiveness and comprehensiveness of the program for independent oversight, reviews, and audits of the RPP;
(6) the effectiveness and comprehensiveness of the process to evaluate the RPP to improve the program and the process to examine problems and incidents at the facility, and
(7) the management of records relating to the RPP.

The NRC inspection program also routinely reviews the RPP at the RINSC facility. The NRC staff reviewed the NRC IRs for the years 2011 through 2016 (Ref. 28), and noted that in 2012, the licensee received a non-cited Severity Level IV violation related to the licensee's failure to adequately control access to a high radiation area, which resulted in a reactor staff member (student trainee) receiving an elevated dose (the elevated dose was 115 mrem [1.15 mSv], well below the 5,000 mrem (50 mSv) occupational dose limit in 10 CFR 20.1201). However, the NRC staff also noted that the licensee implemented appropriate corrective actions to prevent a recurrence following the incident, including staff retraining and a new procedure for entry into the high radiation area. The NRC staff also noted that in 2013, the licensee received another non-cited Severity Level IV violation related to the fact that the RSO was determined not to have the TS-required educational background. The licensee also implemented appropriate corrective actions following this incident, specifically, replacing the RSO with another individual who had met the educational requirements, and not operating the reactor until the new RSO was in place. The NRC staff noted no other significant issues related to the RPP in the 2011 through 2016 IRs.

The NRC staff reviewed the information above, and finds that the RPP presented in the SAR complies with 10 CFR 20.1101, paragraphs (a) and (c), and is implemented in an acceptable manner. The NRC staff also finds that the radiation program is consistent with guidance in ANSI/ANS-15.11-2016, "Radiation Protection at Research Reactor Facilities" (Ref. 44). The NRC staff further finds that the licensee provides reasonable confidence that its commitment to radiation protection in all activities will protect the facility staff, the environment, and members of the public from exposure to radiation in excess of the 10 CFR Part 20 limits. Therefore, based on the information above, the NRC staff concludes that the RINSC radiation protection program is acceptable.

11.1.3 As Low As Reasonably Achievable Program

The regulations in 10 CFR 20.1101(b) require licensees to use procedures and engineering controls to achieve occupational doses and doses to members of the public that are ALARA. SAR Section 11.1.3 describes the RINSC ALARA program and the commitment of the licensee

to achieving doses that are ALARA. The RSO administers the RINSC ALARA program. The RINSC ALARA program includes the following elements:

(1) A training program for individuals using radiation sources so that they can recognize and protect themselves from sources of ionizing radiation;
(2) A comprehensive dosimetry program, including badge monitoring and bioassays (radiation exposure control and dosimetry are discussed in SER Section 11.1.5);
(3) Investigation of any exposures that are above ALARA levels;
(4) Radiation monitoring and surveying of areas where radiation and/or contamination could be present (radiation monitoring and surveying are discussed in SER Section 11.1.4);
(5) Review of effluent releases, and investigation of any releases that are over 10 percent of regulatory limits; and,
(6) Review and audit of the use of radioactive material and the radiation safety program.

The regulations in 10 CFR 20.1101(d) requires that to implement the ALARA requirements of 10 CFR 20.1101(b), licensees shall establish a constraint on air emissions of radioactive material to the environment such that the individual member of the public likely to receive the highest dose will not be expected to receive a dose in excess of 10 mrem (0.1 mSv) in 1 year. As discussed in SER Section 11.1.1, the licensee's historical operation has not resulted in doses that are in excess of this constraint.

The NRC inspection program routinely reviews the effectiveness of the RINSC ALARA program. The NRC staff reviewed the NRC IRs for the years 2011 through 2016 (Ref. 28), and noted that except for the 2012 non-cited violation related to the licensee's failure to adequately control access to a high radiation area, for which the licensee subsequently took appropriate corrective action (discussed in SER Section 11.1.2), there were no significant issues related to the RINSC ALARA program. The NRC staff also reviewed the RINSC annual reports for the years 2009 through 2016 (Ref. 16), and noted that there were no personnel exposures in excess of 10 percent of the 5,000 mrem occupational dose limit in 10 CFR 20.1201, and that estimated public exposures were also less than 10 percent of the 100 mrem public dose limit in 10 CFR 20.1301. These reviews help confirm the effectiveness of the RINSC ALARA program.

The NRC staff reviewed the information above, and finds that the licensee's policies, procedures, and controls for limiting access and personnel exposure provide reasonable assurance that doses to occupational workers and the public will be maintained below regulatory limits and are ALARA. The NRC staff finds that the ALARA program is adequately supported at the facility. The NRC staff also finds that the overall ALARA program is consistent with guidance in ANSI/ANS-15.11-2016, "Radiation Protection at Research Reactor Facilities" (Ref. 44), and complies with 10 CFR 20.1101. Therefore, based on the information above, the NRC staff concludes that the RINSC ALARA program is acceptable.

11.1.4 Radiation Monitoring and Surveying

The regulations in 10 CFR 20.1501(a) state that each licensee shall make, or cause to be made, surveys of areas, including the subsurface, that:

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

(2) Are reasonable under the circumstances to evaluate -

(i). The magnitude and extent of radiation levels; and
(ii). Concentrations or quantities of residual radioactivity; and
(iii) The potential radiological hazards of the radiation levels and residual radioactivity detected.

The regulation in 10 CFR 20.1501(c) requires 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.

According to SAR Section 11.1.4.1, the licensee conducts routine radiation and contamination surveys, described in standard procedures, to evaluate basic radiological conditions at the RINSC. The licensee also conducts radiation monitoring to alert staff and operators to changing radiation conditions. The licensee conducts this monitoring and surveying using numerous fixed and portable radiation monitoring and surveying instruments that it maintains throughout the RINSC facility. SAR Section 11.1.4.2 includes a list of these instruments. As observed by the NRC staff during site visits, five fixed area radiation monitors are located throughout the reactor building. The area radiation monitors measure gamma radiation levels. The licensee also has continuous air monitors (particulate and gaseous), which measure gaseous and particulate activity in the reactor building or in effluents released to the environment. All of these instruments indicate and alarm locally and in the control room. The RMS at the RINSC facility are also evaluated and found acceptable in SER Section 7.7.

The licensee has TS requirements for certain RMS. TS 3.7.1.1 (which is discussed and found acceptable in SER Section 7.7) requires that a stack gas monitor, stack particulate monitor, main floor area monitor, and reactor bridge area monitor be operating when: (1) the reactor is operating, (2) irradiated fuel handling is in progress, (3) experiment handling is in progress for an experiment that has a significant fission product, or gaseous effluent activation product inventory, (4) any work on the core or control rods that could cause a reactivity change of more than 0.60 %$\Delta k/k$ is in progress, or (5) any experiment movement that could cause a reactivity change of more than 0.60 %$\Delta k/k$ is in progress. TS 3.7.1.2 (which is also discussed and found acceptable in SER Section 7.7) specifies setpoint levels for the monitors required by TS 3.7.1.1. Other monitors at the RINSC facility do not have TS requirements.

Portable instrumentation is available to survey areas in the RINSC facility for all types of radiation and radioactive contamination that may be present from facility operations. This includes ion chambers and friskers. During site visits, the NRC staff observed portable instrumentation throughout the facility. In its response to RAIs 11.3 and 11.6 (Ref. 3), the licensee provides information on methods and the frequencies of the radiation surveys. The licensee stated that the routine surveys use calibrated survey meters with appropriate detectors. The licensee also stated that wipe tests are used to detect removable contamination. The wipe tests are counted using appropriate radiation instruments, depending on the isotopes that are thought to be present. The frequencies for routine surveys are determined by an evaluation of the radiological hazards likely to be present in the area, the frequency of routine entry into or use of the area, and ALARA considerations. Typically, areas that routinely contain unsealed gamma emitter sources, or beta and alpha emitters capable of being detected by survey meters, are surveyed at least weekly. Surveys occur more frequently when any operation is likely to produce significant radiation and/or contamination. Survey frequencies are reviewed and approved by the NRSC. The routine surveys are supplemented by surveys taken by individual reactor users of their own work areas, as well as personnel contamination surveys (i.e., “frisks”).

TS 4.7.1, which is discussed and found acceptable in SER Section 7.7, requires periodic testing and calibration of the RMS required by TS 3.7.1.1. Other non-TS required radiation monitoring and surveying equipment is also periodically tested and calibrated, as appropriate. Radiation monitoring and surveying equipment observed by the NRC staff on site visits was labelled with calibration stickers, and the calibration for all observed instruments was up-to-date. Calibration activities are controlled by procedures. TS 6.4.2.4, which is discussed and found acceptable in SER Section 12.3, requires that procedures that are approved by the NRSC be used for surveillance checks, calibrations, and inspections that are required by the TSs, or have a significant effect on reactor safety.

As required by TS 6.8.1.3, records of surveillance activities required by the TSs, including surveillance of radiation monitors, must be maintained for at least 5 years. As required by TS 6.8.1.4, records of facility radiation monitoring surveys must also be maintained for at least 5 years. TSs 6.8.1.3 and 6.8.1.4 are discussed and found acceptable in SER Section 12.6.

As discussed in SER Section 11.1.5, operators and other personnel working at the reactor wear individual radiation dose monitoring badges, as required. The licensee also conducts an environmental monitoring program, which is discussed in SER Section 11.1.7.

The NRC inspection program routinely reviews the effectiveness of the RINSC radiation monitoring and surveying program. The NRC staff reviewed the NRC IRs for the years 2011 through 2016 (Ref. 28), and noted that there were no significant issues related to radiation monitoring and surveying.

The NRC staff reviewed the information above, and finds that the licensee's radiation monitoring and surveying, including the equipment used and the surveillances performed on the equipment, are adequate for detecting the types and intensities of radiation likely to be encountered within the facility and will help ensure compliance with 10 CFR 20.1501(a) and 10 CFR 20.1501(c). Therefore, based on the information above, the NRC staff concludes that the radiation monitoring and surveying program at the RINSC facility is acceptable.

11.1.5 Radiation Exposure Control and Dosimetry

SAR Section 11.1.5 describes the radiation exposure control and dosimetry processes at the RINSC facility. The reactor and facility are designed to minimize radiation exposure to occupational workers and members of the public. The design incorporates shielding that is appropriate for the types of radiation encountered at the facility, and that maintains radiation levels at all points above and outside the reactor pool below 1 mrem (0.01 mSv) per hour. This shielding includes thick concrete, and the water above the reactor core in the pool.

The reactor ventilation system also helps control exposures to RINSC staff and the public. The exhaust system draws in air from near the top of the reactor pool, as well as from experimental areas, which are the locations where most Ar-41 is present. This helps minimize the concentration of Ar-41 within the reactor building, reducing occupational doses. The ventilation system also dilutes the facility effluents and releases them through an elevated stack, increasing dispersion and reducing doses to members of the public. The ventilation system is also evaluated and found acceptable in SER Sections 9.1 and 11.1.1. The reactor is located within a confinement building, and as required by TS 3.5.1 (discussed and found acceptable in SER Section 6.2.1). The ventilation system shall maintain the confinement building pressure at least 0.5 in of water below atmospheric pressure whenever the reactor is operating or other

activities are in progress that could result in a radioactive material release. This helps ensure that any air leakage is into, not out of, confinement, and any radioactive material released to confinement will be released through the stack such that it can be monitored, diluted before release, and adequately dispersed.

The licensee uses entry control to minimize doses to workers and members of the public. Access to the RINSC requires training appropriate to the level of access needed, and the level of potential exposure to radioactive materials. In general, access to high radiation areas is controlled by keeping entry points locked. Keys to these areas are controlled by senior ROs. Radiation workers wear protective equipment, including lab coats, disposable gloves, and protective eyewear, as appropriate, to minimize contamination and exposure to airborne radioactive materials.

SAR Section 11.1.5.6 states that personnel at the RINSC are monitored for radiation exposures. As discussed in SAR Section 11.1.2.3, the RSO is responsible for administering a radiation dosimetry program. According to the SAR, and as observed by the NRC staff during site visits, individual dosimetry is used by all personnel entering areas where radiation and/or radioactive material could be present. The licensee also used extremity monitoring when appropriate. In its response to RAIs 11.4 and 11.7 (Ref. 3), the licensee describes the provisions for the extremity monitoring at the RINSC. The licensee states that it is its policy to assign extremity monitoring to any individual likely to receive a measurable radiation dose to the extremities. The licensee also states that its radiation worker training includes instruction on the proper usage of dosimetry used for extremity monitoring. In response to RAI 11.5 (Ref. 3), the licensee states that bioassays to measure internal dose from ingested or inhaled radionuclides may be required for anyone handling or using unsealed radioactive sources, and are required for individuals likely to receive an annual intake in excess of 10 percent of the applicable limits. TS 6.7.1.7, which is discussed and found acceptable in SER Section 12.5, requires that the RINSC annual report include a summary of annual radiation exposures in excess of 500 mrem received by facility personnel, 100 mrem received by non-staff members, or 10 mrem received by members of the general public. TS 6.8.3.3, which is discussed and found acceptable in SER Section 12.6, requires that records of personnel radiation exposures be retained for the life of the facility.

The NRC staff reviewed the NRC IRs for the years 2011 through 2016 (Ref. 28), and noted that except for the 2012 non-cited violation related to the licensee's failure to adequately control access to a high radiation area, for which the licensee subsequently took appropriate corrective action (discussed in SER Section 11.1.2), there were no significant issues related to exposure control and dosimetry at the RINSC.

As discussed in SER Section 11.1.3, the NRC staff reviewed the RINSC annual reports for the years 2009 through 2016 (Ref. 16), and noted that there were no personnel exposures in excess of 10 percent of the 5,000 mrem occupational dose limit in 10 CFR 20.1201, "Occupational Dose Limits for Adults," and that estimated public exposures were also less than 10 percent of the 100 mrem public dose limit in 10 CFR 20.1301, "Dose Limits for Individual Members of the Public." This helps confirm the effectiveness of the licensee's exposure control and dosimetry.

The NRC staff reviewed the information above regarding the licensee's exposure control and dosimetry, and finds that personnel exposures at the RINSC facility are satisfactorily controlled by the design of the facility, and through the RINSC radiation protection and ALARA programs. The NRC staff also finds that the licensee's personnel dose monitoring complies with 10 CFR 20.1502, "Conditions Requiring Individual Monitoring of External and Internal

Occupational Dose," which requires monitoring of external and internal radiation doses to workers who could receive in excess of 10 percent of applicable 10 CFR Part 20 limits. Therefore, based on the information above, the NRC staff concludes that the radiation exposure control and dosimetry at the RINSC facility are acceptable.

11.1.6 Contamination Control

SAR Section 11.1.6 discusses contamination control at the RINSC. The licensee stated that radioactive contamination is controlled by using specific and detailed written procedures for radioactive and/or contaminated material handling, using trained personnel, and by conducting radiation surveying to detect contamination in a timely manner (SER Section 11.1.4 evaluates and finds acceptable radiation surveying and monitoring, including contamination surveys). After working in contaminated areas, personnel are required to survey themselves when leaving their work area, and again when exiting controlled areas surrounding a contaminated area. As discussed in SER Section 11.1.5, radiation workers wear protective equipment, including lab coats, disposable gloves, and protective eyewear, as appropriate, to minimize contamination and exposure to airborne radioactive materials. The licensee also stated that all work where contamination is considered likely requires oversight by a qualified health physics technician. Contamination events are documented in radiological incident reports, helping to avoid repeating events that caused unplanned contamination.

The NRC staff reviewed the information above. The NRC staff reviewed the NRC IRs for the years 2011 through 2016 (Ref. 28), and noted no significant issues related to contamination control at the RINSC facility, indicating that adequate controls exist to prevent the spread of radiological contamination within the facility. Based on its review of the information above, which indicates that the licensee has sufficient contamination control measures, as well as on the licensee's history of satisfactory contamination control, the NRC staff concludes that the contamination control program at the RINSC facility is acceptable.

11.1.7 Environmental Monitoring

The environmental monitoring that is performed at RINSC is described in SAR Section 11.1.7. Doses outside the facility are monitored at certain locations using optically-stimulated luminescent dosimeters, which are collected and read on a quarterly basis. These dosimeters help the licensee monitor any offsite dose from the facility, either from direct (external) radiation or from radioactive effluents. The licensee does not identify any specific sampling of soil, vegetation, or water as part of its environmental monitoring program. TS 6.7.1.6, which is discussed and found acceptable in SER Section 12.5, requires that the RINSC annual report include a summary of the results of environmental surveys performed outside the facility during the reporting period. This summary shall include the locations of the surveys. TS 6.8.3.2, which is discussed and found acceptable in SER Section 12.6, requires that records of offsite environmental monitoring be retained for the life of the facility.

The NRC staff reviewed the licensee's annual reports for the years 2009 through 2016 (Ref. 16). The reports show that the licensee has environmental dosimeters at 3 locations outside the reactor building: the northeast wall, the demineralizer door, and the heat exchanger door. The licensee states that these areas are in locations where access is limited, and the areas would not be frequented by members of the public. Therefore, the licensee applies occupancy factors for members of the public located in these areas to calculate potential public doses (for the 2009 through 2015 annual reports, the licensee used an occupancy factor of 1 percent; for the 2016 annual report, the licensee used an occupancy factor of 2.5 percent).

Applying the occupancy factors, the highest reported annual dose was 5.73 mrem (0.0573 mSv), measured at the demineralizer door, and reported in the 2016 annual report. The NRC staff finds that given the environmental dosimeters are in locations that are only occasionally occupied by members of the public, and are only occupied for brief periods of time, the occupancy factors that the licensee applies for these locations are reasonable. The NRC staff also finds that the reported doses are less than 10 percent of the 100 mrem public dose limit in 10 CFR 20.1301, "Dose Limits for Individual Members of the Public." The NRC staff also reviewed the NRC IRs for the years 2011 through 2016 (Ref. 28), and noted no significant issues related to the environmental monitoring program at the RINSC facility.

The NRC staff reviewed the information above regarding environmental monitoring at the RINSC facility, including annual operating reports, which indicated that the operation of the reactor had not adversely affected the environment. The NRC staff finds that the environmental monitoring program helps to assess and provide an early indication of any environmental impact caused by the reactor facility operation. Therefore, based on the information above, the NRC staff concludes that the environmental monitoring program at the RINSC facility is acceptable and demonstrate compliance with the dose limits of 10 CFR 20.1301.

11.2 Radioactive Waste Management

The purpose of the radioactive waste management program is to minimize radioactive waste and to help ensure that radioactive waste materials are identified, assessed, controlled, handled, stored, and disposed of in accordance with all applicable regulations, and in a manner that will protect the RINSC staff, the public, and the environment. SAR Section 11.2 discusses the waste management program at the RINSC facility. The licensee's historical performance in these areas, as documented in NRC IRs and the RINSC annual operating reports, in the SAR, as supplemented, and as observed by the NRC staff during site visits, provides documentation that measures are in place to minimize radiation exposure to the RINSC staff and the public and to provide adequate protection against releases of radioactivity to the environment.

11.2.1 Radioactive Waste Management Program

SAR Section 11.2.1 describes the RINSC radioactive waste management program. The RINSC may generate a variety of gaseous, liquid, and solid radioactive wastes and/or effluents. The RINSC may also generate mixed waste (i.e., waste that is radioactive, as well as hazardous and/or biohazardous). Since there are stringent regulatory requirements for wastes containing any of these materials, the licensee has developed a radioactive waste management program to help ensure that those regulatory requirements are met.

All individuals who work with radioactive materials at the RINSC are required to have training approved by the RSO. This training includes instruction on dealing with radioactive waste. The implementation of the ALARA program also encompasses the minimization of the generation of radioactive waste. The NRSC has the authority to consider in advance, and approve or disapprove, the production, procurement, use, and ultimate disposal of radioactive materials at the RINSC. Experiments are designed to avoid unnecessary generation of radioactive material. To minimize the generation of possible mixed wastes, all experiments are also reviewed to avoid the use of hazardous chemicals. When possible, the licensee separates radioactive waste by radioisotope and type, which allows for greater ease of handling. When practical, short-lived radioactive wastes are stored for decay, so that they can be disposed of as non-radioactive waste. All waste disposals are accomplished through the RSO. In general, radioactive wastes generated at the RINSC are collected, processed, and stored in a secure

area within the facility until they are transferred to a licensed broker, processor, or burial site operator.

TSs 6.4.2.5 and 6.4.2.8, which are discussed and found acceptable in SER Section 12.3, requires that procedures that are approved by the NRSC be used for radiation safety activities, and for the receipt, use, and transfer of byproduct material. These TSs encompass procedures used for radioactive waste. TSs 6.2.4.1 and 6.2.4.5, which are discussed and found acceptable in SER Section 12.2, require that the RINSC operations and radiation safety program, including aspects of operations and the radiation safety program related to radioactive waste disposal, be audited at least annually.

The NRC staff reviewed the information above. The NRC staff also reviewed the licensee's annual operating reports for the years 2009 through 2016 (Ref. 16), and the NRC IRs for the years 2011 through 2016 (Ref. 28), and noted that there were no significant issues related to radioactive waste or effluent management, handling, or releases. The NRC staff finds that the licensee's practices demonstrate reasonable assurance that radiological wastes or releases from the facility will not exceed applicable regulatory limits, nor will they pose an unacceptable radiation risk to the environment or the public. The NRC staff also finds that the licensee has adequate procedures 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. Therefore, based on the information above, the NRC staff concludes that the radioactive waste management program at the RINSC facility is acceptable.

11.2.2 Radioactive Waste Controls

SAR Section 11.2.2 discusses radioactive waste controls at the RINSC facility (other portions of SAR Chapter 11 also contain information on radioactive waste controls). The SAR does not indicate that any gaseous wastes are produced at the facility, other than Ar-41, which is released as an effluent. Ar-41 is the only gaseous effluent released in significant quantities during RINSC operations. Controls related to Ar-41 effluents are discussed in detail in SER Section 11.1.1.

Liquid radioactive wastes generated at the RINSC are very limited. As discussed in SER Section 11.1.1, the main source of liquid radioactive waste is primary coolant (i.e., primary coolant that has been removed from the PCS for sampling or other purposes, and must be disposed of). Small quantities of liquid radioactive wastes can also be generated by other activities such as experiments. Liquid radioactive wastes are sampled to measure their radioactivity (using batch sampling of the liquid waste retention tank), and when possible, low-activity aqueous liquid wastes are disposed of by release to sanitary sewerage, as an effluent in accordance with 10 CFR 20.2003, "Disposal by Release into Sanitary Sewerage." Other liquid wastes that cannot be released to sanitary sewerage are packaged and transported offsite for disposal in accordance with applicable NRC and Department of Transportation regulations. Controls related to liquid radioactive waste are also discussed in SER Section 11.1.1.

Two main forms of solid low-level radioactive waste are generated at the RINSC. The first of these is used ion exchange resins. These are ambient air dried for 2 months in an access controlled area prior to disposal. The drying time also allows radionuclides in the resin (particularly sodium-24) to decay, reducing the activity of the resins. The second form of solid low-level radioactive waste is laboratory waste materials, such as irradiated materials,

contaminated tools, and contaminated toweling. The laboratory wastes are accumulated in containers in various work areas, and collected from around the facility by RINSC health physics staff on a weekly basis. Solid wastes are volume-reduced when practical. The wastes are placed in Department of Transportation approved drums. The dose rates on the outside of the drums is measured, and the total drum activities are calculated. The drums are then transported offsite (in accordance with applicable NRC and Department of Transportation regulations) to organizations authorized to receive the material for disposal. Controls related to solid radioactive waste are also discussed in SER Section 11.1.1.

The NRC staff reviewed the information above regarding gaseous, liquid, and solid radioactive waste controls. The NRC staff also reviewed the licensee's annual operating reports for the years 2009 through 2016 (Ref. 16), and the NRC IRs for the years from 2011 through 2016 (Ref. 28), and noted that there were no significant issued related to radioactive waste controls. The NRC staff finds that the licensee's radioactive waste controls demonstrate reasonable assurance that radiological wastes or releases from the facility will not exceed applicable regulatory limits, nor will they pose an unacceptable radiation risk to the environment or the public. The NRC staff also finds that the licensee has adequate procedures 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. Therefore, based on the information above, the NRC staff concludes that the radioactive waste controls at the RINSC facility are acceptable.

11.2.3 Release of Radioactive Waste

Releases of gaseous waste (Ar-41 effluents) are discussed in detail in SER Section 11.1.1. As discussed in SER Sections 11.1.1 and 11.2.2, low-level aqueous liquid wastes are sampled and released to sanitary sewerage as effluents in accordance with 10 CFR 20.2003, "Disposal by Release into Sanitary Sewerage," and other liquid wastes and solid wastes are packaged and transported offsite for disposal in accordance with applicable NRC and Department of Transportation regulations. Based on the information above, and the information discussed in detail and found acceptable in SER Sections 11.1.1 and 11.2.2, the NRC staff concludes that the releases of radioactive waste from the RINSC facility are acceptable.

TSs related to releases of radioactive effluents from the RINSC facility are discussed below.

TS 3.7.2.1 states:

> TS 3.7.2.1 Airborne Effluents
>
> The annual total effective dose equivalent to the individual member of the public likely to receive the highest dose from air effluents will be calculated using a generally-accepted computer program and will not exceed 100 mrem per year.

TS 3.7.2.1 requires the licensee to periodically estimate the maximum public dose from air effluents using an approved methodology, and also requires that this dose not exceed 100 mrem (1 mSv) per year. SER Section 11.1.1 discusses calculations of public Ar-41 dose. The NRC staff finds that TS 3.7.2.1 helps to ensure that doses to members of the public from Ar-41 emissions from the RINSC facility will not exceed the 100 mrem public dose limit in 10 CFR 20.1301, "Dose Limits for Individual Members of the Public," and will therefore not pose

an unacceptable radiation risk to members of the public. Therefore, based on the information above, the NRC staff concludes that TS 3.7.2.1 is acceptable.

