Enclosure 2

Safety Evaluation Report

Rensselaer Polytechnic Institute

Docket No. 50-225

ADAMS Accession No. ML111110690

Safety Evaluation Report Related to the Renewal of Facility Operating License No. CX-22 for the Rensselaer Polytechnic Institute Critical Experiments Facility, Rensselaer Polytechnic Institute

June 2011
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission

ABSTRACT

This safety evaluation report summarizes the findings of a safety review conducted by the staff of the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Reactor Regulation. The NRC staff conducted this review in response to a timely application filed by the Rensselaer Polytechnic Institute (RPI, the licensee) for a 20-year renewal of Facility Operating License No. CX-22 to continue to operate the Rensselaer Polytechnic Institute Critical Experiments Facility (RCF, the facility). In its safety review, the NRC staff considered information submitted by the licensee, past operating history recorded in the licensee's annual reports to the NRC, inspection reports prepared by the NRC staff, and firsthand observations by NRC staff. On the basis of its review, the NRC staff concludes that RPI can continue to operate the RCF for the term of the renewed facility operating license, in accordance with the renewed license, without undue risk to public health and safety, facility personnel, or the environment.

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ABBREVIATIONS AND ACRONYMS

Abbreviation	Definition
ADAMS	Agencywide Documents Access and Management System
AEA	Atomic Energy Act of 1954, as amended
ALARA	as low as reasonably achievable
ANSI/ANS	American National Standards Institute/American Nuclear Society
C	Celsius
CFR	*Code of Federal Regulations*
cm	centimeter(s)
DOE	U.S. Department of Energy
F	Fahrenheit
ft	foot (feet)
in.	inch(es)
ISG	interim staff guidance
kW-hr	kilowatt-hour(s)
m	meter(s)
m^3/hr	cubic meter(s) per hour
MHA	maximum hypothetical accident
mrem	millirem
mrem/hr	millirem per hour
mrem/yr	millirem per year
mSv	millisievert(s)
mSv/hr	millisievert(s) per hour
mSv/yr	millisievert(s) per year
NRC	U.S. Nuclear Regulatory Commission
NSRB	Nuclear Safety Review Board
RCF	Rensselaer Polytechnic Institute Critical Experiments Facility
RNSC	Radiation and Nuclear Safety Committee
RPI	Rensselaer Polytechnic Institute
RSO	Radiation Safety Officer
SAR	safety analysis report
SER	safety evaluation report
SPERT	special power excursion reactor test
SRM	staff requirements memorandum
TLD	thermoluminescent dosimeter
TS	technical specification(s)
W	watt(s)

LIST OF ABBREVIATIONS

Abbreviation	Definition

1 INTRODUCTION

1.1 Overview

By letter dated November 19, 2002, as supplemented by letters dated July 21, July 28, and September 3, 2008; June 28, August 31, October 14, and October 28, 2010; and February 14 and May 9, 2011, the Rensselaer Polytechnic Institute (RPI, the licensee) submitted to the U.S. Nuclear Regulatory Commission (NRC, the Commission) a timely application for a 20-year renewal of the Class 104c Facility Operating License No. CX-22, Docket No. 50-225 for the Rensselaer Polytechnic Institute Critical Experiments Facility (RCF, the facility).

The regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.51(a) state that each license will be issued for a period of time to be specified in the license but in no case to exceed 40 years from the date of issuance. The original facility operating license for the RCF was issued to RPI on August 10, 1965. The license was twice renewed, with the most recent being on December 2, 1983, for a period of 20 years expiring on December 2, 2003. A renewal would authorize continued operation of the RCF for an additional 20 years. Because RPI filed the request for license renewal in a timely manner, until the NRC staff completes action on the renewal request, the licensee is permitted to continue operation of the RCF under the terms and conditions of the existing license, in accordance with 10 CFR 2.109, "Effect of Timely Renewal Application."

The NRC staff conducted its review based on information contained in the renewal application, as supplemented. The renewal application includes the safety analysis report (SAR), proposed technical specifications (TS), the operator requalification plan, the emergency plan, financial qualifications, and responses to NRC staff requests for additional information. The NRC staff also based its review on annual reports of facility operation submitted by the licensee and inspection reports prepared by the NRC staff. The NRC staff conducted site visits to observe facility conditions.

The licensee's application and other materials reviewed by the NRC staff may be examined or copied for a fee at the NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, MD. The NRC maintains the Agencywide Documents Access and Management System (ADAMS), which provides text and image files of the NRC's public documents. Documents related to this license renewal dated on or after November 24, 1999, may be accessed through the NRC's Public Electronic Reading Room on the Internet at http://www.nrc.gov. Those without access to ADAMS who have problems accessing the documents located in ADAMS, or who want to access documents dated before November 24, 1999, may contact the reference staff in the NRC Public Document Room at 1-800-397-4209 or 301-415-4737, or by e-mail to PDR Resources@nrc.gov. Parts of the SAR and the licensee's responses to requests for additional information contain security-related information and are protected from public disclosure. The dates and associated ADAMS accession numbers of the licensee's renewal application and associated supplements are listed in Chapter 7 of this SER.

In conducting its safety review, the NRC staff evaluated the facility against the requirements of the regulations, including 10 CFR Parts 20, 30, 50, 51, 55, 70, 73, and 140; applicable regulatory guides; and relevant accepted industry standards, such as the American National Standards Institute/American Nuclear Society (ANSI/ANS) 15 series. The NRC staff also referred to the guidance contained in NUREG-1537, "Guidelines for Preparing and Reviewing

Applications for the Licensing of Non-Power Reactors," issued February 1996. Because there are no specific accident-related regulations for research reactors, the staff compared calculated dose values for accidents against the requirements in 10 CFR Part 20, "Standards for Protection against Radiation" (i.e., the standards for protecting facility personnel and the public against radiation).

The NRC staff used the focused review process to review the licensee's application for license renewal. In SECY-08-0161, "Review of Research and Test Reactor License Renewal Applications," dated October 24, 2008 (ADAMS Accession No. ML082550140), the NRC staff provided the Commission with information about staff plans to improve the review of license renewal applications for research and test reactors. The Commission issued the staff requirements memorandum (SRM) for SECY-08-0161 on March 26, 2009 (ADAMS Accession No. ML090850159). The SRM directed the staff to streamline the renewal process for research and test reactors, using some combination of the options presented in SECY-08-0161. The focused review process limits review to the most safety-significant aspects of the license renewal application. The SRM directs the NRC staff to implement a graded approach with a scope commensurate with the risk posed by each facility. The graded approach incorporates elements of the alternative safety review approach discussed in Enclosure 1 of SECY-08-0161. In the alternative safety review approach, the NRC staff considers the results of past NRC staff evaluations when determining the scope of the review. A basic requirement, as described in the SRM, is that licensees be in compliance with applicable regulatory requirements.

The NRC developed interim staff guidance (ISG) (ADAMS Accession No. ML092240244) in October 2009 to assist the NRC staff in the review of license renewal applications using a focused license renewal approach. The NRC made a draft of the ISG available for public comment and considered public comments in its development of the final ISG. The NRC staff conducted this review of the RCF using the final ISG.

This safety evaluation report (SER) summarizes the findings of the NRC staff's safety review of the licensee's application and explains the technical details considered in evaluating the radiological safety aspects of continued operation of the RCF. This SER and the environmental assessment and finding of no significant impact, dated June 3, 2011 (ADAMS Accession No. ML102110500), provide the basis for issuance of a renewed license authorizing operation of the RCF at steady-state power levels not to exceed 100 watts (W).

William B. Kennedy, Project Manager, of the NRC's Office of Nuclear Reactor Regulation, Division of Policy and Rulemaking, Research and Test Reactors Licensing Branch, prepared this SER. Other contributors to the safety review include NRC staff members Marcus H. Voth, Spyros Traiforos, Anthony Bowers, Paul V. Doyle, and Jo Ann Simpson.

1.2 Summary and Conclusions on Principal Safety Considerations

On the basis of its safety evaluation, the NRC staff makes the following findings:

- The design and use of the reactor structures, systems, and components important to safety during normal operation, in accordance with the TS, are safe, and safe operation can reasonably be expected to continue.

- The expected consequences of postulated credible accidents and a maximum hypothetical accident (MHA) have been considered. The licensee performed conservative analyses of the most serious credible accidents and the MHA and determined that the calculated potential radiation doses outside the reactor room would not exceed doses in 10 CFR Part 20, for unrestricted areas, and that no fuel damage would occur.

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

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

- The licensee's TS, which provide limits controlling operation of the facility, give reasonable assurance that the facility will be operated safely and reliably. There has been no significant degradation of the reactor since issuance of the original license, and the TS will continue to ensure that there will be no significant degradation of safety-related equipment.

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

- The licensee maintains an emergency plan in compliance with 10 CFR 50.54(q) and Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," to 10 CFR Part 50, "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 reactor operators and the plan for operator requalification are acceptable. These procedures give reasonable assurance that the reactor facility will be operated with competence.

On the basis of these findings, the NRC staff concludes that RPI can continue to operate the RCF in accordance with the Atomic Energy Act of 1954, as amended (AEA), NRC regulations, and Renewed Facility Operating License No. CX-22 without undue risk to public health and safety, facility personnel, or the environment. The issuance of the renewed license will not be inimical to the common defense and security.

1.3 General Facility Description

The RCF is located in Schenectady, NY, adjacent to the southern bank of the Mohawk River. The American Locomotive Company constructed the reactor building and reactor in 1956. RPI assumed operation of the facility in 1964 for the purposes of research and teaching students in the Department of Nuclear Engineering and Science. A stand alone building, constructed primarily of reinforced concrete, houses the reactor, control room, an office, and a small classroom. The reactor room is approximately 9 meters (m) (30 feet (ft)) high and contains a below-grade area that houses the moderator storage tank. The reactor tank is mounted at grade on steel beams that span the below-grade area of the reactor room. The control room is adjacent to the reactor room. Approximately 1 m (3 ft) of concrete separates the control room from the reactor room and provides shielding for facility personnel during reactor operation. A ventilation stack extends above the reactor room to a height of 15 m (50 ft). The ventilation system contains a filter to reduce potential radioactive particulates in the exhaust air.

The reactor uses special power excursion reactor test (SPERT) fuel pins and has a licensed maximum steady-state operating power of 100 W. The reactor core consists of an octagonal lattice of fuel pins, three grid plates, and four boron-enriched control rods. The control rod drive mechanisms contain magnetic clutches that allow the rods to drop into the core by gravity in the event of a power loss or scram signal from the safety system. The core support structure is attached to the bottom of the reactor tank, a 7,600-liter (2,000-gallon) cylindrical stainless steel tank. The reactor uses untreated city water as a neutron moderator, as a coolant, and for radiation shielding. Although the water provides cooling for the fuel, the low reactor power level obviates the need for cooling during normal operation and abnormal conditions. The main functions of the water are neutron moderation and radiation shielding. The reactor tank is equipped with heaters to control the temperature of the moderator. The only penetrations in the reactor tank are a moderator fill line and a fast-dump line in the floor of the tank. The fast-dump line connects to the moderator storage tank and provides a backup reactor shutdown mechanism. There are no experiment facilities external to the reactor tank.

1.4 Shared Facilities and Equipment

The RCF is located in a stand alone building. All building systems are dedicated to supporting operation of the facility. The RCF receives city water and electricity from local distribution networks. These services are not required for safe shutdown of the reactor.

