Safety Evaluation Report Related to the Renewal of Facility Operating License No. R-2 for the Penn State Breazeale Reactor, Pennsylvania State University

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Office of Nuclear Reactor Regulation

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 Pennsylvania State University (PSU, the licensee) for a 20-year renewal of Facility Operating License No. R-2 to continue to operate the Penn State Breazeale Reactor (PSBR, the facility). The facility is located at the PSU campus in University Park, PA. In its safety review, the NRC staff considered information submitted by the licensee (including past operating history recorded in the licensee's annual reports to the NRC), as well as inspection reports prepared by NRC personnel and firsthand observations. On the basis of this review, the NRC staff concludes that PSU can continue to operate the PSBR, in accordance with the renewed license, without posing a significant risk to the health and safety of the public, facility personnel, or the environment.

CONTENTS

CONTENTS (cont.)

CONTENTS (cont.)

CONTENTS (cont.)

LIST OF ABBREVIATIONS

Abbreviation	Definition
ADAMS	Agencywide Documents Access and Management System
AEA	Atomic Energy Act of 1954, as amended
AEC	Atomic Energy Commission
AFRRI	Armed Forces Radiobiology Research Institute
ALARA	as low as reasonably achievable
ALI	annual limit on intake
Am-Be	americium beryllium
ANSI/ANS	American National Standards Institute/American Nuclear Society
Ar-41	argon-41
C	Celsius
CFR	*Code of Federal Regulations*
cm	centimeter
Co-60	cobalt-60
COE	College of Engineering
D_2O	deuterium oxide
DAC	derived air concentration
DCC-X	Digital Control Computer X
DCC-Z	Digital Control Computer Z
DOE	U.S. Department of Energy
EES	emergency exhaust system
EP	emergency plan
ESF	engineered safety feature
F	Fahrenheit
FC	fission chamber
FES	facility exhaust system
FFT	fast flux tube
FNI	fast neutron irradiator
ft	foot
FY	fiscal year
g	acceleration due to gravity
GA	General Atomics
GIC	gamma ion chamber
gpm	gallon per minute
HEU	high-enriched uranium
HVAC	heating, ventilation, and air conditioning
I&C	instrumentation and controls
in.	inch
km	kilometer
kW	kilowatt

LIST OF ABBREVIATIONS

Abbreviation	Definition
kW(t)	kilowatt thermal
l	liter
LCO	limiting condition for operation
LOCA	loss-of-coolant accident
m	meter
m/s	meter per second
MEPD	maximum elemental power density
MHA	maximum hypothetical accident
mi	mile
mi/h	miles per hour
MMI	modified Mercalli intensity
mrem	millirem
mSv	millisievert
MTR	materials testing reactor
MW(t)	megawatt thermal
N-16	nitrogen-16
Na-24	sodium-24
NP	normalized power
NRC	U.S. Nuclear Regulatory Commission
PA	public address (system)
PCMS	protection, control, and monitoring system
PSBR	Penn State Breazeale Reactor
PSRSC	Penn State Reactor Safeguards Committee
PSU	Pennsylvania State University
RPO	Radiation Protection Office
RSEC	Radiation Science and Engineering Center
RSS	reactor safety system
s	second
SAR	safety analysis report
SER	safety evaluation report
SL	safety limit
SNM	special nuclear material
SSC	structures, systems, and components
TEDE	total effective dose equivalent
TS	technical specification(s)
wt%	weight percent
U-235	uranium-235
UIC	University Isotope Committee

LIST OF ABBREVIATIONS

Abbreviation	Definition
UPS	uninterruptible power supply
USGS	U.S. Geological Survey
Zr-H	zirconium hydride
μS/cm	microsiemens per centimeter
$	dollar (of reactivity)

1 INTRODUCTION

1.1 Overview

By letter (and supporting documentation) dated December 6, 2005, as supplemented by letters dated October 31, 2008, and April 2, June 11, September 1, and October 21, 2009, the Pennsylvania State University (PSU, 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. R-2 (NRC Docket No. 50-005). The renewed license would authorize continued operation of the Penn State Breazeale Reactor (PSBR), part of the Radiation Science and Engineering Center (RSEC) located on the PSU campus in University Park, PA. In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 2.109, "Effect of Timely Renewal Application," the current license will not be deemed to have expired until the Commission takes final action on the licensee's 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 (EP), the physical security plan, financial qualifications, and responses to 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 review 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 or 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@nrc.gov.

This safety evaluation report (SER) summarizes the findings of the NRC staff's safety review of the licensee's application. This SER and the environmental assessment and finding of no significant impact, dated November 4, 2009 (ADAMS Accession No. ML092330193), will serve as the basis for issuance of a renewed license authorizing operation of the PSBR at steady-state power levels up to 1 megawatt thermal (MW(t)) and in pulse mode operation with a maximum reactivity insertion of 3.50 dollars ($). In conducting its safety review, the NRC staff evaluated the facility against the requirements of 10 CFR Parts 19, 20, 30, 50, 51, 55, 70, and 73; applicable NRC 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 in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," issued February 1996.

William B. Kennedy from the NRC's Office of Nuclear Reactor Regulation, Division of Policy and Rulemaking, Research and Test Reactors Branch A, prepared this SER. Other contributors to the safety review include NRC staff members William C. Schuster, Anthony Bowers, Paul V. Doyle, and JoAnn Simpson. Under contract to the NRC, William Watkins, James Willison, and James Wallace of Washington Safety Management Solutions, LLC, provided a technical evaluation of the licensee's SAR and TS.

1.2 Summary and Conclusions on Principal Safety Considerations

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

- The design, testing, and performance of the PSBR structures, systems, and components (SSCs) important to safety during normal operation are acceptable. Safe operation of the facility can reasonably be expected to continue.

- The licensee's management organization, training and research activities, and security measures continue to be acceptable. The licensee's management organization is able to maintain and safely operate the reactor, ensure the safe operation of the facility, and protect its special nuclear material (SNM).

- The licensee and the NRC staff have considered the expected consequences of postulated accidents, including a bounding maximum hypothetical accident (MHA), using conservative initiating and mitigating assumptions. The calculated radiation doses resulting from the MHA satisfy the regulatory dose requirements in 10 CFR Part 20, "Standards for Protection Against Radiation," for facility personnel and members of the general public.

- The NRC staff does not expect exposures from radiation and releases of radioactive effluents and wastes from the facility to result in doses or concentrations in excess of the limits specified by 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 and finds they are consistent with as-low-as-reasonably-achievable (ALARA) principles.

- The renewed facility operating license and TS, which state limits controlling the operation of the facility, provide reasonable assurance that the licensee will operate the facility in accordance with the assumptions and analyses in the SAR. No significant degradation of SSCs has occurred, and the TS will continue to provide reasonable assurance that no significant degradation of SSCs will occur.

- The financial data submitted with the application demonstrate that the licensee has acceptable access to sufficient funds to cover operating costs and to eventually decommission the reactor facility.

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

- The licensee's EP provides acceptable assurance that the licensee will continue to be prepared to assess and respond to emergency events.

- Continued operation of the PSBR poses no significant radiological risk to the health and safety of the public, facility personnel, or the environment.

On the basis of these findings, the NRC staff concludes that PSU can continue to operate the PSBR in accordance with the Atomic Energy Act of 1954, as amended (AEA, the Act), NRC regulations, and Facility Operating License R-2 without endangering the health and safety of the public.

1.3 General Facility Description

The PSBR is located in the Breazeale Nuclear Reactor Building on PSU's University Park campus. The facility is built into the side of a hill. PSU constructed the original building containing the reactor and associated systems in 1954. The current facility includes additional classrooms, laboratories, hot cells, and a machine shop. The building is constructed of concrete blocks, bricks, insulated steel and aluminum panels, structural steel, and reinforced concrete and is generally fireproof in nature. The building is constructed on two levels, with the reactor bay and control room on the upper level and the neutron beam laboratory and the mechanical equipment room on the lower level. The reactor control room is located adjacent to the reactor bay.

The reactor uses TRIGA fuel elements and has a maximum steady-state operating power of 1 MW(t). The reactor has a pulse capability with the maximum licensed pulse reactivity insertion resulting in a peak pulse power of approximately 2000 MW(t). The reactor is suspended in an open pool from a movable bridge structure. It can be moved to different locations within the pool and can also be rotated for different experiment configurations. The reactor pool has a number of beam ports and facilities for both thermal and fast neutron irradiations. Control rods are lifted by electromagnets that allow the rods to drop back into the core by gravity in the event of a power loss or scram signal.

Natural circulation of the pool water cools the fuel elements in the reactor core. Heat from the pool is removed through heat exchangers located on the lower level of the facility. A diffuser assembly above the core delays the rise of activation products to the pool surface and reduces dose rates in the reactor bay. The large volume of water above the reactor core provides shielding from radiation emitted by the fuel during operation and shutdown.

The reactor facility is housed within an industrial building with a separate ventilation system. The exhaust fans are tied into the radiation monitoring system so that the detection of high radiation levels will result in the suspension of normal building exhaust and the routing of building exhaust through an emergency exhaust system with roughing, absolute, and charcoal filters.

1.4 Shared Facilities and Equipment

The PSBR water, steam, and electricity are furnished through the campuswide systems. In an adjacent building, the Combustion Engineering Laboratory uses some of the effluent from the secondary side of the reactor heat exchanger. Some water handling facilities within the building are shared. For example, a storage tank can be used for either the reactor pool water or the cobalt-60 (Co-60) facility pool water. The PSBR has its own dedicated heating and ventilation system. The licensee's application describes the shared facilities and equipment and their use by other university entities. As described in the PSBR SAR and elsewhere in this SER, shared facilities and equipment cannot affect either safe operation or safe shutdown of the reactor.

1.5 Comparison with Similar Facilities

General Atomics (GA) designed the TRIGA reactors and since 1958, when the first one started, 60 TRIGA reactors have been built and safely operated in over 20 countries. The PSBR was the first to convert a materials testing reactor core to a TRIGA core. Although all the fuel elements in most other TRIGA reactors have the same fuel loading by weight percent (wt%), the PSBR core has fuel elements with a mixture of fuel loadings by wt%. Also, the PSBR core has a hexagonal fuel arrangement rather than the more widely used circular fuel arrangement. The core location in a large pool is different than the more common location in a tank. The ability to move or rotate the core is a unique feature of the PSBR. While these features differentiate the PSBR from other TRIGA reactors and give rise to some unique operations issues, the basic design criteria and safety features of the PSBR are consistent with similar facilities. Accordingly, the safe operation of similar facilities is an indication that the PSBR can be safely operated.

1.6 Summary of Operation

The licensee has operated the PSBR in accordance with Facility Operating License No. R-2 and has established procedures to facilitate experiments, research, and education. Educational uses include student laboratory exercises and student operator training. According to the 2005, 2006, and 2007 annual reports submitted to the NRC, PSU operated the facility for approximately 2,000 hours per year, with the reactor critical between 840 and 1,040 hours per year. Each year has between 150 and 200 pulses of the reactor. These values represent expected annual facility operation during the period of the renewed license.

1.7 Compliance with the Nuclear Waste Policy Act of 1982

Section 203(b)(1)(B) of the Nuclear Waste Policy Act of 1982 specifies that the NRC may, as a precondition to issuing or renewing an operating license for a research reactor, require the applicant to have reached an agreement with the U.S. Department of Energy (DOE) for the disposal of high-level wastes and spent nuclear fuel. In accordance with a letter from DOE (R.L. Morgan) to the NRC (H. Denton) dated May 3, 1983, it has been determined that all universities operating nonpower reactors have entered into a contract with DOE that provides that DOE retain title to the fuel and that DOE is obligated to take the spent fuel, high-level waste, or both, for storage or reprocessing. Because PSU has entered into such a contract with DOE, it has satisfied the applicable requirements of the Nuclear Waste Policy Act of 1982.

1.8 Facility Modifications and History

The U.S. Atomic Energy Commission (AEC) issued Facility Operating License No. R-2 for the operation of the PSU reactor in July, 1955. The initial power level was limited to 100 kilowatts thermal (kW(t)). The reactor achieved criticality on August 15, 1955. In 1960, the authorized maximum operating power level was increased to 200 kW(t).

Increased use caused PSU to replace the original core with a TRIGA reactor core and control system in 1965. The new core achieved criticality in December 1965. The new core had a maximum steady-state power level of 1 MW(t) and included a pulse capability allowing a peak pulse power of approximately 2000 MW(t).

The original TRIGA core was fueled with 8.5 wt% fuel. Change No. 5 to the TS, dated April 14, 1972, allowed a core modification to use a mixture of 8.5 wt% and 12 wt% fuel. PSU replaced the graphite thermal column with a deuterium oxide (D_2O) tank in 1971 and installed a second generation D_2O tank in 1997. In 1994, PSU modified the reactor bridge to allow for north-south, east-west, and rotational movement of the reactor core.

In 1991, PSU replaced the original GA control console with a new analog-digital control system. The new console has two independent systems for the safety, protection, control, and monitoring functions. Chapter 7 of this SER contains further details of the control system.

2 SITE CHARACTERISTICS

2.1 Geography and Demography

2.1.1 Geography

The PSBR is located at north latitude 40º48′15″ and west longitude 77º51′15″. The corresponding Universal Transverse Mercator coordinates are Zone Number 18, Northing 4520939 m, and Easting 259236 m. The PSBR facility is located on the PSU University Park campus in State College, PA, with Hastings Road to the north, Biggler Road to the south, and University Drive to the east. The PSBR is located in Centre County in the central part of the State, in Nittany Valley and near Bald Eagle, Nittany, and Tussey mountains. Residential and commercial areas border the campus.

The operations boundary is the reactor building. A chain-link fence marks the reactor site boundary. Access is through a locked gate. PSU owns and controls the land adjacent to the PSBR. U.S. Routes 220 and 322 pass within 2.4 kilometers (km) (1.5 miles (mi)) of the site. Plans exist for Route 220 to become Interstate 99 when missing sections are completed.