TS 4.7.2.1 states:

> TS 4.7.2.1 Airborne Effluents
>
> The annual total effective dose equivalent to the individual member of the public likely to receive the highest dose from air effluents shall be calculated annually.

TS 4.7.2.1 requires that the dose calculation required by TS 3.7.2.1 be conducted annually. The licensee provided clarification (Ref. 58) as to how TS 4.7.2.1 allows the licensee to ensure it does not exceed the 10 mrem dose constraint in 10 CFR 20.1101(d) due to Ar-41 emissions, given that the TS only requires the licensee to check the dose once per year, and by the time the dose is checked the exceedance could have already occurred. The licensee stated that it monitors Ar-41 releases more often than annually, and the stack radiation monitor readings, which give an indication of measured Ar-41 releases, are noted and recorded during every reactor shift. Trends in these data would give an indication if there were the potential for the constraint to be exceeded in one year. The NRC staff finds that by requiring the annual calculations to be performed in conjunction with the other Ar-41 monitoring activities that the licensee performs, TS 4.7.2.1 helps ensure that annual doses to members of the public will not exceed the 10 mrem constraint in 10 CFR 20.1101(d) or the 100 mrem annual public dose limit in 10 CFR 20.1301. Therefore, based on the information above, the NRC staff concludes that TS 4.7.2.1 is acceptable.

TS 3.7.2.2 states:

> TS 3.7.2.2 Liquid Effluents
>
> All liquid effluent discharges shall be within regulatory limits in accordance with 10 CFR 20, appendix B, table 3.

TS 3.7.2.2 requires that liquid effluent discharges from the RINSC facility to the sanitary sewer have radionuclide concentrations that are within the 10 CFR Part 20, Appendix B, Table 3, concentration limits for sewer releases. The NRC staff finds that by requiring that the radionuclide concentrations in sanitary sewer discharges be below these limits, TS 3.7.2.2 helps ensure that any public dose from these releases is within 10 CFR Part 20 limits. Therefore, based on the information above, the NRC staff concludes that TS 3.7.2.2 is acceptable.

TS 4.7.2.2 states:

> TS 4.7.2.2 Liquid Effluent Sampling
>
> The liquid waste retention tank discharge shall be batch sampled and the gross activity per unit volume determined to be less than the limits set in 10 CFR Part 20 before release.

TS 4.7.2.2 requires that sampling be conducted before each release of liquid radioactive effluents to ensure that the release is in compliance with TS 3.7.2.2. The NRC staff finds that by requiring the licensee to verify that the concentrations of radioactive materials released are

within 10 CFR Part 20, Appendix B, Table 3, limits before any release, TS 4.7.2.2 helps ensure that any public dose from these releases is within 10 CFR Part 20 limits. Therefore, based on the information above, the NRC staff concludes that TS 4.7.2.2 is acceptable.

TS 6.7.1.5, which is discussed and found acceptable in SER Section 12.5, requires that the RINSC annual report summarize gaseous and liquid effluents released from the RINSC facility. TS 6.8.3.1, which is discussed and found acceptable in SER Section 12.6, requires that the licensee maintain records of gaseous and liquid radioactive effluents released to the environs for the life of the facility.

11.3 Conclusions

Based on its review of the information in the SAR, as supplemented by responses to RAIs, observations of the licensee's operations, review of annual operating reports, and the results of the NRC inspection program, the NRC staff concludes the following regarding the licensee's radiation protection and radioactive waste management programs:

- The RINSC RPP complies with the requirements in 10 CFR 20.1101(a) and 10 CFR 20.1101(c), is acceptably implemented, and provides reasonable assurance that the RINSC staff, the public, and the environment are protected from unacceptable radiation exposures. The RPP is acceptably equipped and staffed with trained individuals. The RINSC management is committed to radiation safety and has defined a program with appropriate lines of authority and communication to allow radiation protection staff to carry out the program.
- The RINSC ALARA program is supported by the highest levels of management and complies with the requirements of 10 CFR 20.1101(b). The radiation protection and radioactive material controls at the RINSC facility provide reasonable assurance that radiation doses to the facility personnel and the public and effluent releases to the environment will be ALARA.
- Facility design and procedures limit the production and release of Ar-41 and N-16, and control the potential for occupational and public radiation exposures. Conservative calculations of the quantities of these gases released into restricted and unrestricted areas, and the results of personnel dosimetry and area radiation monitoring, provide reasonable assurance that doses to the RINSC staff and the public will be below applicable 10 CFR Part 20 limits. Liquid radioactive effluents from the facility are also controlled and released in accordance with applicable 10 CFR Part 20 limits. The systems and procedures provided for control of radioactive effluents, when operated and used in accordance with the TSs, are acceptable to help ensure that releases of radioactive materials from the facility are within the limits of NRC regulations and are ALARA.
- The specified surveillance and other TS requirements related to effluents provide the necessary controls to help ensure continued normal operation of the RINSC facility will not pose a significant risk to the health and safety of the public or the environment, and the dose from RINSC operations would be below the applicable regulatory limits of 10 CFR Part 20.
- The licensee has adequately identified and described potential radiation sources. The licensee also sufficiently controls radiation sources.

- The radiation monitoring and surveying program at the RINSC facility helps ensure compliance with 10 CFR 20.1501. The results of the radiation surveys carried out at the facility, doses to the persons issued dosimetry, and the results of the environmental monitoring program help confirm that the implementation of the radiation protection and ALARA programs are effective.
- The RINSC program for contamination control meets all regulatory requirements, and helps ensure the control of radioactive contamination, so that there is reasonable assurance that the health and safety of the facility staff and the public, and the environment will be protected.
- The facility radioactive waste management program provides reasonable assurance that radioactive waste produced at the facility will be controlled and handled in accordance with applicable regulations, and its release will not pose an unacceptable radiation risk to the environment or the public.

The NRC staff reviewed the RINSC RPP and radioactive waste management program as described in the SAR, as supplemented. The NRC staff finds that the licensee implements adequate and sufficient measures to minimize radiation exposure to facility workers and the public. Therefore, the NRC staff concludes that there is reasonable assurance that the RINSC radiation protection and radioactive waste management programs will provide acceptable radiation protection to RINSC staff, members of the public, and the environment.

12. CONDUCT OF OPERATIONS

The conduct of operations involves the administrative aspects of facility operation (Section 6 of the TSs), and the facility emergency, security and operator requalification plans. The administrative aspects of facility operations are the facility organization, training, operational review and audits, procedures, required actions, and records and reports.

12.1 Organization

The R-95 license is issued to the State of Rhode Island. The RIAEC, an agency of the State of Rhode Island, has responsibility for the safe operation of the reactor. Organizational control of the RINSC is delegated to the Director of the RIAEC. Reactor operations is the responsibility of the Assistant Director for Operations who reports to the RIAEC Director. An organization chart is presented in the Figure 6-1 of the TSs (Ref. 60) and shown below as Figure 12-1 in this SER. The Reactor Supervisor, licensed Senior Reactor Operators (SRO) and Reactor Operators (RO) all report through the Assistant Director for Operations to the RIAEC Director.

The radiation safety organization has a reporting chain independent of reactor operations. The Assistant Director for Radiation and Reactor Safety reports to the RIAEC Director. The radiation protection organization has the authority to interdict or terminate safety-related activities.

SAR Chapter 12 describes the responsibilities for the management levels which reflect the operation and policies of the reactor facility. TS 6.1, “Organization,” identifies the minimum qualified staff to safely operate and shutdown the reactor. As described in TS 6.1.4, “Selection and Training of Personnel,” the selection and training of the RIAEC Director is in accordance with ANSI/ANS-15.4-2007, “Selection and Training of Personnel for Research Reactors”, (Ref. 17). Licensed ROs and SROs are trained in accordance with 10 CFR Part 55.

TS 6.1 states:

> 6.1 Organization
>
> 6.1.1 Organization Structure
>
> The Rhode Island Nuclear Science Center (RINSC) Reactor shall be licensed to the State of Rhode Island. The Rhode Island Atomic Energy Commission is the state agency that shall have responsibility for the safe operation of the reactor. The Governor of the state shall appoint five Commissioners to the Rhode Island Atomic Energy Commission (RIAEC) who shall have the authority to recommend the selection of a Director, and appoint individuals to the Nuclear and Radiation Safety Committee (NRSC). The Director shall be the organizational head, and shall be responsible for the reactor facility license. The Assistant Director for Operations shall be responsible for the reactor programs and operation of the facility. The Assistant Director for Radiation and Reactor Safety shall be responsible for the safety programs of the facility. The RINSC staff shall operate and maintain the facility. The Nuclear and Radiation Safety Committee (NRSC) shall be an independent review and audit committee. Figure 6.1 shows the organization chart.

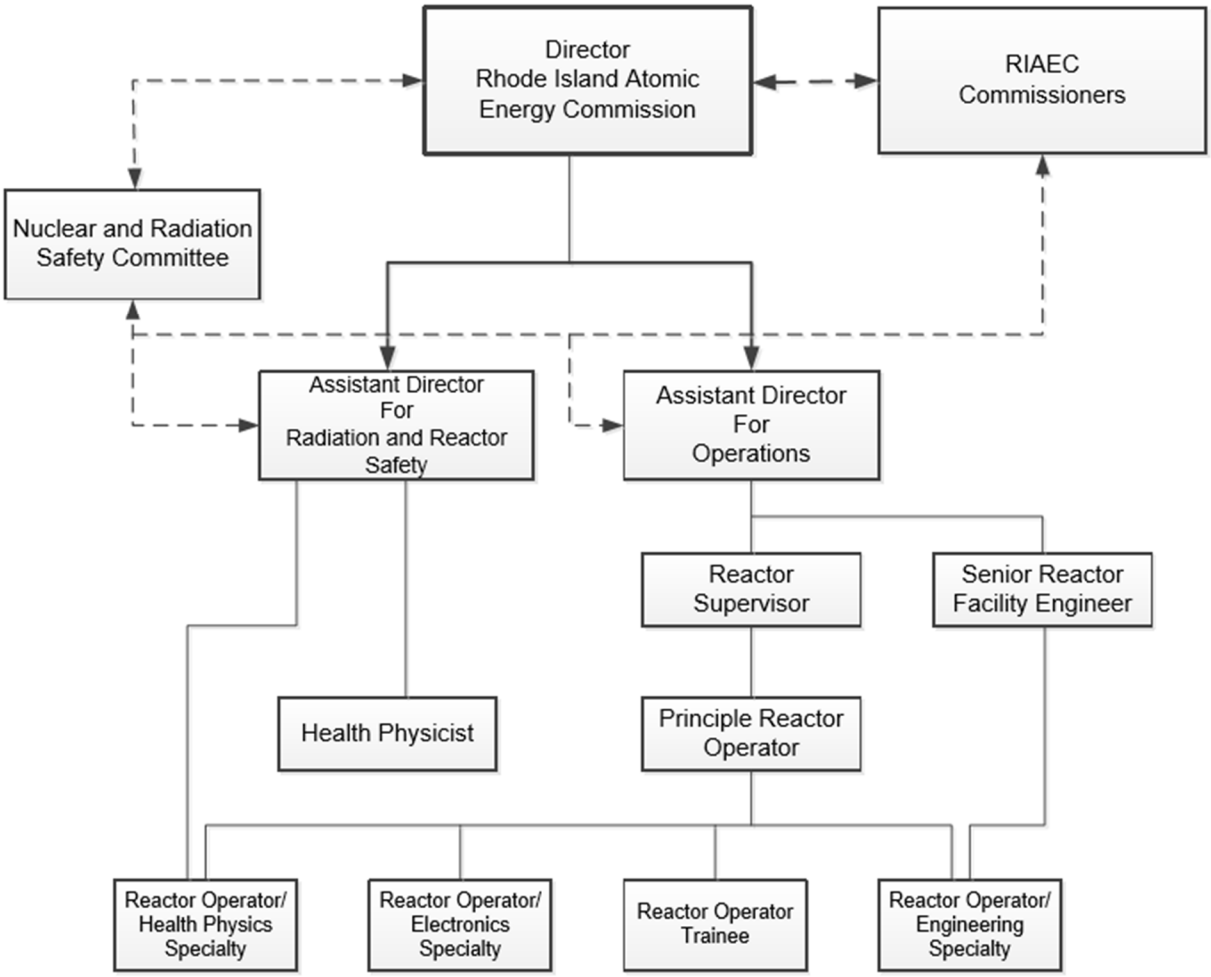

Figure 12-1 Rhode Island Atomic Energy Commission Organization Chart (Reproduced from Figure 6-1 of TS [Ref. 60])

6.1.2 Responsibility

6.1.2.1 Rhode Island Atomic Energy Commission (RIAEC)

The Rhode Island Atomic Energy Commission is the state agency that serves as the liaison between the State of Rhode Island, and the federal regulating authority. RIAEC, led by the Director, shall have the ultimate responsibility for the RINSC Reactor license. The RIAEC Commissioners provide the general direction for the utilization of the facility.

6.1.2.2 Director

The Director of the RIAEC is the organization head, and shall be responsible for the license, and for developing and directing all of the administrative and technical programs. The Director shall be responsible for ensuring facility compliance with federal and state licenses and regulations, and for all

activities in the reactor facility which may affect reactor operations or involve radiation hazards. This individual is level 1 management.

6.1.2.3 Assistant Director for Operations

The Assistant Director for Operations shall be responsible for implementing the operations programs and managing the operation of the RINSC facility. The Assistant Director shall be responsible for ensuring that operation of the reactor is compliant with the provisions of the RINSC License and Technical Specifications. This individual is level 2 management.

6.1.2.4 Assistant Director for Radiation and Reactor Safety

The Assistant Director for Reactor and Radiation Safety shall be responsible for implementing and managing the Radiation Safety Program. The Assistant Director shall ensure that that the public and facility personnel are safeguarded from undue exposure to radiation, and that the facility is compliant with federal and state radiation safety regulation. This individual is level 2 management.

6.1.2.5 Reactor Supervisor

The Reactor Supervisor shall be responsible for the day to day operation of the facility. This individual is level 3 management.

6.1.2.6 Senior Reactor Operators

The Senior Reactor Operator on duty during reactor operations shall be responsible for directing the licensed activities of Reactor Operators. The Senior Reactor Operator shall ensure that the operability of the reactor is compliant with the RINSC License and Technical Specifications during operation, and that any experiments performed during operation have been reviewed and approved by the NRSC, and are installed in accordance with any limitations prescribed by the approved experiment. The Senior Reactor Operator shall also ensure that experimenters follow facility procedures.

6.1.2.7 Reactor Operators

The Reactor Operator on duty during reactor operations shall be responsible for manipulating the controls of the reactor. The Reactor Operator shall direct the actions of Reactor Operator Trainees, and ensure that the reactor is operated within the limits of the RINSC Technical Specifications.

6.1.3 Staffing

6.1.3.1 Minimum Staffing Requirements

6.1.3.1.1 The minimum staffing requirements when the reactor is not secured there shall be a licensed Reactor Operator in the control room.

6.1.3.1.2 The minimum staffing requirements when all of the shim safety rods are not fully inserted into the core shall be two individuals present in the facility:

6.1.3.1.2.1 A Reactor Operator in the control room, and

6.1.3.1.2.2 A second individual present in the facility that has security access to the confinement building and is capable of scramming the reactor, initiating a facility evacuation, and notifying RINSC staff members and appropriate response agencies.

6.1.3.1.3 If the Senior Reactor Operator on duty is not serving as the Reactor Operator or the second individual present in the facility, they shall be readily available on call.

6.1.3.2 A Senior Reactor Operator shall be present in the facility as defined in section 5.1 during any of the following operations:

6.1.3.2.1 The initial reactor start-up and approach to power for the day,

6.1.3.2.2 Fuel element, reflector element, or control rod core position changes,

6.1.3.2.3 Recovery from an unscheduled shutdown or an unscheduled power reduction in excess of 25%.

6.1.3.3 Staff Contact List

6.1.3.3.1 A staff contact list that includes management, radiation safety, and other operations personnel shall be available in the control room for use by the Reactor Operator.

6.1.4 Selection and Training of Personnel

6.1.4.1 Qualification

6.1.4.1.1 Rhode Island Atomic Energy Commissioners

The RIAEC Commissioners shall be aware of the general operational and emergency aspects of the reactor facility.

6.1.4.1.2 Director

At the time of the appointment to the position, the Director shall have a minimum of six years of nuclear experience. The individual shall have a Bachelor of Science degree or higher in an engineering or scientific field, or an equivalent combination

of education and experience. The degree may fulfill up to four years of the six years of nuclear experience required.

6.1.4.1.3 Assistant Director for Operations

At the time of the appointment to the position, the Assistant Director shall have a minimum of six years of nuclear experience. The individual shall have a Bachelor of Science degree or higher in an engineering or scientific field, or an equivalent combination of education and experience. The degree may fulfill up to four of the six years of nuclear experience required.

6.1.4.1.4 Assistant Director for Radiation and Reactor Safety

At the time of the appointment to the position, the Assistant Director shall have a minimum of three years of health physics experience. The individual shall have a Bachelor of Science degree or higher in an engineering or scientific field, or an equivalent combination of education and experience. The degree may fulfill up to two years of the three years of nuclear experience required.

6.1.4.1.5 Reactor Supervisor

At the time of the appointment to the position, the Reactor Supervisor shall have a minimum of three years of nuclear experience, and have the training to satisfy the requirements for being a licensed Senior Reactor Operator. A maximum of two years of full time academic training may be substituted for two of the three years of nuclear experience.

6.1.4.1.6 Senior Reactor Operators

Senior Reactor Operators shall be licensed pursuant to 10 CFR Part 55.

6.1.4.1.7 Reactor Operators

Reactor Operators shall be licensed pursuant to 10 CFR Part 55.

6.1.4.2 Initial Training and Licensing

Personnel that require a Reactor Operator or Senior Reactor Operator license shall be trained in accordance with the facility Operator Training Program.

6.1.4.3 Re-Qualification and Re-Licensing

As a condition of maintaining their operating licenses, Reactor and Senior Reactor Operators shall participate in the facility Operator Re-Licensing Program.

6.1.4.4 Medical Certification

Facility senior management shall certify that the health of each Reactor Operator and Senior Reactor Operator is such that they will be able to perform their assigned duties. This certification shall be maintained in accordance with 10 CFR Part 55.23.

TS 6.1.1 establishes the organizational structure of the RINSC staff. This specification helps ensure that the RINSC organization structure is properly delineated and understood. The staff notes that on TS Figure 6.1 (SER Figure 12-1) solid and dotted lines are not defined. The licensee confirmed (Ref 58) that the lines are defined as shown in ANSI/ANS-15.1-2007, where solid lines are reporting lines and dotted lines are communication lines. The RINSC organizational structure described in this specification and shown in RINSC TS Figure 6.1, is consistent with the guidance in ANSI/ANS-15.1-2007, Section 6.1.1, as accepted in NUREG-1537. Based on the information above, the NRC staff concludes that TS 6.1.1 is acceptable.

TS 6.1.2 establishes what positions have the responsibility for implementing the RINSC license including the TSs. This helps to ensure that key positions in the organizational structure understand this responsibility. The NRC staff finds that the organizational responsibilities described in this specification are consistent with the guidance in ANSI/ANS-15.1-2007, Section 6.1.2, as accepted in NUREG-1537. Based on the information above, the NRC staff concludes that TS 6.1.2 is acceptable.

TS 6.1.3.1 describes the minimum staffing necessary to safely operate the RINSC reactor. When the reactor is not secure, a licensed operator needs to be in the control room. A second person capable of performing the actions listed in TS 6.1.3.1.2.2 needs to be present at the facility when all of the shim safety rods are not fully inserted. If the shim safety rods are not fully inserted, the reactor is not secure. The regulations in 10 CFR 50.54(k) 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.” The NRC staff finds that the specification implements this requirement, the guidance in ANSI/ANS-15.1-2007 Section 6.1.3 item 1 as accepted in NUREG-1537, and the additional guidance in NUREG-1537 regarding operator designations. Based on the information above, the NRC staff concludes that TS 6.1.3.1 is acceptable.

TS 6.1.3.2 requires an SRO to be present for certain reactor operations. The regulations in 10 CFR 50.54(m)(1) state that:

> 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 unscheduled significant reduction in power, and refueling, or as otherwise prescribed in the facility license.

The licensee has defined a significant reduction in power as a reduction greater than 25 percent. The NRC staff finds that the specification meets the requirements in 10 CFR 50.54(m)(1) and is consistent with the guidance in ANSI/ANS-15.1-2007, Section 6.1.3, Item 3 as accepted in NUREG-1537. Based on the information above, the NRC staff concludes that TS 6.1.3.2 is acceptable.

TS 6.1.3.3 requires administrative control for a contact list of personnel in management, radiation safety and operations available in the control room for the operating staff. The NRC staff finds that the specification implements the guidance in ANSI/ANS-15.1-2007, Section 6.1.3, Item 2 as accepted in NUREG-1537. Based on the information above, the NRC staff concludes that TS 6.1.3.3 is acceptable.

TS 6.1.4 requires administrative control for qualification, training, requalification and medical certification of personnel. The training of personnel is accomplished using the guidance of ANSI/ANS-15.4-2007. The facility requalification program is evaluated and found acceptable in SER Section 12.10. The medical certification in accordance with 10 CFR 55.23, "Certification," of licensed operators is the responsibility of facility senior management. Based on the information above, the NRC staff concludes that TS 6.1.4 is acceptable.

The NRC staff reviewed TS 6.1 and finds that the contained specifications are consistent with the requirements of 10 CFR 50.36, "Technical Specifications," and are consistent with the guidance of NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 6.1 is acceptable.

12.2 Review and Audit Activities

Independent review and audit functions are performed by the NRSC. TS 6.2.1, "Nuclear Radiation Safety Committee (NRSC) Composition and Qualifications," defines the membership of the NRSC, which is comprised of qualified members from both within and outside the RIAEC, and advises the Director on all matters or policy pertaining to safety. TS 6.1.1 requires that members be appointed by the RIAEC. TS 6.2, "Review and Audit," outlines meeting frequencies, quorums, frequencies of audits, and lists audit activities.

TS 6.2 states:

6.2 Review and Audit

6.2.1 Nuclear and Radiation Safety Committee (NRSC) Composition and Qualifications

6.2.1.1 Composition

The NRSC shall be comprised of a minimum of seven individuals:

6.2.1.1.1 The Director

6.2.1.1.2 The Assistant Director for Operations

6.2.1.1.3 The Assistant Director for Radiation and Reactor Safety

6.2.1.1.4 Four members that are not RIAEC commissioners or staff

6.2.1.2 Qualification

The collective qualification of the NRSC members shall represent a broad spectrum of expertise in science and engineering.

6.2.1.3 Alternates

Qualified alternates may serve in the absence of regular members.

6.2.2 Nuclear and Radiation Safety Committee Charter

The NRSC shall have a written Charter that specifies:

6.2.2.1 Meeting frequency of not less than once per year.

6.2.2.2 Quorum shall consist of a minimum of four (4) members, including the Assistant Director for Radiation and Reactor Safety or designee, and the Director or Assistant Director for Operations.

6.2.2.3 NRSC Minutes shall be reviewed and approved at the next committee meeting.

6.2.3 Review Function

All review results will be documented in the NRSC meeting minutes. The NRSC shall review the following items:

6.2.3.1 Proposed changes to the Technical Specifications,

6.2.3.2 Violations of the Technical Specifications,

6.2.3.3 Proposed changes to the License,

6.2.3.4 Violations of the License,

6.2.3.5 Proposed changes to the NRSC Charter,

6.2.3.6 10 CFR Part 50.59 evaluations,

6.2.3.7 New procedures,

6.2.3.8 Major changes to procedures that have safety significance,

6.2.3.9 Violations of procedures that have safety significance,

6.2.3.10 New experiments,

6.2.3.11 Operating abnormalities that have a safety significance, and

6.2.3.12 Reportable occurrences.

6.2.4 Audit Function

The audit function is normally performed in conjunction with a scheduled NRSC meeting. The non-RIAEC staff members of the NRSC shall audit the following items either before or after the meeting and identify any discrepancies for resolution:

6.2.4.1 Reactor operations shall be audited at least annually to verify that the facility is operated in a manner consistent with public safety and within the terms of the facility license.

6.2.4.2 The Operator Re-Qualification Program shall be audited at least biennially,

6.2.4.3 The Emergency Plan and Emergency Plan Implementing Procedures shall be audited at least biennially,

6.2.4.4 Actions taken to correct any deficiencies found in the facility equipment, systems, structures, or methods of operation that could affect reactor safety shall be audited at least annually, and

6.2.4.5 The Radiation Safety Program shall be audited at least annually.

6.2.4.6 Results of the audit will be captured in the NRSC meeting minutes

TS 6.2.1 delineates the NRSC composition and qualification requirements. The NRC staff finds that the requirements in this specification are consistent with the guidance provided in ANSI/ANS-15.1-2007, Subsections 6.2.1 and 6.2.3, as accepted in NUREG-1537. Based on the information above, the NRC staff concludes that TS 6.2.1 is acceptable.

TS 6.2.2 delineates the NRSC charter and rules. The NRC staff finds that the requirements in this specification are consistent with the guidance provided in ANSI/ANS-15.1-2007, Section 6.2.2, as accepted in NUREG-1537. Based on the information above, the NRC staff concludes that TS 6.2.2 is acceptable.