1.5 Compliance with the Nuclear Waste Policy Act of 1982

Section 302(b)(1)(B) of the Nuclear Waste Policy Act of 1982 specifies that the NRC may require, as a precondition to issuing or renewing an operating license for a research or test reactor, that the applicant have entered into an agreement with the U.S. Department of Energy (DOE) for the disposal of high-level radioactive wastes and spent nuclear fuel. In a letter dated May 3, 1983, R.L. Morgan, of DOE, informed H. Denton, of the NRC, that universities and other government agencies operating nonpower reactors had entered into contracts with DOE providing that DOE retains title to the fuel and is obligated to take the spent fuel, or high-level waste, or both, for storage or reprocessing. An e-mail dated May 3, 2010 (ADAMS Accession No. ML101250570), from James Wade of DOE to Paul Doyle (of the NRC) reconfirms this obligation with respect to the fuel at the RCF (DOE Contract No. 78202, valid August 1, 2008-August 31, 2013). By entering into such a contract with DOE, RPI has satisfied the applicable requirements of the Nuclear Waste Policy Act of 1982.

1.6 Facility Modifications and History

The NRC renewed Facility Operating License No. CX-22 on December 2, 1983, extending the license expiration date to December 2, 2003. NUREG-1023, “Safety Evaluation Report Related to the Renewal of the Operating License for the Critical Experiment Facility of the Rensselaer Polytechnic Institute,” issued October 1983, detailed the NRC staff’s safety evaluation of the license renewal. The most significant facility modification since the renewal was conversion of the reactor to use low enriched uranium SPERT fuel. As part of the NRC order modifying the facility operating license, the NRC issued a safety evaluation dated July 7, 1987, that updated NUREG-1023 to reflect the changes in the reactor design and TS necessary for the conversion. Since the conversion, there have been several modifications to the license and TS, including changes to the experiment program, reactor control system, and administrative controls.

1.7 Emergency Planning

The NRC staff reviewed the licensee's emergency plan, dated August 2004, against NUREG-0849, "Standard Review Plan for the Review and Evaluation of Emergency Plans for Research and Test Reactors," issued October 1983; Regulatory Guide 2.6, "Emergency Planning for Research and Test Reactors," Revision 1, issued March 1983; ANSI/ANS-15.16, "Emergency Planning for Research Reactors," issued 1982; and NRC Information Notice 97-34, "Deficiencies in Licensee Submittals Regarding Terminology for Radiological Emergency Action Levels in Accordance with the New Part 20," issued June 1997, and the requirements of 10 CFR 50.34(b)(6)(v). Based on its review, the NRC staff finds that the emergency plan is in accordance with the guidance and regulations. Additionally, the licensee has demonstrated the ability to make changes to the emergency plan in accordance with 10 CFR 50.54(q). Accordingly, the NRC staff concludes that the emergency plan provides reasonable assurance that the licensee can respond appropriately to a variety of emergency situations and that the emergency plan will be adequately maintained during the period of the renewed license.

1.8 Operator Training and Requalification Program

The NRC staff reviewed the licensee's operator training and requalification program, dated May 2011, against all applicable regulations (10 CFR 50.54(i)–(l) and 10 CFR Part 55, "Operators' Licenses") and guidance contained in ANSI/ANS-15.4, "Selection and Training of Personnel for Research Reactors," issued 1988. Based on its review against the guidance and regulations, the NRC staff finds that the program satisfies the regulatory requirements and is in conformance with the guidance. Based on this finding, the NRC staff concludes that the licensee's operator training and requalification program provides reasonable assurance that the licensee will have technically qualified reactor operators and senior reactor operators.

1.9 Financial Considerations

1.9.1 Financial Ability To Operate the Facility

Under 10 CFR 50.33(f)(2), the application to renew or extend the term of any operating license for a nonpower reactor shall include the financial information that is required in an application for an initial license. Accordingly, RPI submitted its projected annual operating costs for the RCF for the 5-year period 2011-2015. Projected operating costs were broken down into labor and other costs (primarily utilities). Total projected costs range from $33,000 in 2011 to approximately $41,000 in 2015. According to the licensee, RPI's Mechanical, Aerospace, and Nuclear Engineering Department budget will provide funds to cover operating costs. The NRC staff reviewed the projected operating costs and projected sources of funds for the RCF and found them to be reasonable.

Based on its review, the staff finds that RPI has demonstrated reasonable assurance of obtaining the funds necessary to cover the estimated costs of operation for the RCF for the period of the license. Accordingly, the NRC staff concludes that RPI has met the financial qualifications requirements pursuant to 10 CFR 50.33(f) and is financially qualified to hold the renewed license for the RCF.

1.9.2 Financial Ability To Decommission the Facility

The regulation at 10 CFR 50.33(k) requires that an application for an operating license for a utilization facility contain information to demonstrate how reasonable assurance will be provided that funds will be available to decommission the facility. The regulation at 10 CFR 50.75(d) requires that each nonpower reactor applicant for or holder of an operating license shall submit a decommissioning report that contains a cost estimate for decommissioning the facility, identification of the funding method(s) to be used to provide funding assurance for decommissioning, and a description of the means of adjusting the cost estimate and associated funding level periodically over the life of the facility.

The licensee stated that the anticipated costs for decommissioning are $444,554 (2010 dollars), with costs broken down by labor, waste disposal, other (equipment and supplies), and a 25-percent contingency factor. According to the licensee, the cost estimate is based upon reasonably conservative assumptions about characterization needs and waste volume. In reviewing the cost estimate, the NRC staff took into consideration experience at other facilities with similar construction and operational history and finds that the decommissioning cost estimate for the RCF is reasonable. According to the licensee, it will review the decommissioning cost estimate annually and will update its cost estimate using the following methodology: labor costs will be adjusted based upon changes in New York State hourly wages for Specialty Trades Contractors as reported by the New York State Department of Labor; waste disposal costs will be based upon current disposal rates at EnviroCare (owned by EnergySolutions as of the date of this SER); and costs for other items will be adjusted by changes in United States Bureau of Labor Statistics Consumer Price Index.

By letter dated August 31, 2010 (ADAMS Accession No. ML102790045), as supplemented on October 14, 2010 (ADAMS Accession No. ML103070074), RPI submitted a request to change its method of providing financial assurance from a bank certificate of deposit to the self-guarantee method, as allowed by 10 CFR 50.75(e)(1)(iii) for non-profit entities such as universities. The NRC staff reviewed the documentation submitted by RPI and, by letter dated December 9, 2010 (ADAMS Accession No. ML103160251), approved RPI's request to use the self-guarantee method for providing financial assurance for decommissioning. The findings of the letter remain valid for license renewal, and are restated below:

> The NRC staff reviewed the self-guarantee agreement and corroborating documentation from RPI to cover the cost of decommissioning the RPI RCF and finds that the self-guarantee agreement meets or exceeds the financial test criteria for a non-profit university that issues bonds, that it is acceptable for providing financial assurance, and that it is in accordance with the provisions of Appendix E to 10 CFR Part 30.
>
> However, pursuant to Section II.C.3 of Appendix E to 10 CFR Part 30, after the initial financial test, RPI, "[m]ust repeat passage of the test within 90 days after the close of each succeeding fiscal year," and under regulation Section II.C.3 of Appendix E to 10 CFR Part 30:
>
>> If the licensee no longer meets the requirements ... the licensee must send notice to the NRC of its intent to establish alternative financial assurance as specified in the NRC regulations. The notice must be sent by certified mail, return receipt requested, within 90 days after the end of the fiscal year for which the year end financial data show that the licensee no longer meets the financial test requirements.

The NRC staff reviewed RPI's information on financial assurance as described above and finds that the decommissioning cost estimate is reasonable; the self-guarantee method for providing financial assurance for decommissioning is acceptable; and RPI's means of adjusting the decommissioning cost estimate periodically over the life of the facility are reasonable. The NRC staff notes that any adjustment of the decommissioning cost estimate must incorporate, among other things, changes in costs due to the availability of disposal facilities.

1.9.3 Foreign Ownership, Control, or Domination

Section 104d of the Atomic Energy Act of 1954, as amended (AEA), prohibits the NRC from issuing a license under Section 104 of the AEA to "any corporation or other entity if the Commission knows or has reason to believe it is owned, controlled, or dominated by an alien, a foreign corporation, or foreign government." The NRC regulation at 10 CFR 50.38, "Ineligibility of Certain Applicants," contains language to implement this prohibition. According to the licensee, RPI is a State of New York corporation principally doing business in New York and is not owned, controlled, or dominated by an alien, foreign corporation, or foreign government. The licensee provided the names, addresses, and citizenship of RPI's trustees and officers, all of whom are U.S. citizens. The NRC staff does not know or have reason to believe otherwise.

1.9.4 Nuclear Indemnity

The NRC staff notes that RPI currently has an indemnity agreement with the Commission, and said agreement does not have a termination date. Therefore, RPI will continue to be a party to the present indemnity agreement following issuance of the renewed license. Pursuant to 10 CFR 140.71, "Scope," RPI, as a nonprofit educational institution licensee, is not required to provide nuclear liability insurance. The Commission will indemnify RPI for any claims arising out of a nuclear incident under the Price-Anderson Act (Section 170 of the AEA) and in accordance with the provisions under its indemnity agreement pursuant to 10 CFR 140.95, for up to $500 million and above $250,000. Also, RPI is not required to purchase property insurance under 10 CFR 50.54(w).

1.9.5 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 RPI RCF and, when necessary, to shut down the facility and carry out decommissioning activities.

2 REACTOR DESCRIPTION

2.1 Summary Description

The RCF is a critical facility, which is a subset of nonpower reactors. Currently, it is the only critical facility licensed by the NRC. The reactor is primarily used for reactor physics and radiation protection experiments and demonstrations. The reactor is licensed to operate at steady-state power levels up to 100 W. According to the licensee, the reactor is usually operated at a power level of less than 1 W. The reactor does not have any pulse capability. The reactor uses SPERT fuel consisting of uranium dioxide pellets clad in stainless steel. The fuel is enriched in uranium-235 to approximately 4.8 percent by weight. The core is contained in a stainless steel tank located in the reactor room. Because of the low power level, the reactor does not require cooling during operation. According to the licensee, the water contained in the reactor tank during operation serves primarily as a neutron moderator and radiation shield.

2.2 Reactor Core

The reactor core is located in a stainless steel tank and is mounted on four posts secured to the bottom of the tank. All structural components are stainless steel. The core has a vertically-tiered structure. The bottom tier is the lower fuel support plate, which rests on a carrier plate mounted on the four core support posts. The lower support plate has holes to position the bottom ends of the fuel pins. The second tier is a middle plate that has holes through which the fuel pins pass. The middle plate provides lateral support for the fuel pins. The third tier is the upper fuel pin lattice plate. The tiered structure is secured by tie rods extending the height of the core. The plates are penetrated by four vertical holes that contain the control rods. The control rods are located at the periphery of the core. The RCF currently has provisions for the use of two core arrangements. One arrangement is an octagonal lattice of fuel pins with no vacant fuel positions inside the core periphery, and the other is a similar arrangement but containing a vacant section of pin positions in the center of the core. Both arrangements are required to meet the operating limits specified in the TS. TS 5.3, “Reactor Core and Fuel,” provides requirements for core design and core component arrangement. These requirements are consistent with the licensee’s analyses of reactor operation and transients.