The PSBR SAR contains sufficient information to appropriately characterize the geography of the area surrounding the PSBR. Based on the NRC staff's review of the information presented by the licensee and maps of the University Park area, there is reasonable assurance that no geographic features exist that make the site unsuitable for continued operation of the PSBR.

2.1.2 Demography

According to the Centre Regional Planning Agency documents, the immediate vicinity around PSU includes Patton Township, Ferguson Township, College Township, Harris Township, and State College Borough. In the 2000 Census, these population units contained a total of 79,406 residents. The majority of this population resides within a 5-mile radius of the PSBR. State College Borough, which contains the PSBR, is already mostly developed, and the Centre Regional Planning Agency expects a 7 percent population increase by 2030. The regional population over the past 30 years has increased by approximately 8000 each decade and is expected to increase 28 percent by 2030. The outlying townships account for most of the expected regional population growth. As stated in the PSBR SAR, the nearest public residential area is approximately 357 meters (m) (1170 feet (ft)) from the facility.

According to the PSBR SAR, PSU has indicated that it intends to keep the peak student population fairly constant at 42,000 students. The student population is highest during the fall and spring semesters. The PSBR SAR states that the nearest campus residential area is the Eastview Terrace Dormitories, located approximately 128 m (420 ft) south of the facility.

The NRC staff verified population data by reviewing the Centre Regional Planning Agency documents. The licensee presented sufficient demographic information to accurately characterize the region surrounding the reactor site and assess the potential radiological impact on the public from operation of the facility. Based on the NRC staff's review of the demography of the area surrounding the facility, there is reasonable assurance that there are no current or projected demographic features that render the site unsuitable for continued facility operation.

2.2 Nearby Industrial, Transportation, and Military Facilities

The area surrounding the PSBR does not contain heavy industry. The major employers are in education, government, light manufacturing, and research and development. The only military facility in the immediate area is a small Army Reserve center 2.4 km (1.5 mi) away.

The main transportation routes are U.S. Routes 220 and 322 to the north of the PSBR. At least 2.4 km (1.5 mi) separates the PSBR from these roads. There is no rail access to the State College area. The NRC staff confirmed this information by reviewing area maps.

The University Park Airport is located 4.5 km (2.8 miles) north of the PSBR and serves both private and commercial aircraft. The airport's main runway is oriented northeast-southwest. Approach and takeoff paths are not over the PSBR. A shorter crosswind runway is oriented north-northwest/south-southeast, but it is not in line with the PSBR. The NRC staff confirmed this information by reviewing area maps and satellite photography.

Based on the character of the local industry and distances involved, the NRC staff concludes that neither local industry, nor transportation, nor military facilities pose a significant risk to the continued safe operation of the PSBR.

2.3 Meteorology

State College, PA, is located in the Valley and Ridge Province, approximately 400 km (250 mi) west of New York City. Weather at the site varies by season, with snow common in the winter. Storm systems predominantly come from the west. Lake-effect snow occurs but is of limited effect because Lake Erie is 320 km (200 mi) away. Typical snow accumulations are 5–15 centimeters (cm) (2–6 inches (in.)). Below freezing precipitation (sleet, snow, and ice storms) is most likely between November and March.

According to information supplied by the licensee and confirmed by the NRC staff, monthly average temperatures range from a low of -3.17 degrees Celsius (degrees C) (26.3 degrees Fahrenheit (degrees F)) in January to a high of 21.8 degrees C (71.2 degrees F) in July. Temperature extremes are a low of -29 degrees C (-20 degrees F) and a high of 38 degrees C (101 degrees F). Average daily humidity ranges between a low of 55 percent in the spring to approximately 80 percent in the summer. Precipitation is distributed throughout the year, with an annual average of 97 cm (38 in.). The maximum recorded precipitation in a 24-hour period was 12.8 cm (5.05 in.) in September 2004. Snowfall averages 119 cm (46.9 in.) annually. The maximum recorded 24-hour snowfall was 70.1 cm (27.6 in.) in both 1993 and 1994. The maximum recorded snowfall for one winter was 261 cm (102.6 in.) in the winter of 1994–1995.

Wind data is available from the PSU weather station. Average wind speeds range from 2–3.6 meters per second (m/s) (4.5–8 miles per hour (mi/h)). Generally, the wind direction is from the west, with the least frequent winds from the southeast.

The PSBR SAR discusses severe weather in the region. It indicates that the rate of thunderstorm incidence in central Pennsylvania is 40/year. It further states that hurricanes have not affected central Pennsylvania to a great extent, because of its distance inland. However, three storms have passed into the area in the last decade after weakening to tropical storms. The NRC staff confirmed the incidence of tropical storms in the area by reviewing maps from the National Hurricane Center.

The 50-year mean recurrence interval for high wind is 31 m/s (70 mi/h) in central Pennsylvania. The strongest wind gust recorded at University Park was 42 m/s (95 mi/h) in 1991. Tornados occur periodically in Pennsylvania, with an annual average number of 15, of which 3 were of F2 intensity or greater on the Fujita scale. Over the period 1950–1995, there were nine tornadoes in Centre County, with two of F2 intensity or greater.

Ice storms occasionally affect the area. From 1982–1990, University Park experienced approximately 120 hours of freezing rain. The 50-year return period is for ice thickness of 1.9 cm (0.75 in.).

The maximum snow loading was calculated by the licensee by summing the maximum 24-hour winter rainfall of 8.26 cm (3.25 in.) and the maximum monthly snowfall of 121 cm (47.5 in.). The maximum snow load was calculated at 2.43 kilopascals (50.75 pounds per square foot). The NRC staff notes that this is consistent with the methodology and values calculated at other nonpower reactors in the region.

Based on the meteorological information supplied by the licensee and the NRC staff's independent review, the NRC staff concludes that the meteorology in the vicinity of the PSBR does not pose any significant risk to the continued safe operation of the reactor.

2.4 Hydrology

State College, PA, is located in the Spring Creek drainage basin. Limestone and dolomite formations underlie this basin. The basin has an area of approximately 453 square kilometers (175 square miles) and discharges through the Milesburg Gap approximately 21 km (13 mi) north of the site.

The PSBR is built into the side of a hill on a dolomite formation. The site is 47 m (154 ft) above the nearest stream, and there is no history of flooding in the area. No probable maximum surge and seiche flooding considerations exist for this site, as no large bodies of water are near the site where a significant storm surge or seiche can form. As the site is not adjacent to a coastal area, tsunami flooding is not considered credible. There are no existing or proposed dams upstream of the site; thus, there are no seismically induced potential dam failure considerations for the site.

The nearest water well to the PSBR facility is approximately 250 m (820 ft) from the site and has a depth of 48 m (156 ft) to the water table. As discussed in Chapter 5 of this SER, the PSBR is a pool reactor with sufficient surrounding pool water to minimize the neutron flux at the pool wall. Neutron flux levels beyond the pool wall are insignificant. Thus, the potential for the creation of direct activation products outside the pool structure is insignificant. As the water table is significantly below the level of the facility, mechanisms for the transport of activation products are not present. Chapter 5 of this SER discusses the design features that prevent pool leakage.

Based on the lack of credible flooding risks, the NRC staff concludes that the local hydrology does not pose a significant risk to the continued safe operation of the PSBR. As discussed in greater detail in Chapter 5 of this SER, the NRC staff evaluated the potential for groundwater contamination and concludes that design features of the facility minimize the potential for contamination of the groundwater.

2.5 Geology, Seismology, and Geotechnical Engineering

Section 2.5 of the PSBR SAR describes the geology in the vicinity of the facility. The PSBR is located on the flank of the Valley and Ridge Province in central Pennsylvania. This province is characterized by severely folded and thrust-faulted sedimentary rocks. The PSBR site is located on the Nittany formation of dolomite with no intervening soil below the foundation. The surrounding rock formations are of limestone and dolomite. There are no known faults at the site nor any that project to the site from the surrounding area. Known faults in the general area are associated with the tectonic folding of the Appalachians. None of the known faults are currently active.

The Valley and Ridge Province has a lower level of seismicity compared to the adjacent areas in the northeastern United States. The historical record for seismic activity in the region of the PSBR extends back to 1724, although the instrumented record is much more recent. The seismic record in Pennsylvania shows few events in the Valley and Ridge Province and those were of relatively low intensity. The largest known seismic event in the Valley and Ridge Province had a magnitude of 3.7 and modified Mercalli intensity (MMI) of 6. Two events of MMI 6 and 7, recorded in the eastern portion of the State, were not seismic in nature but occurred as a result of mine collapses. A probabilistic risk assessment on seismic activity was performed for the Susquehanna Nuclear Power Plant in eastern Pennsylvania, also in the Valley and Ridge Province. That facility is in a zone with a slightly higher potential for earthquakes and is considered an upper bound on seismic effects at the PSBR. This upper bound was a seismic event of magnitude 5.

The U.S. Geological Survey (USGS) updated its seismic hazard maps for the United States based on new seismological, geophysical, and geological information. The USGS analysis shows relatively low ground-motion risk for a broad area surrounding the PSBR site. The NRC staff reviewed the 2008 National Seismic Hazard Map produced by USGS. This map shows only a 2-percent probability that, in 50 years, peak lateral ground acceleration will exceed 0.06 times the acceleration caused by gravity (g). The maximum anticipated ground acceleration for the area is 0.07 g.

The PSBR site is located in an area with no historical high-magnitude earthquakes and with low maximum anticipated ground acceleration. Because the facility is built directly on dolomite with no intervening soil, the potential for liquefaction is essentially nonexistent. Given the relatively low intensity and magnitude of seismic events, the presence of competent rock at the site, and the performance of existing foundations to date, the NRC staff expects the site grounds and foundations to continue to perform satisfactorily for their design conditions. The seismic history supports the conclusion that a catastrophic earthquake at or near the site is unlikely during the life of the facility. Based on the above information, the NRC staff concludes that the geology of the PSBR site is suitable for supporting the reactor building, structure, and systems, and that potentially damaging seismic events are unlikely to occur during the period of the renewed license.

2.6 Conclusions

The NRC staff concludes that the PSBR site has experienced no significant geographical, meteorological, or geological change since the initial siting of the facility, and therefore the site remains suitable for continued operation of the reactor. The infrequency of the occurrence of tornadoes and earthquakes continues to make the site suitable for operation of the reactor. Hazards related to industrial, transportation, and military facilities will not pose a significant risk to the continued safe operation of the facility. The demographics of the area surrounding the reactor have not changed and are not projected to change in any way that discernibly increases the risk to public health and safety from continued operation of the PSBR during the 20-year period of license renewal.

3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.1 Design Criteria

The design criteria require the SSCs related to safe operation and shutdown of the reactor to be able to perform their intended functions as described in the PSBR SAR. The principal safety-related SSCs are the fuel, core support structure, reactor safety system (RSS), reactor pool, and reactor building. The NRC staff evaluated the following specific design criteria for the above-mentioned SSCs during normal operation and credible accident scenarios:

- The fuel design must prevent the release of fission products.
- The core support structure must maintain its orientation, geometry, and structural integrity.
- The RSS must be able to shut down the reactor.
- The reactor pool must provide adequate shielding of radiation emitted from the reactor core and provide for heat removal from reactor components.
- The reactor confinement building must provide a pathway for a controllable radioactive release.
- The reactor confinement building must protect the reactor from external environmental conditions.

With the exception of the fuel and the RSS, the SSCs mentioned above were designed and constructed in accordance with the construction permit issued by AEC in 1955. The reactor core and the associated RSS were designed and installed in 1965. PSU completed upgrades to the RSS and reactor control systems in accordance with Amendment No. 30 to Facility Operating License No. R-2, dated May 16, 1991. Chapter 4 of this SER discusses the design of the fuel, control elements, core support structure, and reactor pool. Chapter 6 of this SER addresses the design of the exhaust system in the reactor confinement building, while Chapter 7 discusses the design of the RSS.

These SSCs have been maintained or changed using license amendments or licensee review processes, including 10 CFR 50.59, “Changes, Tests, and Experiments”; maintenance; and special procedures, as appropriate, in accordance with the Commission’s rules and regulations and Facility Operating License No. R-2, as amended. The NRC staff previously evaluated all amendments to the facility license, and the NRC inspection program verified that the licensee conducted the proper reviews. Chapter 16 of this SER discusses age-related issues. Based on the above and discussions in the referenced chapters of this SER, the NRC staff concludes that the design and construction of safety-related SSCs provide reasonable assurance that the SSCs will continue to meet the design criteria.

3.2 Meteorological Damage

Section 2.3 of this SER presents the meteorology in the vicinity of the PSBR. Meteorological data demonstrate that extreme weather conditions that could affect the structure of the PSBR facility are unlikely. The highest recorded wind speed in the area was a peak wind gust of 42 m/s (95 mi/h) in 1991. The estimated 50-year return peak wind of 31 m/s (70 mi/h) is consistent with recorded historical wind peaks. The reactor building was built to appropriate building codes at the time of construction. It has survived over 50 years of site weather without sustaining any damage. The NRC staff concludes that the design of the structure is consistent with the meteorological conditions that are likely to occur during the lifetime of the facility.

3.3 Water Damage

Section 2.4 of this SER presents the hydrology in the vicinity of the PSBR. There are no bodies of water at an elevation higher than the PSBR site in the immediate vicinity that could flood the site. The site is located well above the water table and the nearest surface water source. Historical high levels of precipitation would not raise the water table to the point of inundating the reactor building structure. Therefore, the NRC staff concludes that water damage poses no significant risk to safe operation or shutdown of the reactor.