TS 6.2.3 establishes a review process for activities that are reviewed by the NRSC. This review is consistent with the management philosophy and structure expressed in TS 6.1. It also implements the guidance from NUREG-1537, Section 6.2.3, by requiring the NRSC to review items pursuant to 10 CFR 50.59, “Changes, Tests, and Experiments.” The NRC staff notes that a review of audit reports is not specifically on the list of items reviewed by the NRSC. This is because the audit is performed in conjunction with a scheduled NRSC meeting. TS 6.2.4.6 requires the results of the audit to be captured in NRSC meeting minutes, which in accordance with TS 6.2.2.3, are reviewed and approved. Based on the information above, the NRC staff concludes that TS 6.2.3 is acceptable.

TS 6.2.4 requires administrative control for the conduct of audits and the reporting of findings. The NRC staff notes that ANSI/ANS-15.1-2007 suggests that deficiencies uncovered during audits that affect reactor safety are immediately reported to Level 1 management. Because the audit function is normally performed in conjunction with a scheduled NRSC meeting and the Director (Level 1 management) is a member of the Committee, the Director will be immediately informed of audit discovered safety significant deficiencies. The NRC staff finds that the

requirements in this specification are consistent with the guidance provided in ANSI/ANS-15.1-2007, Section 6.2.4, and NUREG-1537. Based on the information above, the NRC staff concludes that TS 6.2.4 is acceptable.

The NRC staff reviewed TS 6.2 and finds that the contained specifications that are consistent with the guidance of NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff finds that TS 6.2 is acceptable.

The NRC staff reviewed the licensee's structure for the conduct of review and audit activities. Based on the structure and composition of the NRSC, as well as listed scope and frequency of activities, the NRC staff finds that review and audit activities are consistent with the guidance of NUREG-1537 and ANSI/ANS-15.1-2007 and are therefore, acceptable.

12.3 Procedures

Written approved procedures govern all aspects of the reactor facility's operation and use of special nuclear and byproduct materials. These procedures, whose existence and scope are specified in TS 6.4, "Procedures," encompass, but are not limited to, the following areas:

- startup, operation, and shutdown of the reactor
- core loading, unloading, and fuel handling
- routine maintenance of major components of systems that could have an effect on reactor safety
- surveillance test or calibrations required by TS or have a significant effect on reactor safety
- implementation of Emergency and Security plans
- experiment evaluation and authorization
- radiation control procedures

Required procedures are reviewed by the NRSC and approved by level 1 or 2 management prior to use per TS 6.4.2.

TS 6.4 states:

> 6.4 Procedures
>
> 6.4.1 Written procedures shall be used that are adequate to assure the safe operation of the reactor, but should not preclude the use of independent judgment and action should the situation require such.
>
> 6.4.2 The procedures for the following activities shall be reviewed by the NRSC, and approved by level 1 or level 2 management prior to use:
>
> 6.4.2.1 Startup, operation, and shutdown of the reactor,
>
> 6.4.2.2 Fuel loading, unloading, and movement within the reactor,
>
> 6.4.2.3 Maintenance of major components of systems that could have an effect on reactor safety,

6.4.2.4 Surveillance checks, calibrations, and inspections that are required by the Technical Specifications, or have a significant effect on reactor safety,

6.4.2.5 Radiation safety,

6.4.2.6 Administrative controls for operations, maintenance, and experiments that could affect reactor safety or core reactivity,

6.4.2.7 Implementation of the Emergency and Security plans, and.

6.4.2.8 Receipt, use, and transfer of byproduct material.

The NRC staff reviewed TS 6.4, as well as the scope and approval process for procedures required by TSs at the RINSC and found both to be consistent with the guidance of NUREG-1537 and ANSI/ANS-15.1-2007 and similar to required procedures at other RTR facilities. Based on the information above, the NRC staff finds that the process and methodology provided in the SAR and TSs ensure proper control and review of procedures. Based on the information above, the NRC staff concludes that TS 6.4 is acceptable.

12.4 Required Actions

TS 6.6, "Required actions," contains the actions to be taken by the licensee for SL violations and for reportable occurrences. The RINSC reactor SL is related to fuel cladding temperature. Should the SL be exceeded, TS 6.6.1, "Action to be Taken in the Event of a Safety Limit Violation," requires reactor shutdown, facility management notification, and NRC notification in accordance with TS 6.7.2, "Special Reports."

TS 1.31, "Reportable Occurrence," lists reportable events (TS 1.3.1.1 is violation of the SL subject to TS 6.6.1). TS 6.6.2, "Action to be Taken in the Event of a Reportable Occurrence Other Than a Safety Limit Violation," discusses the actions to be taken: place the facility in safe condition, Director or Assistant Director notification, review by the NRSC, and NRC notification as per TS 6.7.2, "Special Reports." The definition of reportable occurrences gives reasonable assurance that the licensee will report safety significant events. The NRC staff finds that these TSs will help to ensure that the licensee will take the actions that are necessary to protect public health and safety.

TS 6.6 states:

6.6 Required Actions

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

6.6.1.1 The reactor shall be shut down and reactor operations shall not be resumed until authorization is obtained from the NRC.

6.6.1.2 Immediate notification shall be made to the Director and to the NRSC members.

6.6.1.3 Notification shall be made to the NRC in accordance with paragraph 6.7.2 of these specifications.

6.6.1.4 A safety limit violation report shall be prepared. The report shall include:

6.6.1.4.1 A complete analysis of the causes of the event,

6.6.1.4.2 The extent of possible damage to facility components, systems, or structures

6.6.1.4.3 A statement regarding the impact of the event on the facility personnel.

6.6.1.4.4 A statement regarding the impact of the event on the public.

6.6.1.4.5 A description of any radioactive material release to the environment.

6.6.1.4.6 Corrective actions taken to prevent or reduce the probability of recurrence.

6.6.1.5 The safety limit violation report shall be submitted to the NRSC for review and appropriate action.

6.6.1.6 The safety limit violation report shall be submitted to the NRC in accordance with Paragraph 6.7.2 of these specifications in support of a request for authorization to resume reactor operations.

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

6.6.2.1 The reactor shall be shutdown.

6.6.2.2 The Senior Reactor Operator shall be notified promptly and corrective action shall be taken immediately to place the facility in a safe condition until the cause of the reportable occurrence is determined and corrected.

6.6.2.3 The occurrence shall be reported to the Director or Assistant Director.

6.6.2.4 Operations shall not be resumed without authorization from the Director or Assistant Director for Operations.

6.6.2.5 The occurrence, and corrective action taken shall be reviewed by the NRSC during its next scheduled meeting.

6.6.2.6 Notification shall be made to the NRC in accordance with Paragraph 6.7.2 of these specifications.

The licensee has defined a group of incidents as reportable occurrences and has described the required actions it will take if a reportable occurrence occurs (see SER Section 14.1). The definition of reportable occurrence gives reasonable assurance that safety significant events will be reported to the NRC by the licensee. The licensee has also proposed actions to be taken if a SL is violated or other reportable occurrence occurs. The NRC staff finds that these processes will help to ensure that the licensee will take the actions that are necessary to protect the health and safety of the public.

TS 6.6.1 establishes controls over actions to be taken in the event that a SL is violated. It requires reactor shutdown, prompt reporting to the NRC in compliance with the regulations in 10 CFR 50.36(c)(1), a detailed follow-up report, and timely submission of that report. The specification is consistent with the guidance from ANSI/ANS-15.1-2007, Section 6.6.1, as accepted in NUREG-1537. Based on the information above, the NRC staff concludes that TS 6.6.1 is acceptable.

TS 6.6.2 establishes actions to be taken in the event of a reportable occurrence other than violation of the SL as in TS 6.6.1. The NRC staff finds that this specification helps to ensure that abnormal occurrences, other than violation of the SL, as defined in TS 1.31, are reported and that, if necessary, the reactor is shut down until operation is allowed to resume when authorized by the RINSC management. The specification is consistent with the guidance from ANSI/ANS-15.1-2007, Section 6.6.2, as accepted in NUREG-1537. Based on the information above, the NRC staff concludes that TS 6.6.2 is acceptable.

The NRC staff reviewed TS 6.6. The NRC staff finds that TS 6.6 is consistent with the guidance of NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 6.6 is acceptable.

Based on the above review, the NRC staff concludes that the required actions are appropriate and provide reasonable assurance that the facility will respond to the defined occurrences in a manner consistent with maintaining reactor safety and protection of the health and safety of the public.

12.5 Reports

TS 6.7 specifies reports that the licensee is required to make to the NRC. These include an annual operating report and special reports. TS 6.7.1 lists the required contents of the annual operating report including operational history, major maintenance performed, approved changes to the facility, and radioactive effluents. TS 6.7.2 discusses how to file reports for violation of the SL and other reportable occurrences.

TS 6.7 states:

> 6.7 Reports
>
> 6.7.1 Annual Report
>
> A written report shall be submitted annually to the NRC following the 30th of June of each year, and shall include a summary of reactor operating experience. The following information shall be provided as a minimum:
>
> 6.7.1.1 A summary of the number of hours that the reactor was critical for the period, the energy produced for the period, and the cumulative total energy output since initial criticality;
>
> 6.7.1.2 A summary of the unscheduled shutdowns that occurred during the period, the causes of the shutdowns, and if applicable, corrective action taken to preclude recurrence;

6.7.1.3 A summary of any major maintenance performed during the period that has safety significance, and the reasons for any corrective maintenance required;

6.7.1.4 A summary of 10 CFR Part 50.59 safety evaluations made during the reporting period;

6.7.1.5 A summary of the amount of radioactive effluents, and to the extent possible, an estimate of the individual radionuclides that have been released or discharged to the environs outside the facility as measured at or prior to the point of release.
If the estimated average release after dilution or diffusion is less than 25% of the concentration allowed, a statement to this effect is sufficient for the summary.

6.7.1.6 A summary of the results of environmental surveys performed outside the facility during the reporting period that includes the locations of the surveys; and

6.7.1.7 A summary of annual radiation exposures in excess of 500 mrem received by facility personnel, or 100 mrem received by non-staff members, or 10 mrem received by members of the general public.

6.7.2 Special Reports

6.7.2.1 Reporting Requirements for Reportable Occurrences

In the event of a reportable occurrence, the following notifications shall be made:

6.7.2.1.1 Within one working day after the occurrence has been discovered, the NRC Headquarters Operation Center shall be notified by telephone, with written follow-up confirmation within 24 hours, at the number listed in 10 CFR Part 20 Appendix D, and

6.7.2.1.2 Within 14 days after the occurrence has been discovered, a written report that describes the circumstances of the event shall be sent to the NRC Document Control Desk at the address listed in 10 CFR Part 50.4.

6.7.2.2 Other Reporting Requirements

6.7.2.2.1 A written report shall be submitted to the NRC within 30 days after the following occurs:

6.7.2.2.1.1 Permanent changes in the facility organization involving level 1 or 2 personnel.

6.7.2.2.1.2 Significant changes in the transient or accident analysis as described in the Safety Analysis Report

TS 6.7.1 establishes requirements for the submittal of the annual operating report to the NRC. The NRC staff finds that this specification helps to ensure that important information will be provided to the NRC in a timely manner. The specification is consistent with the guidance in ANSI/ANS-15.1-2007, Section 6.7.1, as accepted in NUREG-1537. Furthermore, it implements the specific requirement regarding reporting required by 10 CFR 50.59 as cited in NUREG-1537, Appendix 14.1, Section 6.7.1. Based on the information above, the NRC staff concludes that TS 6.7.1 is acceptable.

TS 6.7.2 establishes requirements for the submittal of special reports. The NRC staff finds this specification helps to establish controls over the reporting of changes to certain analyses or to the RINSC organization. The specification is consistent with the guidance in ANSI/ANS-15.1-2007, Section 6.7.2, and NUREG-1537. Based on the information above, the NRC staff concludes that TS 6.7.2 is acceptable.

The NRC staff reviewed TS 6.7. The NRC staff finds that TS 6.7 is consistent with the guidance of NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 6.7.2 is acceptable.

The licensee has described the content and the timing of the submittal of reports to help ensure that important information will be provided to NRC in a timely manner. The NRC staff concludes there is reasonable assurance that the licensee will report appropriate information regarding routine operation, non-routine occurrences, and changes to the facility and personnel to the NRC in a timely manner.

12.6 Records

SAR Chapter 12 and TS 6.8, “Records,” lists the records required and their retention period for three categories of records:

TS 6.8.1, “Records that shall be retained for at least five years”

TS 6.8.2, “Records that shall be retained for a period of at least one certification cycle”

TS 6.8.3, “Records that shall be retained for the life of the reactor facility”

Records required to be kept for 5 years include: reactor operations, principal maintenance activities, experiments performed, surveillance activities, radiation monitoring surveys, fuel inventories and transfers, changes to procedures and NRSC meeting minutes. Records required to be kept for the life of the facility include: effluent records, off-site environmental monitoring surveys, personnel radiation exposures, drawings of the facility, and reportable occurrences.

TS 6.8 states:

6.8 Records

6.8.1 Records that shall be retained for a period of at least five years

6.8.1.1 Reactor operating records,

6.8.1.2 Principal maintenance activities,

6.8.1.3 Surveillance activities required by the Technical Specifications,

6.8.1.4 Facility radiation monitoring surveys,

6.8.1.5 Experiments performed with the reactor,

6.8.1.6 Fuel inventories and transfers,

6.8.1.7 Changes to procedures, and

6.8.1.8 NRSC meeting minutes, including audit findings.

6.8.2 Records that shall be retained for a period of at least one certification cycle

6.8.2.1 Current Reactor Operator re-qualification records shall be maintained for each individual licensed to operate the reactor until their license is terminated.

6.8.3 Records to be retained for the life of the facility

6.8.3.1 Gaseous and liquid radioactive effluents released to the environs,

6.8.3.2 Off-site environmental monitoring surveys,

6.8.3.3 Personnel radiation exposures,

6.8.3.4 Drawings of the reactor facility, and

6.8.3.5 Reportable occurrences.

TS 6.8 establishes requirements for the retention of records that are required to be retained. Such records include Lifetime Records (TS 6.8.3), Five Year Records (TS 6.8.1), and Operator Licensing Records (TS 6.8.2). Details regarding each category are articulated in the specification. This specification helps to ensure a consistent interpretation of record keeping responsibilities. The specification implements the guidance that is consistent with ANSI/ANS-15.1-2007, Section 6.8.1, as accepted in NUREG-1537. Based on the information above, the NRC staff concludes that TS 6.8 is acceptable.

The NRC staff reviewed the list of record categories in TS 6.8 and finds them to be acceptable. The licensee has described the types of records that will be retained by the facility and the period of retention to ensure that important records will be retained for an appropriate time. Based on the above information, the NRC staff concludes that appropriate records will be retained and stored consistent with the guidance in ANSI/ANS-15.1-2007 and NUREG-1537.

12.7 Emergency Planning

The NRC staff conducted a formal review of the RINSC EP dated March 15, 2007 (Ref. 51), and determined that the RINSC EP is compliant with the following regulations and guidance:

- 10 CFR 50.54(q), “Emergency Plans”
- Regulatory Guide 2.6, “Emergency Planning for Research and Test Reactors” (Ref. 76)
- ANSI-15.16-1982, “Emergency Planning for Research Reactors”
- NUREG-0849, “Standard Review Plan for the Review and Evaluation of Emergency Plans for Research and Test Reactors” (Ref. 77)
- NRC Information Notice 97-34, “Deficiencies in Licensee Submittals Regarding Terminology for Radiological Emergency Action Levels in Accordance with the New Part 20” (Ref. 78)
- NRC Information Notice 92-79, “Nonpower Reactor Emergency Response” (Ref. 79)

The NRC staff routinely inspects the licensee’s compliance with the requirements of the EP, and no violations have been identified in recent years based on the NRC staff’s review of IRs for years from 2011 through 2016 (Ref. 28).

The NRC staff reviewed the revisions to the EP (Ref. 52) since March 15, 2007, and determined that the revisions to the plan did not affect the previous NRC staff findings. The NRC staff finds that the licensee’s EP is in compliance with 10 CFR 50.54(q), “Emergency Plans,” which requires research reactor EPs to adhere to the requirements in Appendix E, “Emergency Planning and Preparedness for Production and Utilization Facilities,” to 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities.” The NRC staff concludes the licensee’s EP provides reasonable assurance that the licensee will follow and maintain the effectiveness of an emergency plan that meets the requirements in 10 CFR Part 50, Appendix E.

12.8 Security Planning

The NRC staff reviewed the RINSC reactor PSP dated May 25, 2016, and determined that it is in compliance with the applicable regulations contained in 10 CFR Part 73, “Physical protection of plants and materials,” as referenced in Regulatory Guide 5.59 “Standard Format and Content for a Licensee Physical Security Plan for the Protection of Special Nuclear Material of Moderate or Low Strategic Significance” (Ref. 80). Changes to the PSP 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 PSP. The NRC staff’s review of the NRC IRs from the years 2011 through 2016 (Ref. 28) for the RINSC facility identified no violations of the PSP requirements.

In addition, in a separate security review, the NRC staff found that the site-specific Compensatory Measures committed to in confirmatory action letter (CAL) No. NRR-02-003, have also been incorporated into the RINSC security plan. Therefore, the NRC issued a letter dated June 28, 2016 (Ref. 50), to close CAL No. NRR-02-003.

Based on its review, the NRC staff finds that the licensee maintains a PSP for the facility and its SNM, in accordance with the requirements of 10 CFR Part 73. Therefore, based on the information above, the NRC staff concludes that there is reasonable assurance that the licensee will continue to provide for the physical protection of the facility and its SNM, and that continued operation of the RINSC reactor will not be inimical to the common defense and security.

12.9 Quality Assurance

RINSC is not required to have a quality assurance program.

12.10 Operator Training and Requalification

The NRC staff reviewed the updated requalification plan for the RINSC reactor dated April 28, 2014 (Ref. 3), and found it to be in accordance with the applicable regulations in 10 CFR Part 55 and consistent with the guidance contained in ANSI/ANS-15.4-2007, "Selection and Training of Personnel for Research Reactor" (Ref. 17). Based on the information above, the NRC staff concludes that the operating training and requalification program is adequate and consistent with the guidance documents.

12.11 Startup Plan

A startup plan is required for a new facility and for license amendments authorizing modifications that require verification of operability before normal operations are resumed. The RINSC reactor has been operating successfully for many years and is not submitting such modifications with this license renewal.

12.12 Conclusions

Based on its review of information above, the NRC staff finds that the licensee has sufficient oversight, management positions and responsibilities structure, and procedures to provide reasonable assurance that the reactor will continue to be managed in a way that will not cause significant risk to public health and safety.

The NRC staff reviewed SAR Chapter 12, as supplemented by responses to RAIs, and the applicable specifications in TS Chapter 6, 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 in NUREG-1537 and ANSI/ANS-15.1-2007. The NRC staff finds that the licensee's proposed conduct of operations in the areas reviewed is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. The NRC staff also reviewed the applicable proposed RINSC TS Chapter 6 against the requirements in 10 CFR 50.36 "Technical Specifications," including 10 CFR 50.36(d)(5) and (7) and finds that the TSs meet the requirements of the regulations.

Based on information above, the NRC staff concludes that the licensee has the appropriate organization, experience levels, and adequate controls through the TSs to provide reasonable assurance that the RINSC is managed and operated in a manner that will not cause significant radiological risk to the facility staff or to members of the public.

13. ACCIDENT ANALYSES

The RINSC SAR, as supplemented, provides a series of accident analyses to demonstrate that the health and safety of members of the public and workers are protected during analyzed reactor transients and other hypothetical accident scenarios. The accident analyses help to justify the SL and LSSS that are imposed on the RINSC reactor through the TSs evaluated and found acceptable in this report. The accident analysis helps ensure that no credible accident could lead to unacceptable radiological consequences to the RINSC staff, the members of the public, or the environment. Additionally, the licensee analyzes the consequences of a MHA, which is considered the worst-case fuel failure scenario for the RINSC reactor that would lead to the maximum (bounding) potential radiation hazard to facility personnel and members of the public from the release of fission products. The results of the MHA are used to evaluate the ability of the licensee to respond to and mitigate the consequences of this postulated radioactive release.

The NRC staff reviewed the licensee's analytical assumptions, methods, and results. In addition, the NRC staff performed certain independent calculations and compared those results with the results obtained by the licensee. As will be demonstrated below, none of the potential accidents considered in the SAR, as supplemented, would lead to unacceptable occupational or members of the public exposure.

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

- MHA;
- Insertion of excess reactivity;
- LOCA;
- Loss of coolant flow;
- Mishandling or malfunction of fuel;
- Experiment malfunction;
- Loss of electrical power;
- External events; and,
- Mishandling or malfunction of equipment.

13.1 Maximum Hypothetical Accident

The licensee has determined that the accident scenario with the greatest potential for radiological consequences from fission products (the MHA) is the failure of a fissionable experiment, which would result in the release of fission products. The licensee provides its MHA analysis in a SAR supplement (Ref. 56). Some information from the SAR and the licensee's fuel failure SAR supplement (Ref. 56) are included to supplement the discussion below.

Accident Scenario

The MHA scenario assumes the failure of a fissionable experiment in air in the reactor confinement. TS 3.8.1.4.3, which is discussed and found acceptable in SER Section 10.3, requires that experiments containing fissionable materials shall be doubly encapsulated, and therefore it is unlikely that a fissionable experiment would fail in such a manner as to cause an uncontrolled release of radioactive material into confinement. However, the MHA scenario conservatively assumes that the encapsulation, and that the entire iodine and noble gas fission product inventory of the experiment is available to be released to the confinement.

TS 3.8.1.4.2, which is also discussed and found acceptable in SER Section 10.3, requires that the maximum quantity of fissionable materials used in an experiment shall be no greater than 87.5 milligrams of uranium-235 equivalent (i.e., 87.5 milligrams of uranium-235, or a quantity of another fissionable material whose irradiation would produce a comparable fission product inventory), and the fission rate in a fissionable material experiment shall be no greater than 2.1×10^{12} fissions per second. These two parameters serve to bind the fission products produced in the experiment to the fission product inventory assumed in the analysis.

The scenario assumes that prior to the MHA, the fissionable experiment has been in the reactor for sufficient time for the experiment, which contains 87.5 milligrams of uranium-235 and has a fission rate of 2.1×10^{12} fissions per second, consistent with TS 3.8.1.4.2, to reach fission product inventory saturation levels. Particulate fission products are assumed not to be released from the experiment, and therefore they are not considered in the licensee's analysis. Gaseous fission products (iodines and noble gases) are assumed to be released from the experiment to the reactor confinement. The licensee assumed that the elevated radiation readings generated by the fuel failure would be detected by installed radiation monitors (required by TS 3.7.1.1, which is discussed and found acceptable in SER Section 7.7), and in response to the alarm, the RO would activate the facility evacuation system, which would, in turn, activate the confinement isolation and emergency exhaust system (TS 4.5.2, which is discussed and found acceptable in SER Section 6.2.1, requires that it be verified that the CVS emergency mode activate when the facility evacuation alarm activates). Activation of this system re-aligns the ventilation system to route confinement air through high-efficiency particulate air (HEPA) and charcoal filters prior to its release to the environment through the facility stack. As required by TS 3.5.1, 4.5.2, and 4.5.3 (which are discussed and found acceptable in SER Section 6.2.1), the facility ventilation system, whether operating in normal or emergency mode, shall maintain the confinement building pressure at least 0.5 in of water below atmospheric pressure whenever the reactor is operating or other activities are in progress that could result in a radioactive material release. This helps ensure that any air leakage is into, not out of, confinement, and any radioactive material released to confinement will be released through the stack such that it can be monitored, filtered and diluted before release, and adequately dispersed.

Actuation of the evacuation system and the confinement isolation and emergency exhaust system will prompt operations personnel to ensure that a total evacuation of the reactor building is accomplished promptly. A conservative 5-minute evacuation time is assumed for the dose calculations for personnel in the building.

The NRC staff reviewed the MHA scenario described above. The NRC staff finds that the licensee's assumption that the entire iodine and noble gas fission product inventory of the failed experiment is available for release to the confinement is conservative, and it bounds any credible fissionable experiment release that could occur. The NRC staff additionally finds that the licensee's assumption that particulates would not be released from the experiment is

reasonable, because the particulate fission products are non-volatile and any quantity released from the experiment would not be significant. The NRC staff also finds that the other MHA scenario assumptions and boundary conditions above would lead to conservative estimates for doses to both occupational workers and members of the public. Therefore, the NRC staff concludes that these assumptions are acceptable.

Radionuclide Inventory

The licensee calculated the saturated radionuclide inventory in the fissionable experiment. The licensee performed its inventory calculation by scaling the saturated inventory of the highest-power fuel plate in the core (see SER Sections 4.6 and 13.5), which is assumed to contain 12.5 grams of uranium-235, to the experiment which is assumed to contain 87.5 milligrams of uranium-235. Given the fission rate of approximately 3×10^{14} fissions per second that the licensee assumed for its fuel plate failure analysis, the fission rate would be approximately 2.1×10^{12} fissions per second in the experiment considered in the MHA analysis, given that the fission density (fissions per gram per second) in the failed fuel plate and the failed experiment would be the same. The licensee calculated the saturated inventories in the experiment for select iodine and noble gas (krypton and xenon) radionuclides. The licensee did not include short-lived noble gases with half-lives of less than about 15 minutes. Table 13-1 summarizes the licensee's calculated iodine and noble gas inventories in the experiment.