The TS allow the licensee to modify the core, including the control rod positions and fuel arrangement. These types of modifications would require detailed analyses by the licensee to ensure that all requirements of the TS (e.g., excess reactivity, reactivity coefficients, shutdown margin) are expected to be satisfied for the new configuration. The licensee is required to make measurements of the “unknown core” to verify that the TS requirements are met before undertaking routine operations with the modified core. TS 4.1, “Reactor Core Parameters,” requires the licensee to calibrate the reactor power instrumentation and determine the core excess reactivity, reactivity worth of the most reactive fuel pin, and shutdown reactivity during the initial testing. Testing these core parameters ensures that the core reactivity limits meet the TS requirements before routine operation with the modified core. Successful completion of the measurements allows the licensee to classify the modified core as a “known core,” as described in TS 1.3, “Definitions,” and to operate the reactor using the new core arrangement. TS 3.9, “Facility-Specific Limiting Conditions for Operation,” requires the licensee to load fuel in an “unknown core” using the inverse multiplication to criticality method in order to minimize the possibility of a reactivity accident. TS 6.2, “Review and Audit,” and TS 6.5, “Experiment Review and Approval,” require review and approval of new core configurations by the Nuclear Safety Review Board (NSRB) before implementation. According to the licensee, this process requires evaluation of the new core configuration to ensure that the analyses in the SAR remain valid for the new configuration.

The NRC staff reviewed the SAR and determined that it adequately describes a core design that provides structural support for the core and ensures a stable and reproducible core configuration. The NRC staff evaluated the design and features of the core support structure and found that it is constructed of suitable material, provides for adequate support of the core components, and contains features for reproducible positioning of core components. The NRC staff evaluated the TS requirements and information in the SAR regarding core configuration changes using the guidance in NUREG-1537 and ANSI/ANS-15.1, "The Development of Technical Specifications for Research Reactors," issued 2007. Based on the evaluation, the NRC staff finds that the TS are consistent with the guidance because they require an appropriate safety review, a conservative approach to criticality, and determination of the appropriate core parameters before routine operation. Based on these findings, the NRC staff concludes that the core and core support structure are adequate for the continued safe operation of the RCF.

2.2.1 Reactor Fuel

Each SPERT fuel pin used in the RCF is made up of 60 sintered uranium dioxide pellets encased in a Type 304 stainless steel tube, capped on both ends with a stainless steel cap and held in place with a chromium-nickel spring. An aluminum oxide insulator is installed between the fuel pellets and stainless steel caps on each end of the pin. Gas gaps to accommodate fuel expansion are also provided at both the upper end and around the fuel pellets. TS 5.3 specifies design requirements related to the fuel. These requirements are consistent with the licensee's analyses of normal reactor operation and transients.

As specified in TS 2.1, "Safety Limit—Fuel Pellet Temperature," the licensee chose a conservative safety limit for the fuel of 1,000 degrees Celsius (C) (1,832 degrees Fahrenheit (F)), which is more than a factor of 2 below the melting point for uranium dioxide reported in the literature, and below the melting point of the stainless steel cladding. As discussed in the SAR and Section 2.5 of this SER, the fuel remains at ambient temperature during normal operation. The NRC staff performed an independent calculation of maximum potential fuel temperature assuming adiabatic heating of the fuel pellets for the entire annual reactor operation allowed by TS 3.2.10. The calculation ignored the heat capacity of the cladding and assumed that all fission energy was deposited in the fuel pellets. TS 3.2.10 allows for 2 kilowatt-hours (kW-hr) of operation per consecutive 365 days. Using this limit, the NRC staff's calculation showed that the fuel temperature would increase approximately 100 degrees C (212 degrees F), leaving a large margin to the already-conservative safety limit of 1,000 degrees C (1,832 degrees F). Based on this calculation and the information in the SAR, the NRC staff concludes that the TS requirements for annual operation provide reasonable assurance that heating of the fuel will not pose a hazard to safe use of the fuel during normal operation.

The SPERT fuel was qualified for use in the RCF as part of the conversion of the reactor to use low-enriched uranium fuel. NUREG-1281, "Evaluation of the Qualification of SPERT Fuel for Use in Non-Power Reactors," issued August 1987, detailed the NRC staff review of the SPERT fuel for the operating conditions common to nonpower reactors such as the RCF. The NRC safety evaluation for conversion of the RCF to use the SPERT fuel, issued by NRC order dated July 7, 1987, detailed the NRC staff review of the use of the fuel specifically at the RCF. The NRC staff reviewed the conclusions made in the conversion SER against the information contained in the renewal application. The NRC staff finds that the expected use of the fuel during the period of the renewed license is consistent with the basis for the prior approval to use the fuel. Based on this finding, the NRC staff concludes that the fuel remains acceptable for use in the RCF in accordance with the TS and the license.

TS 5.4, “Fissionable Material Storage,” requires that the fuel be stored in a storage vault with an infinite neutron multiplication factor less than 0.9. The NRC staff evaluated this requirement against the guidance in ANSI/ANS-15.1 and finds it to be consistent and therefore, acceptable. The licensee’s criticality analysis of the storage vault conservatively assumed that the vault was fully flooded with water. The facility is equipped with a fuel vault criticality detection system, as required by TS 3.7.3.

The NRC staff reviewed the RCF safety analysis and fuel qualification documentation and finds that use of the SPERT fuel in the RCF is adequately supported by testing and analysis. The NRC staff finds that the licensee has specified an appropriate safety limit for the fuel temperature that protects the fuel pellets and the cladding from thermal damage. The NRC staff reviewed the TS requirements related to fuel use and storage and finds that they are consistent with applicable guidance documents. Based on these findings, the NRC staff concludes that there is reasonable assurance that the fuel can meet the design objective of maintaining fuel integrity and can thereby function safely in the reactor without undue risk to public health and safety or facility personnel.

2.2.2 Control Rods

Control of the RCF core is provided by four, nonfuel-follower-type control rods specially designed for use with the SPERT fuel core. These rods are spaced 90 degrees apart at the core periphery. Each control rod consists of a square stainless steel tube, which passes through the core and rests on a hydraulic buffer on the bottom carrier plate of the support structure. Housed in each of the tubes are two neutron-absorber sections, one positioned above the other to provide for a total poison section length of approximately 1 m (3 ft). These absorber sections contain boron enriched in the boron-10 isotope in iron. The absorber sections are clad in stainless steel. All absorber sections are of the same dimensions, nominally 6.6 centimeters (cm) (2.6 inches (in.)) square. Because of the symmetry of the RCF cores and the symmetric placement of the control rods, each of the four rods has approximately the same reactivity effect.

The control rods are positioned using the control rod drive systems. According to the licensee, a selector switch allows the operator to move individual rods or combinations of rods. TS 3.2.9 requires interlocks that prevent control rod withdrawal when certain reactor conditions are not met. These interlock requirements are consistent with the analyses in the SAR, and the NRC staff finds them adequate to prevent unanalyzed rod motions. The control rod systems are designed to be failsafe. A reactor scram or a loss of electrical power deenergizes the magnetic clutches in the drive mechanisms, thereby allowing insertion of the rods into the core by gravity. TS 3.2.6 requires a manual scram button to allow the operator to rapidly insert the control rods. TS 3.2.3 specifies that the time from scram initiation to full insertion of any control rod from a fully withdrawn position shall be less than 900 milliseconds. As discussed in Section 4.2 of this SER, this scram time is adequate to terminate reactivity transients to protect the safety limit on fuel temperature. The reactor control console contains position indicators for each rod, including lights that indicate when the rod is at the top or bottom of its range of motion. The console also displays the current supplied to each magnetic clutch.

The designs of the control rod systems allow safe and reliable control of the reactor power level. The SAR presents the licensee's analyses of the requirements for reactivity control systems. The analyses form the bases for the designs of the control rod systems and TS related to reactivity requirements for the systems. TS 3.2.2 requires a minimum of four operable control rods in the reactor core. Reactor shutdown capability is maintained from the most reactive state with the most reactive control rod stuck in the fully withdrawn position. This requirement is specified in TS 3.2.2, which requires the shutdown reactivity provided by the control rods to be at least 0.70 dollars with experiments in their most reactive state and the highest worth control rod fully withdrawn. These TS requirements satisfy the "stuck rod" criterion found in the guidance in NUREG-1537 and ANSI/ANS-15.1. TS 3.1, "Reactor Core Parameters," limits the total core excess reactivity to a maximum of 0.60 dollars. The licensee's analysis of the worst case reactivity transient assumes instantaneous insertion of 0.60 dollars. For this reason, the TS do not contain a limit on the reactivity insertion rate. As discussed in Section 4.2 of this SER, these limits are adequate to provide reasonable assurance that a reactivity accident will not cause fuel damage.

TS 4.2, "Reactor Control and Safety Systems," contains surveillance requirements for the control rod systems. TS 4.2.1 requires semiannual measurement of the scram time for each control rod. TS 4.2.4 requires daily tests of the control rod interlocks. TS 4.2.6 requires that any system be tested for operability after all modifications, maintenance, or repairs. These TS are consistent with the guidance in ANS/ANSI-15.1, and the NRC staff finds them acceptable to adequately monitor the reactivity control systems. TS 5.3 specifies design parameters for the control rod systems, including control rod materials and configuration, that are consistent with the analyses presented in the SAR.

The NRC staff evaluated the reactivity control and scram systems and compared them to those at other nonpower reactors. The analyses presented in the SAR demonstrate that the control rods have sufficient reactivity worth to meet the TS requirements on shutdown margin and provide acceptable control rod dynamic characteristics for both normal and accident conditions. Based on the discussion and findings presented above, the NRC staff concludes that the reactivity control systems and related TS provide reasonable assurance that the reactivity control systems will allow safe and reliable operation and shutdown of the RCF.

2.2.3 Neutron Moderator and Reflector

The RCF uses light water as a neutron moderator and reflector. The municipal water system supplies the moderator water. According to the licensee, the water does not require filtering because the reactor components are stainless steel and the low reactor power does not create a significant radiation hazard from the activation of impurities in the moderator. TS 3.2.5 specifies that the normal moderator level shall not be greater than 25 cm (10 in.) above the top grid of the core. According to the licensee, this limit is based on having sufficient moderator above the core while preventing the flooding of instrument tubes. Additionally, the moderator level limit ensures that the moderator dump adds negative reactivity within 1 minute of activation, as required by TS 3.2.4. TS 4.2.5 requires visual verification of the moderator height before reactor startup. This surveillance ensures that the requirement of TS 3.2.5 is satisfied before reactor operation.

The moderator dump provides a backup shutdown mechanism in case the control rods fail to shut down the reactor. A stainless steel pipe 15 cm (6 in.) in diameter connects the reactor tank to the moderator storage tank. When the reactor scrams, a valve in the pipe automatically opens, dumping the moderator into the storage tank. As mentioned above, TS 3.2.4 requires the moderator dump to add negative reactivity within 1 minute of activation. TS 4.2.2 requires semiannual measurement of the moderator dump time. TS 3.2.8 permits the senior reactor operator on duty to close the moderator dump valve after a scram if the cause of the scram is known, all control rods are fully inserted, and the reactor is decreasing in power. This allows the operator to retain the moderator in the reactor tank for shielding purposes, while ensuring that the backup shutdown mechanism is not required to shut down the reactor.