3.4 Seismic Damage

As discussed in Section 2.5 of this SER, the probability of seismic hazards in the PSBR vicinity are low. If a seismic event caused significant shaking at the site, the reactor control rods could decouple from their supporting electromagnets and drop into the core. As this is the normal mechanism for rapid reactor shutdown, it poses no safety concern. If seismic activity did not deenergize the control rods and shut down the reactor, the reactor operator could manually shut it down. Disruption of electrical service caused by seismic activity poses no significant risk to the facility. As discussed in Chapter 8 of this SER, electricity is not required to accomplish or maintain safe shutdown of the reactor. An earthquake of sufficient magnitude would cause the reactor pool structure as a whole to shake. If this shaking caused rupture of the pool or connecting piping, coolant could drain out of the reactor pool. As discussed in Chapter 13 of this SER, the licensee analyzed a loss-of-coolant accident (LOCA) and concluded that, under conservative assumptions for core operating history, the maximum fuel temperature would remain below the air-cooled temperature limit. Based on these discussions, the NRC staff concludes that the design of the facility provides reasonable assurance that potential seismic events will not pose a significant radiological risk to the health and safety of the public.

3.5 Systems and Components

The licensee has a preventive maintenance and surveillance program to provide reasonable assurance that the mechanical systems and components and the electrical and instrumentation systems and components important to safety meet the performance requirements of the TS. Section 4.2.1 of this SER discusses the fuel design requirements. Chapter 13 evaluates accident scenarios, and Chapter 16 considers aging issues associated with the fuel. These discussions show that the fuel cladding design basis and related TS are adequate to ensure fuel cladding integrity under all credible circumstances.

Chapter 7 of this SER discusses the design of the instrumentation and control (I&C) systems, including the reactor control system and the RSS. Section 4.2.2 discusses the design of the control rods. These discussions show that the RSS design bases and related TS give reasonable assurance that the RSS will function as designed to ensure safe operation and safe shutdown of the reactor.

3.6 Conclusions

On the basis of the above considerations, the NRC staff concludes that the design and construction of the PSBR are adequate to withstand all credible wind, water, and seismic events associated with the site and ensure safe shutdown if they occur. Safe operation during the period of the current license and routine NRC inspections have verified the design and acceptable performance of safety-related systems and components. The NRC staff also concludes that surveillance activities required by the TS discussed in the above-referenced sections of this SER provide reasonable assurance that the safety-related functions of the facility SSCs will be operable. Accordingly, the NRC staff concludes that the reactor systems and components are adequate to provide reasonable assurance that continued operation will not cause significant radiological risk to the health and safety of the public, licensee personnel, or the environment.

4 REACTOR DESCRIPTION

4.1 Summary Description

The PSBR is a Mark III TRIGA research reactor licensed to operate at a steady-state power level of 1 MW(t) and a peak pulse power of approximately 2000 MW(t). The TRIGA fuel is approximately 20 percent enriched uranium-235 (U-235) and the core contains a mixture of 8.5 wt% and 12 wt% uranium fuel. The reactor is contained in a concrete pool surrounded by a biological shield of earthen backfill, except for the south side, where beam ports penetrate an additional thickness of concrete biological shielding.

Natural convection with water provides core cooling during normal operation. A filtration and demineralizer system maintains water purity at acceptable levels to prevent significant corrosion of reactor components and fuel. The pool water is the primary heat sink and provides radiation shielding, secondary neutron moderation, and neutron reflection. The TRIGA fuel design incorporates a homogeneous mixture of zirconium hydride (ZrH) in the fuel structure to provide primary neutron moderation.

The reactor core support structure is mounted on a movable bridge that enables the core to be positioned at various locations in the reactor pool. Movement of the reactor facilitates more efficient use of experimental facilities such as the fast neutron irradiators (FNIs) and neutron beam ports. Internal core experimental facilities include a central thimble, dry irradiation tubes, and a pneumatic transfer system.

4.2 Reactor Core

The reactor core is located near the bottom of the reactor pool. The suspension tower attached to the moveable reactor bridge supports the core. Upper and lower grid plates hold the fuel elements in place. The grid plates contain holes arranged in a hexagonal pattern for positioning the fuel elements. Each fuel element fits into fixed openings in the grid plates. The fuel is arranged within this hexagonal pattern, with some positions filled by control rods, experimental facilities, and the neutron startup source.

One safety control rod, one shim control rod, one regulating control rod, and one transient control rod control the reactor. The safety, shim, and regulating control rods provide primary control of the reactor. These rods are used to attain criticality on startup, make major changes in the power level of the reactor, and compensate for reactivity changes that occur as a result of changes in xenon reactivity worth, changes in fuel and moderator temperature, and fuel burnup. The reactor bridge supports the control rods and their drive motors.

4.2.1 Reactor Fuel

GA developed the U-ZrH fuel used in the PSBR and other TRIGA reactors with power levels up to 14 MW(t). In 1957, GA developed and began using U-ZrH fuels in the TRIGA reactors. Over 6,000 fuel elements have been fabricated for more than 66 TRIGA reactors in various countries around the world. The U-ZrH fuel has unique safety features, including a large negative prompt temperature coefficient of reactivity, high fission product retention, chemical stability when quenched from high temperatures in water, and dimensional stability over large temperature changes. Over 25,000 pulses have been performed with the TRIGA fuel elements at GA, with fuel temperatures reaching peaks of about 1150 degrees C (2100 degrees F).

GA developed TRIGA fuel based on the concept of inherent safety. It sought a core composition with such a large negative prompt temperature coefficient of reactivity that, if all the available excess reactivity were suddenly inserted into the core, the resulting fuel temperature increase would automatically cause the power excursion to terminate before any core damage resulted. Experiments performed in the late 1950s demonstrated that ZrH possessed the basic mechanism needed to produce this desired characteristic.

The PSBR uses typically sized TRIGA fuel elements clad with 304 stainless steel. The fuel is enriched to slightly less than 20 percent in U-235, and elements contain one of two different uranium wt%s. According to the licensee, the 8.5 wt% U fuel has a hydrogen to zirconium atom ratio of 1.7 to 1.0, and the 12 wt% U fuel has a hydrogen-to-zirconium atom ratio of 1.65 to 1.0. The fuel elements contain graphite reflectors at each end of the fuel and stainless steel top and bottom end fixtures to facilitate configuration within the core support structure.

The primary design objective of PSBR fuel is to maintain fuel integrity under any anticipated operating conditions. This objective is ensured by limiting the maximum fuel temperature, as specified in TS 2.1, "Safety Limit—Fuel Element Temperature." This safety limit (SL) is based on the stress in the cladding remaining below the ultimate stress. An increase in the fuel temperature causes an increase in the hydrogen pressure from the dissociation of ZrH. Data in the literature indicate that the stress in the cladding will remain below the ultimate stress if the temperature of the fuel does not exceed 1150 degrees C (2100 degrees F) and the temperature of the fuel cladding is below 500 degrees C (930 degrees F). TS 2.2, "Limiting Safety System Setting," limits the maximum fuel temperature to 650 degrees C (1200 degrees F), as measured by an instrumented assembly located in a core position representative of the maximum elemental power density (MEPD). This limiting safety system setting provides a safety margin between the operational limit and the safety limit to allow for measurement and analytical uncertainties as well as anticipated operational transients. Accident analyses performed in Chapter 13 of the PSBR SAR demonstrate that operation within the limits of the TS ensures no loss of fuel integrity for any of the anticipated accident scenarios analyzed.

TS 5.1, "Reactor Fuel," describes the design characteristics of the individual TRIGA fuel elements used in the PSBR core, and TS 5.2, "Reactor Core," specifies acceptable types and arrangements of fuel and reflector core components. The operational configuration of the reactor is restricted by TS 3.1.5, "Core Configuration Limitation," which specifies the type of fuel elements allowed, the maximum MEPD, and the maximum normalized power (NP).

The NRC staff reviewed the PSBR safety analysis and found the PSBR fuel design is adequately supported by the GA fuel development program and operational history for U-ZrH fuels. The NRC staff found that research and testing performed on similar U-ZrH fuels supports the TS limits on fuel temperature. Therefore, continued operation, as limited by the TS, offers reasonable assurance that the fabricated fuel can meet the design objective of maintaining fuel integrity and can thereby function safely in the reactor without adversely affecting the health and safety of the public.

4.2.2 Control Rods

The PSBR reactor control system includes three standard control rods (safety, shim, and regulating) and a single transient rod. The standard control rods consist of four sections. The top and bottom sections are graphite, the control section is graphite impregnated with powdered boron carbide, and the follower section is 8.5 wt% U-ZrH fuel. The standard control rods are clad with stainless steel. The transient rod consists of two sections. The top control section is borated graphite, and the bottom section is air-filled follower. The transient rod is clad with aluminum.

The control rods and transient rod are internal to the core periphery in grid positions identical to those that contain the fuel elements. The rod locations are roughly symmetrical about the center of the core. This arrangement helps minimize power peaking in the core that could result from an imbalance in neutron absorption biased toward one region of the core. The licensee's policy of operating the reactor with all rods withdrawn by roughly the same amount from the core also reduces power peaking within the core.

The standard control rods are positioned with motor-driven rack-and-pinion drives connected to the control rod by an electromagnet. The control rod drive systems are all independent, and a malfunction in one would not affect insertion or withdrawal of any other. The control rod and drive positions are shown on the operator display panel. The transient rod uses a pneumatic-electromechanical drive system to allow ejection of a predetermined amount of the transient rod from the core for pulsed operation. A system of limit switches provides position indication for the transient rod.

GA designed the control and transient rod systems to allow safe and reliable control of the reactor power level. Chapter 4 of the PSBR SAR presents the licensee's analysis of the requirements for reactivity control systems. This analysis forms the bases for the designs of the control and transient rod systems and TS related to reactivity requirements for the systems. TS 3.2.1, "Reactor Control Rods," requires a minimum of three 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 3.1.3, "Shutdown Margin," which requires the shutdown margin provided by the control rods to be greater than 0.175% Δk/k (~$0.25) with experiments in their most reactive state and the highest worth control rod fully withdrawn. These TS satisfy the "stuck rod" criterion found in the guidance in NUREG-1537 and ANSI/ANS-15.1, "The Development of Technical Specifications for Research Reactors," issued 2007. TS 3.2.2, "Manual Control and Automatic Control," limits the maximum reactivity addition rate to 0.63% Δk/k/s (~$9.00/s) averaged over full travel. TS 3.1.4, "Pulse Mode Operation," limits the maximum stepped reactivity insertion for pulse operation and the maximum worth of the poison section of the transient rod to 2.45% Δk/k (~$3.50). As discussed in Section 4.5 of this SER, these limits preclude a reactivity accident that could cause fuel damage. TS 3.2.4, "Reactor Safety System and Reactor Interlocks," requires interlocks that preclude control and transient rod motions when certain reactor conditions are not met and an interlock that prevents simultaneous withdrawal of two or more rods. These interlock requirements are consistent with the analyses in the PSBR SAR, and the NRC staff finds them adequate to prevent unanalyzed rod motions.

GA designed the control rod and transient rod systems to be failsafe. A reactor scram or a loss of electrical power deenergizes the electromagnets in the drive mechanisms, thereby allowing insertion of the rods into the core by gravity. A reactor scram or a loss of electrical power deenergizes the three-way air supply solenoid, relieving the pressure in the cylinder so that the transient rod drops into the core by gravity. TS 3.2.6, "SCRAM Time," specifies that the time from scram initiation to full insertion of any control rod from a full up position shall be less than 1 second. As discussed in Section 4.5 of this SER, this scram time is adequate to terminate reactivity transients to protect the SL on fuel temperature.

TS 4.2, "Reactor Control and Safety System," contains surveillance requirements for the control rod and transient rod systems. These include reactivity worth (TS 4.2.1, "Reactivity Worth"), rod drive speed and scram times (TS 4.2.2, “Reactivity Insertion Rate”), control and transient rod interlocks (TS 4.2.4, “Reactor Interlocks”), and transient rod inspection and maintenance (TS 4.2.6, “Transient Rod Test”). 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, “Controls Rods,” specifies design parameters for the control rod systems, including control rod materials and configuration, that are consistent with the analyses presented in the PSBR SAR.

The NRC staff has evaluated the reactivity control and scram systems and compared them to other nonpower reactor designs with similar operating characteristics. The analyses presented in the PSBR 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 PSBR.

4.2.3 Neutron Moderator and Reflector

The PSBR uses the TRIGA U-ZrH fuel in which the ZrH acts as the primary moderator. For many years, many similar research reactors have used ZrH successfully as a neutron moderator. The fuel and moderator are alloyed together, causing any sudden increase in power to heat the fuel and moderator simultaneously. This provides the large prompt negative temperature coefficient characteristic of the U-ZrH fuel.

The light water surrounding the reactor fuel serves as the primary reflector and the reactor coolant. In addition, grid positions not filled by fuel-moderator elements and other core components may be filled by graphite reflector elements. Each graphite reflector element is fully enclosed in aluminum, has the same dimensions as the fuel elements, and is anodized after assembly. A blue anodized finish makes the graphite reflector elements easily distinguishable from fuel-moderator elements. The location of the reflector elements on the core periphery minimizes the potential for adverse effects of swelling on control rod movement.

TS 3.3.5, “Coolant Conductivity Limits,” provides coolant chemistry control, specifying allowable conductivity values to minimize corrosion of primary components and fuel. As discussed in Section 5.4 of this SER, the conductivity limits specified in TS 3.3.5 are appropriate to prevent excessive corrosion of the fuel and core components.

The NRC staff evaluated the neutron moderator and reflector systems presented in the PSBR SAR against the guidance in NUREG-1537 and systems in place at similar research reactors. The NRC staff found that the neutron moderators and reflectors used at the PSBR are comparable to those used at other TRIGA reactors and demonstrate material compatibility with respect to chemical, thermal, and radiation environmental performance. Based on these findings, the NRC staff concludes that continued operation within the requirements of the TS provides reasonable assurance that the moderator and reflector systems designed for this reactor will perform as necessary and will not adversely affect safe reactor operation, prevent safe reactor shutdown, or cause an uncontrolled release of radioactive material to the unrestricted environment.