The NRC staff reviewed the licensee's inventory estimate. The NRC staff finds that the inventory estimate is based on the largest fissionable experiment with the maximum fission rate allowed by TS 3.8.1.4.2. The NRC staff also finds that the licensee's inventory estimate assumes that the experiment has been in the reactor for a long period of time, such that the inventory is saturated. The NRC staff reviewed the fission yields used by the licensee for its calculation against published fission yields (Refs. 29, 31, 47), and finds that although the values vary slightly, the difference is not significant. The NRC staff also reviewed the list of radionuclides considered in the licensee's inventory estimate. The NRC staff notes that because the fissionable experiment could potentially fail during operation, there is the possibility that gaseous fission products could be released to confinement rapidly, with no time for decay. Therefore, the licensee's exclusion of short-lived noble gases isotopes (i.e., those with half-lives of approximately 15 minutes or less) may not be reasonable, particularly for occupational dose calculations. The NRC staff noted 1 noble gas isotope, xenon-138, with a half-life of approximately 14 minutes, and 4 additional noble gas isotopes, krypton-89, krypton-90, xenon-137, and xenon-139, with half-lives of less than 4 minutes, which could be of concern.

For the 4 noble gases with half-lives less than 4 minutes, the NRC staff finds that it is reasonable to exclude them from members of the public dose calculations, given that most of the inventory of these noble gases would have decayed out by the time the gases are realistically dispersed within confinement, travelled through the ventilation system and stack, and reached a receptor outside the facility. However, the half-life of xenon-138 is long enough that it would still be present for approximately 2 hours following its release from an operating experiment. Therefore, the NRC staff considered xenon-138 in its confirmatory calculation of members of the public MHA doses, which is discussed below. In its confirmatory calculation of occupational MHA doses, which is also discussed below, the NRC staff considered all 5 of the short-lived noble gas isotopes discussed above. (The licensee conservatively assumed that for the radionuclides that are considered in its MHA analyses, no decay occurs at any point following the initiation of the accident scenario. In its confirmatory calculations, the NRC staff used the same conservative decay assumption for the radionuclides considered by the licensee and for xenon-138. However, for the 4 noble gas isotopes with half-lives less than 4 minutes,

the NRC staff considered the decay that would occur during the 5-minute stay time for occupational workers in confinement.) Based on the information above, the NRC staff concludes that the licensee's MHA inventory estimate shown in Table 13-1 is acceptable, except as noted above.

Table 13-1 RINSC Estimates of the MHA Nuclide Inventory

Nuclides	Estimate of the saturated inventory in the fissionable experiment (Ci)
I-131	1.57×10^{0}
I-132	2.34×10^{0}
I-133	3.84×10^{0}
I-134	4.08×10^{0}
I-135	3.63×10^{0}
Kr-85m	7.55×10^{-1}
Kr-85	1.62×10^{-1}
Kr-87	1.35×10^{0}
Kr-88	2.07×10^{0}
Xe-133m	1.07×10^{-1}
Xe-133	3.84×10^{0}
Xe-135m	5.96×10^{-1}
Xe-135	3.81×10^{0}

Release Fractions

The licensee assumed that 100 percent of the noble gas and iodine inventories listed in Table 13-1 are released to the confinement air, and are instantaneously and uniformly dispersed in the confinement air. TS 5.5.1, which is discussed and found acceptable in SER Section 6.2.1, requires that the free air volume of the confinement building, in which the material released from the experiment to confinement will be dispersed, shall be 181,955 cubic ft.

The licensee used additional release fraction assumptions in determining the material released from the confinement air to the environment. As discussed above, the licensee also assumed that the confinement isolation and emergency exhaust system is manually activated in conjunction with the release of radioactive material from the pool to confinement and the activation of the confinement RMS. TS 4.5.4, which is discussed and found acceptable in SER Section 6.2.1, requires that the emergency filter bank of the emergency exhaust system be verified to be at least 99 percent efficient for removing iodine. Based on the guidance in RG 1.183 (Ref. 34), the licensee considered that the iodine released from the experiment consists of 43 percent organic and 57 percent elemental iodine, and assumed that only the elemental iodine can be absorbed by the charcoal filter. Therefore, the licensee calculated an additional release fraction of approximately 0.436 for iodine released from confinement to the environment, based on the assumptions that 100 percent of the organic iodine, and 1 percent of the elemental iodine, pass through the charcoal filter. The licensee did not take credit for any other iodine hold-up or plate-out in the reactor building or ventilation system. For noble gases, the licensee assumed that the entire inventory released to the confinement air is also available to be released to the environment. The NRC staff reviewed this information, and also reviewed RG 1.183, and notes that the assumption that iodine released from the experiment is 43 percent organic and 57 percent elemental is typically used for iodine that has passed through water,

such as a reactor pool, before being released to air, since the water scrubs a much larger fraction of inorganic than organic iodine. For iodine that is released directly to air, such as the iodine that is released from the RINSC fissionable experiment, the organic fraction of the iodine would be much smaller. Therefore, the licensee's assumption that the iodine released from the experiment is 43 percent organic, and consequently the assumption of a release fraction of 0.436 for iodine passing through the charcoal filter to the environment, are extremely conservative. The NRC staff finds that the licensee's release fractions from the confinement to the environment are consistent with the RINSC TSs, and are consistent with, or more conservative than, guidance in RG 1.183.

The NRC staff reviewed the information above, and finds that the release fractions assumed by the licensee for its MHA analysis are justifiable, conservative, and consistent with guidance and established practice, and the RINSC TSs. Therefore, based on the information above, the NRC staff concludes that the licensee's release fractions are acceptable.

Atmospheric Dispersion

TS 4.5.5, which is discussed and found acceptable in SER Section 6.2.1, requires that ventilation flow through the emergency filter bank of the emergency exhaust system be verified to be no greater than 1,500 cubic ft per minute. The licensee used this air flow rate, the concentrations of radioactive material in confinement, and the release fractions from confinement to the environment (discussed above) to calculate the release rates of material from the 115 ft (35.052 m) high facility stack.

The licensee used guidance in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants" (Ref. 39), for evaluating atmospheric dispersion for its calculations of doses to members of the public from material released from the facility stack. Because RINSC occupies a coastal site (the stack is located 150 m [492.12 ft] from Naragansett Bay), the licensee assumed that fumigation conditions exist for the entire duration of the release. (RG 1.145 states that for coastal sites, calculations should be performed for fumigation and non-fumigation conditions, and the more conservative results used. The licensee previously supplied calculations (Ref. 54) verifying that fumigation conditions are more conservative for calculations for the RINSC site boundary, maximum dose location, and nearest residence.) Consistent with RG 1.145 guidance for fumigation conditions, the licensee assumed that Pasquill F atmospheric stability conditions exist for the entire duration of the release. The licensee also assumed a wind speed of 1 m (3.2 ft) per second, which is more conservative than the 2 m (6.5 ft) per second wind speed recommended in RG 1.145. Additionally, the licensee assumed that the wind blows in the directions of the receptors for the entire release duration.

The licensee performed calculations of radioactive material concentrations and members of the public doses at the RINSC site boundary, the location of the maximally-exposed member of the public, and the nearest residence. The nearest distance to the RINSC site boundary is 50 m (164 ft) from the stack. The licensee's calculations showed that, for the fumigation conditions considered, a member of the public at the site boundary would receive a dose higher than any member of the public located outside the site boundary, and therefore the site boundary would be the location of the maximally-exposed member of the public. The nearest residences to the facility are located approximately 500 m (1,640 ft) west-northwest and south of the stack.

The licensee assumed that the receptors at the site boundary and nearest residence are present for 2 hours following the start of the accident. The NRC staff finds that given the flow

rate from confinement (1,500 cubic ft per minute as specified in TS 4.5.5), and the free air volume of confinement (181,955 cubic ft as specified in TS 5.5.1), it would take 121 minutes for all of the air containing radioactive material to be released from confinement to the environment, and it would also take 121 minutes for the resulting plume of radioactive material to pass the receptors. Therefore, the NRC staff finds that the licensee's use of a 2 hour stay time is reasonable.

The NRC staff reviewed the information above, and finds that the licensee's atmospheric dispersion calculation methodologies and assumptions, which are used in its calculations of doses to members of the public, are reasonable, justifiable, and conservative. The NRC staff also performed independent analyses to determine whether assuming fumigation or non-fumigation conditions would produce more conservative results, and whether the maximally-exposed member of the public would be located at the site boundary. The NRC staff verified the licensee's determinations that the fumigation conditions are more conservative, and that the maximally-exposed member of the public is located at the site boundary. Therefore, based on the information above, the NRC staff concludes that the licensee's atmospheric dispersion calculation methodologies and assumptions are acceptable.

Dose Calculations

The licensee calculated the potential total effective dose equivalent (TEDE) for an occupational worker in the reactor building. The licensee also calculated the members of the public TEDE at the site boundary (which is also the location of maximum member of the public dose, as discussed above), and at the location of the nearest resident.

As discussed above, the licensee assumed a 5 minute exposure time for the occupational worker in the reactor building. For the occupational dose calculations, the licensee also assumed that the containment ventilation system, including the emergency exhaust system, is off and isolated, such that no radioactive material is vented from the confinement during the 5 minute stay time (a conservative assumption because, as discussed above, the emergency exhaust system would be activated and would reduce the concentrations of radioactive material in the confinement following the initiation of the accident). The licensee also conservatively assumed that no other leakage or decay of radioactive material in the confinement occurs during the 5 minute stay time. The occupational worker is exposed due to submersion in, and inhalation of, airborne radioactive material.

Also as discussed above, members of the public outside the reactor building are assumed to be exposed to radioactive material that leaves the building via the emergency exhaust system and enters the environment through the facility stack. The members of the public are exposed due to submersion in, and inhalation of, airborne radioactive material. No credit is taken for the radionuclide decay inside or outside the confinement.

For the occupational and members of the public dose calculations, the licensee followed the derived air concentration (DAC) and air effluent concentration (AEC) approaches, respectively, that are based 10 CFR Part 20, Appendix B. The NRC staff noted that the DAC approach accounts for the external (submersion) dose from noble gases, and the internal (inhalation) thyroid dose from iodines, but it does not consider the external (submersion) dose or the internal dose to other organs from the iodines. (Because noble gases are non-reactive and do not accumulate in the human body, the inhalation dose from noble gases is negligible). The AEC approach accounts for the external dose from noble gases, and the internal dose to all organs from iodines, but does not consider the external dose from iodines. Additionally, in using the

AEC approach for the members of the public doses, the NRC staff noted that the licensee appears to make some assumptions that are inconsistent with the accepted method of applying this approach. Specifically, for the members of the public dose calculations, the licensee assumes that the AECs for iodines listed in 10 CFR Part 20, Appendix B, Table 2, Column 1, correspond to an internal dose of 100 mrem to the whole body, while the accepted interpretation is that these AECs correspond to an internal dose of 50 mrem to the whole body. The assumption used by the licensee resulted in an overestimation of the members of the public internal dose from iodines by a factor of 2. Additionally, for the members of the public dose calculations, the NRC staff noted that the licensee used an AEC for krypton-88 that appeared to have a misplaced decimal point compared to the AEC from 10 CFR Part 20, Appendix B, Table 2, Column 1 (the licensee used an AEC of 9×10^{-8}, compared to the 10 CFR Part 20, Appendix B, AEC of 9×10^{-9}). The licensee's use of this AEC for krypton-88 resulted in an underestimation of the members of the public external dose from krypton-88 by a factor of 10. For its confirmatory analysis of occupational and members of the public MHA doses, which is discussed below, the NRC staff used an alternate approach that uses the dose conversion factors from Federal Guidance Report (FGR) No. 11 (Ref. 40) for internal (inhalation) dose calculations, and FGR No. 12 (Ref. 41) for external (submersion) dose calculations, to verify that the licensee's total calculated doses remain within 10 CFR Part 20 limits. As discussed below, the NRC staff calculations confirmed that the licensee doses remained within the 10 CFR\ Part 20 limits.

The NRC staff reviewed the information above regarding methodology and assumptions for the licensee's occupational and members of the public dose calculations. The NRC staff concludes that the methodology and assumptions discussed above are generally reasonable and consistent with established practice, and are therefore acceptable, except as noted above.

Occupational Dose Estimates

The licensee calculated occupational MHA doses as discussed above. The licensee's calculated occupational doses include the committed dose equivalent (CDE) to the thyroid (organ dose to the thyroid from radioactive material inhalation), the committed effective dose equivalent (CEDE) (internal whole-body dose from inhalation of radioactive material), the deep dose equivalent (DDE) (external whole-body dose from submersion in radioactive material), and the TEDE (total whole-body dose from internal and external sources). The licensee's calculated occupational doses are shown in Table 13-2.

The NRC staff performed a confirmatory calculation of the MHA occupational doses. The NRC staff used the licensee's MHA source term shown in Table 13-1, and also considered krypton-89, krypton-90, xenon-137, xenon-138, and xenon-139 in its calculation, as discussed above (the NRC staff obtained the U-235 cumulative fission yields for these noble gases from Ref. 47). The NRC staff considered the decay of the 4 noble gases with half-lives less than 4 minutes in its MHA occupational dose calculations, but did not consider the decay of xenon-138 or any other noble gases or iodines. Also as discussed above, the NRC staff used dose conversion factors from FGR No. 11 and FGR No. 12 in place of the licensee's DAC approach. Other aspects of the methodology and assumptions used by the NRC staff for its confirmatory analysis were similar to those used by the licensee for its analysis. The results of the NRC staff confirmatory occupational dose calculations are shown in Table 13-2 alongside the licensee's results. There is some variation in the licensee- and NRC-calculated doses due to the differences in the methodologies and assumptions used. The difference in the calculated values for the DDE is particularly significant, due to the NRC staff calculation's consideration of the 5 additional short-lived noble gas isotopes. However, as Table 13-2 shows, all calculated

doses are below the occupational dose limits in 10 CFR 20.1201, “Occupational Dose Limits for Adults.”

Table 13-2 MHA 5-minute Occupational Dose Estimates in the Restricted Area

Dose Parameters	RINSC-Calculated Dose (mrem)	NRC Confirmatory Calculation (mrem)	10 CFR 20.1201 Dose Limit (mrem)
CDE to the thyroid	49,800	48,900	50,000
CEDE	1,490	1,550	5,000
DDE	74	2,540	5,000
TEDE	1,570	4,100	5,000

Public Dose Estimates

The licensee calculated members of the public MHA doses as discussed above. The licensee’s calculated members of the public doses include the CEDE (internal whole-body dose from inhalation of radioactive material), the DDE (external whole-body dose from submersion in radioactive material), and the TEDE (total whole-body dose from internal and external sources). The licensee’s calculated members of the public doses are shown in Table 13-3.

The NRC staff performed a confirmatory calculation of the MHA members of the public doses. The NRC staff used the licensee’s MHA source term shown in Table 13-1, and also considered Xe-138 in its calculation, as discussed above. Also as discussed above, the NRC staff used dose conversion factors from FGR No. 11 and FGR No. 12 in place of the licensee’s DAC approach. Other aspects of the methodology and assumptions used by the NRC staff for its confirmatory analysis of MHA members of the public doses were similar to those used by the licensee for its analysis. The results of the NRC staff confirmatory members of the public dose calculations are shown in Table 13-3 alongside the licensee’s results. There is some variation in the licensee- and NRC-calculated doses due to the differences in the methodologies and assumptions used. The difference in the calculated values for the DDE is particularly significant, due to the licensee’s use of a DAC for krypton-88 that was inconsistent with the value in 10 CFR 20, Appendix B, and the NRC staff’s consideration of the dose from xenon-138. However, as Table 13-3 shows, all calculated doses are equal to or below the 100 mrem members of the public dose limit in 10 CFR 20.1301, “Dose Limits for Individual Members of the Public.”

Table 13-3 MHA Members of the Public Dose Estimates at the Site Boundary and Nearest Residence

	RINSC-Calculated Doses (mrem)		NRC Confirmatory Calculations (mrem)		
Dose Parameters	**Site Boundary (Location of Maximally-Exposed Member of the Public)**	**Nearest Residence**	**Site Boundary (Location of Maximally-Exposed Member of the Public)**	**Nearest Residence**	**10 CFR 20.1301 Dose Limit (mrem)**
CEDE	97	9.7	65	6.5	100
DDE	3.1	0.3	23	2.3	100
TEDE	100	10	88	8.8	100

The NRC staff noted that although the members of the public dose calculations above were performed for the nearest residence and the RINSC site boundary (the licensee and NRC staff determined that the nearest site boundary would be a higher-dose location that any other location outside the site boundary), there are still other members of the publicly-accessible locations nearer to the reactor building, within the site boundary. Given the parameters of the RG 1.145 fumigation model used for the licensee and NRC staff members of the public dose calculations, the total members of the public doses calculated for locations closer to the stack than the site boundary would be greater than the site boundary doses, assuming that an individual remained at those locations for the entire time it would take a plume of radioactive material to pass. However, the NRC staff notes that publicly-accessible areas within the RINSC site boundary are under the control of the licensee. Although the MHA is not considered to be a credible accident, the NRC staff expects that if such an accident were to occur, the licensee would control access to areas near the reactor building as needed, helping to ensure that member of the public doses from the MHA would remain below the 100 mrem member of the members of the public dose limit in 10 CFR 20.1301. The NRC staff also notes that, as discussed above, the licensee and NRC calculations use extremely conservative assumptions, and also notes that the RG 1.145 fumigation model is typically not intended to be applied for locations very near a release point, and the model could significantly overestimate doses near the release point.

The NRC staff also performed a calculation of the shine DDE dose rate (the external radiation exposure to due to radioactive material suspended in the air of the confinement) for members of the public located at the site boundary and the nearest residence, and at an additional location 10 m (32.8 ft) from the reactor confinement building wall. The NRC staff's shine dose rate calculation used the MicroShield 10.0 computer code, modelling the confinement building as a spherical volume, and considering the radioactive material released to the reactor bay to be uniformly distributed throughout the volume. The NRC staff calculation considered all iodines and noble gases in the licensee's MHA inventory, plus xenon-138. The calculations take no credit for radioactive decay or any reduction in concentration due to leakage or material being exhausted through the ventilation system, but the calculations do take credit for the shielding provided by the 9 in (22.86 centimeter) concrete wall of the building. The NRC staff's calculated dose rates are shown in Table 13-4. The dose rate at the location of the nearest residence is small. Assuming that a member of the public were located at the site boundary for the entire

2-hour period before all airborne radioactive material is exhausted from the confinement, the member of the public could receive a shine DDE of approximately 0.6 mrem from airborne material in confinement (assuming no radioactive decay, and assuming no material is exhausted from confinement until the end of the 2 hour period). The NRC staff noted that the sum of this dose and the licensee's 100 mrem member of the public TEDE calculated and shown in Table 13-3 could slightly exceed the 100 mrem member of the public dose limit in 10 CFR 20.1301. However, the NRC staff also notes that, as discussed above, the licensee used an AEC approach that is inconsistent with the accepted interpretation of the AECs listed in 10 CFR Part 20, Appendix B, which resulted in the licensee's overestimation of the public doses from iodines. The NRC staff calculation used a different, accepted approach to calculate a TEDE of 88 mrem, which remains below the 100 mrem public dose limit in 10 CFR 20.1301 when added to the NRC staff's calculated 0.6 mrem shine DDE. (Both the licensee and NRC staff calculations used other extremely conservative assumptions, as discussed above, including the assumption that 43 percent of the iodine released from the experiment would be organic. The use of a more realistic organic fraction would have significantly increased the fraction of the iodine scrubbed by the emergency exhaust filters, and significantly reduced the public dose estimates for the MHA.)

Table 13-4 MHA Radiation Shine through the Reactor Confinement Building

Parameters	10 Meters from Confinement	Site Boundary	Nearest Residence
Dose rate (mrem per hour)	3.5	0.3	0.0004

The NRC staff reviewed the licensee's MHA dose calculations, and finds that the methodologies and assumptions used by the licensee are reasonable, conservative, and consistent with established industry practices, except as noted above. As discussed above, the NRC staff also performed independent confirmatory calculations of the occupational and member of the public doses from the MHA. The NRC staff finds, based on its review of the licensee's dose calculations, and the results of the NRC staff's confirmatory calculations, that the MHA results demonstrate that the maximum MHA doses are below the occupational dose limits in 10 CFR 20.1201 and the members of the public dose limit in 10 CFR 20.1301. Therefore, based on the information above, the NRC staff concludes that the results of the RINSC MHA are acceptable.

13.2 Insertion of Excess Reactivity

SAR Section 13.2.2, as supplemented by the licensee's response to RAIs 13.7, 13.9, 13.23, and 14.85 (Ref. 3), describes the reactivity insertion accident, which was analyzed for both forced convective cooling and natural convective cooling operation modes using PARET/ANL version 7.5.

The response to RAI 13.7 (Ref. 3) describes the evaluation of a step reactivity insertion accident. The neutronics codes used to generate input for the PARET models were: WIMS/ANL for multi-group neutron cross sections; REBUS-PC (which includes DIF3D as the neutronics solver) for power density information, and VARI3D (which also includes DIF3D as the neutronics solver for real and adjoint flux), to provide the reactor kinetics delayed neutron fractions, decay constants, and prompt neutron lifetime. Data on reactor power distribution is

provided in SAR Section 4.5 (Ref. 2), and data on the reactor kinetics parameters and reactivity feedback coefficients is provided in the licensee's response to RAI 4.10 (Ref. 3).

The FC transients are assumed to take place under the following assumptions (TSs, Revised Section 2.1.1 in RAI 14.36, "Safety Limits in the Forced Convection Mode"), provided in Table 13-5 below:

Table 13-5 Forced Convection Transient Analysis Assumptions

Measured Parameter	Analysis Values Used	TS 2.2.2 (LSSS forced convection)
Reactor Power - P (MWt)	2.3	2.3
Coolant Flow - m gallons per minute(gpm)	1740	1560
Water Height – H (ft, inches)	23 ft 9.1 inches	23 ft 7 inches
Coolant Temperature –T_0 (°F)	123	122

The NC transients are assumed to take place under the following assumptions (TSs, Revised Section 2.1.2 in RAI 14.52, "Safety Limits in the Natural Convection Mode"), provided in Table 13-6 below:

Table 13-6 Natural Convection Transient Analysis Assumptions

Measured Parameter	Analysis Values Used	TS 2.2.1 (LSSS natural convection)
Reactor Power - P (kW)	125	115
Water Height – H (ft, inches)	23 ft 9.1 inches	23 ft 7 inches
Coolant Temperature –T_0 (°F)	128	127

The period trip at 4 seconds is assumed to fail. The power trip is functional. The time delay for control blades to begin to move after a trip is assumed to be 100 milliseconds. The time to full insertion is the maximum allowed of 1.0 second (TS 3.2.2).

<u>Case 1: Rapid Insertion of 0.6% Δk/k Reactivity from Very Low Power</u>

The bounding step reactivity insertion accident for the RINSC reactor assumes an insertion of reactivity of 0.6 percent Δk/k from an initial reactor power level of 10 Wt, coolant flow rate of 1,740 gpm, pool coolant height of 23.758 ft (7.24 m), and a reactor core coolant outlet temperature of 123 °F (50.6 °C). According to the licensee's PARET/ANL analysis provided in its response to RAI 13.7 (Ref. 3), the measured peak reactor power would increase from 10 Wt to 2.423 MWt after the reactor power trip at 2.3 MWt scrams the reactor after 10.179 seconds (period trip is assumed to fail). The peak fuel centerline temperature is calculated to be 79.8 °C (175.6 °F) with a fuel clad temperature of 79.1 °C (174.3 °F). This temperature is 450 °C (842 °F) below the SL.

The NRC staff finds that TS 2.2.2 helps to ensure that the analyzed fuel temperatures are maintained bounding and not achieved due to the LSSS for the high-power scram which will initiate at 2.3 MWt. The NRC staff notes that the licensee performed this analysis at 1,740 gpm, whereas the LSSS flow limit is 1,560 gpm. Although this analysis is not truly at limiting conditions, it is the NRC staff's conclusion that the flow rate difference will not impact on the resulting fuel temperature analysis results which indicates significant temperature margin to the SL.

Case 2: Slow Insertion of 0.02 % Δk/k /Second Reactivity from Very Low Power

The response to RAI 13.7 (Ref. 3) also describes the evaluation of a slow ramp reactivity insertion accident under forced cooling mode. The reactivity insertion accident for 0.02 percent Δk/k/sec, which corresponds to the reactivity limit TS 3.2.3 for a single blade. In this case, the withdrawal accident is initiated from a subcritical condition (reactor power of 10 Wt) and a coolant flow rate of 1740 gpm, and with the accident, is terminated by reactor power scram at 2.3 MWt, and due to time associated with the negative reactivity insertion of the control blade of 100 milliseconds, the reactor power reaches a peak of 2.509 MWt at 32.298 seconds. The results of the licensee's analysis indicate that the peak fuel centerline temperature is calculated to be 79.1 °C (174 °F) with a fuel clad temperature of 78.9 °C (174 °F). This temperature is 450 °C (842 °F) below the SL.

The NRC staff finds that TS 2.2.2 helps to ensure that the analyzed fuel temperatures are maintained bounding and not achieved due to the LSSS for the high-power scram which will initiate at 2.3 MWt. Based on the information above, the NRC staff concludes that the resulting fuel temperature analysis results indicates significant temperature margin to the SL.

Case 3: Slow Insertion of 0.02 % Δk/k / Second Reactivity from 1.8 MWt Power

and

Case 4: Slow Insertion of 0.02 % Δk/k /Second Reactivity from 2.2 MWt Power

The response to RAI 13.7 (Ref. 3) also describes the evaluation of slow ramp reactivity insertion accidents under forced cooling mode. The analyses document ramp reactivity insertions with an insertion rate of 0.02 percent Δk/k/sec at 1.8 MWt and 2.2 MWt. These analyses show that a high-power scram at 2.3 MWt is initiated at approximately 6.774 and 2.498 seconds, respectively. Maximum fuel centerline temperatures are calculated to be 76.7 and 75.9 °C (170 and 168.6 °F). The maximum corresponding fuel clad temperatures are 75.9 and 75.1 °C (168.6 and 167.2 °F). This temperature is 450 °C (842 °F) below the SL.