The NRC staff evaluated the neutron moderator system presented in the SAR against the guidance in NUREG-1537 and systems in place at other research reactors. The NRC staff finds that the neutron moderator used at the RCF demonstrates material compatibility with respect to chemical, thermal, and radiation environment performance. The NRC staff finds that the moderator dump provides an acceptable backup shutdown mechanism and that the TS governing its operation are consistent with the guidance. Based on these findings, the NRC staff concludes that continued operation within the requirements of the TS provides reasonable assurance that the moderator system will perform as necessary and will not adversely affect safe reactor operation or cause an uncontrolled release of radioactive material to the unrestricted environment.

2.2.4 Neutron Startup Source

The facility operating license authorizes RPI to possess a plutonium-beryllium neutron source for use in connection with operation of the reactor. The source is used to provide neutrons for the startup of the reactor. TS 3.2.6 requires an operable startup channel during reactor operation that is interlocked with the control rods. TS 3.2.9 requires that the interlock prevent control rod withdrawal beyond the fully inserted position unless the measured neutron count rate in the core is greater than 2 counts per second. These requirements ensure that there are sufficient neutrons to allow a safe reactor startup before the operator inserts a significant amount of positive reactivity into the reactor core. The source is normally stored in a paraffin shield container. Section 3.1.1 of this SER discusses radiation hazards associated with the neutron source. The source has a drive mechanism that allows the reactor operator to raise and lower the source remotely. The NRC staff reviewed the neutron startup source description and requirements against the guidance in NUREG-1537 and ANSI/ANS-15.1 and the use of neutron sources at other nonpower reactors. The NRC staff finds that source and its operation are comparable to those used at other research reactors and the TS requirements are consistent with the guidance. Based on these findings and the long history of safe operation of the neutron source at the RCF, the NRC staff concludes that continued use of the source is acceptable.

2.3 Reactor Tank

The reactor tank is a cylindrical stainless steel tank that is open on the top. The tank is approximately 2.1 m (7 ft) high and 2.1 m (7 ft) in diameter. The tank has a capacity of 7,600 liters (2,000 gallons). The tank walls are approximately 1 cm (0.4 in.) thick. The only penetrations in the tank are the moderator dump and fill lines in the bottom of the tank. The reactor tank is mounted at grade on steel beams that span the below-grade area of the reactor room. A stairway provides access to a deck that surrounds the reactor tank at the level of the top of the tank. TS 5.2, “Reactor Coolant System,” specifies design requirements for the reactor tank, including capacity and material.

2.4 <u>Biological Shield</u>

The water in the reactor tank, the reactor tank walls, and the reactor building walls provide shielding from direct radiation from the reactor core. The reactor tank contains approximately 0.9 m (3 ft) of water between the edge of the reactor core and the tank walls. TS 3.2.8 permits the senior reactor operator on duty to close the moderator dump valve after a scram to retain the moderator for shielding purposes. The reactor building walls are constructed of poured concrete and have a minimum thickness of 0.3 m (1 ft). The wall separating the control room from the reactor room is 0.9-m (3-ft) thick to provide additional shielding for the reactor operator. The control room is equipped with a gamma-sensitive radiation monitor that would alert the operator to significantly elevated radiation levels in the control room. As discussed in Chapter 3 of this SER, the licensee's records show that personnel exposures have been below detectable levels for many years, and licensee measurements indicate that the shielding is adequate to keep doses at the site boundary within the regulatory limits. Based on the historically-low personnel doses and licensee measurements outside the reactor building, the NRC staff concludes that the biological shielding at the RCF is adequate to protect public health and safety and facility personnel from direct radiation from the reactor.

2.5 <u>Nuclear Design</u>

2.5.1 Normal Operating Conditions

According to the licensee, the RCF is normally operated with "Core A," which has a solid lattice (no vacant fuel positions inside the core periphery) of fuel pins. The licensee may also operate the reactor using "Core B," which is similar to Core A, but contains vacant fuel positions at the center of the core. The Core A lattice is octagonal. Four control rods are located 90 degrees apart at the core periphery. Moderator temperature is normally at the ambient temperature in the reactor room. Electric heaters may be used to raise the water temperature if ambient temperature falls below the limit for minimum moderator temperature of 10 degrees C (50 degrees F) specified by TS 3.1.4. According to the licensee, normal excess reactivity is between 0.10 dollars and 0.35 dollars depending on the core configuration and moderator temperature. Total control rod worth is approximately 2 dollars. The control rod speed is normally about 7.6 cm per minute (3 in. per minute). According to the licensee, this gives a maximum reactivity insertion rate of about 0.003 dollars per second. The licensee also stated that the reactor is usually operated at a steady-state power level of less than 1 W for short periods of time. At this power level, there is no significant heating of the fuel or moderator, and temperature-related reactivity feedback effects are not observed in connection with changing power level. According to the licensee's annual reports submitted to the NRC, average annual operation was about 0.008 kW-hr for the past 10 years.

The NRC staff reviewed the normal operating conditions at the RCF and finds that normal operations are well within all operating limits discussed in Section 2.5.3 and elsewhere in this SER. Based on this finding, the NRC staff concludes there is reasonable assurance that normal operation of the RCF will be conducted in a safe and conservative manner.

2.5.2 Reactor Core Physics Parameters

The neutron lifetime and effective delayed neutron fraction presented in the SAR are 12.2 microseconds and 0.00765, respectively. According to the licensee, the moderator temperature coefficient of reactivity is negative for all temperatures greater than about 33 degrees C (91 degrees F). While the coefficient is positive below this temperature, TS 3.1.2

requires that the integrated reactivity added between ambient (10 degrees C (50 degrees F)) and 38 degrees C (100 degrees F) be no more than 0.15 dollars. TS 3.1.2 also requires the temperature coefficient to be negative above 38 degrees C (100 degrees F). Both cores authorized for use at the RCF satisfy these requirements. According to the licensee, the void coefficient of reactivity is always negative. This is required by TS 3.1.3. The reactivity coefficient requirements must be satisfied for any new core configuration.

The NRC staff reviewed the information in the SAR and the requirements of the TS related to reactor core physics parameters against the guidance in NUREG-1537 and ANSI/ANS-15.1. The NRC staff finds that the licensee appropriately accounted for the reactor physics parameters in the requirements of the TS and the supporting analyses in the SAR. The NRC staff finds that the TS requirements are consistent with the guidance and are therefore, acceptable. Based on these findings, the NRC staff concludes that the core physics parameters support safe operation of the reactor.

2.5.3 Operating Limits

TS 2, "Safety Limits and Limiting Safety System Settings," and TS 3, "Limiting Conditions for Operation," specify the RCF operating limits. Surveillance requirements related to the operating limits appear in TS 4, "Surveillance Requirements." Design requirements are specified in TS 5, "Design Features." The SAR contains additional description and discussion of these limits and requirements. Many of the operating limits and surveillance requirements are evaluated elsewhere in this SER in the section with greatest relevance to the specific limits and requirements. Discussion of design features is included in the relevant sections of this SER and is not discussed in detail in this section.

TS 2.2, "Limiting Safety System Settings," limits the maximum reactor power to 100 W. This is consistent with the maximum power authorized by the facility operating license. TS 2.2 also limits the reactor period to periods longer than 5 seconds. Although a 5-second period may not be manually controllable by the reactor operator given the normal control rod drive speed and control rod reactivity worth, this requirement ensures that the reactor safety system will terminate reactivity transients before the safety limit is exceeded. Section 4.2 of this SER provides a detailed discussion of a worst case reactivity transient. The licensee's transient analysis demonstrates that the limits specified by TS 2.2 are adequate to prevent fuel damage.

TS 3.1 requires that the excess reactivity not exceed 0.60 dollars. This is consistent with the licensee's analysis of the worst case reactivity insertion accident and limits excess reactivity to preclude prompt criticality. As defined in TS 1.3, the limit on excess reactivity appropriately includes the reactivity of all moveable experiments.

TS 3.2, "Reactor Control and Safety Systems," specifies reactivity limits and control rod requirements. These limits and requirements are consistent with the licensee's analysis and descriptions in the SAR, including the assumptions of the reactivity transient analysis. TS 3.2.6, TS 3.2.7, and TS 3.2.9 specify requirements for operability of the reactor safety system during reactor operation and other facility evolutions, such as fuel movement and experiment loading. TS 3.2.6 requires redundant safety channels that monitor reactor power, a safety channel that monitors reactor period, and a safety channel that monitors neutron levels during reactor startup. An interlock prevents control rod withdrawal unless the reactor startup channel detects more than two counts per second. This requirement ensures that the nuclear instrumentation is functioning and sufficient neutrons are present in the core before reactor startup. TS 3.2.10 limits the integrated reactor power for any 365 consecutive days to 2 kW-hr. This limit is fundamental to many of the licensee's analyses concerning radiation protection, potential radiation dose to a maximally exposed member of the public, and radioactive effluents. As discussed in Chapter 3 of this SER, this limit ensures that all offsite doses and effluents are a small fraction of the applicable regulatory limits.

The TS include TS 3.3, "Coolant Systems," TS 3.4, "Containment or Confinement," TS 3.5, "Ventilation Systems," and TS 3.6, "Emergency Power"; however, these TS do not contain any requirements. As previously stated, the reactor does not require cooling because of the low licensed power level and large thermal reservoir of the fuel. This adequately justifies not including TS requirements for a cooling system. TS 5.1, "Site and Facility Description," requires a reactor building and specifies the design features of the building. However, the licensee's analyses do not assume any confinement or containment function associated with the reactor building or ventilation systems. This adequately justifies the omission of TS requirements for confinement, containment, or ventilation. The RCF does not require electrical power to achieve or maintain safe shutdown. The licensee's analyses do not make any assumptions based on the availability of emergency power. This adequately justifies the omission of TS requirements for emergency power.

TS 3.7, "Radiation Monitoring," specifies required radiation monitoring equipment and facility evolutions that require the equipment to be operating or operable and available to the facility personnel. Chapter 3 of this SER discusses these requirements.

TS 3.8, "Experiments," specifies requirements related to the experiment program and individual experiments. These requirements are supported by administrative controls, including review, approval, and written procedures, specified in TS 6.2, TS 6.4, "Procedures," and TS 6.5. Section 4.6 of this SER discusses these requirements.

TS 3.9 contains requirements that the licensee has categorized as unique to the RCF. These include requirements for fuel transfers, familiarity of personnel with radiation protection practices, and fuel loading. These requirements are discussed elsewhere in this SER.

TS 4 specifies surveillance requirements for the operating limits, as appropriate. Requirements not discussed elsewhere in this SER include calibration of the safety system channels, tabulation of integrated reactor power, and circumstances that allow the licensee to waive surveillances. TS 4.3, "Coolant Systems," requires annual calibration of the reactor power and reactor period safety channels. This requirement is consistent with the guidance in ANSI/ANS-15.1 and is acceptable. TS 4.7, "Radiation Monitoring," requires the licensee to tabulate integrated reactor power on a quarterly basis as long as no three consecutive quarters exceed 1.5 kW-hr. This specification provides reasonable assurance that the licensee will remain cognizant of the integrated reactor power and not inadvertently exceed the limit of 2 kW-hr. Given the normal annual operation of the facility of 0.008 kW-hr, the NRC staff finds that quarterly tabulation is conservative and the requirement is acceptable. TS 4.2.8 allows the licensee to waive surveillance requirements on an instrument, component, or system when it is not required to be operable. TS 4.2.8 requires that the surveillance be completed successfully before the instrument, component, or system is declared operable. This TS allows the licensee operational flexibility and avoids unnecessary work without compromising safety. This TS is consistent with the guidance in ANSI/ANS-15.1 and is therefore, acceptable.