4.2.4 Neutron Startup Source

The PSBR uses an americium-beryllium (Am-Be) neutron source to provide neutron indication for reactor instrumentation during reactor startup. A double encapsulated stainless steel container provides encapsulation for the source material. An aluminum source holder with dimensions similar to a standard fuel element is used to position the source in any fuel element position in the core. A daily sensitivity check of the safety system's wide-range neutron detection channel is performed by removing the source holder from the core, as required by TS 4.2.3, "Reactor Safety System."

The americium isotope used in the PSBR source has a 433-year half-life that produces a neutron when the alpha particle from its radioactive decay is absorbed by beryllium. These materials are a common combination for neutron sources, and the long half-life provides a constant neutron source for startup reactor instrumentation indication. Operation in the PSBR has demonstrated adequate source strength for providing a functional neutron-induced signal for reactor startup. The heat generated in the source during full-power operation of the PSBR is sufficiently low that adequate cooling is provided by conduction to the primary coolant.

The proper functioning of the neutron source is integral to the RSS to ensure an adequate neutron-induced signal on the reactor instrumentation before startup, as specified in TS 3.2.4. TS 4.2.3 and TS 4.2.4 specify surveillance requirements for the RSS and reactor interlocks involving the neutron source.

The NRC staff evaluated the Am-Be neutron source in use at the PSBR and found it comparable to those used at similar licensed research reactors. The NRC staff concludes this type of neutron source is acceptable for the PSBR. Operational history has demonstrated the design is adequate for providing sufficient startup neutrons and source-range indication for reliable reactor startup. Continued use of the Am-Be source in accordance with the applicable TS and procedures provides reasonable assurance that the neutron source and holder design can operate safely and reliably.

4.2.5 Core Support Structure

The PSBR reactor core is supported by an aluminum suspension tower attached to a trolley and reactor bridge that allows the reactor core to be moved north-south, east-west, and rotated. These movements are controlled by a high-gear-ratio hand wheel. Vise clamps, chains, and padlocks prevent movement during reactor operation.

The suspension tower supports the core assembly and grid plates, a central thimble, an ionization chamber for the power range monitor, a fission chamber for the wide-range monitor, other auxiliary reactor power detectors, the nitrogen-16 (N-16) diffuser system plumbing and spray nozzle, an external neutron source, a pneumatic transfer system terminus, vertical irradiation tubes, and core lights. The aluminum bottom grid plate supports the weight of the fuel and has holes arranged in a hexagonal pattern for fuel element positioning. Cooling water holes in the bottom grid plate provide a path for natural convective coolant flow. An aluminum safety plate suspended below the bottom grid plate prevents control rods from dropping out of the core. The aluminum top grid plate facilitates proper positioning of fuel elements, control rods, experiments, and in-core measuring instruments.

TS 3.1.5 specifies a 4.3-cm (1.7-in.) center line grid spacing to ensure proper and consistent fuel element configuration. Section 5.4 of this SER discusses the TS 3.3.5 requirement for proper chemistry control to prevent coolant corrosion of the core support structures. TS 5.2 specifies acceptable core configurations as an arrangement of TRIGA fuel elements positioned in the reactor grid plates.

The NRC staff reviewed the PSBR safety analysis and determined the core support structure section adequately describes the design that provides structural support for the core and ensures a stable and reproducible core configuration for all anticipated conditions throughout the reactor life cycle. The NRC staff evaluated the design and features of the core support structure and found that it is constructed of appropriate material, provides for adequate support of the core components, and contains features for reproducible positioning of core components. The NRC staff found that the core support structure design allows sufficient natural convection coolant flow and is compatible with the coolant and radiation environment. Based on these findings, the NRC staff concludes that the core support structure is adequate for the continued safe operation of the PSBR.

4.3 Reactor Pool

The PSBR reactor pool measures approximately 9.1 m (30 ft) long, 4.3 m (14 ft) wide, and 7.3 m (24 ft) deep. The pool is constructed of steel-reinforced concrete with a volume of approximately 2.7×10^5 liters (l) (71,000 gallons). The pool walls are lined with a watertight polyurea coating. Seven neutron beam ports penetrate the PSBR reactor pool wall. There are two cooling system penetrations approximately 3 m (17 ft) above the pool floor (approximately 1.8 m (6 ft) below the surface of the pool water), with the return pipe terminating in a perforated pipe along the pool bottom. The primary cleanup system has two penetrations approximately 3 m (10 ft) above the pool floor, with the inlet piping near the water's surface (surface skimmer inlet) and the outlet piping just below the surface of the pool.

A worst case siphoning of the pool water caused by a break in the cleanup system pump room piping (or cooling system pump room piping) could only siphon to the lowest point in the demineralizer room, which would leave approximately 1.2 m (4 ft) of water above the reactor core. Pool-level-low alarms required by TS 3.3.1, "Coolant Level Limits," would alert the operator of the break, allowing initiation of actions to mitigate the water loss. Reactor pool water that leaked into the demineralizer room would be collected in the waste tank in the evaporator building. A large loss of pool water could exceed the tank capacity, for which case the operator receives a high-level alarm. As discussed in Section 13.3 of this SER, a LOCA caused by a worst case malfunction in the pool barrier or primary system piping would not result in fuel damage.

TS associated with the reactor pool include TS 3.3.1, which requires a minimum of 5.5 m (18 ft) of water above the bottom grid plate for reactor operation; TS 3.3.2, "Detection of Leak or Loss of Coolant," which requires an alarm and corrective action when the pool water level drops 26 cm (10.2 in.) from full; and TS 3.3.4, "Pool Water Supply for Leak Protection," which requires a source flow rate of at least 6.3 liters/second (100 gallons/minute (gpm)) to the pool. Surveillance TS associated with the reactor pool include TS 4.3.1, "Fire Hose Inspection," which requires an annual inspection of two dedicated fire hoses that provide an emergency water supply to the pool; and TS 4.3.4, "Pool Water Level Alarm," which requires a monthly operability check of the pool-water-level alarm.

The NRC staff reviewed the PSBR safety analysis and determined that the reactor pool section adequately describes the reactor pool design and that appropriate measures and features exist to minimize the potential for a loss of pool integrity, which could lead to a loss of coolant or other malfunction. The NRC staff found that reactor penetrations and piping are appropriately designed to prevent uncovering of the core by siphoning. As discussed in Section 13.3 of this SER, a LOCA would not result in fuel damage. Recent inspection testing, concrete repair, and liner upgrades documented in the licensee's application provide reasonable assurance that the pool is capable of withstanding the corrosion and radiation environment for the period of the renewed license. Based on these findings and discussion, the NRC staff concludes that the reactor pool design and associated TS are adequate for continued safe operation of the PSBR.

4.4 Biological Shield

The reactor pool water and steel-reinforced concrete pool walls provide biological shielding for the PSBR. The pool walls are thick, steel-reinforced concrete backed by earthen fill on three sides. On the south side of the reactor pool, an additional thickness of high-density concrete provides extra shielding for the neutron beam laboratory. Except for the normally inaccessible areas inside and immediately adjacent to the beam ports, radiation levels in the neutron beam laboratory from the operation of the reactor are below levels measurable with standard radiation survey equipment. TS 3.3.1 requires a minimum of 5.5 m (18 ft) of water above the top of the bottom grid plate, providing sufficient shielding to protect personnel during operation at full power. An alarm sounds when the water level drops to 5.56 m (18.25 ft) above the top of the bottom grid plate. By procedure, this alarm condition requires the operator to shut down the reactor. The N-16 diffuser pump mitigates the potential exposure to N-16, discussed in more detail in Section 5.6 of this SER.

The NRC staff reviewed the PSBR biological shield design and found that it is consistent with other licensed research reactors of similar design and maximum licensed power level. As discussed in Section 11.1.5 of the PSBR SAR, operational exposures to reactor operations personnel have historically been very low and have seldom exceeded an annual total effective dose equivalent (TEDE) of 0.5 millisievert (mSv) (50 millirem (mrem)). Based on the above finding and the historical personnel doses, the NRC staff concludes that continued operation within the requirements of the TS and the PSBR radiation protection program offers reasonable assurance that the shield design will maintain exposures below the limits of 10 CFR Part 20.

4.5 Nuclear Design

4.5.1 Normal Operating Conditions

All core loadings undergo evaluation by licensee staff and approval of the facility director. Computer analysis of core loadings ensure the MEPD does not exceed 24.7 kW(t) and the maximum NP does not exceed 2.2. TS 3.1.5 requires these limits, and accident analyses in Chapter 13 of the PSBR SAR demonstrate no loss of fuel integrity when these limits are met. The MEPD and NP for a typical core design (Core Loading # 52) provided in the PSBR SAR are 16.54 kW(t) and 1.724, respectively. These values are well within the limits of the TS.

The example provided in the PSBR SAR, Core Loading # 52, shows that the shutdown margin is $3.31, with the reactor critical at 50 watts and the most reactive rod stuck in the full out position. The licensee appropriately calculated the shutdown margin as the difference between the reactivity needed to achieve criticality and the reactivity worth of the most reactive control rod. The calculated shutdown margin greatly exceeds the TS minimum of $0.25. TS 3.1.2, "Reactivity Limitation," limits the maximum excess reactivity of the core above cold, clean, critical to 4.9% Δk/k (~$7.00), including the reactivity effects of samarium poison, experiments, and experimental facilities. For a typical core loading described in the PSBR SAR, the total reactivity worth of the four control rods is $12.80. The total reactivity required to make the core critical at 50 watts is $7.29, leaving an excess reactivity of $5.51. This is well within the limit of $7.00 specified in TS 3.1.2.

The control rod drop times and reactivity insertion rates for Core Loading # 52 demonstrate reactivity control and dynamic characteristics of the core within the limits of the TS and accident analyses in Chapter 13 of the PSBR SAR. The drop times for the three standard control rods are all below 0.5 second, and the transient rod is 0.658 second, all of which are within the 1 second requirement of TS 3.2.6. The maximum reactivity insertion rate for any of the four control rods is $0.141/second, and the sum of the maximum rates for the safety, shim, and regulating rods is $0.384/second. These are within the limit of $0.90/second specified in TS 3.2.2. The transient rod worth of Core Loading # 52 is $2.88, which is within the $3.50 limit of TS 3.1.4.

The potential for reactivity transients resulting from in-core components that could be flooded or voided is addressed by analyses of the reactivity worth of all such facilities for Core Loading # 52. Historically, the maximum single reactivity worth for an experimental facility was for a 1.5 ft×1.5 ft air-filled aluminum box with a worth of $0.80. This reactivity worth bounds the potential reactivity changes caused by flooding for the experimental facilities and is well within the $3.50 limitation of TS 3.7, “Limitations of Experiments.”

Other TS related to the normal operating conditions of the reactor core include surveillance requirements for reactivity worth (TS 4.2.1), reactivity insertion rate (TS 4.2.2), and transient rod operability (TS 4.2.6). Design specifications are provided for the reactor core (TS 5.2) and control rods (TS 5.3). These and previously discussed TS in this section are consistent with the guidance in ANSI/ANS-15.1. The NRC staff found that the analyses presented in the PSBR SAR adequately justify these TS. Based on the above considerations, the staff concludes that the licensee has adequately analyzed expected normal reactor operation during the period of the renewed license.

4.5.2 Reactor Core Physics Parameters

The neutron lifetime and effective delayed neutron fraction presented in the PSBR SAR are 38 microseconds and 0.007, respectively. The average prompt temperature coefficient is -1.4 E-4 Δk/k/ degrees C. These values are presented as representative of standard TRIGA cores. The NRC staff reviewed the core physics parameters presented in the PSBR SAR and found the values to be representative of the parameters for similar research reactors, although direct comparison is not possible because of the mixed-core design of the PSBR. The safety analysis for the PSBR uses a direct measurement of fuel temperatures, and uncertainties in the reactor physics parameters are appropriately accounted for in the overall measurement uncertainties discussed in the licensee’s renewal application.

4.5.3 Operating Limits

The PSBR SAR provides reactivity analyses for Core Loading # 52. This sample core configuration demonstrates: (1) an excess reactivity of $5.51, which is within the $7.00 limit of TS 3.1.2; and (2) a shutdown margin of $3.31, which is within the $0.25 limit of TS 3.1.3. The shutdown margin requirement for the PSBR requires this margin, with all experiments in their most reactive state and the highest reactivity worth control rod fully withdrawn. This analysis demonstrates a readily operable reactor with a modest excess reactivity fully controlled under normal operating conditions.

The PSBR TS contain limits on the control rod system that provide for safe and reliable control of the reactor power level. TS 3.2.2 limits the reactivity addition rate to 0.63% Δk/k/s (~$0.90/s) averaged over full travel. Chapter 13 of the PSBR SAR contains an analysis of a ramp reactivity insertion that appropriately uses this TS limit as the reactivity insertion rate. As discussed in Section 13.2.1 of this SER, the NRC staff found the licensee's analysis reasonable and appropriate. Based on this finding, the NRC staff concludes that the limit on the reactivity addition rate is acceptable and adequately justified.

TS 3.1.4 limits the maximum stepped reactivity insertion for pulse operation and the maximum worth of the poison section of the transient rod to 2.45% Δk/k (~$3.50). The analysis in Chapter 13 of the PSBR SAR shows that a $3.50 pulse, starting from the pool ambient temperature, and a reactor power of 1 kW, corresponds to a maximum fuel temperature of 1,095 degrees C (2,000 degrees F). This is below the SL of 1,150 degrees C (2,100 degrees F). TS 3.2.4 requires an interlock that prevents pulsing from reactor power levels greater than 1 kW. As described in Section 13.2.1 of this SER, the NRC staff found the licensee's analysis appropriate and reasonable. Based on this finding, the NRC staff concludes that the maximum stepped reactivity insertion limit for pulse operation is acceptable and adequately justified. The NRC staff finds that the design, functional description, and TS requirements related to the transient rod system offer reasonable assurance that pulses will be limited to values that will not cause fuel damage.