The NRC staff finds that TS 2.2.2 helps to ensure that the analyzed fuel temperatures are maintained bounding and not achieved due to the LSSS for the high-power scram which will initiate at 2.3 MWt. Based on the information above, the NRC staff concludes that the resulting fuel temperature analysis results indicates significant temperature margin to the SL.

Case 5: Rapid Insertion of 0.6% Δk/k Reactivity from 100 kWt under Natural Convection Cooling

The response to RAI 13.7 (Ref. 3) also describes a reactivity insertion accident for the RINSC reactor during NC mode operation assuming an insertion of reactivity of 0.6 percent Δk/k from an initial power of 100 kWt, pool coolant height of 23.758 ft (7.24 m), and a reactor core outlet

temperature of 128 °F (53.3 °C). According to the licensee's PARET/ANL analysis, the measured peak reactor power would increase from 100 kWt to 404 kWt after reactor power trip at 125 kWt scrams the reactor after 0.036 seconds (period trip is assumed to fail). The peak fuel centerline temperature is calculated to be 65.7 °C (150.3 °F) with a fuel clad temperature of 65.7 °C (150.3 °F). This temperature is 450 °C (842 °F) below the SL.

The NRC staff finds that TS 2.2.1 helps to ensure that the analyzed fuel temperatures are maintained bounding and not achieved due to the LSSS for the high-power scram which will initiate at 115 kWt. Based on the information above, the NRC staff concludes that the resulting fuel temperature analysis results indicates significant temperature margin to the SL which is 530 °C (986 °F).

13.2.1 Step (Rapid) Reactivity Insertion Accident

The NRC staff has reviewed the results of the step reactivity insertion analyses presented by the licensee for this accident and finds that Case 1 is the limiting case for this analysis, which is assumed to be caused by an experiment that is moved within the core while at LP. The NRC staff finds that the peak fuel temperature remains below the SL of 530 °C, and concludes that no failure of the fuel element cladding or fission product release would be expected under any mode of operation.

13.2.2 Ramp (Slow) Reactivity Insertion Accident

In response to RAI 13.7 (Ref. 3), the licensee provided 5 cases of ramp reactivity addition analyses (see SER Section 13.2.1 above). The ramp or slow reactivity insertion analysis assumes the failure of the control system that results in a control blade moving in an uncontrolled manner. The ramp reactivity insertion is initiated from LP and is terminated by the reactor safety system (the minimum reactor period scram is assumed to fail).

The NRC staff reviewed the ramp reactivity analyses and finds that Case 2 is the most limiting scenario as it results in the maximum fuel centerline and fuel clad temperatures, which are 79.1 °C and 65.7 °C, (174.3 and 150.3 °F) respectively. The NRC staff finds that these temperatures are well below the fuel blister temperature (SL) of 530 °C (986 °F). The NRC staff finds that the licensee's analysis of the slow reactivity insertion accidents used TS 3.2.1.2 value of 0.02 percent $\Delta k/k$ reactivity insertion per second starting at a low reactor power (Case 2) and near licensed power of 2 MWt during FC mode operation (Case 3 and 4). The NRC staff finds that the bounding slow reactivity insertion accident is initiated at LP and is terminated by the reactor safety system (minimum reactor period scram is assumed to fail) after a reactor power of 2.5 MWt is reached at 32.198 seconds.

<u>Conclusions</u>

The NRC staff reviewed the results of the ramp reactivity insertion analyses presented by the licensee and finds that in all five scenarios, the peak fuel temperature remains below the SL of 530 °C (986 °F). Based on the information provided above, the NRC staff concludes that no failure of the fuel element cladding or fission product release would be expected from a reactivity insertion accident, under any mode of operation.

13.3 Loss of Coolant Accident

The LOCA analysis is presented in SAR Section 13.2.3, and supplemented by the responses to RAIs 10.1, 10.2, and 13.10 through 13.17 (Ref. 3). In its response to RAI 13.14, the licensee stated that the entire LOCA analysis supplied in the SAR is replaced by the material provided in the RAI responses listed above, and in this SER is referred to as the LOCA Analysis. The LOCA Analysis, performed by Argonne National Laboratory, developed a correlation of fuel plate temperatures versus time following the complete uncovering of the core due to the loss of the cooling water (Decay Heat Power Model), and then the licensee established an acceptable pool drain time so that the fuel plate temperatures would remain above the blister temperature to preserve the fuel cladding integrity (Pool Drain Time Model).

Decay Heat Power Model

The LOCA Analysis states that the decay heat model provides a steady state solution of the heat transfer equations for a partially submerged core using the highest powered fuel plate in the core. Heat generated in the assembly above the waterline is conducted down along the length of fuel element into the pool of water remaining. This heat, along with the heat generated in the submerged portion of the assembly, causes the water to boil and produce steam that rises up through the exposed surfaces of the fuel plates. A coupled pair of ordinary differential equations are derived – one to represent the axial distribution of the fuel plate temperature and one to represent the axial distribution of the steam temperature.

The LOCA Analysis demonstrates that as long as a portion of the fuel remains submerged in water at a level that is no lower than the elevation of the bottom of the 8-in beam ports (the lowest pool penetration), and the decay heat power is no greater than 0.827 percent of the analyzed reactor power (2.3 MW), then there is sufficient cooling capacity to prevent the fuel cladding temperature from reaching the SL temperature, where fuel cladding integrity is challenged. As stated in the LOCA Analysis, an acceptable fuel and cladding temperature limit not to be exceeded under any conditions of operation is 530 °C (986 °F); this is the RINSC SL in TS 2.1. The time to reach 0.827 percent of full power decay heat, provided by the values in Table 1 of the LOCA Analysis is between 15,000 (4.16 hours) and 20,000 seconds (5.5 hours).

The licensee utilized the data in the response to RAI 10.2 (Ref. 3), which states that the time at which the power decays to 0.827 percent is 16,232 seconds (4.5 hours). The licensee then evaluated all credible pool drain scenarios to ensure than any potential drain event would not result in the core being uncovered sooner than 4.5 hours after reactor shut down. The NRC staff had previously evaluated this approach in the safety evaluation report for the HEU to LEU conversion and found it acceptable.

Pool Drain Time Model

The licensee described the drain time model in the response to RAI 13.11 (Ref. 3). The licensee explained that the given the core geometry, the relative location of the beam tubes to the core center, the cross-sectional area of the pool, and the size of the break that the drain time derived from Bernoulli's Equation can be expressed as:

$$t = \frac{2A_1}{C_d A}\sqrt{\frac{h_i}{2g}}\left[1 - \frac{\sqrt{h_f}}{\sqrt{h_i}}\right] \qquad \text{Equation 13-1}$$

where the stated parameters have the values corresponding to the licensee's facility, as stated in Table 13-7 below.

Table 13-7 Pool Drain Time Parameters

Variable	Representation	Value
A	Combined area of the break	multiple values evaluated ft^2
A_1	Surface area of the pool	150 ft^2
C_d	Break loss coefficient	0.61
g	gravitational acceleration	32.2 ft/s
h_i	initial pool water height	28.06 ft
h_f	final pool water height	4.8333 ft

In its response to RAI 13.12 (Ref. 3), the licensee provided the basis for the initial and final water heights used in the LOCA analysis. The coolant level at which the scram occurs is 23.54 ft (7.17 m) above the top of the active core, which in the LOCA event is taken to be the top of the fuel. This level is the minimum pool level that is permitted by the LSSS, while operating at any FC power level, in accordance with TS 2.2.1.2 or TS 2.2.2.2. The LOCA analysis assumption indicates that the water level starts at the TS minimum and draining halts when the water falls to the level that is at the top of the fuel box. Since this level is above the 8-in beam port level discussed above, it ensures that the fuel remains completely covered by pool coolant, and the time to drain is less, so it provides a conservative assumption for the LOCA analysis. Based on its review of this methodology the NRC staff finds that the equations are properly derived and the assumptions used for the parameters are appropriate given the descriptions of the facility in the SAR.

LOCA Scenarios Considered

According to SAR Section 10.2.1.1, the RINSC reactor pool has six beam ports located at mid-core elevation, two that are 8-in in diameter and 4 that are 6-in in diameter, and a 6-in through-port which traverses the pool below the bottom of the fuel (see Figure 4-1).

The licensee stated that the LOCA scenario is based on the assumption that a beam port tube is severed. In its response to RAI 13.10, the licensee provided the following justification for the LOCA scenario by indicating that the only open penetrations into the pool are the rabbit tubing, through port, and beam port tubes. Dropping something into the reactor pool, and shearing the through port is not considered to be a credible accident scenario because it runs underneath the thermal column extension. As a result, the beam ports are used for the LOCA analysis, with the assumption that the beam port flange fitting of the largest beam ports is sheared off and it drains as does the drain tube for that beam port.

In the response to RAI 10.3 (Ref. 3), the licensee stated that portions of the thermal column, which is a significant structural member, extend over the through port thus protecting it from damage due to dropped loads. The reactor structure when in the HP end of the pool also protects the through tube from damage from dropped objects. During a site visit the NRC staff viewed the penetration of the through-port at the concrete face of the pool structure, and observed that it has a bolted flange which is nearly flush with the concrete face. The NRC staff

finds that the configuration of the through-port does not provide an opportunity for a dropped load to shear an opening for the release of pool water. The licensee explained during the site visit that use of the through-port includes a shutoff valve to mitigate any potential leakage. The NRC staff concludes that for the through-port to be a credible source for a LOCA, multiple failures would be required which is beyond the licensing basis for RINSC. For this reason, the NRC staff finds that there is no credible event that could drain the pool via the through-port.

In its response to RAI 13.10, the licensee stated that the rabbit tubing enters through the pool wall at an elevation that is close to the top of the pool. The NRC staff reviewed the design of the pool structure for the rabbit tubing, and observed the layout during a site visit. The NRC staff finds that the rabbit tubes do indeed enter the pool near the top of the pool elevation, and the NRC staff concludes that a failure of the rabbit tubing due to a shearing event inside the pool would not lead to a significant loss of pool water.

In response to RAI 13.10 (Ref. 3), the licensee stated that the design basis accident for RINSC is the in-pool shearing of a beam port, and each beam port has a 1 in diameter drain line, and a 1 in diameter vent line. These lines merge into a single 1-in line at an elevation below the bottom of the beam port, extend out to the confinement edge of the reactor pool wall, and go through a manual drain valve. On the outlet side of the drain valve there is a one-half in diameter orifice plate welded onto the drain line. The 1-in drain line connects to a 2 in diameter drain line that is common to all of the beam ports, as well as the through port. This common line also connects to the 1-in drain line for the through port, and the 5 in off gas vent for the thermal column. The common drain line is well below the elevations of the beam ports, through port, and thermal column, and it connects to a 5-in drain line that extends into the basement underneath the reactor. The 5-in line is capped with a pipe tee that is positioned vertically. The upward facing vent ultimately ties into the Off-Gas System. The downward facing drain diameter is reduced to a 2-in line that empties into a 5-gal bucket that is open to air. This is illustrated below in Figure 13-1 below provided by the licensee in response to RAI 10.2 (Refs. 3, 6).

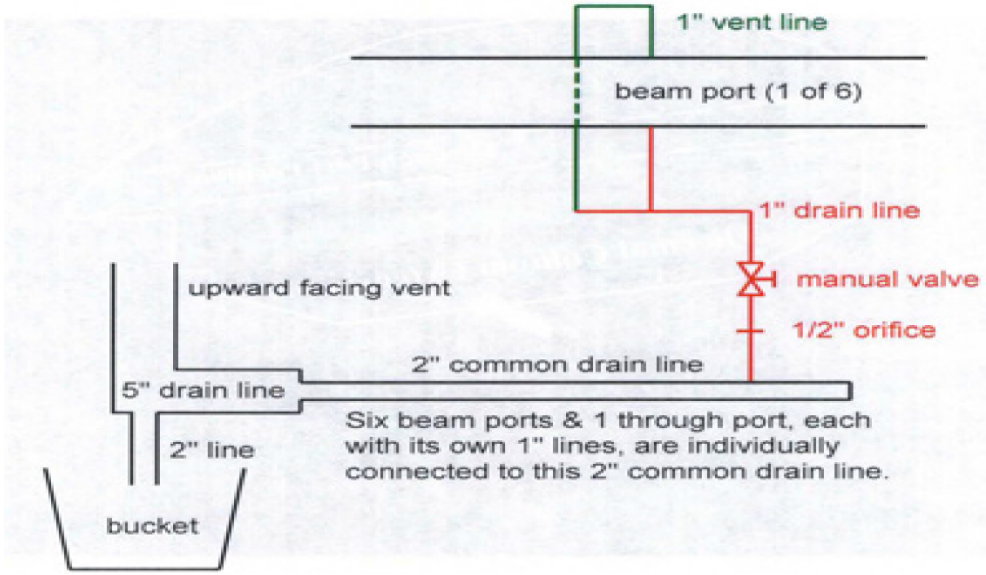

Figure 13-1 Experimental Drain System

The licensee stated in the response to RAI 13.10 (Ref. 3), that if an 8-in diameter beam port is sheared, it will flood. The water will have two possible pathways: (1) through the beam port drain line which is a ½ in diameter hole; or (2) though the opening in the beam port flange fitting which according to TS 3.9.3.1 is allowed to be as large as 1.25 in in diameter. The most constrictive point of the beam port drain line is the one-half inch diameter orifice on the drain line. Since this opens into progressively larger pipe diameters the limiting diameter for the drain line (.5 in) determines the maximum flow rate through the drain line.

In response to RAI 10.2 (Ref. 3), the licensee assumed that a rupture of the beam port will cause water to drain from both of these openings simultaneously. The NRC staff reviewed and observed the beam port configuration and finds that the licensee's LOCA analysis appropriately identifies the limiting scenario for the LOCA event as described above.

Licensee Analysis

In response to RAI 10.2 (Ref. 3), the licensee combined the use of the Decay Heat Power Model with the Pool Drain Time Model and showed that if an allowable drain time of 16,232 seconds (4.5 hours) is assumed, then the total drain area that corresponds is 1.68 in^2. Since the drain line contribution is fixed at ½ in diameter that means that the maximum allowable opening for the beam tube flange fitting is 1.37 in in diameter.

The licensee provides controls in TS 3.9.3.1, Specification 1 that requires that "Each beam port shall have no more than an area of 1.25 in^2 open to confinement during reactor operation." Thus, according to the licensee, since the opening allowed is less than 1.37 in in diameter then the drain time expected if the beam port flange fitting and the drain line both allow pool water to drain will be longer than the required 4.5 hours to prevent fuel blistering.

Confirmatory Analysis

The NRC staff has reviewed the analysis supplied for the fuel temperature from decay heat and finds that it is based on acceptable methods and assumptions and is suitably conservative for use on the RINSC facility. Similarly, the NRC staff has reviewed the drain time model and finds that it is also acceptable and appropriate for use for the RINSC facility.

The NRC staff has utilized the drain time model in conjunction with the 4.5 hours assumed for the drain time and confirmed the calculation of 1.37 in diameter as a maximum total opening. In addition, the NRC staff has performed an independent calculation of pool drain time assuming that the drain line (1/2 in diameter) and the beam port fitting (1.25 in diameter) both fail at the same time. The resulting drain time is 5.3 hours at 81.40 gpm average flow rate. The NRC staff finds that the drain flow areas and drain times calculated by the licensee are acceptable.

Furthermore, the NRC staff notes that several important points are relevant to this drain time. The confirmatory drain time average flow rate calculated is 81.40 gpm. The first point is that the makeup water system operates normally on a float switch which, when it detects a loss of pool level of 1, it automatically opens a valve that allows water from the primary makeup water system (SER Section 5.5.1) to flow into the pool. This is an on-demand system connected to city water through a filtration system and it supplies water at 5 gpm. Secondly, any drop in pool level below the TS minimum results in an alarm to the operator at the reactor console and to security, which then notifies the RINSC staff by telephone. Third, the AWSS (SER Section 5.7) is also capable of providing a source of water to replenish the pool inventory. This system activates manually and is capable of supplying as much as 60 gpm to the inventory. Fourth,

SAR Section 9.3.2 describes the availability of the fire hose which is connected to city water. For these reasons, the NRC staff finds that there are diverse systems capable of indicating, and mitigating a LOCA event and there is sufficient time to employ any of the several available means for adding water to the pool.

LOCA Dose Analysis

Although a LOCA would not result in fuel failure, the NRC staff noted that during the course of a LOCA, if the reactor core becomes completely uncovered, the fission products in the fuel in the core would constitute an unshielded gamma-ray source. Therefore, the NRC staff performed an additional analysis to estimate the external (shine) dose rates at selected locations inside and outside the reactor confinement following a complete LOCA (which would be highly unlikely to occur, as discussed above). The NRC staff conservatively assumed that the reactor had been operating continuously for 1 year preceding the LOCA, and that the LOCA occurs instantaneously (the reactor also shuts down at the same instant the LOCA occurs). The core is considered to be a point source of radiation at the bottom of the reactor pool, and all gamma rays emitted from the core are assumed to have an energy of 1 MeV (a conservative assumption, since most of the decay photons from the core will have energies less than 1 MeV, and therefore will result in lower doses). The calculation considered attenuation of the photons in air, and, for locations outside the confinement, attenuation in the building walls and truck door. The calculation evaluated the doses at 5 locations: (1) the reactor bridge (within confinement); (2) the door to the control room (within confinement); (3) next to the truck door (within confinement); (4) next to the truck door (outside confinement); and (5) at 50 meters (164 ft) from the outside of the truck door, near the RINSC site boundary (outside confinement). Except for the reactor bridge location, most of the dose at these locations results from radiation scattered off the ceiling of the reactor building. The NRC staff conservatively assumed the ceiling to be a thick concrete slab in order to maximize the backscattered radiation. The NRC staff calculated the dose rate near the RINSC site boundary 1 hour following reactor shutdown. For the other locations, the NRC staff calculated the dose rates 10 seconds, 1 hour, 7 hours, 1 day, 1 week, and 1 month following reactor shutdown. The results of the NRC staff's LOCA dose calculation are shown in Table 13-8.

Table 13-8 NRC Staff Calculated LOCA External Dose Rates

	Dose Rates at Locations				
Time Following Reactor Shutdown	**Reactor Bridge (rem/hour)**	**Control Room Door (mrem/hour)**	**Inside Truck Door (mrem/hour)**	**Outside Truck Door (mrem/hour)**	**Site Boundary (mrem/hour)**
10 seconds	137,000	6,030	2,340	2,330	-
1 hour	37,100	1,640	640	630	1.9
7 hours	22,800	1,010	390	390	-
1 day	16,300	720	280	280	-
1 week	8,710	340	150	150	-
1 month	4,790	210	82	82	-

The NRC staff's calculated doses show that the dose rate on the reactor bridge would be very high if the core were completely uncovered and therefore, the reactor bridge would be inaccessible (given that an instantaneous LOCA is not a credible accident, if personnel were on the reactor bridge during the initiation of a LOCA, there would be sufficient time to evacuate the personnel from the bridge). The dose rates in other areas within the confinement, such as the control room and near the truck door, where most exposure is from scattered radiation, would also be significant. However, given that in any credible LOCA scenario the pool would drain over the course of several hours or longer (when the pool is partially drained, dose rates would still be elevated, but would be well below the values in Table 13-8), and that it would only take 5 minutes or less to evacuate confinement (see SER Section 13.1), any dose to occupational workers prior to their evacuation would be low, and well below the 5,000 mrem occupational dose limit in 10 CFR 20.1201. Additionally, even if a complete LOCA were to occur, the dose rates would be low enough that personnel could occupy areas within confinement for brief periods of time, if necessary, to perform actions needed for recovery operations. Personnel would be able to operate the AWSS values discussed above to increase the makeup flow of water to the pool, because these values are in a location that would have dose rates that are below those in the control room (due to additional shielding). Based on the information above, the NRC staff finds that occupational doses due to shine from the reactor core during a LOCA accident could be maintained below the 5,000 mrem occupational dose limit in 10 CFR 20.1201, and would allow recovery operations to be performed.

The NRC staff's calculated dose rate near the RINSC site boundary 1 hour after the instantaneous LOCA and reactor shutdown is 1.9 mrem per hour. However, given the time it would take for the core to become uncovered due to coolant drainage (at least several hours), any LOCA dose rate at the site boundary would be less than 1.9 mrem per hour because the inventory of the core would have had additional time to decay. The NRC staff notes that even if the core becomes uncovered in 1 hour, and there were no additional radioactive decay of the core following the first hour after shutdown, a member of the public could remain at the RINSC site boundary over for over 50 continuous hours before the 100 mrem dose limit in 10 CFR20.1301 would be exceeded. The area within the RINSC site boundary is under the control of the licensee, and the NRC staff expects that the licensee would control access to any area within the site boundary, as needed, to help ensure that member of the public doses remain below the 100 mrem public dose limit in 10 CFR 20.1201 during any LOCA or other accident conditions. Additionally, the RINSC site is located on the University of Rhode Island Narragansett Bay Campus, and therefore the area outside the RINSC site boundary, extending a significant distance from the site boundary, is under State of RI control. Therefore, the licensee could also control access to this area, as needed, should elevated dose rates exist. Based on the information above, the NRC staff finds that member of the public doses due to shine from the reactor core during a LOCA accident could be maintained below the 100 mrem members of the public dose limit in 10 CFR 20.1301.

Conclusion

The NRC staff reviewed the licensee's methodology, and checked the licensee's calculations for numerical accuracy and the validity of the assumptions. Since the drain time for the TS controlled conditions is 5.3 hours, the NRC staff finds that the results of a LOCA, as described in the scenario would not result in fuel being uncovered or loss of pool water as a heat removal source within the 4.51-hour time interval where damage to the fuel could result. Therefore, the NRC staff finds that the LOCA would not result in fuel failure or loss of fuel cladding integrity as the fuel temperature would not exceed the SL of 530 °C (986 °F). The NRC staff also performed calculations of the occupational and member of the public dose rates that could exist

due to the loss of core shielding during a LOCA, and finds that the calculations demonstrate that a LOCA at the RINSC would not result in doses that are in excess of applicable 10 CFR Part 20 limits. Based on its review, the NRC staff concludes that a loss of coolant, as analyzed, would not pose an undue risk to the health and safety of the members of the public.

13.4 Loss of Flow Accident

Forced convection is required for reactor power levels above 100 kWt. In this mode, cooling water flow through the reactor is from the top of the reactor to the bottom. In the event of a reactor scram and loss of forced flow, the water direction in the core will reverse and start to flow upward as NC of the water begins cooling the fuel. The licensee analyzed the loss-of-flow accident (LOFA) caused by a loss of power to the primary coolant pumps, and for other mechanical failures that interrupt cooling flow.

13.4.1 Loss of Electrical Power to the Primary Pumps

The NRC staff noted that the operating parameters cited in SAR Section 13.2.4.1 LOFA analysis did not match the limiting conditions in the TSs. In the response to RAIs 13.18 and 13.19 (Ref. 3), the licensee described the results of new LOFA analyses using RELAP5. In these analyses, the licensee used the trip setpoints values that are consistent with the TS LSSS for FC (TS 2.2.2). In these analyses, the licensee listed the initial conditions (limiting trip values), pump flow coast down, limiting conditions on the reactivity insertion and its timing, and the timing when the NC gate valve opens. For the case when the gate valves open (9 second after trip), the licensee's calculations show a maximum fuel clad temperature of 115.62 °C (240.12 °F), with a peak fuel temperature of 115.73 °C (240.31 °F) and a coolant saturation temperature of 115.90 °C (240.62 °F). These results indicate the coolant peak temperature to be close to the onset of sub-cooled boiling. The licensee added that at a higher initial power level, if sub-cooled boiling occurs, coolant heat transfer would increase and would limit the fuel and clad temperature rise.

For the case when the two gate valves fail to open, the licensee stated that the peak clad temperature is slightly lower (115.61 vs. 115.62 °C), and occurs at the same timeframe (at 9.41 seconds). The reason for the lower peak temperature when the gate valves do not open can be attributed to the changes in the amount of flow that is drawn out of the outlet duct. As shown in Figure 13-2, the flow rate near 9 seconds after the event initiation indicates that the outlet duct upward flow rate immediately after the gate valves open is slightly higher than what would be obtained by extrapolating the flow rate from before the gate valves open. The slightly increased coolant being drawn out of the outlet plenum causes slightly less up-flow through the fuel channels and slightly higher temperatures in the fuel channels than if the gate does open. However, it is desirable to have the gates open for long-term NC cooling.