The NRC staff reviewed all operating limits and associated surveillance requirements, including those not explicitly mentioned above, against the guidance in NUREG-1537 and ANSI/ANS-15.1 and requirements at other research reactors. The NRC staff finds that the TS requirements meet or exceed the recommendations of the guidance and are comparable to those at other research reactors. The NRC staff also finds that the operating limits are consistent with the licensee's analyses in the SAR. Based on these findings, the NRC staff concludes that the operating limits and related surveillance requirements provide reasonable assurance that the facility can continue to be operated safely during the period of the renewed license.

2.6 Thermal-Hydraulic Design

Because of the low licensed power level of the RCF, no significant thermal-hydraulic considerations are associated with operation of the reactor. The reactor does not have an engineered cooling system, and the reactor fuel and moderator do not significantly change temperature during operation. Some moderator circulation may occur as a result of natural convection in the coolant, but this would not create any appreciable coolant velocity. As previously discussed, the water in the reactor tank is primarily a neutron moderator and radiation shield.

2.7 Conclusions

Based on the review of the information in the SAR and the above findings, the NRC staff concludes that the licensee has adequately described the bases and functions of the reactor design to demonstrate that it can be safely operated and shut down from any operating condition or accident assumed in the safety analysis. The NRC staff concludes that the reactor systems provide adequate control of reactivity, containment of coolant, and barriers to the release of radioactive material, as well as sufficient radiation shielding for the protection of facility personnel and the public. The NRC staff further concludes that the nuclear design and operating limits required by the TS are adequate to ensure fuel integrity and that continued operation of the RCF within the limits of the TS and the facility license will not result in undue risk to public health and safety, facility personnel, or the environment.

3 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

3.1 Radiation Protection

3.1.1 Radiation Sources

The primary radiation source at the RCF is the reactor fuel. During normal operations, the fuel is shielded by the water in the reactor tank and the thick concrete walls of the reactor room. Normally, no personnel enter the reactor room during reactor operation. According to the licensee, the maximum dose rate outside the reactor room during full power operation is 0.27 millisievert per hour (mSv/hr) (27 millirem per hour (mrem/hr)). This dose rate is measured at the personnel door to the reactor room, and facility personnel do not normally occupy this area during reactor operation. The licensee stated that, after reactor shutdown, the highest radiation level is approximately 0.05 mSv/hr (5 mrem/hr) directly over the reactor tank. According to the licensee, the fuel can be safely handled shortly after shutdown. The fuel contains some long-lived fission products; however, contact doses from the fuel are typically less than 0.01 mSv/hr (1 mrem/hr). When not in the core, the fuel is stored in a shielded rack and monitored with an area radiation monitor sensitive to gamma rays that also serves as an inadvertent criticality monitor. This monitor is required by TS 3.7.

Another solid radiation source is the plutonium-beryllium neutron source. The source is stored in a shielded housing in the reactor tank. According to the licensee, the source generates a dose rate of approximately 0.12 mSv/hr (12 mrem/hr) at the point of nearest access. This is reduced by a factor of 4 when the reactor tank is filled with moderator. Other than the fuel, activated core components, and neutron startup source, there are few solid radiation sources at the RCF. According to the licensee, the RCF experiment program consists mostly of reactor physics experiments. The reactor does not irradiate a significant number of samples that are removed from the core and handled by facility personnel. The licensee uses activation foils to perform power measurements and calibrations. According to the licensee, the foils are not highly activated and do not require shielded containers for transfer from the reactor room to the gamma-spectroscopy area in the RCF. The licensee reuses the foils after decay in storage. The NRC staff reviewed the information contained in the SAR against the guidance in NUREG-1537 and finds that the SAR contains sufficient information to provide a reasonable understanding of the solid radiation sources at the RCF.

According to the licensee, small quantities of short-lived isotopes are created in the moderator during reactor operation. Because of their short half-lives and low concentrations, these isotopes do not pose a significant radiation hazard to facility personnel. Sampling of the moderator before discharge has shown that radioactivity in the liquid is consistent with background levels. No other liquid radiation sources exist at the RCF. The NRC staff reviewed the information contained in the SAR against the guidance in NUREG-1537 and finds that the SAR contains sufficient information to provide a reasonable understanding of the liquid radiation sources at the RCF.

According to the licensee, there are no significant sources of airborne radioactivity at the RCF during routine operation. Reactor operation may produce some argon-41 in the reactor room. As discussed in NUREG-1023, infinite operation of the reactor at 100 W would not generate argon-41 concentrations greater than the occupational or effluent concentration limits specified in Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage," to 10 CFR Part 20, "Standards for Protection against Radiation." The NRC staff reviewed the basis for this evaluation and determined that there have been no changes in facility operation that would invalidate the results. TS 3.2.10 limits annual operation to 2 kW-hr, or about a factor of 400 less than the infinite operation assumption in NUREG-1023. This ensures that the licensee satisfies the dose constraint of 0.1 millisievert (mSv) (10 millirem (mrem)) for air emissions of radioactive material specified in 10 CFR 20.1101(d). Based on the annual average power generation of 0.008 kW-hr per year reported by the licensee in the last 10 annual reports to the NRC, the argon-41 concentration would be approximately five orders of magnitude below the regulatory limit for air effluents concentrations specified in Table 2 of Appendix B to 10 CFR Part 20. This is consistent with the licensee's statement that there are no significant sources of airborne radioactivity at the RCF during routine operation. The NRC staff reviewed the information contained in the SAR against the guidance in NUREG-1537 and finds that the SAR contains sufficient information to provide a reasonable understanding of the airborne radiation sources at the RCF.

Based on the above discussion and findings, the NRC staff concludes that the description and characterization of the radiation sources at the RCF are reasonable for a research reactor of this type and power level and that this information provides sufficient detail to evaluate the radiation protection program and controls described in the remainder of this chapter of the SER.

3.1.2 Radiation Protection Program

Section 11.1.2 of the SAR describes the radiation protection program required by 10 CFR 20.1101, "Radiation Protection Programs." According to the licensee, the primary purpose of the radiation protection program is to "assure radiological safety for all University personnel and the surrounding community." This program includes the stated policy to employ the ALARA concept in all operations at the RCF. TS 6.3, "Radiation Safety," requires the Radiation and Nuclear Safety Committee (RNSC) and Radiation Safety Officer (RSO) to implement the radiation protection program for the RCF. According to the licensee, the RSO conducts routine contamination surveys, maintains and monitors personnel exposure records, and trains the RCF personnel to conduct routine radiation protection activities, such as radiation surveys. All facility personnel receive initial training and annual refresher training from the RSO. The RSO also provides advice to the Facility Director and Operations Supervisor on matters of radiation safety. TS 6.1, "Organization," requires the RSO to be organizationally independent of the reactor operations group and to have interdiction responsibility and authority. These requirements are consistent with the guidance in ANS/ANSI-15.1 and provide reasonable assurance that radiation safety decisions are independent of operational considerations.

TS 6.9, "Operating Records," requires the licensee to retain records relating to personnel dosimetry and effluents for the life of the facility, consistent with 10 CFR 20.2106, "Records of Individual Monitoring Results," and 10 CFR 20.2107, "Records of Dose to Individual Members of the Public." TS 6.9 also requires the licensee to retain facility surveys for a minimum of 5 years. This exceeds the requirements of 10 CFR 20.2103, "Records of Surveys," which only requires the licensee to retain these records for 3 years; therefore, the NRC staff finds the requirement to be acceptable.

The NRC staff reviewed the structure and strategy of the radiation protection program for the RCF and finds it to be consistent with the guidance in NUREG-1537 and ANSI/ANS-15.11, "Radiation Protection at Research Reactor Facilities," issued 2009, and the applicable regulatory requirements. Based on this finding, the NRC staff concludes that the RPI radiation protection program provides reasonable assurance that facility operation will not pose a significant risk to facility personnel or members of the public and that radioactive materials will be handled safely.

3.1.3 ALARA Program

RPI has a defined ALARA policy for exposure to radiation, established by the university Provost and implemented by the RNSC and RSO. The policy states that operations are to be conducted in a manner to maintain all radiation exposure consistent with the ALARA principle. TS 6.3 requires the RCF to have an ALARA program. The main tenet of the policy is that individual exposures remain well within applicable regulatory limits. As required by TS 6.3, management at all levels, as well as each individual worker, must take an active role in minimizing radiation exposure. The RNSC and the RSO are responsible for implementation of the ALARA program. Furthermore, the RNSC and the RSO routinely review personnel exposures at the RCF to ensure that they are maintained ALARA. As discussed in Section 3.1.5 of this SER, doses to facility personnel are usually below the detectable threshold. The NRC staff reviewed the information in the SAR and the TS requirements concerning the RPI ALARA program against the guidance in NUREG-1537 and ANSI/ANS-15.11 and finds that the program is consistent with the guidance. Based on this finding and the historically low radiation doses to facility personnel, the NRC staff concludes that the ALARA program provides reasonable assurance that the licensee will minimize personnel exposure to radiation.

3.1.4 Radiation Monitoring and Surveying

Fixed radiation monitoring at the RCF consists of a gamma area monitoring system. The gamma monitoring system consists of four monitors that use Geiger-Mueller detectors. The monitors are located in the control room, equipment hall, fuel storage area, and near the top of the reactor tank. The fixed radiation monitoring system also functions as an inadvertent criticality monitoring system. Based on the expected radiation hazards at the facility described in the SAR, the NRC staff finds that the monitor types and monitoring locations are acceptable. According to the licensee, all monitors have visual and audible alarms and display in the control room. The RCF is also equipped with a continuous air monitor that draws air from near the top of the reactor tank. This monitor has local visual and audible alarms. The RCF radiation monitoring system is comparable to those used in other low-power research reactors. In addition to the fixed radiation monitoring system, the licensee maintains a portable survey meter and a thin-window Geiger-Mueller detector for additional monitoring during fuel or experiment loading or unloading.

TS 3.7.1 and TS 3.7.4 require the fixed radiation monitoring system and continuous air monitor to be operating during reactor operation. TS 3.7.1(b) allows temporary replacement of an area radiation monitor with an equivalent portable unit. TS 3.7.2 requires the fixed radiation monitoring system to be operating and the portable instruments to be available during fuel and experiment loading and unloading. TS 3.9 requires facility personnel to monitor all operations and evolutions with appropriate radiation instrumentation. The NRC staff reviewed these requirements against the guidance in NUREG-1537 and ANSI/ANS-15.1 and finds them to be consistent with the guidance and therefore acceptable. TS 4.7 requires daily to monthly channel tests and semiannual calibrations of the area radiation monitoring system. TS 4.7 also requires annual calibration of the portable radiation monitoring equipment. The NRC staff reviewed these surveillance requirements against the guidance in ANSI/ANS-15.1 and finds them to be consistent with the guidance and therefore, acceptable.