Other TS related to the operating limits of the reactor include surveillance requirements for excess reactivity (TS 4.1.2, "Reactor Excess Reactivity"); those previously discussed in Section 4.5.1 include reactivity worth (TS 4.2.1), reactivity insertion rate (TS 4.2.2), and transient rod operability (TS 4.2.6). TS 5.3 contains the control rod design specifications. TS 3.2.4 requires one operable fuel temperature scram channel (scram at less than or equal to 650 degrees C (1,200 degrees F)) and two operable reactor power scram channels (scram at less than or equal to 110 percent of maximum reactor operational power, not to exceed 1.1 Megawatt) during steady-state and square wave modes of reactor operation. TS 3.2.4 also requires that the fuel temperature scram channel be operable during pulse mode operation. These fuel temperature and reactor power limits are consistent with those used in the licensee's accident analyses presented in Chapter 13 of the PSBR SAR. TS 4.1.1, "Reactor Power Calibration," TS 4.2.3, and TS 4.2.5, "Overpower SCRAM," contain surveillance requirements for the fuel temperature and reactor power scram channels. With the exception of TS 4.1.1, the TS discussed in this section are consistent with the guidance in ANSI/ANS-15.1, and the NRC staff finds them acceptable. TS 4.1.1 specifies a longer surveillance interval for the reactor power calibration than that recommended in ANSI/ANS-15.1. The NRC staff previously approved the longer surveillance interval in Amendment No. 37 to Facility Operating License No. R-2, dated October 14, 2004 (ADAMS Accession No. ML042790477), and continues to find it acceptable.

The thermal-hydraulic analysis of the PSBR is based on the MEPD and NP restrictions of 24.7 MW(t) and 2.2, respectively, specified in TS 3.1.5. The accident analyses discussed in Chapter 13 of this SER used these worst case power density and distribution factors, which are considered sufficiently conservative that continued operation within the TS provides reasonable assurance that reactor operation will not result in a loss of fuel integrity or undue radiological hazards to facility personnel, members of the public, or the environment.

The NRC staff reviewed the PSBR safety analysis and determined that the nuclear design section adequately describes the nuclear design characteristics necessary to ensure safe and reliable operation under normal operating conditions. The NRC staff found that the reactor core physics parameters are consistent with reactors of similar design, and the operating limits specified in the TS will ensure fuel cladding integrity. The NRC staff concludes that the nuclear design of the PSBR, as limited by the TS, provides reasonable assurance that the facility can continue to be operated safely during the period of the renewed license.

4.6 Thermal-Hydraulic Design

The PSBR has a licensed maximum steady-state operating power of 1 MW(t) and is cooled by natural convection. The limiting parameters related to the structural integrity of the fuel are peak fuel temperature and cladding temperature. The reactor is designed to maintain the peak fuel temperature at less than 1,150 degrees C (2,100 degrees F), while limiting the cladding temperature to 500 degrees C (930 degrees F), as specified in TS 2.1. Above this temperature, hydrogen pressure within the fuel rod may reach a level that could compromise the fuel cladding integrity. The licensee presented results from calculations in the literature that demonstrate that the steady-state maximum fuel element power allowed by TS 3.1.5 is at least 50 percent below that which would result in a critical heat flux. As long as it is not exceeded, cladding temperatures will remain below the value of 500 degrees C (930 degrees F) referenced in the basis for TS 2.1. For pulse operations, the licensee presented results from calculations in the literature that demonstrate that the maximum cladding temperature reaches only 180 degrees C (360 degrees F) for a pulse with a peak fuel temperature of almost 1,000 degrees C (1,830 degrees F), well below the SL.

The NRC staff reviewed the thermal-hydraulic data and analyses presented by the licensee and found that the PSBR thermal-hydraulic characteristics will ensure fuel integrity under all analyzed conditions. The NRC staff also found that the related limits in the TS ensure that the critical heat flux will not be exceeded and that cladding temperatures will be consistent with those assumed in the basis for the SL. Based on these findings, the NRC staff concludes that the thermal-hydraulic design, as limited by the TS, is adequate for continued safe operation of the PSBR.

4.7 Conclusions

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 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. The nuclear and thermal-hydraulic design and operating limits required by the TS are adequate to ensure fuel integrity. Therefore, the NRC staff concludes that continued operation of the PSBR within the limits of the TS and the facility license will not result in undue risk to the health and safety of facility personnel, the public, or the environment.

5 REACTOR COOLANT SYSTEMS

5.1 Summary Description

The PSBR reactor coolant system uses light water in an open concrete pool to provide natural convection cooling of the reactor core. The pool water is filtered and demineralized using a cleanup system designed to maintain conductivity at less than 5 microsiemens per centimeter (μS/cm). A 1-MW heat exchanger in the secondary cooling system is capable of removing the heat from full-power reactor operation. The cooling system maintains the pool water temperature sufficiently cool to minimize evaporative losses and to prevent thermal degradation of the demineralizer system.

5.2 Primary Coolant System

Section 5.2 of the PSBR SAR describes the design and construction of the primary coolant system. The primary coolant system allows continuous full-power operation of the reactor while maintaining the pool water within an acceptable temperature range, as specified in TS 3.3.6, "Coolant Temperature Limits." The primary purpose of the temperature limit of 60 degrees C (140 degrees F) specified by TS 3.3.6 is to preserve the integrity of the demineralizer resin in the cleanup system. By procedure, an alarm at this temperature requires corrective action by the reactor operator.

Short-term operation of the PSBR is possible without the heat removal capability of the primary coolant system because of the large heat sink capacity of the 2.7×10^5 l (71,000 gallons) of pool water. Confirmatory calculations by the NRC staff estimate a heatup rate of approximately 3 degrees C/hour (5.4 degrees F/hour) for the bulk pool temperature, thereby allowing several hours of operation with no cooling of the pool before reaching the temperature limit specified in TS 3.3.6. An adequate pool level, as required by TS 3.3.1, provides for this cooling capacity, as well as adequate shielding and LOCA protection. Section 11.1.1.3 of the PSBR SAR discusses in detail radiation levels above the pool during full-power operation. These levels are considered acceptable and, in conjunction with the radiation exposure control program discussed in Section 11.1.5 of the PSBR SAR and the ALARA program required by TS 3.6.4, "As Low As Reasonable Achievable (ALARA)," should ensure personnel exposures remain below the limits in 10 CFR 20.1201, "Occupational Dose Limits for Adults." Section 13.1.3 of the PSBR SAR provides a detailed LOCA analysis and demonstrates no loss of fuel integrity for the worst case credible LOCA scenario. The initial pool level assumptions in the LOCA analysis are consistent with TS 3.3.1 mentioned above and TS 3.3.2, which requires a pool level alarm and corrective action if the level falls below 26 cm (10.2 in.) below full pool level.

Administrative controls and design features minimize the potential for a loss of coolant from the pool. Isolation valves in the demineralizer room allow pool isolation from the primary coolant and cleanup systems. Section 4.3 of this SER discusses penetrations of the pool walls from these systems. A worst case siphoning of the pool water caused by a break in one of these systems could only siphon to the lowest point in the demineralizer room, which would leave approximately 1.2 m (4 ft) of water above the reactor core. In this case, reactor shutdown required by TS 3.3.1 and continued coverage of the core would prevent a loss of fuel integrity. In the event of pool leakage, TS 3.3.4 requires a makeup water source of at least 378 l/min (100 gpm). The NRC staff finds that these design features make it very unlikely that a LOCA could uncover the core.

Section 5.4 of this SER discusses the chemical environment of the primary coolant system. TS 3.3.5 requires the electrical conductivity of the primary coolant to be less than 5 μS/cm, and TS 4.3.3, "Pool Water Conductivity," contains an appropriate surveillance requirement on this limit. This environment maintains within an acceptable level the corrosion of the fuel, core components, and the primary coolant loop structure, which are constructed of aluminum and stainless steel, and minimizes the potential for activated contaminants to become a radiological hazard.

TS 3.6.1, "Radiation Monitoring Information," requires radiation monitoring above the pool area, and TS 3.3.3, "Fission Product Activity," requires an air particulate monitor that can detect fission product leakage from the fuel. These requirements provide further assurance of adequate shielding protection and early detection of a loss of fuel cladding integrity.

The surveillance requirements in TS 4.3.1; TS 4.3.2, "Pool Water Temperature"; TS 4.3.3; TS 4.3.4; and TS 4.6.1, "Radiation Monitoring System and Evacuation Alarm"; require periodic measurement, inspection, calibration, and operability checks for the parameters addressed. These requirements are consistent with the guidance for coolant system surveillance provided in ANSI/ANS-15.1 and NUREG-1537. TS 5.6, "Reactor Pool Water Systems," requires the reactor core to be cooled by natural convection water flow.

The NRC staff evaluated the primary coolant system and determined that the licensee presented sufficient evidence to demonstrate that heat can be removed from the fuel under all possible operating conditions to preclude loss of fuel integrity from failure related to thermal stress. As discussed in Section 5.4 of this SER, the NRC staff determined that the TS requirements on coolant chemistry are adequate to protect fuel integrity and other reactor components from corrosion damage. The NRC staff found that the TS require pool water levels and radiation monitoring that are adequate to shield personnel from direct radiation from the core and maintain personnel exposures below the limits in 10 CFR 20.1201. Based on these findings, the NRC staff concludes that continued operation in accordance with the TS provides reasonable assurance that the primary system can perform all intended functions, as described in the PSBR SAR.

5.3 Secondary Coolant System

Section 5.3 of the PSBR SAR describes the design and construction of the secondary coolant system. This system is designed to transfer a heat load of at least 1 MW(t) from the primary coolant system to the environment. The secondary cooling water is pumped from Thompson Pond, a small body of water near the PSBR site, through the primary-to-secondary heat exchangers and then to a storm sewer that returns the water to Thompson Pond. As discussed in the NRC staff's environmental assessment of the license renewal, the PSBR maintains the appropriate permit for this water usage.

The secondary system is designed to operate at a higher pressure than the primary system to prevent or acceptably mitigate the potential for an uncontrolled release of reactor pool water to the unrestricted environment through a primary-to-secondary leak. Additional design features isolate the primary side when the systems are shut down to prevent any potential leakage to the secondary system. Instrumentation monitors the primary-to-secondary differential pressure and sends an alarm message to the operator console if the differential pressure falls below an acceptable level. Since a secondary-to-primary leak may increase the pool water level, a high-level switch in the reactor pool alarms at the operator control console to indicate a potential leak.

The licensee uses periodic samples to monitor the secondary coolant for radioactivity, and an auxiliary operating procedure specifies the frequency of sampling, limits, and required corrective actions. According to the licensee, concentration limits specified in the procedure are sufficiently low that any leakage of pool water to the environment would not exceed the concentration limits for disposal by the release into sanitary sewerage specified in 10 CFR 20.2003, “Disposal by Release into Sanitary Sewerage.”

The NRC staff evaluated the PSBR secondary coolant system and found that its design and heat exchanger capacity, as described in the PSBR SAR, are adequate for removing any heat load required by reactor operation. The NRC staff found that the system design and constraints on the heat exchanger’s differential pressure are appropriate to minimize the potential for leakage of pool water to the secondary system. Additionally, the NRC staff found that the licensee’s monitoring of secondary water for radioactivity should allow sufficient time for corrective action to mitigate any leakage. Based on these findings, the NRC staff concludes that the secondary system is adequate for the continued safe operation of the PSBR and should not pose a significant risk of an uncontrolled release of radioactive material into the environment.

5.4 Primary Coolant Cleanup System

Section 5.4 of the PSBR SAR describes the design and construction of the primary coolant cleanup system. The design basis of this system is to help preclude corrosion of the fuel and other core and primary system components, as well as prevent activated contaminants from becoming a radiological hazard. The primary coolant cleanup system uses particulate filters and a demineralizer to maintain the conductivity below 5 μS/cm, as specified in TS 3.3.5. Conductance probes monitor demineralizer inlet and outlet conductivity and provide a high-conductivity alarm in the control room. TS 4.3.3 requires daily measurement and recording when the reactor is operated and monthly intervals when shut down. The conductivity limit and surveillance frequency are consistent with similar licensed nonpower reactors and are considered acceptable for monitoring coolant conductivity at the PSBR.

The TS do not contain specific pH limits for the primary coolant. However, the mix of demineralizer resins used in the cleanup system maintains a slightly acidic pH to minimize corrosion of the aluminum and stainless steel components in the reactor pool and primary system. According to the licensee, the average pH reading over the 3-year period of October 2002 through September 2005 was 5.43, with a low of 5.12 and a high of 5.71 during the period. This pH range is consistent with the recommended operational parameters found in the literature for minimizing corrosion in aluminum and low-temperature stainless steel systems. A PSBR procedure also requires periodic samples of pool water to monitor pH, gross radioactivity, and tritium.

Pool water quality is normally very high, such that radiation levels in the vicinity of the demineralizer are not significant. Moreover, the demineralizer room is accessible only to authorized reactor staff, occupancy time is minimal, and occupancy time could be further restricted if unusual radiation levels were present. The demineralizers are currently replaced approximately every 5 years. The spent resin and particulate filters are treated, packaged, and disposed of as solid radioactive waste.

The NRC staff evaluated the PSBR primary coolant cleanup system and found its design comparable to those at similar research reactors. The NRC staff found that the coolant chemistry limits are consistent with the values recommended in the literature for the types of materials in contact with the primary coolant. Section 5.2 of this SER addressed the potential for leaks from the system and found this did not pose a significant hazard. Based on these findings, the NRC staff concludes that continued operation of the primary coolant cleanup system, in accordance with the TS, should maintain corrosion of the fuel, core components, and the primary coolant loop at acceptable levels.