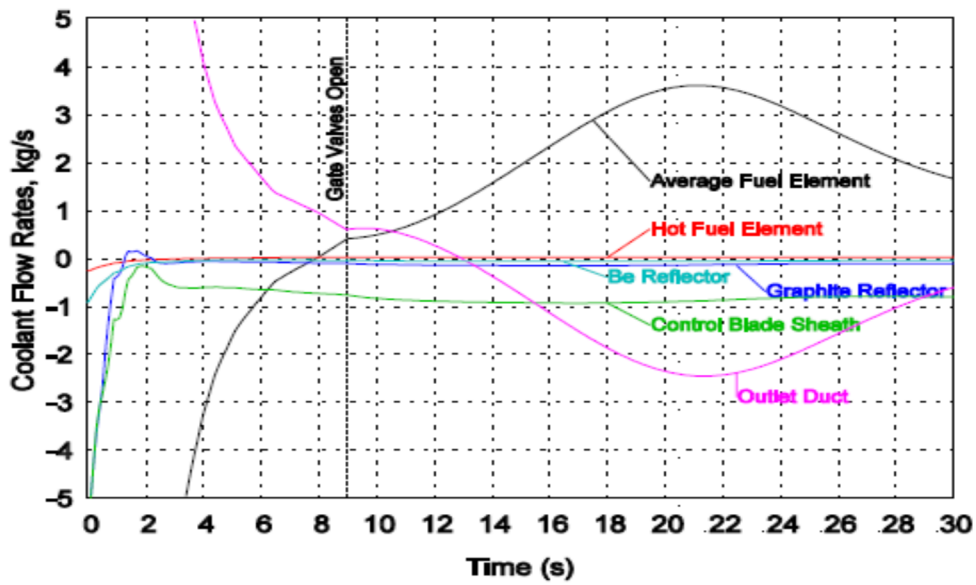

Figure 13-2 Flow rates for the LOFA

The NRC staff has reviewed the new LOFA analysis and finds that the assumptions used in the analysis are acceptable. The licensee's steady-state FC T-H analysis (Ref. 3) establishes initial conditions that are slightly different than those used in the LOFA analysis. Table 13-9 summarizes the initial conditions used in the LOFA and those used in the steady-state T-H analysis.

Table 13-9 Initial Steady-State Conditions

Parameters	LOFA Initial Steady-State Condition	Steady-State Thermal-Hydraulic Analysis
Reactor Power (MWt)	2.3	2.4
Total Pump Flow (gpm)	1740	1580
Height of Water above the top of the core (ft)	23.76	23.54.
Primary coolant outlet temperature (°F)	123	125

The NRC staff notes that if the licensee were to analyze the LOFA with the initial conditions from the steady-state thermal-hydraulic analysis, the fuel cladding temperature would be expected to be slightly higher (estimated at 10 to 15 °C) than the value cited above because of the increase in power and reduction of flow. However, the NRC staff finds that there is significant margin to the SL in the provided analysis, and that these changes to the initial conditions would result in cladding and fuel temperatures that are well below the SL of 530 °C (986 °F). Based on the information described above, the NRC staff finds that the LOFA results are acceptable. The NRC staff concludes that the LOFA analysis indicates that no fuel element damage would occur due to loss of electrical power to the primary pumps.

13.4.2 Failure of a Pump or Other Component in the Primary Coolant System

SAR Section 13.2.4.2 describes the failure of a pump or other component in the PCS. The event starts with natural circulation flow being directed upward through the core and through the open top plenum to the pool. The coolant gate valves are held closed by forced circulation during that operational mode and opened by gravity when NC occurs. An automatic scram occurs if one of the gate valves is opened during FC mode of operation (TS 3.2.1, Table 3.1). Annual inspection of the gate valves in accordance with TS 4.2.5.5, helps ensure operation (open) on a loss of coolant flow. Failure of an isolation or check valve in the primary cooling loops could result in a reduction of the coolant flow without causing a scram (each loop has a flow capacity of 1,950 gpm). However, the coolant flow will be reduced to the alarm setpoint of nominal flow, and will alert the operator to the low-flow condition and give an operator the opportunity to take corrective action to resolve the abnormal condition and shutdown the reactor, if needed.

As discussed above, the total loss of flow is evaluated using RELAP5, which determines that the maximum fuel clad temperature due to failure of the coolant gates to open is 115.61 °C (240.1 °F), which is below the blister temperature (SL) of 530 °C (986 °F).

The NRC staff reviewed the accident sequences stated for the LOFA accidents, and finds that the scenarios results in a scram signal, which will shut down the reactor, as required by TS 3.2.4, and limits the fuel clad temperature to below the SL of 530 °C (986 °F). Based on the information above, the NRC staff concludes that the licensee's LOFA scenario was properly analyzed and the results are acceptable.

13.5 Mishandling or Malfunction of Fuel

The licensee provided its analysis of a fuel failure in a SAR supplement (Ref. 56) and superseded the information provided in SAR Section 13.2.1, as supplemented by RAI responses 13.2 through 13.5 (Ref. 3). However, some information from other sections of the SAR is included to supplement the discussion below.

The licensee initially determined that its fuel failure scenario was the MHA for the RINSC. However, the licensee subsequently determined its fissionable experiment failure scenario (see SER Section 13.1) to be the MHA (Ref. 58).

Accident Scenario

The fuel failure scenario assumed that the fuel plate with the highest power is damaged under water. The failure could be mechanistic (i.e., cladding failure due to fuel mishandling) or non-mechanistic (i.e., cladding failure (fuel malfunction) due to overheating). However, a mechanistic scenario is more realistic at the RINSC, because as the LOCA analysis in SER Section 13.3 demonstrates, the peak cladding temperature remains below the blister temperature of the cladding for any credible accident. The fuel plate is assumed to lose the entire surface of its cladding on one side, and fission fragments in the fuel plate are released into the reactor pool. The amount of total activity in that plate that would be available to be released would depend on the temperature of the fuel, and the surface area of the fuel that is exposed. The licensee stated that the fuel temperature, even during full power operation, is low, and therefore diffusion from the fuel matrix would be essentially zero. Consequently, the only fission products that would be released would be those that are within the range of fission

fragment recoil from the denuded surface of the fuel, due to the kinetic energy associated with fission fragment recoil.

The scenario assumes that prior to the fuel failure, the reactor has been operated for sufficient time for the core to reach fission product inventory saturation levels. The fuel failure in the pool is assumed to result in an instantaneous release of fission products from the fuel, and the released fission products are assumed to become uniformly mixed throughout the pool. Any particulate fission products released from the fuel are assumed to remain in the coolant, and therefore they are not considered in the licensee's analysis because they would not significantly contribute to the doses. The gaseous fission products (iodines and noble gases) that are released from the fuel plate were considered in the licensee's analysis, because the licensee assumed that the gaseous fission products can be released from the pool to the reactor confinement. The licensee assumed that the elevated radiation readings generated by the fuel failure would be detected by installed radiation monitors (required by TS 3.7.1.1, which is discussed and found acceptable in SER Section 7.7), and, in response to the alarm, the RO would activate the facility evacuation system, which would, in turn, activate the confinement isolation and emergency exhaust system (TS 4.5.2, which is discussed and found acceptable in SER Section 6.2.1, requires that it be verified that the CVS emergency mode activate when the facility evacuation alarm activates). Activation of this system re-aligns the ventilation system to route confinement air through HEPA and charcoal filters prior to its release to the environment through the facility stack. As required by TS 3.5.1, 4.5.2, and 4.5.3 (which are discussed and found acceptable in SER Section 6.2.1), the facility ventilation system, whether operating in normal or emergency mode, shall maintain the confinement building pressure at least 0.5 in of water below atmospheric pressure whenever the reactor is operating or other activities are in progress that could result in a radioactive material release. This helps ensure that any air leakage is into, not out of, confinement, and any radioactive material released to confinement will be released through the stack such that it can be monitored, filtered and diluted before release, and adequately dispersed.

Actuation of the evacuation system and the confinement isolation and emergency exhaust system will prompt operations personnel to ensure that a total evacuation of the reactor building is accomplished promptly. A conservative 5-minute evacuation time is assumed for the dose calculations for personnel in the building.

The NRC staff reviewed the fuel failure scenario described above. The NRC staff finds that the licensee's assumption that fission products in the range of fission product recoil could be released to the pool, but that fission products in the remainder of fuel matrix would not be released, is reasonable because the insertion of reactivity analysis in SER Section 13.2, and the LOCA analysis in SER Section 13.3, demonstrate that no credible reactivity transient or LOCA accident at the RINSC facility would cause the RINSC fuel safety limit temperature to be exceeded. Therefore, these accidents would not result in cladding damage, or melting of the fuel which could cause fission product release from the remainder of the fuel matrix. The only accident scenario that could cause a significant release from the fuel would be an accident such as a fuel handling accident that mechanistically damaged the cladding and caused a release of fission products from the range of fission product recoil only. The NRC staff also finds that the licensee's assumption that the failed fuel plate loses the entire surface of its cladding on one side is conservative because it is consistent with severe damage to the fuel, and it bounds any credible fuel damage event that could occur. The most likely damage from a fuel handling accident would be localized and very small (e.g., a chip or scratch). The NRC staff finds that any malfunction of the fuel that could occur, such as fuel plate swelling, bowing, or leaks, would lead to consequences that are smaller than those analyzed in the licensee's fuel failure

scenario, because any radioactive release would be smaller and would occur more slowly. The NRC staff additionally finds that the licensee's assumption that particulates would remain in the pool is reasonable, because the particulate fission products are non-volatile and would be highly soluble in the large volume of pool water. The NRC staff also finds that the other fuel failure scenario assumptions and boundary conditions above would lead to conservative estimates for doses to both occupational workers and members of the public. Therefore, the NRC staff finds that these assumptions are acceptable.

Radionuclide Inventory

The licensee calculated the saturated radionuclide inventory in the highest power fuel plate. As discussed in SER Section 4.6, when the reactor is operating at full power, the power in the highest power plate is 9.653 kWt. Using this fuel plate power, and the cumulative fission yields for uranium-235, the licensee calculated the saturated inventories for select iodine and noble gas (krypton and xenon) radionuclides. The licensee did not include short-lived noble gases with half-lives of less than about 15 minutes. Table 13-10 summarizes the licensee's calculated iodine and noble gas inventories in the highest-power fuel plate.

The NRC staff reviewed the licensee's inventory estimate and finds that the inventory estimate is conservative because it is based on the highest power fuel plate in the core, and it assumes that the core (and the highest power fuel plate) have been operated for a long period of time such that the inventory is saturated. The NRC staff reviewed the fission yields used by the licensee for its calculation against published fission yields (Refs. 29, 31, 47), and finds that although the values vary slightly, the difference is not significant. The NRC staff also reviewed the list of radionuclides considered in the licensee's inventory estimate. The NRC staff noted that because the fuel failure scenario can be assumed to be a mechanistic fuel failure (i.e., a fuel handling accident), as discussed above, and fuel in the core is not moved during reactor operation, the fuel failure scenario would not occur during reactor operation. There would be a delay between the shutdown of the reactor and a potential fuel failure scenario, and therefore, the licensee's exclusion of very short-lived noble gases isotopes (i.e., those with half-lives of approximately 4 minutes or less) is reasonable. However, the licensee also excluded xenon-138 (Xe-138), which has a half-life of approximately 14 minutes. The NRC staff noted that the half-life of Xe-138 is long enough that it would still be present in the core inventory for approximately 2 hours following shutdown. Therefore, the NRC staff considered Xe-138 in its confirmatory calculation of occupational and members of the public fuel failure scenario doses, which is discussed below. Although short-lived noble gases are excluded from calculations discussed in this section because they are assumed to have decayed away by the time the accident occurs, the licensee and NRC staff calculations discussed in this section conservatively assume that for the radionuclides that are considered in the analyses, no decay occurs following the initiation of the accident scenario. Based on the information above, the NRC staff finds that the licensee's fuel failure inventory estimate shown in Table 13-10 is acceptable, except as noted above.

Table 13-10 RINSC Estimates of the Fuel Failure Scenario Nuclide Inventory

Nuclides	Estimate of the saturated inventory in the highest power fuel plate (Ci)
I-131	2.25×10^{2}
I-132	3.35×10^{2}
I-133	5.48×10^{2}
I-134	5.82×10^{2}
I-135	5.18×10^{2}
Kr-85m	1.08×10^{2}
Kr-85	2.31×10^{1}
Kr-87	1.92×10^{2}
Kr-88	2.95×10^{2}
Xe-133m	1.53×10^{1}
Xe-133	5.49×10^{2}
Xe-135m	8.51×10^{1}
Xe-135	5.45×10^{2}

Release Fractions

The licensee assumed that the noble gases and iodines that are within the range of fission product recoil from the denuded surface of the fuel plate are released to the reactor pool. The licensee assumed a recoil range of 1.37×10^{-3} centimeters, which is the range of fission fragment recoil in aluminum as stated in NUREG/CR-2079, "Analysis of Credible Accidents for Argonaut Reactors" (Ref. 38). Using this recoil range, and the dimensions of the fuel plate, the licensee calculated that the release fraction for noble gases and iodines from the failed fuel plate to the pool is 0.027. The NRC staff finds that the recoil range assumed by the licensee is appropriate for aluminum matrix fuel such as the fuel used in the RINSC reactor, and that the licensee's method for calculating the release fraction using the recoil range is reasonable and consistent with established practice.

The licensee assumed that the noble gases and iodines released to the pool are instantaneously and uniformly mixed throughout the pool water. The licensee also assumed that the noble gases released to the pool are all instantaneously released to the confinement air (i.e., the release fraction of noble gases from the pool to the confinement air is 1). The licensee used the guidance in RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Ref. 34), to determine the retention of the iodine in the pool water. Based on the guidance in RG 1.183, Appendix B, the licensee stated that because the RINSC pool has more than 23 ft of water above the active fuel, 99.5 percent of the iodine released to the pool is retained in the pool, and only 0.5 percent of the released iodine enters the confinement air. Therefore, the licensee assumed that the release fraction of iodines from the pool to the confinement air is 0.005. Similarly to the noble gases, the licensee assumed that the release of iodines from the pool to the confinement air would occur instantaneously. This is conservative because in actuality, the iodines would be released from the pool slowly due to pool water evaporation, allowing time for the iodines to decay. The NRC staff finds that the licensee's release fractions for noble gases and iodines from the pool to the confinement air are reasonable, conservative, and consistent with the guidance in RG 1.183.

Table 13-11 summarizes the licensee's release fraction assumptions for noble gases and iodine from the fuel to the pool and from the pool to the confinement air. The licensee assumed that the noble gases and iodines that are released to confinement are instantaneously and uniformly dispersed throughout the confinement air. TS 5.5.1, which is discussed and found acceptable in SER Section 6.2.1, requires that the free air volume of the confinement building, in which the material released from the pool to confinement will be dispersed, shall be 181,955 cubic ft.

Table 13-11 Noble Gas and Iodine Release Fractions from the Fuel Plate to Confinement

Group	Release Fraction from the Fuel Plate to the Pool	Release Fraction from the Pool to the Confinement	Total Release Fraction from the Fuel Plate to Confinement
Noble gases	0.027	1.0	0.027
Iodine	0.027	0.005	1.35×10^{-4}

The licensee uses additional release fraction assumptions in determining the material released from the confinement air to the environment. As discussed above, the licensee also assumed that the confinement isolation and emergency exhaust system is manually activated in conjunction with the release of radioactive material from the pool to confinement and the activation of the confinement RMS. TS 4.5.4, which is discussed and found acceptable in SER Section 6.2.1, requires that the emergency filter bank of the emergency exhaust system be verified to be at least 99 percent efficient for removing iodine. Based on the guidance in RG 1.183, the licensee considered that the iodine released from the pool to the confinement consists of 43 percent organic and 57 percent elemental iodine, and assumed that only the elemental iodine can be absorbed by the charcoal filter. Therefore, the licensee calculated an additional release fraction of approximately 0.436 for iodine released from confinement to the environment, based on the assumptions that 100 percent of the organic iodine, and 1 percent of the elemental iodine, pass through the charcoal filter. The licensee did not take credit for any other iodine hold-up or plate-out in the reactor building or ventilation system. For noble gases, the licensee assumed that the entire inventory released to the confinement air is also available to be released to the environment. The NRC staff finds that the licensee's release fractions from the confinement to the environment are conservative and are consistent with guidance in RG 1.183.

The NRC staff reviewed the information above, and finds that the release fractions assumed by the licensee for its fuel failure scenario analysis are justifiable, conservative, and consistent with guidance and established practice, and the RINSC TSs. Therefore, based on the information above, the NRC staff finds that the licensee's release fractions are acceptable.

Atmospheric Dispersion

For its analysis of member of the public doses from the fuel failure scenario, the licensee used similar methodologies and assumptions as the MHA regarding the emergency exhaust system and stack, atmospheric dispersion, and the locations and stay times for exposed members of the public. The NRC staff reviewed this information and found it acceptable, as discussed in SER Section 13.1.

Dose Calculations

The licensee calculated the potential TEDE for an occupational worker in the reactor building. The licensee also calculated the members of the public TEDE at the RINSC site boundary (which is also the location of maximum member of the public dose for accident releases from the RINSC stack, as discussed in the MHA analysis SER Section 13.1), and at the location of the nearest resident.

As discussed above, the licensee assumed a 5-minute exposure time for the occupational worker in the reactor building. For the occupational dose calculations, the licensee also assumed that the containment ventilation system, including the emergency exhaust system, is off and isolated, such that no radioactive material is vented from the confinement during the 5-minute stay time (a conservative assumption because, as discussed above, the emergency exhaust system would be activated and would reduce the concentrations of radioactive material in the confinement following the initiation of the accident). The licensee also conservatively assumed that no other leakage or decay of radioactive material in the confinement occurs during the 5-minute stay time. The occupational worker is exposed due to submersion in, and inhalation of, airborne radioactive material.

Similar to the MHA analysis in SER Section 13.1, members of the public outside the reactor building are assumed to be exposed to radioactive material that leaves the building via the emergency exhaust system and enters the environment through the facility stack. The members of the public are exposed due to submersion in, and inhalation of, airborne radioactive material. No credit is taken for the radionuclide decay inside or outside the confinement.

For the occupational and members of the public dose calculations, the licensee followed the DAC and AEC approaches, respectively, that are based 10 CFR Part 20, Appendix B. The NRC staff noted that the DAC approach accounts for the external (submersion) dose from noble gases, and the internal (inhalation) thyroid dose from iodines, but it does not consider the external (submersion) dose or the internal dose to other organs from the iodines. Because noble gases are non-reactive and do not accumulate in the human body, the inhalation dose from noble gases is negligible. The AEC approach accounts for the external dose from noble gases, and the internal dose to all organs from iodines, but does not consider the external dose from iodines. Additionally, in using the AEC approach for the member of the public doses, the NRC staff noted that the licensee appeared to make some assumptions that are inconsistent with the accepted method of applying this approach. Specifically, for the members of the public dose calculations, the licensee assumed that the AECs for iodines listed in 10 CFR Part 20, Appendix B, Table 2, Column 1, correspond to an internal dose of 100 mrem to the whole body, while the accepted interpretation is that these AECs correspond to an internal dose of 50 mrem to the whole body. The assumption used by the licensee resulted in an overestimation of the member of the public internal dose from iodines by a factor of 2. Additionally, for the member of the public dose calculations, the NRC staff noted that the licensee used an AEC for krypton-88 that appeared to have a misplaced decimal point compared to the AEC from 10 CFR Part 20, Appendix B, Table 2, Column 1 (the licensee used an AEC of 9×10^{-8}, compared to the 10 CFR Part 20, Appendix B, AEC of 9×10^{-9}). The licensee's use of this AEC for krypton-88 resulted in an underestimation of the member of the public external dose from krypton-88 by a factor of 10. For its confirmatory analysis of occupational and the member of the public fuel failure scenario doses, which is discussed below, the NRC staff used an alternate approach that uses the dose conversion factors from FGR No. 11 (Ref. 40) for internal (inhalation) dose calculations, and FGR No. 12 (Ref. 41) for external (submersion) dose calculations, to verify that the licensee's total calculated doses remain within 10 CFR Part 20 limits.

The NRC staff reviewed the information above regarding methodology and assumptions for the licensee's occupational and member of the public dose calculations. The NRC staff finds that the methodology and assumptions discussed above are generally reasonable and consistent with established practice, and are therefore acceptable, except as noted above.

Occupational Dose Estimates

The licensee calculated occupational doses as discussed above. The licensee's calculated occupational doses include the CDE to the thyroid (organ dose to the thyroid from radioactive material inhalation), the CEDE (internal whole-body dose from inhalation of radioactive material), the deep dose equivalent (DDE) (external whole-body dose from submersion in radioactive material), and the TEDE (total whole-body dose from internal and external sources). The licensee's calculated occupational doses are shown in Table 13-12.

The NRC staff performed a confirmatory calculation of the fuel failure scenario occupational doses. The NRC staff used the licensee's fuel failure scenario source term shown in Table 13-10, and also considered Xe-138 in its calculation, as discussed above (the NRC staff obtained the U-235 fission yield for Xe-138 from Ref. 47). Also as discussed above, the NRC staff used dose conversion factors from FGR No. 11 and FGR No. 12 in place of the licensee's DAC approach. Other aspects of the methodology and assumptions used by the NRC staff for its confirmatory analysis were similar to those used by the licensee for its analysis. The results of the NRC staff's confirmatory occupational dose calculations are shown in Table 13-12 alongside the licensee's results. There is some variation in the licensee- and NRC-calculated doses due to the differences in the methodologies and assumptions used. However, as Table 13-12 shows, all calculated doses are below the occupational dose limits in 10 CFR 20.1201.

Table 13-12 Fuel Failure Scenario 5-minute Occupational Dose Estimates in the Restricted Area

Dose Parameters	RINSC-Calculated Dose (mrem)	NRC Confirmatory Calculation (mrem)	10 CFR 20.1201 Dose Limit (mrem)
CDE to the thyroid	961	944	50,000
CEDE	29	30	5,000
DDE	285	480	5,000
TEDE	314	510	5,000

Public Dose Estimates

The licensee calculated the member of the public doses as discussed above. The licensee's calculated member of the public doses include the CEDE (internal whole-body dose from inhalation of radioactive material), the DDE (external whole-body dose from submersion in radioactive material), and the TEDE (total whole-body dose from internal and external sources). The licensee's calculated public doses are shown in Table 13-13.

The NRC staff performed a confirmatory calculation of the fuel failure scenario member of the public doses. The NRC staff used the licensee's fuel failure scenario source term shown in

Table 13-10, and also considered Xe-138 in its calculation, as discussed above. Also as discussed above, the NRC staff used dose conversion factors from FGR No. 11 and FGR No. 12 in place of the licensee's DAC approach. Other aspects of the methodology and assumptions used by the NRC staff for its confirmatory analysis of fuel failure scenario member of the public doses were similar to those used by the licensee for its analysis. The results of the NRC staff's confirmatory member of the public dose calculations are shown in Table 13-13 alongside the licensee's results. There is some variation in the licensee- and NRC-calculated doses due to the differences in the methodologies and assumptions used. However, as Table 13-13 shows, all calculated doses are below the 100 mrem members of the public dose limits in 10 CFR 20.1301.

Table 13-13 Fuel Failure Scenario Member of the Public Dose Estimates at the Site Boundary and Nearest Residence

	RINSC-Calculated Doses (mrem)		NRC Confirmatory Calculations (mrem)		
Dose Parameters	**Site Boundary (Location of Maximally-Exposed Member of the Public)**	**Nearest Residence**	**Site Boundary (Location of Maximally-Exposed Member of the Public)**	**Nearest Residence**	**10 CFR 20.1301 Dose Limit (mrem)**
CEDE	1.9	0.2	1.3	0.1	100
DDE	12	1.2	46	4.6	100
TEDE	14	1.4	47	4.7	100

The NRC staff noted that although the member of the public dose calculations above were performed for the nearest residence and the RINSC site boundary (the licensee and NRC staff determined that the nearest site boundary would be a higher-dose location that any other location outside the site boundary), there are still other publicly-accessible locations nearer to the reactor building, within the site boundary. However, the NRC staff expects that for these areas, which are under control of the licensee, the licensee would control access to help ensure that the members of the public doses remain below the 100 mrem member of the public dose limit in 10 CFR 20.1301 during accident conditions (see SER Section 13.1).

The NRC staff also performed a calculation of the shine DDE dose rate (the external radiation exposure due to radioactive material suspended in the air of the confinement) for members of the public located at the site boundary and the nearest residence, and at an additional location 10 m (32.8 ft) from the confinement building wall. The NRC staff's shine dose rate calculation used the MicroShield 10.0 computer code, modelling the confinement building as a spherical volume, and considering the radioactive material released to the reactor bay to be uniformly distributed throughout the volume. The NRC staff calculation considered all iodines and noble gases that the licensee considered to be released to the confinement air, plus xenon-138. The calculations take no credit for radioactive decay or any reduction in concentration due to leakage or material being exhausted through the ventilation system, but the calculations do take credit for the shielding provided by the 9 in (22.86 centimeter) concrete wall of the building. The NRC staff's calculated dose rates are shown in Table 13-14. The dose rate at the location of the nearest residence is small. Assuming that a member of the public were located at the site

boundary for the entire 2-hour period before all airborne radioactive material is exhausted from the confinement, the member of the public could receive a shine DDE of approximately 0.8 mrem from airborne material in confinement (assuming no radioactive decay, and assuming no material is exhausted from confinement until the end of the 2-hour period). The NRC staff notes that when this additional dose is added to the licensee- and NRC-calculated member of the public TEDEs shown in Table 13-13, the total dose is still well below the 100 mrem members of the public dose limit in 10 CFR 20.1301.

Table 13-14 Fuel Failure Scenario Radiation Shine through the Reactor Confinement Building

Parameters	10 Meters from Confinement	Site Boundary	Nearest Residence
Dose rate (mrem per hour)	4.6	0.4	0.0008

The NRC staff reviewed the licensee's fuel failure scenario dose calculations, and finds that the methodologies and assumptions used by the licensee are reasonable, conservative, and consistent with established industry practices, except as noted above. As discussed above, the NRC staff also performed independent confirmatory calculations of the occupational and member of the public doses from the fuel failure scenario. The NRC staff finds, based on its review of the licensee's dose calculations, and the results of the NRC staff confirmatory calculations, that the fuel failure scenario results demonstrate that the maximum fuel failure doses are below the occupational dose limits in 10 CFR 20.1201 and the members of the public dose limit in 10 CFR 20.1301, and are bounded by the MHA discussed in SER Section 13.1. Therefore, based on the information above, the NRC staff concludes that the results of fuel failures at the RINSC are acceptable.