As discussed in Section 3.1.6 of this SER, the licensee conducts monthly contamination surveys of the facility. These surveys typically do not detect activity above background levels. According to the licensee, radiation surveys inside and outside the reactor building while the reactor is operating constitute an experiment performed at the RCF. Section 3.1.5 of this SER discusses personnel monitoring, and Section 3.1.7 of this SER covers environmental monitoring.

Based on its review, the NRC staff concludes that the installed and available radiation detection and air monitoring equipment is of the proper type and located appropriately to detect and quantify expected radiation at the RCF. Furthermore, the NRC staff concludes that the programs to use and maintain the equipment, as well as the frequency of surveys, satisfy the requirements of 10 CFR 20.1501(a) and (b) and provide reasonable assurance that doses to personnel will be kept below the limits specified in 10 CFR 20.1201, "Occupational Dose Limits for Adults."

3.1.5 Radiation Exposure Control and Dosimetry

The RCF is located in a fenced area with limited access. This meets the definition of a controlled area given in 10 CFR 20.1003, "Definitions." This fence is normally locked and access controlled by RCF personnel. Access to the building requires training appropriate to the expected level of access to radioactive materials. During operation, access to the reactor room is restricted, except for radiation surveys and maintenance checks authorized by the Operations Supervisor. Normally, opening the door to the reactor room causes a scram, which prevents personnel from inadvertently entering the room with the reactor operating. TS 3.2.6 requires the scram function to be operable during reactor operation, except for temporary bypass to allow access for radiation surveys or maintenance checks. During shutdown, radiation levels in the facility decrease significantly. As discussed in Section 11.1 of the SAR, the licensee uses administrative controls to maintain personnel doses ALARA.

According to the SAR, all personnel entering the areas where radiation and radioactive material could be present use individual dosimeters. The RCF dosimeter processor is certified by the National Voluntary Laboratory Accreditation Program, as 10 CFR 20.1501(c) requires. According to the licensee, personnel dosimeters consistently show no detectable radiation dose. This assertion is consistent with the low annual average power generation and general absence of radiation sources other than the fuel. The measured personnel doses demonstrate compliance with the annual occupational dose limit of 50 mSv (5,000 mrem) specified in 10 CFR 20.1201.

Based on the above discussion and review against the applicable regulatory requirements, the NRC staff concludes that the exposure control and dosimetry program provides reasonable assurance that the licensee will adequately monitor personnel and control radiation exposures to facility personnel below the limits in 10 CFR Part 20, Subpart C, "Occupational Dose Limits."

3.1.6 Contamination Control

Typically, contamination at research reactors results from the experiment program, maintenance activities, or spills from reactor systems. According to the licensee, the RCF experiment program focuses on reactor physics and radiation protection experiments to support a laboratory class and demonstrations for visiting students. Aside from the activation foils, the RCF does not frequently irradiate samples that are removed from the reactor. Because of the low reactor power, the foils are not highly activated and do not pose a risk of contamination.

Based on the nature of the experiment program as described in the SAR, the NRC staff finds that it poses no significant risk of contamination of the facility. Because of the low reactor power, the reactor core components and reactor equipment, other than the fuel, are not significantly activated during routine operation. Similarly, only minimal activation of impurities in the reactor moderator occurs during operation. Based on these considerations, the NRC staff finds that maintenance activities and spills from reactor systems pose no significant risk of facility contamination.

As discussed in the SAR, the licensee conducts a monthly contamination survey of the facility. According to the licensee, the survey routinely shows no detectable contamination. TS 3.7.2 requires a portable survey instrument capable of detecting contamination to be available during in-core loading and unloading of experiments and fuel. TS 3.9 requires facility personnel familiar with monitoring techniques to monitor all facility operations, including in-core loading and unloading of experiments and fuel, with appropriate radiation and contamination monitoring equipment. TS 6.7, "Required Actions in the Event of a Reportable Occurrence," requires the licensee to report the discovery of loose surface contamination to the Facility Director, the NSRB, and the NRC. Further, the NSRB is required to review and concur on all corrective actions taken to prevent recurrence of the contamination. These TS requirements are more conservative than or consistent with the guidance in ANSI/ANS-15.1, and the NRC staff finds them to be acceptable. TS 6.9 requires the retention of contamination survey records for 5 years. This exceeds the 3-year retention period required by 10 CFR 20.2103.

Based on the lack of significant risks of contamination, the historic lack of detectable contamination, and the conservative TS requirements, the NRC finds that the contamination survey and control program is acceptable. Based on this finding, the NRC staff concludes that the licensee's program for contamination control provides reasonable assurance that contamination at the facility will not pose a significant risk to public health and safety, facility personnel, or the environment.

3.1.7 Environmental Monitoring

The licensee conducts an environmental monitoring program to measure radiation dose at locations around the RCF. Dose measurements are made quarterly using thermoluminescent dosimeters (TLDs). The monitoring program consists of four measurements at the exclusion area boundary and two measurements at the site boundary. An additional measurement for control purposes is taken approximately 1.6 kilometer (1 mile) away. This type of environmental monitoring program is typical for a low-power research reactor. The RCF does not discharge significant amounts of airborne radioactive effluents, and the primary source of offsite radiation from the facility is direct radiation from the reactor fuel. Based on the lack of significant quantities of airborne effluents, the NRC staff finds that the licensee's TLD-based environmental monitoring program is acceptable to determine the potential radiological impact of RCF operation on the surrounding environment. As discussed in Section 3.1.4 of this SER, the licensee maintains a continuous air monitor to assess airborne radiation levels in the reactor room during reactor operation.

During the annual reporting periods from January 1, 2000, to December 31, 2009, measured doses at the site boundary were below the level detectable by the TLDs. According to the licensee, the minimum detectable level is 0.1 mSv (10 mrem) and the measurement frequency is quarterly. Conservatively assuming that the actual doses were equal to the minimum detectable doses, the annual dose to a member of the public would be 0.4 millisievert per year (mSv/yr) (40 millirem per year (mrem/yr)). This demonstrates compliance with the annual dose limit for a member of the public of 1 mSv/yr (100 mrem/yr) set by 10 CFR 20.1301(a)(1). Based on the annual average power generation of 8 watt-hours per year reported by the licensee, the expected annual dose would be less than 0.01 mSv/yr (1 mrem/yr). This is consistent with the TLD measurements being below the detectable level. TS 6.9 requires retention of records of the TLD monitoring results for the life of the facility and this satisfies the records retention requirement of 10 CFR 20.2107.

The licensee performed measurements of radiation levels at the point of nearest public access with the reactor operating at 13 W. The licensee extrapolated the results to operation at 100 W and determined a maximum dose rate of 0.013 mSv/hr (1.3 mrem/hr). This dose rate demonstrates compliance with the dose rate limit of 0.02 mSv/hr (2 mrem/hr) for members of the public specified in 10 CFR 20.1301(a)(2). TS 3.2.10 limits annual reactor operation to 2 kW-hr per year. This requirement limits the maximum annual dose to a maximally exposed member of the public to 0.26 mSv (26 mrem). This demonstrates compliance with the dose limit of 10 CFR 20.1301(a)(1). The licensee's methods for demonstrating compliance with 10 CFR 20.1301, "Dose Limits for Individual Members of the Public," meet the requirements of 10 CFR 20.1302, "Compliance with Dose Limits for Individual Members of the Public."

The NRC staff reviewed the environmental monitoring program against the guidance in NUREG-1537, ANSI/ANS-15.11, and the applicable requirements of 10 CFR Part 20. The NRC staff finds that the monitoring program is consistent with the guidance and satisfies the applicable regulatory requirements. Based on this finding, the NRC staff concludes that the licensee's environmental monitoring program provides reasonable assurance that radiation from the facility will be detected, recorded, and reported in a manner that protects public health and safety and the environment.

3.2 Radioactive Waste Management

According to the licensee, routine operation of the RCF generates no radioactive waste that requires periodic disposal. The experiment program generates no significant quantities of radioactive material, other than fission products in the reactor fuel and some activation products in the core support structure. The activation foils used for power calibrations are not considered radioactive waste because they are allowed to decay in storage and then reused. As discussed in Section 3.1.1 of this SER, the RCF generates insignificant quantities of gaseous and liquid radioactive materials. TS 4.7 requires sampling of the moderator before discharge. This ensures that moderator discharges will meet the regulatory requirements in Appendix B to 10 CFR Part 20. Because of the limit on annual reactor operation of 2 kW-hr per year specified by TS 3.2.10, the RCF will not need to ship spent fuel or receive new fuel during the period of the renewed license. As discussed in Section 1.5 of this SER, the reactor fuel will be returned to DOE in accordance with the Nuclear Waste Policy Act of 1982 upon eventual decommissioning of the facility. Based on these considerations, the NRC staff concludes that radioactive waste at the RCF poses no significant risk to public health and safety, facility personnel, or the environment.

3.3 Conclusions

The NRC staff concludes that the RCF radiation protection and ALARA programs, radiation monitoring and surveying, and exposure control and dosimetry provide reasonable assurance that doses to facility personnel will be maintained below the regulatory limit and ALARA. The NRC staff concludes that the licensee's environmental monitoring and radioactive waste management programs provide reasonable assurance that doses to members of the public will be kept below the regulatory limit and ALARA. Additionally, the staff concludes that the licensee's radioactive waste management program provides reasonable assurance that radioactive wastes will be handled and disposed of in accordance with applicable regulations and should not have a significant impact on the environment.

4 ACCIDENT ANALYSES

4.1 Maximum Hypothetical Accident

The accident scenario for the RCF with the greatest potential radiological consequences is the failure of an experiment that contains radioactive material. TS 3.8.7 authorizes the licensee to conduct encapsulated experiments that contain radioactive material. TS 3.8.10 limits the amount of radioactive material in experiments such that a complete release would not result in a concentration in the reactor room greater than two times the limits in 10 CFR Part 20, Appendix B, Table 2 for air effluents. As the basis for this limit, the licensee analyzed the radiological consequences of an experiment failure and demonstrated that radiation doses to facility personnel and members of the general public would be below the applicable regulatory limits. According to the licensee, experiments that contain significant amounts of radioactive material that could become airborne in the event of an experiment failure are rarely, if ever, performed at the RCF.

The licensee made the following assumptions about the experiment failure; the NRC staff evaluation of each assumption follows the description of the assumption:

- The licensee assumed that an experiment fails, releasing all gaseous, volatile, and particulate activity to the reactor room air, and that the material instantaneously mixes with the air in the reactor room. The assumption of a complete release of all gaseous, volatile, and particulate activity to the reactor room air is consistent with Regulatory Guide 2.2, "Development of Technical Specifications for Experiments in Research Reactors," and is therefore, acceptable. The assumption of instantaneous mixing is conservative because it does not account for radioactive decay during the finite time it would take for the material to escape the experiment encapsulation and mix with the air in the reactor room. Based on this conservatism, the NRC staff finds this assumption to be acceptable.

- The licensee assumed no atmospheric dispersion during the release period. This assumption is conservative because it ignores the reduction in the effluent concentration due to dispersion from an elevated release or building wake effects from a ground level release. The assumption is also conservative because it ignores natural variations in wind speed and direction that would occur over the period of the release. Given the licensee's assumed release rate (discussed below), the release would last for a period of months, allowing adequate time for significant variations in wind speed and direction. Based on the conservatism of ignoring atmospheric dispersion, the NRC staff finds this assumption to be acceptable.