5.5 Primary Coolant Makeup Water System

Section 5.5 of the PSBR SAR describes the design and construction of the primary coolant makeup water system. Primary coolant is lost from the reactor pool by evaporation. Sources of makeup water include recycled condensate from the reactor bay air conditioner, demineralized water from the liquid waste evaporator system, the university water system, and water from Thompson Pond. Under normal circumstances, makeup water other than recycled air conditioner condensate would be processed through the primary coolant cleanup system. Under emergency conditions, water could be added directly to the reactor pool from Thompson Pond through a fire hose connection to the primary system, from the university water system through pool floor drains, and from the university water system using a fire hydrant or pump truck. TS 3.3.4 requires the availability of at least 378 l/min (100 gpm) from any of the emergency supplies. TS 4.3.1 contains surveillance requirements for the inspection of two dedicated fire hoses for emergency makeup water supply. The NRC staff considers these TS requirements adequate to ensure that makeup water will be available to compensate for minor leaks to maintain an acceptable pool level. Section 13.3 of this SER addresses a loss of coolant from a severe accident.

5.6 Nitrogen-16 Control System

Radiation exposure from N-16 in the reactor bay is mitigated by the use of a pump to diffuse the N-16 generated near the reactor into the bulk pool water. This allows sufficient time for the N-16 (7-second half-life) to decay before reaching the pool surface. The pump automatically energizes at approximately 200 kW(t) and does not significantly affect the natural convection cooling of the core.

According to the PSBR SAR, during the years from 2000 to 2004, dosimetry measurements for the control room and reactor bay indicated quarterly averages of 0.067 mSv (6.7 mrem) and 0.33 mSv (33.3 mrem), respectively. Higher levels were recorded in 2004, reflecting experiments conducted in that year, in which the N-16 diffuser pump was turned off. According to the licensee, significant N-16 levels are not present in the primary or demineralizer systems because of the decay time in the bulk pool water before entering these systems.

TS 3.6.1 requires operable radiation monitoring equipment in the reactor bay during reactor operation. TS 4.6.1 ensures that the operability requirements will be met during reactor operation. Radiation exposure is managed under the universitywide radiation protection program, and the TS contain appropriate requirements and surveillance for maintaining personnel doses consistent with ALARA principles. Based on the historical dose data and the radiation monitoring requirements specified in the TS, the NRC staff concludes that the design features and administrative controls associated with the N-16 control system provide reasonable assurance that exposure from N-16 will be minimized and will not pose a significant risk to PSBR personnel.

5.7 Conclusions

Based on the above considerations, the NRC staff concludes that the design of the PSBR cooling systems, as described in the PSBR SAR, is adequate for the removal of heat generated during continuous full-power reactor operation and for the removal of decay heat after shutdown from extended full-power operation. The NRC staff concludes that the coolant systems contain appropriate features to protect personnel from excessive radiation hazards, minimize corrosion of system components and fuel, and prevent or detect losses of coolant. The NRC staff further concludes that the coolant systems in place at the PSBR are sufficient for continued safe operation of the reactor within the related limits of the facility license and TS.

6 ENGINEERED SAFETY FEATURES

6.1 Confinement

The PSBR engineered safety feature (ESF) consists of the reactor bay confinement. In the event of an accident, the confinement system would mitigate an inadvertent release of radioactivity to the environment. Analyses in Chapter 13 of the PSBR SAR indicate that even without the benefit of the confinement system, doses from the MHA would be below the regulatory limits for members of the public specified in 10 CFR 20.1301, "Dose Limits for Individual Members of the Public."

The reactor bay of the PSBR serves as a confinement designed to limit the exchange of effluents with the external environment through controlled pathways. The reactor bay has a minimum volume of 1,933 cubic meters (68,260 cubic feet) and is maintained at a negative pressure by either the facility exhaust system (FES) or the emergency exhaust system (EES). TS 5.5, "Reactor Bay and Exhaust Systems," contains the design requirements for the reactor bay and exhaust systems. Surveillance requirements in TS 4.4, "Confinement," and TS 4.5, "Facility Exhaust System and Emergency Exhaust System," ensure that reactor bay doors are properly secured and checks are performed to verify operability of the exhaust systems.

During normal operation, the FES controls the flow of air through the reactor bay to minimize radiation exposures to workers from radioactive gases, primarily argon-41 (Ar-41). The system consists of two exhaust fans and associated ductwork. TS 3.5, "Engineered Safety Features—Facility Exhaust System and Emergency Exhaust System," requires operation of at least one of the fans when the reactor is not secured or when irradiated fuel or a fueled experiment with significant fission product inventory is being moved outside containers, systems, or storage areas. Leakage—for example, around doors—supplies fresh air to the reactor bay. Fans on the roof of the facility exhaust air out of the reactor bay. Historical measurements discussed in Section 11.1.1.1 of the PSBR SAR show an average annual concentration of Ar-41 of less than or equal to 2.7 percent of the air effluent concentration limit in 10 CFR Part 20, Appendix B.

The EES is actuated manually or automatically whenever the evacuation alarm is initiated. The evacuation alarm can be initiated by any one of three reactor bay radiation monitors, two air monitors, or the Co-60 irradiator bay or neutron beam laboratory radiation monitors. The evacuation alarm secures the FES and actuates the EES. Air enters the EES through a screened opening in the reactor bay wall and then passes through a prefilter, an absolute filter, and a carbon filter before being exhausted through the stack.

The MHA analyzed in Chapter 13 of the PSBR SAR assumes the rupture of a fuel element cladding in air, releasing volatile fission products to the reactor bay. One of the air or radiation monitors in the reactor bay would detect the accident, thereby initiating an evacuation alarm and actuating the EES. According to the licensee, the EES would rapidly remove the activity from the reactor bay, and 92 percent of the TEDE in the unrestricted area would be received in the first hour. Calculations by the licensee show that the dose to the PSBR staff for a 1-minute evacuation time is about 10.38 mSv (1,038 mrem) TEDE, and the 24-hour TEDE dose to the unrestricted area is about 0.26 mSv (26 mrem), both of which are below the respective regulatory limits of 10 CFR 20.1201 and 10 CFR 20.1301. The dose to the unrestricted area is conservatively calculated without credit for the carbon filter in the EES. Successful filtering by the carbon filter would further mitigate the release of radioactive material to the environment and significantly reduce the potential offsite dose consequences. Testing, surveillance provisions and intervals, and related TS ensure that, if required, the confinement ESF will be available and operable to mitigate accident consequences. Based on the above discussion, the NRC staff finds that the system design and related TS demonstrate that the confinement ESF will control radiological exposures or releases to within the exposure limits of 10 CFR Part 20 and ALARA.

6.2 Conclusions

Based on the above discussion and findings and the evaluation of the MHA presented in Chapter 13 of this SER, the NRC staff concludes that the confinement design and related TS provide reasonable assurance that airborne radioactive material generated at the PSBR during normal operation and potential accidents will not pose a significant risk to facility personnel, the public, or the environment.

7 INSTRUMENTATION AND CONTROL SYSTEMS

7.1 Summary Description

Chapter 7 of the PSBR SAR describes the instrumentation and control (I&C) systems. PSU installed the current system in 1991 as a replacement for the original GA TRIGA system and further updated the hardware and software in 2004. According to the licensee, it has designed the control systems to operate in four different modes: manual, automatic, square wave, or pulse. The console uses a hardwired analog system for safety-related functions. All nonsafety-related functions are implemented with computer controls. All safety functions are met with hardwired analog equipment, with those functions then duplicated in the software system.

Two independent systems provide the safety, protection, control, and monitoring functions. The RSS provides the scram and operational interlock functions required by the TS. The RSS is hardwired, making no use of any software-programmable equipment or devices containing embedded microprocessors. The second system is the protection, control, and monitoring system (PCMS), which is fully computerized using digital control and monitoring. The PCMS uses two computers, Digital Control Computer X (DCC-X), with its interface equipment, and Digital Control Computer Z (DCC-Z). DCC-X performs nonsafety-related functions necessary for operation. DCC-Z has no control function and only provides a monitoring function.

The RSS and PCMS receive data on the reactor power level through two separate instruments. A fission chamber is used for wide-range power monitoring and has a range of 10^{-10} percent to 200 percent of 1 MW(t). A gamma ion chamber (GIC), with a range of 1 to 120 percent of steady-state power, provides input to the pulse power monitor to a maximum of 2000 MW(t). Two thermocouples installed in an instrumented fuel element also provide input to the RSS and PCMS. Analog display devices wired directly from the analog reactor power and temperature instrumentation show reactor conditions. In addition, the display of the DCC-X computer provides the operator with information about the state of the reactor.

7.2 Design of Instrumentation and Control Systems

Section 7.2.1 of the PSBR SAR contains the design criteria for the PSBR I&C systems, as follows:

(1) The RSS is separated from the PCMS through use of buffered devices and by physical separation to the extent possible within the console.

(2) The RSS is completely hardwired and does not contain any software programmable devices with embedded microprocessors for signal processing or actuation functions.

(3) The RSS logic is designed to fail safe on loss of power.

(4) Any enhancements, such as redundancy, to the safety functions or reactor protection functions are done through the PCMS via a DCC-X SCRAM input.

(5) The safety functions are designed to meet the single failure criterion for failures in the RSS crediting both the operable portions of the RSS and the PCMS to mitigate the failure consequences.

(6) The PCMS, consisting of the DCC-X and its interface equipment and DCC-Z, is designed to fail conservative through use of extensive self-tests and a watchdog.

(7) DCC-Z, the monitoring computer, does not perform any control actions and is buffered from the control computer, DCC-X, by use of one way data communications. All connections to external monitoring computer systems are via DCC-Z and hence these systems are also buffered from the PCMS DCC-X computer.

(8) DCC-X is designed to provide all reactor protection, control and monitoring functions necessary for safe operation.

(9) Where practical, design of human interfaces employ consistency in operation methodologies, equipment organization, labeling schemes, etc. to maintain an ergonomic interface for operation and maintenance.

A review of the PSBR SAR by the NRC staff indicates that the SAR text acceptably reflected the above design criteria relating to design bases, system descriptions, and performance objectives of the I&C systems.

I&C-related TS are derived from the analysis in Chapter 13 of the PSBR SAR, which describes the systems credited to prevent or mitigate releases. The accidents analyzed in Chapter 13 of the PSBR SAR include an MHA, a loss of coolant, and reactivity insertions. An associated consequence analysis includes crediting selected controls (e.g., equipment and safety management programs) to prevent or mitigate the release. The credited controls form the bases for the selection of TS. Some of the credited controls are fuel cladding and fuel properties, pool water, radiation monitors, fuel temperature and power monitoring and interlocks, control rods, maximum power operating history, emergency exhaust system (filters, fans, and motor start controls), confinement, equipment response to loss of power, and controls on the type of experiments.

Table 7-1 below summarizes applicable I&C equipment and performance objectives contained in the limiting conditions for operation (LCOs) specified by the TS. The requirements of the TS provide reasonable assurance that the systems will perform their safety functions when actuated.

Table 7-1 TS Summary

TS Number	Description	Safety Function
3.2.3	LCO—Required operability requirements for applicable operating modes: Element temperature Linear power (power and wide range) Log power Period Pulse peak power	Provide parametric signals to control systems and interlocks
3.2.4	LCO—Specify minimum number of channels and interlocks that must be operable: <u>Channels</u> Fuel temp High power (two required) Detector power supply Scram bar on console Preset timer Watchdog circuit <u>Interlocks</u> Source level Pulse mode inhibit Transient rod Shim, safety, and regulating rod Simultaneous rod withdrawal	Provides sufficient assurance that I&C channels and interlocks are available to perform applicable safety functions
3.3.2	LCO—Detection of leak or loss of coolant (26 cm pool level drop)	Ensures sufficient coolant is available to protect the safety limit
3.3.3	LCO—Requires an air particulate monitor to operate in the reactor bay	Ensures detection of fission product in the reactor bay and provides alarm and time for protective action
3.3.5	LCO—Coolant conductivity (5 μS/cm)	Minimizes activation of contaminates and precludes corrosion of fuel cladding
3.3.6	LCO—Primary coolant maximum temperature limit (60 degrees C (140 degrees F))	Maintains coolant temperature to protect demineralizer resins
3.6.1	LCO—Requires number of operable channels for the area radiation monitor, continuous air monitor, and beamhole laboratory monitor	Ensures sufficient radiation monitoring information is available for operator action, if radiation is detected
3.6.2	LCO—Requires the evacuation alarm to be operable	Ensures personnel are alerted and evacuated when a potential hazard exists
3.6.3	LCO—Ar-41 concentration limit	Protects the public from Ar-41 exposure

The TS contain surveillance intervals for testing and calibrating I&C system components and functions. The NRC staff reviewed these intervals and found them to be consistent with the guidance found in ANSI/ANS-15.1 and the intervals used at similar research reactors. Based on this finding, the NRC staff concludes that the specified intervals provide reasonable assurance that I&C component failure and degradation will be detected in a timely manner and that specified calibration frequencies are adequate to prevent significant drift in instrument setpoints and detection ranges.

7.3 Reactor Control System

Chapter 7.3 of the PSBR SAR describes the reactor control system. The PCMS controls the reactor. The RSS shares sensors and signals with the PCMS. All signal connections between the RSS and the PCMS are buffered through relays for digital signals or isolators for analog signals. The PCMS validates operation of the RSS by performing the same logic as the RSS. If there is a failure of that validation, a scram request is issued to the RSS, causing a scram. The primary function of the PCMS is to provide the reactor operator with the information and control capability to safely operate and shut down the reactor. Instruments include sensors and controls associated with power, period, cooling system, ventilation, radiation, and experimental facilities. Sensors and processors include detection and control of flux level, wide-range fission detection (source level to 200 percent full power), pulse power, linear power, fuel temperature, control rod permissives, control rod interlocks, and safety trips. There are four modes of operation: manual, auto, square wave, and pulse.

The control system includes an automatic reactor power reduction feature initiated by high fuel temperature, high reactor power (reactor power high), excessive power spread (between the fission chamber and the GIC) and reactor operation inhibit. Power is reduced by driving all rods, except for the transient rod, into the core at high speed. Reactor operation inhibits include the following conditions: the keyswitch is off, radiation hazard from the neutron beam ports exists, both east and west bay or air radiation trips are defeated, pool temperature is high, and the reactor bay truck door is open.