13.6 Experiment Malfunction

The licensee discussed and analyzed experiment malfunctions in responses to RAI 13.1 and RAI 13.7 (Ref. 3), and in a SAR supplement (Ref. 56). The licensee identified potential experiment malfunctions that could result in either unanticipated reactivity transients, or releases of radioactive material. TS 3.1.1.3, which is discussed and found acceptable in SER Section 10.3, limits the reactivity worth of experiments at the RINSC. The basis for TS 3.1.1.3 is the licensee's evaluation of accidents involving insertion of excess reactivity, which is discussed and found acceptable in SER Section 13.2. TS 3.8.1.4.1, which is also discussed and found acceptable in SER Section 10.3, requires that the quantity of fissionable materials used in experiments shall not cause the experiment reactivity worth limits to be exceeded. Since TS 3.1.1.3 and TS 3.8.1.4.1 limit the reactivity worth of experiments such that any unanticipated reactivity transient would be less severe than the transients analyzed in SER Section 13.2, the NRC staff finds that experiment malfunctions that could result in unanticipated reactivity transients are bounded by the analyses in SER Section 13.2.

The licensee's SAR supplement (Ref. 56) analyzed fissionable experiment failures, which could result in releases of radioactive material. The licensee has determined that a fissionable experiment failure accident is the MHA for the RINSC (Ref. 58). TS 2.1.4.2, which is discussed and found acceptable in SER Section 10.3, limits the maximum quantity of fissionable materials and the fission rate in fissionable experiments. The basis for TS 3.8.1.4.2 is the licensee's MHA analysis, which is discussed and found acceptable in SER Section 13.1. Since TS 3.8.1.4.2 places limitations on fissionable experiments such that the fission product inventory of any

fissionable experiment would be below the inventory analyzed in the MHA, the NRC staff finds that fissionable experiment failures are bounded by the MHA.

TS 3.8.1.1 requires that corrosive materials be doubly contained in corrosion resistant containers, and TS 3.8.1.2 requires that highly water reactive materials not be placed in the reactor. TS 6.5 imposes requirements for the review and approval of experiments at the RINSC. TSs 3.8.1.1, 3.8.1.2, and 6.5 are discussed and found acceptable in SER Section 10.3. The NRC staff finds that these TSs help prevent damage to the reactor from experiment malfunctions other than those discussed above, and also help ensure that safety considerations related to experiments are adequately reviewed before the experiment is performed.

The NRC staff reviewed the information above. The NRC staff finds that the licensee has proper controls established for experiments to minimize the likelihood or consequences of experiment malfunctions. The NRC staff also finds that the performance of experiments within the restrictions of the TSs provides reasonable assurance that the potential consequences of experiment malfunctions would be bounded by those evaluated and found acceptable in the MHA and insertion of reactivity accident analyses. Therefore, based on the information above, the NRC staff concludes that the results of experiment malfunctions at RINSC are acceptable.

13.7 Loss of Electrical Power

The loss of normal electrical power is an anticipated event for the RINSC and the licensee would not expect this event to cause an accident. According to SAR Section 8.3, reactor shutdown is passive and fail safe in that, if normal power is lost, the control rods automatically fall into the core by gravity, thereby shutting down the reactor. In addition, upon a power loss, no forced cooling system is available and the event will be similar to the loss of flow discussed in Section 13.4.1; essentially, any residual heat from the core is dissipated into the pool water and eventually into the reactor room air space. No TS requires building power when the reactor is shut down. Therefore, since the reactor is automatically shut down when all power is lost, there are no requirements for electrical power to maintain the reactor in a safe condition.

Heat from the core is dissipated into the pool water and eventually into the reactor room air space. Therefore, since the reactor is automatically shut down when all power is lost, there are no requirements for electrical power to maintain the reactor in a safe condition.

However, the RINSC facility is equipped with an emergency generator that will supply selected equipment during abnormal losses of power. These include: (1) emergency exhaust system, (2) emergency evacuation system, (3) dilution blower, emergency lighting, communication equipment, and (4) multiple electrical outlets. The main purpose of the emergency electrical power source is to ensure that power is available to confinement system components that are necessary to make certain that the confinement system is able to perform its intended function in the event of an electrical power outage (TS 3.6). Quarterly operability tests verify that the emergency power system starts in the event of a facility power outage (TS 4.6), with a fuel tank that is at least 5 percent full during monthly tests. This ensures that there is sufficient fuel to power the emergency generator under full load for approximately 30 hours.

On the basis of these design factors, the NRC staff concludes that there is reasonable assurance that a loss of normal electrical power would not pose an undue risk to members of the public health and safety, facility personnel, or the environment.

13.8 External Events

According to SAR Section 13.2.7, tornadoes and floods are very rare in the area of the RINSC reactor. The building is designed to withstand the wind storms that occur infrequently during the hurricane season. The RINSC is located in a low seismic activity area. According to SAR Section 2.5.2, only a few earthquakes of modified Mercalli intensity (MMI) V or greater have been centered within RI during the past 200 to 300 years. A recent Seismic Vulnerability Study of RI identifies the strongest earthquake as a MMI level VI, with a possibility of MMI VII earthquake. The reactor building is a reinforced concrete, low-rise building that is designed to the requirements of the Uniform Building Code for Zone 2 structures and thus can be expected to have an acceptable response to earthquakes, with no damage expected. In addition, the reactor is submerged in a pool of water surrounded by a concrete biological shield and support structure which is not physically adjacent to any of the exterior walls. The thick monolithic structure housing the core provides sufficient protection for credible external events.

SAR Section 2.2 provides a review of the nearby industrial, transportation, and military facilities, and concludes that because of the separation distance the potential for any impacts from these facilities to be very low. In addition, the facility location and its security minimizes the potential for any human controlled events affecting the facility. Therefore, these events are not considered to be viable causes of accidents for the reactor facility.

The NRC staff reviewed the potential for external events as described in SAR Chapter 2 and in SER Chapter 3 and finds there is reasonable assurance that no external event would pose an unacceptable risk to the health and safety of the public.

13.9 Mishandling and Malfunctioning of Equipment

According to the SAR Section 13.2.8, the potential exists for a release of radioactive water to the environment following a pool leak. Water leaking from the pool would drain to the reactor room and from there to the lower levels of the facility and into the campus water drain network. Reactor water would be diluted by other sources in the campus network before entering the public drain system and eventually Narragansett Bay. In the response to RAI 2.2 (Ref. 3), the licensee identified the main source of the pool contamination as Tritium, with a concentration of 3.0×10^{-4} μCi/ml. This concentration is well below the allowed concentration in Table 3 of Appendix B to 10 CFR Part 20, which can be discharged to sewer system. Because the pool leak will get further dilution by the surface water in the campus network, there would be no significant potential facility impact on the groundwater. In addition, the NRC staff finds that TS 3.3.2 helps to detect a primary to secondary system leak by testing the secondary coolant analysis for the presence of Sodium-24, which is produced by the activation of the aluminum structural materials in the primary pool, and a small concentration is present in the primary coolant during reactor operation. More than a small concentration in the secondary coolant would alert the operators to a potential a primary to secondary system leak, and corrective actions to minimize any potential for leakage to the environment.

The NRC staff reviewed the potential release pathway and source terms and finds that the consequences of the release of the pool water to the surface water or groundwater to be negligible.

13.10 Conclusions

The NRC staff has reviewed the accident analyses presented in the SAR and in RAI responses, and finds the licensee has considered a sufficient range of accident categories and analyzed limiting scenarios for each category to bound all credible accidents for RINSC reactor. Based on its review, 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 fission product release of a fueled experiment, or 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 RINSC staff or radiation exposure to a member of the public in excess of the applicable 10 CFR Part 20 limits.

- The licensee has generally employed appropriate methods in performing the accident and consequence analysis.

- The review of the calculations, including assumptions, demonstrated that a LOCA would not result in unacceptable fuel element temperatures.

- External events that would lead to fuel failure are unlikely.

- The accident analysis confirms the acceptability of the licensed power of 2.0 MWt, including the response to anticipated transients and accidents.

- The accident analysis confirms the general acceptability of the assumptions and methods stated in the individual analyses provided in the SAR, as supplemented. In some cases, the NRC staff applied enhancements to its methodology for its confirmatory calculations.

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

14. TECHNICAL SPECIFICATIONS

In this section of the SER, the NRC staff provides its evaluation of the licensee's proposed TSs. The TSs for the RINSC reactor define specific features, characteristics, and conditions required for the safe operation of the RINSC. The TSs are explicitly included in the renewal license as Appendix A. The NRC staff reviewed the format and content of the TSs for consistency with the guidance in Chapter 14, "Technical Specifications," of NUREG-1537, Appendix 14.1, "Format and Content of Technical Specifications," and ANSI/ANS-15.1-2007 (Ref, 15).

The NRC staff specifically evaluated the content of the proposed TSs to determine whether they meet the requirements in 10 CFR 50.36, "Technical Specifications," to include SLs, LSSSs, limiting conditions for operation (LCOs), Surveillance Requirements (SRs), Design Features, and Administrative Controls. The NRC staff also relied on NUREG-1537 (Ref. 11) to perform its review. The NRC requires each licensee for a license to operate a non-power reactor to develop TSs that state the limits, operating conditions, and other requirements imposed on the facility operation to protect the environment and the health and safety of the facility staff and the members of the public, in accordance with 10 CFR 50.36. The TSs are typically derived from the facility descriptions and safety considerations in the SAR and represent a comprehensive envelope of safe operation. The SER sections where the TS was evaluated are only referenced in this Chapter if the TS was evaluated previously in the SER.

14.1 Introduction

The licensee provided the scope, format, and applicable definitions in Section 1.0 of the proposed RINSC TS. The licensee proposed definitions to be generally consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. The licensee's proposed TSs include minor modifications to, and some additional facility-specific, definitions.

TS 1.0 states:

> Scope
>
> This document constitutes the Rhode Island Nuclear Science Center (RINSC) Technical Specifications for Facility License number R-95 as required by 10 CFR Part 50.36 and supersedes all prior Technical Specification revisions and/or amendments. This document includes the "bases" to support the selection and significance of the specifications. Each bases is included for information purposes only, they are not part of the Technical Specifications and do not constitute limitations or requirements to which the licensee must adhere.
>
> Format
>
> These specifications are formatted to NUREG-1537 and ANSI/ANS-15.1-2007.

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 with acceptable accuracy to known values of the parameter that the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip, and shall be 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, or by comparison of the channel with other independent channels or systems measuring the same parameter.

1.4 Channel Test

A channel test is the introduction of a signal into the channel for verification that it is operable.

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.

1.6 Control Rod

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.

1.7 Core Configuration

The core configuration includes the number, type, and arrangement of fuel elements, reflector elements, and control rods occupying the core grid.

1.8 Excess Reactivity

Excess reactivity is that amount of reactivity that would exist if all of the control rods were moved to the maximum reactive condition from the point where the reactor is exactly critical when the core is in the reference core condition with the maximum allowed experiment worth installed.

1.9 Experiment

An experiment is any operation that is designed to investigate non-routine reactor characteristics, or any material or device not associated with the core configuration or the reactor safety systems that is intended for irradiation within the pool or an experimental facility. Hardware that is rigidly secured to a core or shield structure so as to be part of its design to carry out experiments is not normally considered to be an experiment.

1.10 Experimental Facility

An experimental facility is any structure or device which is intended to guide, orient, position, manipulate, or otherwise facilitate a multiplicity of experiments of similar character.

1.11 Explosive Material

Explosive material is any material determined to be within the scope of Title 18, United States Code, Chapter 40, “Importation; Manufacture, Distribution and Storage of Explosive Materials,” and any material classified as an explosive by the Department of Transportation in the Hazardous Material regulations (Title 49 CFR, Parts 100-199).

1.12 Fixed Experiment

A fixed experiment is any experiment, experimental apparatus, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining forces shall be substantially greater than other forces to which the experiment might be subjected that are normal to the operating environment of the experiment, or that can arise as a result of a credible malfunction.

1.13 Limiting Conditions for Operation (LCO)

The limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the reactor.

1.14 Limiting Safety System Setting (LSSS)

Limiting Safety System Settings are settings for automatic protective devices related to those variables having significant safety functions, and chosen so that automatic protective action will correct an abnormal situation before a safety limit is exceeded.

1.15 May

The word “may” is used to denote permission, neither a requirement nor a recommendation.

1.16 Mode of Operation

Mode of operation refers to the type of core cooling that is employed while the reactor is operating. The two modes of operation are forced convection cooling mode which supports reactor operation up to 2 MW_t, and natural convection cooling mode which supports reactor operation up to 100 kW_t.

1.17 Moveable Experiment

A moveable 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.

1.18 Operable

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

1.19 Operating

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

1.20 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.

1.21 Reactivity Worth of an Experiment

The reactivity worth of an experiment is the maximum absolute value of the reactivity change that would occur as a result of:

1.21.1 Insertion or removal from the core,

1.21.2 Intended or anticipated changes in position, or

1.21.3 Credible malfunctions that alter experiment position or configuration.

1.22 Reactor Operating

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

1.23 Reactor Operator

A reactor operator is an individual who is licensed under 10 CFR Part 55 to manipulate the controls of the RINSC reactor.

1.24 Reactor Operator Trainee

A reactor operator trainee is an individual who is authorized to manipulate the controls of the RINSC reactor under the direct supervision of a licensed operator.

1.25 Reactor Safety Systems

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

1.26 Reactor Secured

The reactor is secured when under optimal conditions of moderation and reflection either:

1.26.1 There is insufficient moderator available in the reactor to attain criticality, or

1.26.2 There is insufficient fissile material present in the reactor to attain criticality, or

1.26.3 The following conditions exist:

1.26.3.1 All four shim safety blades and the regulating rod are fully inserted or other safety devices are in the shutdown position, as required by technical specifications, AND;

1.26.3.2 The master switch is in the off position and the key is removed from the lock, AND;

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

1.26.3.4 No experiments are being moved or serviced that have a reactivity worth of greater than 0.6%$\Delta k/k$ when moved.

OR

For the purpose of centering Shim Safety Blade armatures only, with the key in the master switch and the master switch in the ON position, the reactor is secured if ALL of the following conditions are met:

1.26.3.5 The only task being performed by the reactor operator is centering the Shim Safety Blade armatures, AND;

1.26.3.6 The control room door remains closed and locked while the reactor operator is performing the centering operation, AND;

1.26.3.7 The RO does not leave the pool top level of the confinement building, AND;

1.26.3.8 The RO maintains a visual line of sight to the control room door and the top of the stairwell leading to the pool top level, AND;

1.26.3.9 The RO notifies the individual logged as the second person in the facility (ref TS requirement 6.1.3.1.2.2) when leaving the control room and when returning to the control room, AND;

1.26.3.10 The scram relays are NOT reset, AND;

1.26.3.11 The Shim Safety Blade magnets are de-energized, AND;

1.26.3.12 ALL Shim Safety Blades indicate they are on the bottom.

1.27 Reactor Shutdown

The reactor is shut down if it is subcritical by at least 0.75%Δk/k in the reference core condition with the reactivity of all installed experiments included.

1.28 Readily Available on Call

Readily available on call shall mean that the individual is aware that they are on call, can be contacted within ten minutes, and is within a 30 minute driving time from the reactor building.

1.29 Reference Core Condition

The condition of the core when it is at ambient temperature and the reactivity of xenon is less than 0.2%Δk/k.

1.30 Regulating Rod (RR)

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

1.31 Reportable Occurrence

A reportable occurrence is any of the following:

1.31.1 A violation of the safety limit,

1.31.2 An uncontrolled or unplanned release of radioactive material which results in concentrations of radioactive materials inside or outside the restricted area in excess of the limits specified in 10 CFR Part 20,

1.31.3 Operation with a safety system setting less conservative than the limiting safety system setting established in the Technical Specifications,

1.31.4 Operation in violation of a limiting condition for operation established in the Technical Specifications,

1.31.5 A reactor safety system component malfunction or other component or system malfunction which could, or threaten to, render the safety system incapable of performing its intended safety functions unless the cause is due to maintenance,

1.31.6 An uncontrolled or unanticipated change in reactivity in excess of 0.75%$\Delta k/k$ from which the cause is unknown,

1.31.7 Abnormal and significant degradation of the fuel cladding,

1.31.8 Abnormal and significant degradation of the primary coolant boundary, or the confinement boundary,

1.31.9 An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or threatens to cause the existence or development of an unsafe condition in connection with the operation of the facility.

1.32 Restricted Area

Restricted areas are areas in which access is limited by the licensee for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials.

1.33 Safety Channel

A safety channel is a channel in the reactor safety system.

1.34 Safety Limits

Safety limits are limits on important process variables which are found to be necessary to reasonably protect the integrity of the principal barrier which guard against the uncontrolled release of radioactivity. The principal barrier is the fuel element cladding.

1.35 Scram Time

Scram time is the elapsed time between the initiation of a scram signal and the time when the blades are fully inserted in the core.

1.36 Senior Reactor Operator

A senior reactor operator is an individual who is licensed under 10 CFR Part 55 to manipulate the controls of the RINSC reactor and to direct the licensed activities of reactor operators.

1.37 Shall

The word “shall” is used to denote a requirement.

1.38 Shim Safety Blade (SSB)

A shim safety blade is a control rod of high reactivity worth used primarily to make course adjustments to power level, and to provide a means for very fast reactor shutdown by having scram capability.

1.39 Should

The word “should” is used to denote a recommendation.

1.40 Shutdown Margin

Shutdown Margin is the minimum shutdown reactivity necessary to provide confidence that under reference core conditions, the reactor can be made subcritical by means of the control and safety systems starting from any permissible operating condition including maximum experiment worth and with the most reactive Shim Safety Blade and the Regulating Rod in their most reactive positions and that the reactor shall remain subcritical without further operator action.

1.41 Site Boundary

That line beyond which the land is not owned, leased, or otherwise controlled by the licensee.

1.42 Surveillance Activities

Surveillance activities are activities that are performed on a periodic basis for the purpose of verifying the integrity and operability of facility infrastructure and equipment which provides confidence that these components will perform their intended functions.

1.43 Surveillance Intervals

Maximum intervals are to provide operational flexibility, not to reduce frequency. Established frequencies shall be maintained over the long term. Allowable surveillance intervals shall not exceed the following:

1.43.1 5 years (interval not to exceed 6 years).

1.43.2 2 years (interval not to exceed 2 1/2 years).

1.43.3 Annual (interval not to exceed 15 months).

1.43.4 Semiannual (interval not to exceed 7 1/2 months).

1.43.5 Quarterly (interval not to exceed 4 months).

1.43.6 Monthly (interval not to exceed 6 weeks).

1.43.7 Weekly (interval not to exceed 10 days).

1.43.8 Daily (shall be done during the calendar weekday).

1.44 True Value

The true value is the actual value of a parameter.

1.45 Unscheduled Shutdown

An unscheduled shutdown is any unplanned shutdown of the reactor that is not associated with testing or check out operations, which is caused by:

1.45.1 Actuation of the reactor safety system,

1.45.2 Operator error,

1.45.3 Equipment malfunction, or

1.45.4 Manual shutdown in response to conditions that could adversely affect safe operation.

1.46 Water Reactive Material

A material that explodes; violently reacts; produces flammable, toxic or other hazardous gases; or evolves enough heat to cause auto-ignition or ignition of combustibles upon exposure to water or moisture.

The NRC staff reviewed the scope and format of the TSs and finds that they are consistent with NUREG-1537 and ANSI/ANS-15.1-2007 with the exception of TS 1.26, discussed below.

TS 1.26, “Reactor Secured,” provides an additional set of conditions to allow operational flexibility while maintaining safety and compliance with the regulations. The definition of reactor secured typically requires the master switch to be in the off position and the key removed from the lock. TS 1.26.3.2 contains that requirement.

At the RINSC facility, turning off the master switch also removes power from the console. Anytime the power is removed from the console, a new startup checkout list must be performed. The design of the shim safety rods at the RINSC reactor design requires them to be centered (realigned) before reactor startup if they have been scrammed. If they are not centered, the shim safety rods can rub against the shroud they travel in during start up and get knocked loose from the control rod drive electromagnets causing an unwanted reactor scram. Reactor scrams

and a runback are performed as part of the startup checkout procedures. This requires the rods to be realigned after the startup checkout procedure is completed and prior to starting the reactor. If the key is removed to satisfy the secured requirements in TS 1.26.3.2, the just completed checkout would become invalid. The licensee's definition would allow the shim safety rods to be realigned without a second licensed operator present in the control room.

The licensee developed TS 1.26.3.5 through 1.26.3.12 to achieve an equivalent level of safety and security as removing the key. The NRC staff reviewed these TSs and finds that for the limited time it takes to center the shim safety blade armatures, the actions of: locking the control room door, only going to the pool top level area, maintaining visual sight of the control room door and the top of the stairwell, and contacting the second person required to be present at the facility, achieve the equivalent security as removing the key from the lock.

The NRC staff also finds that for the limited time required for the action ensuring the scram relays are not reset, the shim safety blades are de-energized, and all shim safety blades indicate they are on the bottom, the same safety level is achieved as turning off the master switch and removing the key. Based on the information above, the NRC staff concludes that the alternative wording for "Reactor Secured" provides acceptable safety and security consistent with the intention of the definition in the guidance in NUREG-1537 and ANSI/ANS-15.1-2007.

The NRC staff also reviewed the definition of reactor secured and the ability of the RO to center the shim safety blade armatures against the regulatory requirements in 10 CFR 50.54(j) and 10 CFR 50.54(k). The regulations in 10 CFR 50.54(j) state that "Apparatus and mechanisms other than controls, the operation of which may affect the reactivity or power level of a reactor shall be manipulated only with the knowledge and consent of an operator or senior operator licensed pursuant to part 55 of this chapter present at the controls." The process of centering the shim safety blade armatures does not change the reactivity or the power level of the reactor. The controls are fully inserted in the core when the centering activity occurs. For this reason, the NRC staff finds that the TS for reactor secured satisfies the regulations in 10 CFR 50.54(j).

The regulations in 10 CFR 50.54(k) states "an operator or senior operator licensed pursuant to part 55 of this chapter shall be present at the controls at all times during the operation of the facility." As stated in TS 1.22, "Reactor Operating," the reactor is operating any time it is not secured or shut down. When all of the shim safety blades are fully inserted, the reactor meets the subcriticality requirements of shut down as defined by TS 1.27. TS 1.26.3.2 requires the master switch to be in the off position and the key removed. The alternative actions, described above, meet the safety and security intention of placing the master switch in the off position and removing the key, which means the reactor would be in a secured condition. Since the reactor is secured and shutdown, it is not operating. Since the reactor is not operating, an operator is not required to be at the controls as required by 10 CFR 50.54(k). The NRC staff finds that the TS for reactor secured satisfy the regulatory requirements in 10 CFR 50.54(k).

The NRC staff reviewed the other TS definitions and finds that they are either facility specific or are standard definitions used in research reactor TSs, enhance the clarity of the TSs, and are consistent with NUREG-1537 and ANSI/ANS-15.1-2007. On this basis, the NRC staff concludes that TS 1.1 through 1.46 are acceptable.

14.2 Safety Limits and Limiting Safety System Settings

TS 2.1 and TS 2.2 are evaluated and found acceptable in SER Section 4.5.3, “Operating Limits.”

14.3 Limiting Conditions for Operation

14.3.1 TS 3.1 Core Parameters

TS 3.1.1, “Reactivity Limits,” including TS 3.1.1.1, “Core,” and TS 3.1.1.2, “Control Rods,” is evaluated and found acceptable in SER Section 4.5.3, “Operating Limits.”

TS 3.1.1.3, “Experiments,” is evaluated and found acceptable in SER Section 10.3, “Experiment Review.”

TS 3.1.2, “Core Configuration Limits,” is evaluated and found acceptable in SER Section 4.5.1, “Normal Operating Conditions.”

14.3.2 TS 3.2 Reactor Control and Safety System

TS 3.2.1, 3.2.2 and 3.2.3 are evaluated and found acceptable in Section SER 4.2.2, “Control Blades.”

TS 3.2.4 is evaluated and found acceptable in SER Section 7.4, “Reactor Protection System.”

14.3.3 TS 3.3 Coolant System

TS 3.3.1, “Primary Coolant System,” including TS 3.3.1.1, “Primary Coolant Conductivity,” and TS 3.3.1.2, “Primary Coolant Activity,” is evaluated and found acceptable in SER Section 5.4, “Primary Coolant Cleanup System.”

TS 3.3.2, “Secondary Coolant System,” is evaluated and found acceptable in SER Section 5.3, “Secondary Coolant System.

14.3.4 TS 3.4 Confinement System

TS 3.4, “Confinement System,” including TS 3.4.1, is evaluated and found acceptable in SER Section 6.2.1, “Confinement System.”

14.3.5 TS 3.5 Confinement Ventilation System

TS 3.5, “Confinement Ventilation System,” including TS 3.5.1, is evaluated and found acceptable in SER Section 6.2.1, “Confinement System.”

14.3.6 TS 3.6 Emergency Power System

TS 3.6, “Emergency Power System,” including TS 3.6.1, is evaluated and found acceptable in SER Section 8.2, “Emergency Electrical Power Systems.”

14.3.7 TS 3.7 Radiation Monitoring System and Effluents

TS 3.7.1, “Radiation Monitoring Systems,” including TS 3.7.1.1, “Required Radiation Monitoring Systems,” and TS 3.7.1.2, “Radiation Monitoring System Alarm Set Points,” is evaluated and found acceptable in SER Section 7.7, “Radiation Monitoring Systems.”