- The licensee assumed that the airborne radioactive material escapes the reactor building at a rate equal to the breathing rate for "Reference Man" under conditions of "light work" of 1.2 cubic meters per hour (m^3/hr), as specified in Appendix B to 10 CFR Part 20. According to the licensee, this assumption is limiting because (1) a higher release rate would decrease the average release concentration, and (2) a lower release rate necessitates dilution to increase the volume of the released material to match the breathing rate. The NRC staff agrees that a release rate equal to the breathing rate is limiting when combined with the other assumptions discussed above and is therefore acceptable. However, the NRC staff notes that the breathing rate chosen by the licensee is for conditions of "light work," and not for members of the general public. According to Appendix B to 10 CFR Part 20, the breathing rate for members of the general public is approximately 0.8 m^3/hr.

The licensee analyzed the experiment failure to determine the potential radiation dose to the maximally-exposed member of the public. In calculating the potential dose to a maximally-exposed member of the public, the licensee assumed that the airborne radioactive material escapes the reactor building at a rate of 1.2 m^3/hr. The release rate is driven by the natural chimney effect of the building. According to the licensee, a release rate of 1.2 m^3/hr would require a temperature differential between the building interior and the outside atmosphere of approximately 0.01 degrees Kelvin. Higher temperature differentials would generate higher release rates, and decrease the accident consequences as discussed above. Using the release rate of 1.2 m^3/hr and the building volume of 1,019 cubic meters, the licensee calculated the average release concentration over the period of 1 year to be 0.1 times the initial concentration in the reactor building. Given that TS 3.8.10 limits the concentration in the reactor room to 2 times the air effluent limit, the licensee calculated that the accident would result in a maximum potential radiation dose of 0.1 mSv (10 mrem), which is below the limit of 1 mSv (100 mrem) specified in 10 CFR 20.1301. The TS requirement also ensures that radiation exposure to facility personnel will not exceed 1 mSv (100 mrem), which is below the annual limit of 50 mSv (5000 mrem) specified in 10 CFR 20.1201.

The NRC staff performed check calculations of the licensee's analysis and independent calculations to confirm the conservatism of the analysis. The NRC staff used a lower breathing rate in the independent calculations to account for the difference between the breathing rate for "Reference Man" under conditions of "light work" and the breathing rate for a member of the general public specified in Appendix B to 10 CFR Part 20. Although the lower breathing rate results in a slightly higher calculated potential radiation dose of 0.14 mSv (14 mrem), the result is still a small fraction of the regulatory limit of 1 mSv (100 mrem) specified in 10 CFR 20.1301. Based on the independent calculations, the NRC staff finds that the conservatism in the assumption of no atmospheric dispersion would far outweigh the nonconservatism introduced by the breathing rate. Given that the nearest permanent residence is several hundred meters from the facility and the release lasts for months, the NRC staff used Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," to estimate that atmospheric dispersion would reduce the potential radiation dose by at least a factor of 10. The NRC staff did not perform detailed calculations of a realistic dispersion coefficient for the RCF site taking into account changes in wind speed and direction because the conservatively-estimated dose reduction factor of 10 is adequate to show that the potential off-site radiation dose is a small fraction of the regulatory limit.

The NRC staff reviewed the licensee's analysis of the MHA against the guidance in NUREG-1537 and the applicable requirements of 10 CFR Part 20. Based on its review against the guidance in NUREG-1537, the NRC staff finds that the licensee made conservative assumptions (with one exception) and used appropriate methods for calculating the maximum potential radiation exposure to a member of the public. Based on its review against the requirements of 10 CFR Part 20, the NRC staff finds that the licensee's analysis demonstrates compliance with the radiation dose limits to members of the public. The NRC staff also finds that the limit on airborne radioactive material specified in TS 3.8.10 demonstrates compliance with the annual dose limit for facility personnel. Based on these findings, the NRC staff concludes that the MHA will not pose an undue risk to public health and safety, facility personnel, or the environment.

4.2 Insertion of Excess Reactivity

Chapter 13 of the SAR describes a reactivity insertion accident resulting from an unsecured experiment malfunction. The licensee made the following assumptions about the transient; the NRC staff evaluation of each assumption follows the description of the assumption:

- The licensee hypothesized that failure of an unsecured experiment caused a step insertion of 0.60 dollars of positive reactivity into the core. This assumption is consistent with the reactivity limit of 0.60 dollars for moveable or unsecured experiments specified by TS 3.8.6. This assumption is more conservative than the reactivity insertion rate limits specified by TS 3.8.5. This assumption is also more conservative than the reactivity limit of 0.20 dollars for a single fuel pin specified by TS 3.2.1. This assumption is consistent with the total core excess reactivity limit of 0.60 dollars specified by TS 3.1. Because the licensee assumed a reactivity insertion that bounds the reactivity limits specified in the TS, the NRC staff finds the assumption to be acceptable.

- The licensee assumed that the reactor was operating at 200 W, which includes a factor of 2 for cumulative uncertainties associated with instrumentation calibration. This assumption is consistent with the limiting safety system setting for reactor power of 100 W specified by TS 2.2. According to the licensee, the calibration methods for the nuclear instrumentation are accurate to within 30 percent, and instrument response is accurate to within 5 percent. Therefore, the assumed 100-percent uncertainty is conservative. Because the licensee assumed a reactor power level that is consistent with the requirements of the TS and the license and conservatively accounts for uncertainty, the NRC staff finds the assumption to be acceptable.

- The licensee assumed failure of the ion chamber signal to safety channel PP2. TS 3.2.6 requires the channel to be operable during reactor operation. TS 2.2 requires this channel to scram the reactor at a reactor period of 5 seconds or greater or a reactor power level of 100 W or less. This assumption is the most conservative assumption for safety channel failure because safety channel PP2 provides the only short-period scram and has the lowest setpoint of the three reactor power scram channels. Based on this conservatism, the NRC staff finds this assumption to be acceptable.

- The licensee assumed that one of the two remaining high-power scram channels required by TS 3.2.6 would cause a scram at a reactor power level of 1,800 W. According to the licensee, the scram occurs at 1,800 W because the channels trip at 90 percent of the range scale, and 200 W corresponds to 10 percent of the highest range scale for the channels. Because the licensee assumed reactor safety system operation that is consistent with the requirements of the TS, the NRC staff finds the assumption to be acceptable.

- The licensee assumed a scram time of 1.5 seconds for the control rods to fully insert from the fully withdrawn position. TS 3.2.3 requires the scram time to be less than or equal to 900 milliseconds. Because the licensee assumed a scram time that is more conservative than the TS requirement, the NRC staff finds this assumption to be acceptable.

- The licensee assumed that the control rods instantaneously insert 1 dollar of negative reactivity at the end of the scram time. This assumption is consistent with the minimum shutdown reactivity requirement of 1 dollar specified by TS 3.2.2 for four operable control rods. This assumption is conservative because, realistically, the control rods would insert reactivity over the entire length of travel according to the integral worth of the rods. Because the licensee assumed shutdown reactivity that is consistent with the requirements of the TS and control rod reactivity insertion that is conservative compared to realistic reactor behavior, the NRC staff finds the assumption to be acceptable.

According to the licensee, the reactivity insertion results in a prompt increase in reactor power to 600 W, followed by a steady increase in power to 1,800 W on a period of 3 seconds. At 1,800 W, the safety system generates a scram signal. The power continues to rise on a 3-second period until 1.5 seconds later, when the control rods have inserted 1 dollar of negative reactivity. As stated in Section 13.2 of the SAR, the peak power during the transient is 3,050 W and the total energy deposited in the core is approximately 10 kilojoules. According to the licensee, the transient results in a fuel temperature rise of less than 0.1 degrees C (0.2 degrees F). For this reason, the licensee ignored temperature and void reactivity feedback effects in the analysis. The NRC staff agrees that, given the minimal increase in temperature, reactivity feedback effects would have no significant impact on the transient analysis. The licensee conducted the analysis using the FRKGB computer code, which was benchmarked against Gaussian, Nordheim-Fuchs, and SPERT-type power bursts.

The NRC staff performed independent calculations of the transient using the core characteristics provided in Table 13.1 of the SAR and simplified analytical methods. The licensee provided justification for the core characteristics in the SAR, and based on a review against the guidance in NUREG-1537, the NRC staff finds them to be reasonable for the reactor design. The NRC staff's calculations resulted in values similar to those presented by the licensee for reactor period, peak reactor power, total energy released, and fuel temperature increase during the transient. Based on the similarity of the results of the independent calculations, the NRC staff finds the results of the licensee's calculations to be reasonable. The NRC staff also finds that the fuel temperature will remain a small fraction of the fuel temperature safety limit of 1,000 degrees C (1,832 degrees F) specified by TS 2.1 and thus not result in the uncontrolled release of radioactive material from fuel failure.

Based on the findings discussed above, the NRC staff concludes that operation in accordance with the requirements of the TS provides reasonable assurance that the worst case reactivity transient will not pose an undue risk to public health and safety, facility personnel, or the environment.

4.3 Loss of Moderator

The RCF reactor tank is a stainless steel tank approximately 1 cm (0.4 in.) thick. Other than the connections for the moderator dump and fill lines in the tank floor, the reactor tank has no penetrations. According to the licensee, a controlled loss of moderator from the reactor tank (through the moderator fast dump line) is a backup reactor shutdown mechanism. As discussed in Chapter 2 of this SER, the purpose of the water in the core tank is to provide neutron moderation and radiation shielding, not to cool the fuel. The reactor does not require any cooling during operation and therefore, does not require any cooling during shutdown when the heat load is a fraction of the operating heat load.

According to the licensee, the loss of moderator does not result in an accident situation at the RCF. In the case of an uncontrolled loss of moderator, the safety concerns are increased radiation levels on and off site and the potential release of radioactive materials in the reactor moderator. As discussed in Chapter 3 of this SER, the moderator does not contain significant amounts of radioactive material. According to the licensee, sampling of the moderator has never shown levels of radioactivity above the detectable limit of a few disintegrations per minute per liter of moderator. Therefore, a loss of moderator should not cause significant radioactive contamination of the facility or the environment.

The loss of the water shielding above and around the core would increase the radiation levels in the reactor room and control room. In the event of a loss of moderator, the reactor would shut down automatically. According to the licensee, the highest radiation levels in the reactor room after shutdown are measured by a radiation detector located approximately 1.5 m (5 ft) above the top of the reactor. Approximately 25 cm (10 in.) of water shielding normally exists between the top of the reactor core and the radiation detector. The licensee stated that the dose rate at this location decreases from 0.05 mSv/hr (5 mrem/hr) shortly after shutdown to 0.003 mSv/hr (0.3 mrem/hr) well past shutdown. The licensee stated that 1 m (3 ft) of reinforced concrete provides shielding for the control room from the reactor. This thickness of concrete provides more gamma ray attenuation than the water shielding above the core. Accordingly, the NRC staff finds that the dose rate in the control room after a loss of moderator would be a fraction of the 0.05 mSv/hr (5 mrem/hr) measured above the reactor core. As discussed in Section 3.1.4 of this SER, TS 3.7 requires an area radiation monitor in the control room. This monitor would alert facility personnel to abnormally high radiation levels and allow the licensee to take appropriate actions in accordance with the facility emergency plan. Based on these findings, the NRC staff concludes that the RCF design features and requirements of TS 3.7 provide reasonable assurance that a loss of moderator will not pose an undue risk to facility personnel.