The control system requests the following scrams to the RSS system:

- high reactor power (fission chamber or GIC)
- pulse timer timed-out
- high radiation from any of seven monitors in the reactor bay or the neutron beam laboratory
- opening of the reactor bay truck door
- both east and west facility exhaust fans off
- interlock validation failure
- rod velocity signal failure
- rod motor overspeed
- square wave termination request
- emergency evacuation button
- remote scram pushbutton (one of four)

The control system includes a “watchdog” function that is wired to the RSS. The circuit includes a timer and redundant relays. The control system self-checks and, if a check fails or the software seizes, the timer times out and the relays open and initiate a scram. The control system also includes input/output cards that signal conditions to both the processor and the actuation devices. The cards also include isolation devices to prevent unacceptable interaction with the RSS.

Manual controls exist for the east and west facility exhaust fans, the N-16 diffusion pump, the closed-circuit television camera and monitor for the neutron beam laboratory, and the pneumatic tube system described in Section 10.2.6 of this SER.

TS related to the PCMS contain safety requirements and parameters based on the analysis in the SAR. Associated surveillances provide confidence that the reactor I&C systems will perform when required.

The NRC staff evaluated the reactor control system described in the SAR, conducted a site visit to observe the I&C equipment, and draws the following conclusions:

- There are sufficient channels, range, and sensitivity to detect reactor power and period over all ranges of operation at the PSBR to safely operate and shut down the reactor.
- The TS requirements and surveillances provide adequate assurance that the reactor control I&C systems will actuate when required.

The SAR analyzes all normal operating mode characteristics of the reactor facility and establishes the relationships between the operation modes and the PCMS functions. The PCMS provides all necessary information to the operator and control mechanisms to maintain control over the full range of normal operations, for each mode of operation.

Based on its review of the information presented in the PSBR SAR, the NRC staff concludes that the PCMS provides adequate sensor accuracy and reliability. The NRC staff determined that the PCMS provides sufficient, redundant interlocks to control and limit radiological releases to the reactor operator staff and the public, including the potential for a single active component failure.

7.4 Reactor Protection System

Chapter 7.4 of the PSBR SAR describes the RSS, which comprises the following hardware:

- a fission chamber (FC), a pre-amp, and a signal processor that measure power over a 10-decade range with log of fraction of full power, linear (linear on the PCMS displays, percent full power on a square root scale bargraph on the wide-range monitor front panel), and log-rate signal outputs
- a GIC with its amplifier that provides linear power and pulse range power and a fuel temperature signal processor that provides two fuel temperature signal outputs
- a hardwired RSS that uses relay logic to perform a scram
- operational interlock functions

The RSS includes instrumentation to provide the following information to the operator:

- a measure of the flux level and rate at the wide-range fission detector from source level (shutdown) to 200 percent of full-power reactor operation
- a measure of the pulse power at the GIC from 0 to 2000 MW(t)
- a measure of the linear power at the GIC from 0 to 120 percent of full power
- a measure of the fuel temperature in degrees C
- the permission to operate control rods
- all required safety trips and operational rod interlocks

According to the licensee, the equipment is designed to provide reliable neutron flux measurement from reactor shutdown to reactor full-power level (10 decades). It is designed to measure neutron flux with the detector in a high-gamma-radiation and electrical noise environment, to measure the fuel temperature, and to provide permission and annunciators for rod movement. The RSS is entirely hardwired analog equipment and has no embedded microprocessors.

The scram and control logic assembly contains relays necessary for safety trips from the wide-range and power-range channels, from manual scrams, and from automatic scram functions provided by the PCMS. It also contains relays necessary for reactor rod movement operation, both manually and automatically. A scram is carried out by releasing the control rods from their drive mechanisms, allowing them to drop under gravity into the core. The regulating, shim, and safety rods are connected to their drives by electromagnets. The transient rod is coupled to its drive by air pressure applied to its cylinder through a solenoid valve. The scram logic opens relay contacts to deenergize the electromagnets and the solenoid valve.

As described in Section 7.4.2 of the PSBR SAR the SCRAM logic fails safe on loss of power or open circuit. According to the licensee, two separate circuits contain the voting logic for redundant parameters so that any single short between two points in the system can affect only one of the redundant parameters. The PSBR SAR describes the arrangement of parameters into the two circuits is as follows:

SCRAM Circuit #1	SCRAM Circuit #2
DCC-X Watchdog Trip	DCC-X SCRAM
FC Power High	GIC Power High
FC Bias Voltage Low	GIC Bias Voltage Low
Manual SCRAM	Keyswitch Off
Fuel Temperature High	2 Spare Scrams
Pulse Timer SCRAM	
2 Spare Scrams	

Interlocks are in place for the transient rod system to ensure appropriate operational conditions before air pressure is available to the system. The transient rod scram overrides all other permissive conditions for operation of the system. According to the licensee, the interlock logic is designed to fail safe on loss of power. Contacts must close to allow the application of air to the transient rod. Air may be applied to the transient rod if the transient rod is at its lowest point, if air has been previously applied, if the reactor is in pulse mode and the reactor is at a low initial power, or if the reactor is in square-wave mode.

Control logic for the regulating, shim, and safety rods is similar. Interlocks for these rods are a combination of redundant software interlocks and inhibit blocks from the motor controllers. The reactor is scrammed on interlock validation failure. The software interlocks in the PCMS make use of separate end-of-travel contacts that are configured to fail safe on loss of power.

TS 3.2.3, "Reactor Control System," and TS 3.2.4 provide key safety requirements and parameters based on the safety analysis. Associated surveillances specified in the TS provide confidence that I&C systems will be maintained such that they can perform their intended functions when required.

The NRC staff evaluated the RSS described in the PSBR SAR and conducted a site visit to observe the I&C equipment. Based on its review and firsthand observations, the NRC staff found that there are sufficient channels to detect reactor power and fuel temperature and to actuate the scram functions, and these channels have appropriate range and sensitivity. The NRC staff found that the transient rod and rod drive permissive interlocks preclude unanalyzed rod motions. Additionally, the NRC staff found that the RSS is sufficiently isolated from the PCMS and that the RSS is sufficiently redundant to shut down the reactor in the event of a single active component malfunction. Based on these findings and the requirements contained in the related TS, the NRC staff concludes that the design and operation of the RSS are adequate to ensure that the RSS will automatically shut down the reactor when required.

7.5 Engineered Safety Feature Actuation Systems

As discussed in Section 6.1 of this SER, the reactor bay confinement is the ESF in the PSBR. The two systems that provide reactor bay confinement are the FES and EES. The objective of these systems is to mitigate the consequences of the release of airborne radioactive materials. TS 3.5 requires that, whenever the reactor is not secured, at least one of the two facility exhaust fans be in operation. During normal operation, the FES runs continuously and requires no actuation other than energizing the system before reactor operation. Upon initiation of any alarms from TS-required radiation monitors or a signal from the manually operated evacuation initiate button on the reactor console, the PCMS secures the FES and actuates the EES. A remotely located EES control panel also allows actuation of the system and monitoring of system status.

The NRC staff reviewed the information presented in the PSBR SAR and found that the redundant ESF actuation systems provide for automatic and manual actuation at diverse locations. The NRC staff found that the credited controls are robust and environmentally qualified to function if ESF actuation is required to mitigate radioactive releases as described in the SAR. Based on these findings, the NRC staff concludes that the ESF actuation systems are acceptable.

7.6 Control and Console Display Instruments

According to the PSBR SAR, the control console houses both the RSS and PCMS with the exception of field transducers and actuator devices. The console comprises left-wing, center, and right-wing sections, with the wings angled inward to aid in reading their displays from a position in front of the center section. The left-wing section contains the RSS analog displays and testing controls. The center section contains the DCC-X and DCC-Z display monitor and keyboard interfaces and the rod control switches. The right-wing section contains the hardwired alarm lights and infrequently used switches. The control console provides displays of TS parameters and is accessible for calibration, checking, and self-testing functions. The console includes locking functions to preclude unauthorized operation. Two matching, freestanding control room racks contain the radiation monitoring equipment, neutron beam laboratory closed-circuit television displays, and other equipment.

The NRC staff compared the general arrangement and types of controls and displays provided by the control console to those at similar research reactors and found that the designs are similar. The NRC staff observed the control console during a site visit and found that it provides the reactor operator with the types of information and controls necessary to facilitate reliable and safe operation of the reactor. Based on these findings, the NRC staff concludes that the control console is acceptable for continued operation of the PSBR.

7.7 Radiation Monitoring Systems

Section 7.7 of the PSBR SAR describes the radiation monitoring systems at the PSBR. Eight radiation monitors provide inputs to the PCMS. All but two monitors (air east and air west) have readout modules located in an instrumentation panel in the control room. For these monitors, an alarm results in a warning light on the monitor, an alarm message from the PCMS, and an evacuation initiation issued by the PCMS. Radiation at the alert level is also indicated by a warning light on the monitor, and a message through PCMS, but without the evacuation signal. The air east and air west monitors have local bells and flashing red light alarms that the control room operator can observe. Bar or trend graphs on the PCMS provide current radiation readings for all eight monitors. The 5-kW uninterruptible power supply (UPS) provides electrical power.

TS 3.3.3 requires at least one air particulate monitor to be operable to detect fission product materials that would indicate a fuel element leak. TS 3.6.1 specifies the number of monitors that must be operable for the reactor to be operated. This LCO ensures that reactor operators have radiation monitoring information to enhance their awareness of radiation safety while the reactor is operating.

The NRC staff reviewed the description of the radiation monitoring systems in the PSBR SAR and found that the types of monitors and detection ranges are appropriate for the expected radiological environment at the PSBR. The NRC staff evaluated the monitor locations and found that they are at the locations where radiation levels would be expected during normal reactor operations. The NRC staff also finds that these locations are appropriate for the experiments conducted and the potential upset conditions at the PSBR. Based on the above findings, the NRC staff concludes that the radiation monitoring systems are adequately designed to provide information about the magnitude of the radiation fields of greatest interest in the reactor building and to alert personnel to the existence of any abnormally elevated radiation fields.

7.8 Conclusions

Based on the above discussion, the NRC staff concludes that the nuclear and nonnuclear I&C systems are adequately designed and implemented to provide safe and reliable startup, operation, and shutdown of the reactor during normal operation. The NRC staff concludes that the RSS is adequate to protect the SL on fuel temperature and maintain the reactor in a state as analyzed in the accident analysis. The NRC staff also concludes that the radiation monitoring system is adequately designed, that monitors are appropriately located for the potential hazards at the PSBR, and that the system provides reasonable assurance that facility personnel will be aware of area radiation levels.

8 ELECTRICAL POWER SYSTEMS

8.1 Normal Electrical Power Systems

Section 8.1 of the PSBR SAR describes normal electrical power for the facility. The Allegheny Power Company supplies electricity to the facility through a three-phase transformer located on-site. A UPS acts as a power filter and supplies power to equipment needed for reactor operation. This equipment includes radiation monitors throughout the facility, the reactor console, the rod drive motors and magnets, and auxiliary equipment. Large electrical equipment and cables are located away from I&C sensors, conductors, and actuation devices to minimize electromagnetic interference. A diesel generator starts automatically to supply backup power to the UPS and other equipment if normal facility power fails. The diesel generator provides power for continued operation of radiation monitoring equipment, reactor instrumentation, and facility equipment. A four-battery bank contained in the UPS provides additional short-term backup power.

The TS contain no direct requirements for normal electrical power during reactor operation. Multiple TS contain equipment operability requirements for reactor operation that necessitate the availability of normal electrical power. If normal power and backup power fail, the control rod magnets would deenergize, dropping the control rods into the core and safely shutting down the reactor without operator action. The TS contain no requirements for building power or diesel generator backup power when the reactor is secured. The NRC staff found that this is consistent with the guidance contained in ANS/ANSI-15.1. The NRC staff compared the PSBR normal electrical power system to those in place at other research reactors and found it to be similar in design and operation.

8.2 Emergency Electrical Power Systems

The PSBR does not have any TS requirements for emergency electrical power. Electrical power is not required to shut down the reactor, nor is it required to maintain the reactor in a safe shutdown condition. Equipment that is independent of facility power supplies makeup water for maintaining an adequate water level in the reactor pool. As discussed in Chapter 13 of this SER, the licensee did not take credit for the mitigating effects of the emergency exhaust system in the analysis of the MHA, and the maximum calculated doses resulting from the MHA are within the limits specified in 10 CFR Part 20. Accordingly, since emergency power is not required to maintain doses to members of the public or facility personnel below the regulatory limits, the NRC staff concludes that emergency electrical power is not required for the PSBR.

8.3 Conclusions

The NRC staff reviewed the design bases and functional characteristics related to the normal and backup power supplies at the PSBR. On the basis of its review, the NRC staff concludes that the normal electrical power system at the PSBR facility provides reasonable assurance of adequate operation. The NRC staff further concludes that emergency electrical power is not required for safe reactor shutdown or prevention or reduction of uncontrolled releases of radioactive material to satisfy the dose requirements of 10 CFR Part 20.

9 AUXILIARY SYSTEMS

9.1 Heating, Ventilation, and Air Conditioning Systems

Section 9.1 of the PSBR SAR describes the design and construction of the heating, ventilation, and air conditioning systems (HVAC) at the PSBR. A dedicated reactor bay air conditioner heats and cools the air in the reactor bay and control room. This unit also recirculates and dehumidifies reactor bay air, providing a habitable environment for equipment and personnel. The air from this unit is not mixed with air from inside any other part of the reactor building or from outside the building. Steam unit heaters provide supplemental heating, as needed. The condensate from the reactor bay air conditioner can be piped into the reactor pool as an additional source of makeup water.

The neutron beam laboratory uses a separate air conditioner and heating system. Neither heating nor cooling is provided to the demineralizer room. University power plants located at the east and west ends of the campus supply steam for the heating system. The reactor bay and neutron beam laboratory HVAC systems are independent of the reactor bay facility exhaust system.