TS 3.7.2, “Effluents,” including TS 3.7.2.1, “Airborne Effluents,” and TS 3.7.2.2, “Liquid Effluents,” is evaluated and found acceptable in SER Section 11.2.3, “Release of Radioactive Waste.”

14.3.8 TS 3.8 Experiments

TS 3.8.1, “Experiment Materials,” including TS 3.8.1.1 through 3.8.1.4, is evaluated and found acceptable in SER Section 10.3, “Experiment Review.”

TS 3.8.2, “Experiment Failures or Malfunctions,” including TS 3.8.2.1 through 3.8.2.3, is evaluated and found acceptable in SER Section 10.3, “Experiment Review.”

14.3.9 TS 3.9 Reactor Core Components

TS 3.9.1, “Beryllium Reflectors,” is evaluated and found acceptable in SER Section 4.2.3, “Neutron Moderator and Reflector.”

TS 3.9.2, “Low Enriched Uranium Fuel,” is evaluated and found acceptable in SER Section 4.2.1, “Reactor Fuel.”

TS 3.9.3 “Experimental Facilities,” including TS 3.9.3.1 and TS 3.9.3.2, is evaluated and found acceptable in SER Section 10.2.8, “Flux Trap.”

14.4 Surveillance Requirements (SR)

14.4.1 TS 4.0 Surveillance Requirements

TS 4.0 states:

> Surveillance requirements may be deferred during periods when the reactor is shutdown (except as noted in table 4.1, Technical Specification Surveillance Deferral Summary, below); however, they shall be completed prior to reactor start up unless reactor operation is required to perform the surveillance. Such surveillance shall be completed as soon as practical after reactor start up is complete.
>
> Any additions, modifications, or maintenance performed on any of the systems or components addressed by these Technical Specifications shall be made and tested in accordance with the specifications the systems were originally designed and fabricated to or approved by the 10 CFR 50.59 review and approval process. A system shall not be considered operable until it is successfully tested.

Table 4.1 Technical Specification Surveillance Deferral Summary

	Technical Specification SR	Can Be Deferred During Shutdown (Y/N)	Required Prior to Reactor Operation (Y/N)
1	4.1.1.1 Core Reactivity Limit	Y	Y
2	4.1.1.2 Control Rod Reactivity Limit	Y	Y
3	4.1.1.3 Experiment Reactivity Limit	N	N
4	4.1.2 Core Configuration Limit	Y	Y
5	4.2.1 Shim safety drop times	Y	Y
6	4.2.2 Shim safety interlock/reactivity insertion rate	Y	Y
7	4.2.3 Reactor safety and safety related instrumentation	Y	Y
8	4.2.4 Reactor safety and safety related instrumentation for 2 MW mode of operation	Y	Y
9	4.2.5 Reactor safety and safety related instrumentation scrams, and interlocks	Y	Y
10	4.2.6 Reactor safety and safety related instrumentation channel calibration	Y	Y
11	4.3.1.1 Primary Coolant Conductivity	Y	N
12	4.3.1.2 Primary Coolant Activity	Y	N
13	4.3.1.3 Primary Coolant Level Inspection	N	Y
14	4.3.2.1 Secondary Coolant Activity	Y	N
15	4.4.1 Confinement System Operability	N	Y
16	4.4.2 Confinement System Operability	N	Y
17	4.4.3 Confinement System Operability	N	Y
18	4.5.1 Confinement Ventilation System Operability	N	Y
19	4.5.2 Confinement Ventilation System Operability	N	Y
20	4.5.3 Confinement Ventilation System Operability	N	Y
21	4.5.4 Emergency Filter Bank	N	Y
22	4.5.5 Emergency Filter Bank Flow	N	Y
23	4.6.1 Emergency Power System	N	Y
24	4.6.2. Emergency Power System	N	Y
25	4.6.3. Emergency Power System	N	Y
26	4.7.1.1 Radiation monitors	Y	Y
27	4.7.1.2 Radiation monitors	Y	Y
28	4.7.2.1 Airborne Effluents	N	N
29	4.7.2.2 Liquid Effluent Sampling	N	N
30	4.8.1 Experiments	Y	Y
31	4.9.1 Beryllium Reflector Elements	N	Y
32	4.9.2 Fuel Elements	N	Y
33	4.9.3.1 Experimental Facility Configuration	Y	Y
34	4.9.3.2 Accessing an Experimental Facility	N	N

TS 4.0 helps ensure that surveillances are accomplished in a planned and organized manner. The NRC staff finds that TS 4.0 requires that any additions, modifications, or maintenance performed on any of the systems or components addressed by the TSs be made and tested in accordance with the original design and fabrication specifications or approved in accordance

with the regulations 10 CFR 50.59. The NRC staff finds that this specification requires that a system not be considered operable until it is successfully tested. The NRC staff finds that TS Table 4.1 lists the TS surveillances that can be deferred during shutdown and required prior to operation. The NRC staff reviewed TS 4.0 and TS Table 4.1, and finds that this specification helps ensure that the surveillances, which are important to maintaining the integrity of the RINSC systems are properly maintained during normal operation and extended shutdown periods, prior to resuming reactor operation. The NRC staff also finds that TS 4.0 is consistent with the guidance provided in NUREG-1537, Appendix 14.1, Section 4.0. Based on the review of TS 4.0, the NRC staff concludes that TS 4.0 is acceptable

14.4.2 TS 4.1 Core Parameters

TS 4.1.1, “Reactivity Limit,” is evaluated and found acceptable in SER Sections 4.5.3, “Operating Limits,” and 10.3, “Experiment Review.”

TS 4.1.2, “Core Configuration Limit,” is evaluated and found acceptable in SER Section 4.5.1, “Normal Operating Conditions.”

14.4.3 TS 4.2 Reactor Control and Safety Systems

TSs 4.2.1 and 4.2.2 are evaluated and found acceptable in SER Section 4.2.2, “Control Blades.”

TSs 4.2.3 – 4.2.6 are evaluated and found acceptable in SER Section 7.4, “Reactor Protection System.”

14.4.4 TS 4.3 Coolant Systems

TSs 4.3.1.1, “Primary Coolant Conductivity,” and 4.3.1.2, “Primary Coolant Activity,” are evaluated and found acceptable in SER Section 5.4, “Primary Coolant Cleanup System.”
TS 4.3.1.3, “Primary Coolant Level Inspection,” is evaluated and found acceptable in SER Section 5.2, “Primary Coolant System.”

TS 4.3.2.1, “Secondary Coolant Activity,” is evaluated and found acceptable in SER Section 5.3, “Secondary Coolant System.”

14.4.5 TS 4.4 Confinement System

TS 4.4, “Confinement System,” is evaluated and found acceptable in SER Section 6.2.1, “Confinement System.”

14.4.6 TS 4.5 Confinement Ventilation System

TS 4.5, “Confinement Ventilation System,” is evaluated and found acceptable in SER Section 6.2.1, “Confinement System.”

14.4.7 TS 4.6 Emergency Power System

TS 4.6, “Emergency Power System,” is evaluated and found acceptable in SER Section 6.2.1, “Confinement System.”

14.4.8 TS 4.7 Radiation Monitoring System and Effluents

TS 4.7, “Radiation Monitoring System and Effluents,” is evaluated and found acceptable in SER Sections 7.7, “Radiation Monitoring Systems,” and 11.2.3, “Release of Radioactive Wastes.”

14.4.9 TS 4.8 Experiments

TS 4.8.1, “Experiments,” is evaluated and found acceptable in SER Section 10.3, “Experiment Review.”

14.4.10 TS 4.9 Facility Specific Surveillance

TS 4.9, “Facility Specific Surveillance,” is evaluated and found acceptable in SER Sections 4.2.1, “Reactor Fuel,” 4.2.3, “Neutron Moderator and Reflector,” and 10.2.8, “Flux Trap.”

14.5 Design Features

The RINSC Design Features are discussed in Section 5.0 of the prosed RINSC TS.

14.5.1 TS 5.1 Site and Facility Description

TS 5.1.1 is evaluated and found acceptable in SER Section 2.1.1, “Geography.”

TS 5.1.2 is evaluated and found acceptable in SER Section 6.2.1, “Confinement System.”

14.5.2 TS 5.2 Reactor Fuel

TS 5.2.1 and 5.2.2 are evaluated and found acceptable in SER Section 4.2.1, “Reactor Fuel.”

14.5.3 TS 5.3 Reactor Fuel Storage

TS 5.3.1, TS 5.3.2, and TS 5.3.3 are evaluated and found acceptable in SER Section 9.2, “Handling and Storage of Reactor Fuel.”

14.5.4 TS 5.4 Reactor Core

TS 5.4.1, TS 5.4.2, and TS 5.4.3 are evaluated and found acceptable in SER Section 4.2, “Reactor Core.”

14.5.5 TS 5.5 Confinement (Reactor) Building

TS 5.5.1 is evaluated and found acceptable in SER Section 6.2.1, “Confinement System.”

14.5.6 TS 5.6 Reactor Pool

TS 5.6.1 is evaluated and found acceptable in SER Section 4.3, “Reactor Pool.”

14.5.7 TS 5.7 Confinement Building Ventilation

TS 5.7.1 is evaluated and found acceptable in SER Section 6.2.1, “Confinement System.”

14.6 Administrative Controls

The RINSC Design Features are evaluated and found acceptable in the following sections of this report.

TS 6.1, "Organization," is evaluated and found acceptable in SER Section 12.1, "Organization."

TS 6.2, "Review and Audit," is evaluated and found acceptable in SER Section 12.2, "Review and Audit Activities."

TS 6.3, "Radiation Safety," is evaluated and found acceptable in SER Section 11.1.2, "Radiation Protection Program."

TS 6.4, "Procedures," is evaluated and found acceptable in SER Section 12.3, "Procedures."

TS 6.5, "Experiment Review and Approval," is evaluated and found acceptable in SER Section 10.3, "Experiment Review."

TS 6.6, "Required Actions," is evaluated and found acceptable in SER Section 12.4, "Required Actions."

TS 6.7, "Reports," is evaluated and found acceptable in SER Section 12.5, "Reports."

TS 6.8, "Records," is evaluated and found acceptable in SER Section 12.6, "Records."

14.7 Conclusions

The NRC staff reviewed and evaluated the RINSC TSs as part of its review of the LRA. Specifically, the NRC staff evaluated the content of the TSs to determine whether the TSs meet the requirements in 10 CFR 50.36. Based on its review, the NRC staff concludes that the RINSC TSs are acceptable for the following reasons:

- To satisfy the requirements in 10 CFR 50.36(a), the licensee provided proposed TSs with the application for license renewal. As required by the regulation, the TSs (other than those for administrative controls) included appropriate summary bases. Those summary bases are included in the TSs, but shall not be part of the TSs as required by 10 CFR 50.36(a)(1).

- The RINSC reactor is a facility of the type described in 10 CFR 50.21(c); therefore, as required by 10 CFR 50.36(b), the facility operating license will include the TSs. To satisfy the requirements in 10 CFR 50.36(b), the licensee provided proposed TSs derived from analyses in the RINSC reactor license renewal SAR, as supplemented by responses to RAIs.

- The proposed TSs acceptably implement the recommendations of NUREG-1537 (Ref. 11) and ANSI/ANS-15.1-2007 (Ref. 15), by using definitions that are acceptable.

- To satisfy the requirements in 10 CFR 50.36(c)(1), the licensee proposed TSs that specify a SL for the fuel temperature and LSSSs for the reactor protection system to ensure that the SL is not reached.

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

- The proposed TSs contain surveillance requirements that satisfy the requirements in 10 CFR 50.36(c)(3).

- The proposed TSs contain design features that satisfy the requirements in 10 CFR 50.36(c)(4).

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

The NRC staff reviewed the proposed RINSC TSs and finds the proposed TSs acceptable. The NRC staff concludes that normal operation of the RINSC reactor within the limits of the proposed TSs will not result in radiation exposures in excess of the limits specified in 10 CFR Part 20, "Standards for Protection Against Radiation," for members of the public or radiation worker reactor staff. The NRC staff concludes that the proposed TSs provide reasonable assurance that the RINSC reactor 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 for accident scenarios discussed in SER Chapter 13, "Accident Analyses," and the conduct of activities by the licensee will not endanger the facility staff or members of the public.

15. FINANCIAL QUALIFICATION

15.1 Financial Ability to Operate the Rhode Island Nuclear Science Center Reactor

The regulations in 10 CFR 50.33(f) includes the financial requirements for nonelectric utility licenses. The regulation states, in part, that:

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

The RIAEC does not qualify as an "electric utility," as defined in 10 CFR 50.2, "Definitions." Further, pursuant to 10 CFR 50.33(f)(2), the application to renew or extend the term of any operating license for a non-power reactor shall include the financial information that is required in an application for an initial license. Therefore, the NRC staff has determined that the RIAEC 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. The RIAEC 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, the RIAEC must submit estimates of the total annual operating costs for each of the first 5 years of facility operations from the time of the expected license renewal date and indicate the source(s) of funds to cover those costs, consistent with the requirements in 10 CFR 50.33(f)(2).

In a response to an RAI dated September 16, 2013 (Ref. 8), the RIAEC submitted its projected operating costs for the RINSC reactor for each of the fiscal years (FY) 2015 through FY 2019. The projected operating costs for the RINSC are estimated to range from $1,689,034 in FY 2015 to $1,901,021 in FY 2019. The cost of operating the facility and funding from the state of Rhode Island have been consistent over the years, so the financial evaluation may reliably be reviewed as valid for a period of up to three years. The RIAEC's primary source of funding to cover operating costs is provided by an annual appropriation from the Rhode Island Legislature. In addition, other sources of revenue include federal grants and payments from the University of Rhode Island for radiation safety services. The RIAEC expects that these funding sources will continue for the above-referenced FYs. The NRC staff reviewed the RIAEC's estimated operating costs and projected source of funds to cover those costs and found them to be reasonable.

The RINSC 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 regulations in 10 CFR 50.21(c), provide 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 licensee confirmed (Ref. 58) that less 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 NRC staff reviewed the proposed conduct of commercial activities at the RINSC. Because the regulations in 10 CFR 50.21(c) require that the majority of the RINSC operating costs be funded by non-commercial uses and the cost of conducting commercial

activities at the RINSC is less than 50 percent of the total cost of operating the facility, the NRC staff concludes that the RINSC license can be renewed as a Section 104.c license.

Based on its review, the NRC staff finds that the RIAEC has demonstrated reasonable assurance of obtaining the necessary funds to cover the estimated facility operation costs for the period of the license. Accordingly, the NRC staff has determined that the RIAEC has met the financial qualifications requirements pursuant to 10 CFR 50.33(f) and is financially qualified to engage in the proposed activities regarding the RINSC.

15.2 Financial Ability to Decommission the RINSC Reactor

The NRC has determined that the requirements to provide reasonable assurance of decommissioning funding are necessary to ensure the adequate protection of public health and safety. The regulations at 10 CFR 50.33(k) require that an application for an operating license for a utilization facility contain information to demonstrate how reasonable assurance will be provided and that funds will be available to decommission the facility. The regulations at 10 CFR 50.75(d) require that each non-power reactor applicant for or holder of an operating license shall submit a decommissioning report which contains a cost estimate for decommissioning the facility, an indication of the funding method(s) to be used to provide funding assurance for decommissioning the facility 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 specified in 10 CFR 50.75(e)(1).

In the RAI responses dated September 16, 2013 (Ref. 8), the RIAEC estimated the decommissioning costs for the RINSC at approximately $21 million in 2015 dollars. The decommissioning cost estimate predominantly summarized costs by labor, waste disposal, the annual costs associated with safe storage (SAFSTOR) for a 20-year period after the RINSC ceases operations, other items (e.g., energy, equipment, supplies), and a 31.5 percent contingency factor. In the January 19, 2010 (Ref. 46), and September 16, 2013, submittals, the RIAEC enclosed the worksheets showing the costs estimate and labor rates for RIAEC's base case scenario, where craft hours are only for removal and packaging of materials, and labor rates are estimates based on the Rhode Island Department of Labor and "Rent-a-tech" rates.

After permanent reactor shut down, the RIAEC plans to remove fuel from the core as soon as possible and ship fuel off site in accordance with Department of Energy, NRC, and Department of Transportation regulations. A contractor will be hired to perform dismantlement. In order to estimate costs for Class A wastes, the RIAEC received a quote from a radioactive waste broker. For Class B and C wastes, the RIAEC estimated costs based on Barnwell disposal site rates. The RIAEC will perform surveys of areas cleared of reactor components to potentially release the site.

The RIAEC compared the decommissioning cost estimate for the RINSC to other facility decommissioning costs or estimates, which were: Penn State, Georgia Tech, MIT, Watertown, Virginia, and Ohio State. As stated in the January 19, 2010, RAI response, the RIAEC averaged the range of costs from these reactors and the range of the RIAEC's base case calculation. The RIAEC's base case calculation weighed similar reactors, such as Virginia's, more heavily in the estimate. The RIAEC observed a difference of some 31.5 percent between the two estimates, which is higher than the 25 percent contingency requested by the NRC staff.

To obtain an estimate of costs of decommissioning as envisioned with initial deconstruction, removal of Class A wastes, and subsequent removal of Class B and C wastes when they decay to Class A wastes, RIAEC began with the base case analysis discussed above and computed costs out over 20 years. Since 2001, the average U.S. Bureau of Labor Statistics, Consumer Price Index averaged about 2.85 percent. However, the RIAEC believes that an inflationary rate of 5 percent would more accurately reflect the costs associated with decommissioning for the projected 20 years before undertaking final disposition. In addition, in order to provide a cost estimate in 2015 dollars, the RIAEC provided present values of total costs using a discount rate of 2.85 percent, the recent historical Consumer Price Index. Finally, according to the RIAEC, and based in documentation provided within the application, the recent decommissioning contingency amount included in the RIAEC's annual budget is $30 million. The RIAEC indicates this allows for extremes in the calculations shown plus a cushion for returning the site to readiness for another use. Based on the NRC staff's review of the information submitted by the RIAEC regarding decommissioning of the RINSC, the NRC staff finds that the decommissioning cost estimate submitted by the RIAEC is reasonable.

The RIAEC elected to use a statement of intent (SOI) to provide financial assurance, as allowed by 10 CFR 50.75(e)(1)(iv), for a Federal, State, or local government licensee. The SOI must contain a cost estimate for decommissioning and indicate that funds for decommissioning will be obtained when necessary. The RIAEC provided a SOI, dated September 16, 2013 (Ref. 8), stating that the signator will "…request and obtain these funds sufficiently in advance of decommissioning to prevent delay of required activities." The SOI is for $30,000,000.

To support the SOI and the RIAEC's qualifications to use a SOI, the application stated that the RIAEC is a state agency established under Rhode Island General Law 42-27 for matters relating to nuclear power and included documentation which corroborates this statement. The application also provided information supporting the RIAEC's representation that the decommissioning funding obligations of the RIAEC are backed by the full faith and credit of the State of Rhode Island. The RIAEC also provided documentation on September 16, 2013 (Ref. 8) verifying that Dr. Clinton Chichester, Chairman of the RIAEC, the signator of the SOI, is authorized to execute contracts on behalf of the RIAEC.

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

15.3 Foreign Ownership, Control, or Domination

Section 104d of the Atomic Energy Act (AEA), as amended, prohibits the NRC from issuing a license under Section 104 of the AEA to "any corporation or other entity if the Commission knows or has reason to believe it is owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government." The NRC regulations at 10 CFR 50.38, "Ineligibility of Certain Applicants," contain language to implement this prohibition. According to the application, the RIAEC is a State of Rhode Island government licensee and is not owned,

controlled, or dominated by an alien, a foreign corporation, or a foreign government. The NRC staff finds this acceptable.

15.4 Nuclear Indemnity

Pursuant to the requirements of the Price-Anderson Act (Section 170 of the AEA) and the NRC's implementing regulations at 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," the licensee currently has an indemnity agreement with the Commission that will terminate when Facility Operating License No. R-95 expires, provided all radioactive material has been removed from the location and transportation of radioactive material from the location has ended. Therefore, the RIAEC will continue to be a party to the present indemnity agreement following issuance of the renewed license. Under 10 CFR 140.71, the RIAEC, as a State government licensee, is not required to provide nuclear liability insurance. The Commission will indemnify the RIAEC for any claims arising out of a nuclear incident under the Price-Anderson Act, Section 170 of the AEA, as amended, and in accordance with the provisions under its indemnity agreement pursuant to 10 CFR 140.95 "Appendix E - Form of indemnity agreement with nonprofit educational institutions," up to $500 million. In addition, because RINSC reactor is a research reactor, the licensee is not required to purchase property insurance otherwise required by 10 CFR 50.54(w).

15.5 Conclusion

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 RINSC and, when necessary, to shut down the facility and carry out decommissioning activities. In addition, the NRC staff concludes there are no problematic foreign ownership or control issues or insurance issues that would preclude the issuance of a renewed license. In addition, the NRC staff finds that the applicable provisions of 10 CFR Part 140 have been satisfied. Based on its review, the NRC staff concludes that the licensee is financially qualified to engage in licensed activities during the renewal period.

16. OTHER LICENSE CONSIDERATIONS

16.1 Prior Use of Components

As detailed in previous sections of this SER, the NRC staff concludes that the continued operation of the RINSC reactor will not pose a significant radiological risk to the members of the public. 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 RINSC reactor include the fuel and cladding and reactor safety systems. The original fuel was replaced during the HEU to LEU fuel conversion in 1993. In addition, graphite and beryllium reflectors were added during the conversion to LEU fuel (Ref. 23).

SER Section 4.2.1 describes the NRC staff review and finding of acceptability of the reactor fuel. TS 3.9.2 requires an annual visual inspection of fuel assemblies for defects. Further, this TS requires that each fuel assembly be verified to fit appropriately into the core box. TS 3.9.1 places a limit of 10^{22} neutrons/cm^2 on the accumulated neutron flux to the beryllium reflector. A determination of the exposure to the reflector is required annually by TS 4.9.1. As stated in SER Chapter 4, continued operation, as limited by these TSs, offers reasonable assurance the fuel and reflector can meet the design objective of maintaining fuel integrity, which supports the safe operation of fuel in the reactor and protection of the public health and safety.

Additional considerations supporting the continued use of the fuel include the following:

- The design of the in-pool structures and components minimizes the chance for mechanical impact.
- Reactor components are contained in the core box and have support structures above the fuel.
- Fuel handling requires specially designed tools that do not come into contact with the cladding surface.
- TS 3.3.1 places requirements on the conductivity of the primary coolant. TS 4.3.1 specifies surveillance intervals for the chemical properties of the coolant. These TSs adequately ensure that no significant corrosion of the fuel cladding or support structure has occurred or will occur.

The design of the reactor safety system (e.g., safety channel circuitry and shim safety blade electromagnets) helps to preclude accidents as a result of failure of system components. Failure or removal for maintenance of safety-related I&C components results in a safe reactor shutdown. TS 4.2 specifies surveillance requirements of the reactor control and safety system. The NRC staff evaluated TS 4.2 in Chapter 7 of this SER and found it to be acceptable. The NRC staff concluded that any degradation of system components will be detected and addressed. Additionally, the RINSC reactor staff performs regular preventive and corrective maintenance and replaces system components, including upgrades and modifications, as necessary. The ongoing facility modernization further demonstrates the licensee's commitment to managing system aging.

The NRC staff conducts inspections (Ref. 28) of the RINSC reactor including review of facility modification, corrective actions, and maintenance programs. The NRC staff's review of the inspection results revealed that the facility was being maintained properly and that degraded equipment was repaired or replaced in a timely manner.

The NRC staff finds that there is no indication of significant degradation of the instrumentation and components, and there is strong evidence that the RINSC facility staff will remedy any future degradation with prompt corrective action. Corrective action would be facilitated by required TS surveillance activities, which would detect degradation before it could affect reactor operations or safety. The NRC staff did not consider prior utilization of other systems and components because degradation would occur gradually, be readily detectable, and would not affect the likelihood of accidents.

16.2 Medical Use of the RINSC Reactor

The licensee does not use the RINSC reactor for medical irradiations involving the use of special nuclear material (reactor fuel) for medical therapy.

16.3 Conclusions

The NRC staff reviewed the prior use of reactor components as well as the aging of safety components. Based on its review, the NRC staff concludes that there has been no significant degradation of reactor components to date. Further, the in the TSs provide reasonable assurance that the reactor components will continue to be SRs adequately monitored for degradation of systems and components during the license renewal period.

17. CONCLUSIONS

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

- The application for license renewal dated May 3, 2004, as supplemented on January 19, February 4, August 6, August 18, September 3, September 8, November 8, November 26, December 7, and December 14, 2010; January 24, February 24, and July 15, 2011; March 15, September 16, and December 19, 2013; February 24, April 28, and June 30, 2014; August 7 and August 11, 2015; and January 20, February 26, March 1, April 21, July 20, October 6, November 1, November 14, December 1, December 8, December 13, and December 15, 2016, complies with the standards and requirements of the Atomic Energy Act (AEA), and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations* (10 CFR).

- The facility will operate in conformity with the application, the provisions of the AEA of 1954, as amended, and the rules and regulations of the NRC.

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

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

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

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

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

- The issuance of this license is in accordance with 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," of the NRC's regulations, and all applicable requirements have been satisfied, as documented by the Environmental Assessment and Finding of No Significant Impact published in the *Federal Register* on January 5, 2017 (82 FR 1364), which concluded that renewal of the RINSC license will not have a significant effect on the quality of the human environment.

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

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