In the event of a loss of moderator, the loss of the water shielding around the core would increase the radiation levels at the restricted area boundary and the exclusion area boundary. According to the licensee, the reinforced concrete exterior walls of the reactor room have a minimum thickness of approximately 0.3 m (1 ft). TS 5.1 specifies a minimum distance of 9 m (30 ft) from the reactor to the restricted area boundary and 15 m (50 ft) from the reactor to the exclusion area boundary. The NRC staff performed independent calculations to estimate the potential radiological consequences for a member of the public at the restricted area and exclusion area boundaries. The NRC staff used differences in shield material, shield thickness, and source-to-receptor distance to obtain dose reduction factors of approximately 50 at the restricted area boundary and 150 at the exclusion area boundary. Applying the dose reduction factor of 50 to the licensee's measured dose rate shortly after shutdown yields a dose rate at the restricted area boundary of 0.001 mSv/hr (0.1 mrem/hr). The NRC staff finds that this dose rate demonstrates compliance with the limit of 0.02 mSv/hr (2 mrem/hr) specified in 10 CFR 20.1301(a)(2) for members of the public in unrestricted areas. Applying the dose reduction factor of 150 to the licensee's measurements yields a dose rate of 0.0003 mSv/hr (0.03 mrem/hr) shortly after shutdown and 0.00002 mSv/hr (0.002 mrem/hr) well past shutdown at the exclusion area boundary. In the case that the licensee did not take action to restore the water shielding, the dose rate would continue to decrease as the fission products decayed. The NRC staff estimated that the total dose over the course of a year at the exclusion area boundary would be less than 0.02 mSv (2 mrem). The NRC staff finds that this dose demonstrates compliance with the annual dose limit of 1 mSv (100 mrem) specified in 10 CFR 20.1301(a)(1) for individual members of the public.

Based on the discussion and findings presented above, the NRC staff concludes that a loss of moderator would not pose a significant risk of an uncontrolled release of radioactive material to the environment nor excessive radiation levels in the facility and area surrounding the reactor site. The NRC staff further concludes that a loss of moderator at the RCF poses no undue risk to public health and safety, facility personnel, or the environment.

4.4 Loss of Coolant Flow

The RCF does not have an engineered cooling system. The TS do not require coolant flow during operation of the RCF. Therefore, this accident does not apply to the RCF.

4.5 Mishandling or Malfunction of Fuel

The licensee did not perform a detailed analysis of a release of fission products from a damaged fuel element because normal operation does not generate significant quantities of fission products and there are no credible mechanisms for fuel damage. According to the licensee, the fission product inventory in a fuel element is insufficient to cause a significant off-site hazard. The NRC staff evaluated a fission product release as part of its safety evaluation dated July 7, 1987, supporting the order to convert the RCF to use low enriched uranium fuel. In its safety evaluation, the NRC staff used assumptions that are more conservative than the operating limits in the TS. The NRC staff found that the radiological consequences of the failure of the cladding of several fuel pins would be a small fraction of the limits for members of the public specified in 10 CFR Part 20. As part of this license renewal review, the NRC staff performed independent calculations of a fission product release. The NRC staff confirmed the finding reached in the safety evaluation supporting the conversion order. The NRC staff also confirmed that the assumptions used in the conversion safety evaluation remain applicable to the operation of the RCF assumed in this SER and conservative. The NRC staff review also confirmed that the RCF emergency plan contains appropriate provisions for mitigating the consequences of such an accident. Based on these confirmations, the NRC staff finds that a postulated fission product release does not pose an undue risk to public health and safety, facility personnel, or the environment.

TS 3.2.1 limits the maximum reactivity worth of any fuel pin to 0.20 dollars. This ensures that reactivity accidents as a result of mishandling or malfunction of fuel are bounded by the reactivity insertion accident discussed in Section 4.2 of this SER, which assumes a reactivity insertion of 0.60 dollars.

Based on the findings discussed above, the NRC staff concludes that operation in accordance with the requirements of the TS provides reasonable assurance that mishandling or malfunction of the RCF fuel will not pose an undue risk to public health and safety, facility personnel, or the environment.

4.6 Experiment Malfunction

TS 3.8.6 limits the maximum reactivity worth of a moveable or nonsecured experiment to 0.60 dollars. The reactivity insertion accident discussed in Section 4.2 of this SER assumes a worst case experiment failure. TS 3.8 specifies additional requirements for experiments. TS 3.8.1 requires experiments to have written procedures that permit good understanding of the safety aspects of the experiment. Additionally, the NSRB and Operations Supervisor must review and approve the procedure before performance of the experiment. TS 6.2, TS 6.4, and TS 6.5 specify additional review and approval requirements for experiments. These requirements ensure that experiments are adequately understood and documented before implementation, and that new experiments do not require an amendment to the facility operating license. TS 3.8.2, TS 3.8.3, TS 3.8.8, and TS 3.8.9 specify requirements to preclude experiments from damaging the reactor, fuel pins, or other experiments and from interfering with control rods or safety channels. These requirements ensure that experiments will not exacerbate reactor accidents or cause damage to the fuel pin cladding. TS 3.8.4 requires experiments to be designed to withstand power transients. This requirement ensures that experiments are designed to withstand all credible reactor conditions. TS 3.8.7 and TS 3.8.10 specify additional requirements for experiments that contain radioactive material. As discussed in Section 4.1 of this SER, these requirements ensure that the failure of an experiment that contains radioactive material will not result in doses to facility personnel or members of the public in excess of the applicable dose limits in 10 CFR 20.1301 and 10 CFR 20.1201. The NRC staff reviewed the TS requirements for experiments against the guidance in ANSI/ANS-15.1 and found that the requirements are consistent with the guidance and therefore, acceptable.

Based on the findings discussed above, the NRC staff concludes that performance of experiments in accordance with the requirements of the TS provides reasonable assurance that malfunction of experiments will not pose an undue risk to public health and safety, facility personnel, or the environment.

4.7 Loss of Normal Electrical Power

The loss of normal power is an anticipated event for the RCF and would not be expected to cause an accident. Reactor shutdown is passive and fail safe in that, if normal power is lost, the control rods automatically fall into the core by gravity and the moderator dumps, thereby shutting down the reactor. As discussed in Section 4.3 of this SER, loss of the moderator does not pose an undue risk to public health and safety, facility personnel, or the environment. No TS require 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. 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 public health and safety, facility personnel, or the environment.

4.8 Mishandling or Malfunction of Equipment

The RCF is equipped with an overhead crane. According to the licensee, the crane is periodically inspected according to industry standards. According to the licensee, administrative controls exist that restrict lifting loads over the reactor tank when the reactor tank contains fuel. Additionally, the senior reactor operator on duty supervises all crane operation. The NRC staff finds that these administrative controls provide reasonable assurance that a malfunction of the crane will not result in damage to the fuel.

TS 3.2.2 requires that the reactor be subcritical by at least 0.70 dollars with the most reactive control rod fully withdrawn. TS 3.2.2 also requires that all control rods be operable. This ensures that a control rod malfunction will not prevent a reactor shutdown. This also satisfies the reactivity insertion accident discussed in Section 4.2 of this SER, which assumes the worst case malfunction of the reactor safety system. The NRC staff reviewed these requirements against the guidance in ANSI/ANS-15.1 and found that they are consistent with or more conservative than the guidance and therefore, acceptable.

TS 6.2 and TS 6.4 specify requirements related to maintenance and replacement of equipment at the RCF. The requirements include NSRB review and approval of proposed changes to equipment to ensure that an amendment to the facility operating license is not required, and review and approval of written procedures for maintenance activities that could have an effect on reactor safety. The NRC staff reviewed these requirements against the guidance in ANSI/ANS-15.1 and found that they are consistent with the guidance and therefore acceptable.

Based on the above findings, the NRC staff concludes that the administrative precautions and TS requirements to preclude the mishandling or malfunction of equipment provide reasonable assurance that mishandling or malfunction of equipment will not cause damage to the fuel.

4.9 Conclusions

The licensee analyzed an MHA and found the radiological consequences to be below the applicable regulatory limits for occupational doses and doses to members of the general public. The NRC staff evaluated the licensee's assumptions and methods of calculating doses and finds them to be conservative and appropriate. The licensee analyzed a variety of credible, although unlikely, accident scenarios and found the consequences to be bounded by the MHA. The NRC staff evaluated the accident scenarios and assumptions and concludes that the licensee has analyzed an appropriate spectrum of credible accidents for the RCF and that the MHA bounds the consequences of other credible accidents. The licensee has shown that other credible accidents at the RCF do not have any significant offsite radiological consequences. Accordingly, the NRC staff concludes that accidents at the RCF will not pose an undue risk to public health and safety, facility personnel, or the environment.

5 TECHNICAL SPECIFICATIONS

The NRC staff evaluated the TS as part of its review of the application for renewal of Facility Operating License No. CX-22. The TS define certain features, characteristics, and conditions governing the operation of the RCF. The renewed license explicitly includes the TS as Appendix A. The NRC staff reviewed the format and content of the TS for consistency with the guidance found in ANSI/ANS-15.1 and NUREG-1537. The NRC staff specifically evaluated the content of the TS to determine if they meet the requirements in 10 CFR 50.36, “Technical Specifications.” The NRC staff concluded that the RCF TS meet the requirements of the regulations based on the following findings:

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

- The RCF is a facility of the type described in 10 CFR 50.21(c), and therefore, as required by 10 CFR 50.36(b), the facility license will include the TS. To satisfy the requirements of 10 CFR 50.36(b), the licensee provided TS derived from analyses in the RCF SAR.

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

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

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

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

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

Based on discussions between the NRC project manager and the facility director, the NRC staff modified several of the proposed TS submitted by the licensee. The modifications included changes to correct typographical errors and did not modify the intent or technical content of the TS. The NRC project manager explained the modifications to the licensee, and the facility director agreed to the modifications. Additionally, the facility director reviewed the changes with NSRB, and the NSRB agreed to the changes.

Based on the findings presented above, the NRC staff concludes that the TS are acceptable and meet the applicable regulatory requirements. As discussed elsewhere in this SER, the NRC staff concluded that normal operation of the RCF within the limits of the TS will not result in radiation exposures in excess of the limits specified in 10 CFR Part 20 for members of the general public or facility personnel. The NRC staff also concludes that the TS provide reasonable assurance that the facility will be operated as analyzed in the SAR, and that adherence to the TS will limit the likelihood of malfunctions and the potential accidents.

6 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 November 19, 2002, as supplemented by letters dated July 21, July 28, and September 3, 2008; June 28, August 31, October 14, and October 28, 2010; and February 14 and May 9, 2011, complies with the standards and requirements of the AEA and the Commission's rules and regulations set forth in 10 CFR.

- The facility will operate in conformity with the application, as well as with the provisions of the AEA and the rules and regulations of the Commission.

- There is reasonable assurance that (1) the activities authorized by the renewed license can be conducted at the designated location without endangering public health and safety, and (2) such activities will be conducted in compliance with the rules and regulations of the Commission.

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

- The issuance of the renewed license will not be inimical to the common defense and security or public health and safety.

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