During a site visit, the NRC staff observed that the reactor bay HVAC equipment is physically separated from the EES. This provides confidence that a mechanical malfunction of the HVAC equipment would have no impact on EES operation. According to the licensee, operation of the reactor bay HVAC system would have no adverse effects on the distribution of airborne radioactive materials in the case of a release to the reactor bay. The NRC staff agrees with this assertion, because the reactor bay HVAC system does not create a new discharge pathway to the environment. Because of their low safety significance, there are no TS related to the HVAC systems.

9.2 Handling and Storage of Reactor Fuel

New fuel is physically examined upon receipt and checked for any surface contamination by the Radiation Protection Office (RPO). Accepted fuel is then placed into appropriate storage locations within a controlled access area. Only trained and qualified operators can manipulate fuel, using a specially designed tool. According to the licensee, the fuel handling tool design contains mechanisms to indicate that the fuel is secured to the tool. These mechanisms protect against dropping fuel elements and reduce the likelihood of mechanical damage to the fuel during handling. In accordance with the PSBR radiation protection program, PSU uses ALARA practices to receive, inspect, and store fuel.

TS 5.4, “Fuel Storage,” requires that the k_{eff} of stored fuel elements be less than 0.8 for all allowed configurations and conditions of moderation. The PSBR uses fuel storage racks based on a GA design to meet the 0.8 k_{eff} requirement. The NRC staff finds the TS requirement acceptable to ensure that adequate margin to criticality exists for all fuel in storage. Natural convection cooling by water provides adequate cooling for any stored fuel. TS 3.3.5 requires controls on coolant chemistry to limit corrosion of the fuel cladding. The coolant chemistry controls are appropriate to ensure that corrosion will be minimal and will not cause unacceptable degradation of the fuel cladding that could lead to a release of radioactive material from the fuel. Additionally, limiting the corrosion-induced dispersion of activation products into the coolant helps maintain personnel doses ALARA.

TS 3.4, “Confinement,” TS 3.5.b, and TS 4.4 give LCOs and surveillance requirements that ensure confinement integrity and operability whenever the reactor is not secured, or when the licensee moves fuel or a fueled experiment with significant fission product inventory outside containers, systems, or storage areas. TS 4.1.3, “TRIGA Fuel Elements,” provides surveillance requirements for inspection of fuel elements being placed in the core for the first time, periodic inspections while fuel is in use, and inspection upon removal of fuel from service. These requirements are consistent with the guidance of ANSI/ANS-15.1 and with TS requirements at other research reactors that use similar fuel elements and similar fuel storage methods. Accordingly, the NRC staff finds them acceptable.

Based on the above discussion, the NRC staff finds that the controls to receive, store, and transfer fuel are adequate to preclude inadvertent criticality and unauthorized fuel movement and to minimize the risk of mechanical or chemical damage to the fuel. Based on these findings, the NRC staff concludes there is reasonable assurance that handling and storage of fuel at the PSBR will not pose a significant risk to the health and safety of the public or facility personnel.

9.3 Fire Protection Systems and Programs

As described in the PSBR SAR, the reactor building is constructed of concrete block, structural steel, bricks, aluminum panels, and reenforced concrete. These materials are fireproof in nature. During a site visit, the NRC staff observed minimal flammable material in the reactor bay.

The fire protection program consists of both detection and mitigation equipment and includes extinguishers, smoke alarms, pull stations, smoke detectors, and fuse-activated sprinklers. The NRC staff observed this equipment throughout the facility. Smoke detector alarms indicate in the control room, lobby entrance, and off site. Alarms are powered by normal and backup power supplies. The NRC staff finds that the detection and alarm systems are adequate to provide confidence that a fire within the facility will be readily detected and that facility personnel will be promptly alerted. The Alpha Fire Company of State College, located approximately 2.5 km (1.5 mi) from the reactor building, provides the facility with fire fighting support. A fire hydrant is located outside the reactor site boundary fence, approximately 100 m (330 ft) from the southwest corner of the building.

The reactor is controlled by four control rods, three of which are held out of the core by electromagnets. The other rod is held out by compressed air supplied through an electrically operated solenoid valve. As described in Chapter 8 of this SER, in the event of a power loss (caused by fire or otherwise), the rods would drop into the core, safely shutting down the reactor without operator action.

TS 6.3, “Operating Procedures,” requires the facility to have a procedure for fire or explosion. A PSBR emergency procedure fulfills the requirement. The procedure provides guidance to the reactor operator staff for response to a fire alarm and classifies all rooms in the building according to their potential fire hazard. The procedure also describes the locations of alarm pull stations, smoke detectors, sprinkler systems, and fire extinguishers. The training plan for operators and senior operators at the PSBR requires an annual oral exam on all emergency procedures.

Based on firsthand observations by the NRC staff and the above discussion, the NRC staff concludes that the fire protection program is adequate to detect and combat fires. The NRC staff concludes that fire will not prevent the reactor from shutting down nor cause a fire-related release of radioactive material to the unrestricted environment greater than that considered in the MHA. Additionally, the NRC staff concludes that the facility contains adequate controls to initiate personnel evacuation in the event of a fire.

9.4 Communications Systems

Telephones are in place throughout the facility. According to the PSBR SAR, the telephone system has a dedicated UPS system capable of powering the phones for short periods of time in the event of a building power failure. Communication over the facility's public address (PA) system is possible using these phones. According to the licensee, the main facility UPS powers the PA system, and therefore communication using the control room microphone should be available at all times. The building evacuation alarm operates over the PA system. TS 3.6.2, "Evacuation Alarm," requires the building evacuation alarm to be operable and audible to all facility personnel whenever the reactor is operating. Telephones independent of the facility telephone system are also available in the reactor control room and other areas of the building and operate on telephone company voltage. Two-way radios provide an additional means of communication among reactor operations staff and other support organizations during an emergency.

Based on the above discussion, the NRC staff finds that the PSBR has diverse and redundant means of communication within the facility and offsite. Based on this finding, the NRC staff concludes that the communications systems are adequate to provide communications between the control room and all other PSBR locations, communications between operators and health protection staff, and communications to summon emergency assistance.

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

A PSBR administrative procedure describes materials considered to be under Facility Operating License No. R-2 and areas where these materials may be located. These materials are primarily reactor fuel, reactor core components and support structures, the neutron startup sources, and other materials transported to and from specific facility areas designated in the PSBR procedure. Section 4.2.1 of this SER discusses reactor fuel. Section 4.2.4 of this SER discusses the neutron startup sources. Other materials may include customer samples awaiting shipment and transfer to customer licenses, or an experimental apparatus used in the reactor or neutron beam laboratory that needs to be taken to other areas for research and development or maintenance and repair.

The university also possesses a broad byproduct material license separate from the reactor license. Samples activated by the reactor may be transferred to the byproduct license. Authority for approval of the use of radioactive materials under the university licenses rests with the University Isotope Committee (UIC). Byproduct material from the reactor is only released to a person having a valid UIC authorization or to another NRC license. The RPO monitors the storage and use of all radioactive material at the PSBR, including material under the reactor license. Section 9.6.2 and Chapter 11 of this SER address radioactive liquid waste.

The PSBR facility license permits the receipt, possession, and use of limited quantities of SNM in the form of 9 kilograms (~20 pounds) of contained uranium-235 at enrichments less than 20 percent for reactor fuel and up to 50 grams (~1.8 ounce) of high-enriched uranium (HEU) for fission detectors and fission foils. TS related to SNM include fuel element surveillance requirements (TS 4.1.3), fuel design features and storage (TS 5.1 and TS 5.4), staffing requirements during fuel handling (TS 6.1.3, "Staffing"), procedural requirements (TS 6.3), and requirements on records (TS 6.7.1, "Records to be Retained for at Least Five Years").

The NRC inspection program verifies that the licensee properly uses and maintains procedures related to SNM. The staff most recently reviewed selected procedures related to SNM during an inspection in August 2008 (NRC Inspection Report No. 50-5/2008-201, ADAMS Accession No. ML082130382). The inspection concluded that the licensee satisfied the procedural requirements.

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

9.6 Other Auxiliary Systems

9.6.1 Air Compressors

Two air compressors supply compressed air for all facility needs. In the normal configuration, the large air compressor supplies air to the reactor transient rod line and to the line that goes to the remainder of the building, with the small compressor in a standby mode. Should the large compressor fail to operate, the small compressor operates automatically to supply all transient rod and building air needs. The small compressor can also be used if the other compressor is taken out of service. Air from both compressors is filtered for particulates and oil and is treated by an air dryer. System components are automatically relieved of moisture accumulation. The control room contains an indicator of low air pressure.

The staff has reviewed the facility compressed air system and backup. The NRC staff found that the design of the air compressor system is consistent with other research reactors and appears adequate to perform its function. Furthermore, the operation of the air compressors should not degrade the function of the RSS, cause an accident, or result in the uncontrolled release of radioactive material to the environment. Because compressed air is not needed to prevent malfunctions or reactor accidents, initiate safe reactor shutdown, or prevent an uncontrolled release of radioactive material, the NRC staff concludes the system is acceptable.

9.6.2 Evaporator—Liquid Radioactive Waste Treatment

Section 9.7.2 of the PSBR SAR describes the evaporator system. The evaporator is used to remove water from liquid radioactive waste, minimizing the quantity of material that must be disposed of as waste. According to the PSBR SAR, the facility does not normally generate liquid waste in quantities that need evaporation. City water is processed through the evaporator to provide a source of distilled makeup water for the reactor and Co-60 pools. On occasion, reactor pool water that remains in the holdup tank following pool water transfers is pumped to the evaporator building floor tank and later evaporated for pool makeup water. Waste residue from the evaporation process would be removed from the evaporator, further solidified as needed, and disposed of by the RPO.

The NRC staff acknowledges that the evaporator is not normally used at the PSBR. However, the controls to operate the evaporator and dispose of solid low-level waste are adequate when implemented as described in the ALARA policy, in Section 11.1.3 of the PSBR SAR. Operation of the evaporator system will not result in an accident, interfere with the safe shutdown of the reactor, or cause the uncontrolled release of radioactive material to the environment.

9.6.3 Reactor Bay Overhead Crane

The reactor bay is equipped with a 3-ton overhead crane supported by the building structure. The major function of the crane is to move experiments or experimental facilities, such as the fast flux tube (FFT) and FNI shield plugs. It can also be used to position the pool divider gate. PSBR standard operating procedures require the reactor to be moved away from the above-mentioned experimental facilities when the shield plugs are removed, to eliminate radiation streaming from the reactor core. This action minimizes the risk of dropping an object or failing the crane structure with the reactor under the crane.

The NRC staff reviewed the design and operation of the crane and found it adequate to perform lift functions at the PSBR. Furthermore, procedures for use of the crane identify the conditions under which it may have an effect on the reactor and provide appropriate limitations to give reasonable assurance that the reactor is protected from operation of the crane.

9.7 Conclusions

Based on the above discussions, the NRC staff concludes that the auxiliary systems at the PSBR support safe operation of the facility and aid in the safe shutdown of the reactor. The TS provide reasonable assurance that fuel elements are appropriately handled and that there is no significant risk to the health and safety of the public from the storage and movement of fuel.

10 EXPERIMENTAL FACILITIES AND PROGRAMS

10.1 Summary Description

The PSBR is used primarily for education and research. Examples of current applications are neutron activation analysis, neutron radiography, neutron radioscopy, neutron transmission measurements, and neutron depth profile measurements. The reactor is suspended in a pool and can be moved to different locations and rotated to different axial orientations. Experimental facilities allow radiation from the reactor to interact with different materials at different positions in the pool. The TS provide limitations for the effect on reactivity of all experiments and the means for technical and safety reviews of experiments and experimental facilities.

10.2 Experimental Facilities

The list below includes experimental facilities at the PSBR:

- seven beam ports
- the D_2O thermal column
- vertical irradiation tubes
- the FFT
- the FNI
- the central thimble
- a pneumatic transfer system
- an instrument bridge
- two hot cells
- an Ar-41 production facility
- nonreactor irradiation facilities

10.2.1 Beam Ports

Section 10.2.1 of the PSBR SAR describes the design and construction of the beam ports. The PSBR contains seven beam ports, ranging in size from 8 to 18 cm (3 to 7 in.) inside diameter and are all located at the south end of the reactor pool. Five of the beam ports are directed toward a point offset from one of the normal core locations. The other two beam ports are directed toward a position farther into the pool. The beam ports are at three different elevations corresponding to the center of the fuel length, the bottom of the fuel, and 2.5 cm (9 in.) below the fuel.

At the time of the NRC staff's review, only two beam ports were configured for use. The other beam ports were sealed and shielded to protect personnel. Six of the seven beam ports have doors of steel and poured lead. A stepped aluminum shield filled with high-density concrete provides additional shielding. The seventh beam port is configured with a three-beam collimator, with each beam 1 cm^2 (0.16 in^2) in cross section. Neutrons passing through Beam Ports 4 and 7 must first pass through the D_2O thermal column described in the following section of this SER. Neutrons to Beam Port 7 also pass through a graphite box adjacent to the D_2O thermal column that is used to increase the neutron-to-gamma ratio.

10.2.2 Deuterium Oxide Thermal Column

Section 10.2.2 of the PSBR SAR describes the design and construction of the D_2O thermal column. This structure is an aluminum tank located in the reactor pool. The tank is fixed in position and cannot be moved. The reactor can be moved so that it is adjacent to the tank. This configuration optimizes the thermal neutron flux through Beam Port 4. The tank is connected by aluminum tubes to an expansion tank at the surface of the reactor pool. At this location, operators can check the level of fluid in the D_2O tank and identify any leaks. The D_2O thermal column is filled and emptied according to procedures and in the presence of health physics personnel.

The licensee has analyzed leakage of D_2O into the reactor pool and found that it decreases excess core reactivity. Operation of the reactor adjacent to the D_2O thermal column has a minor effect on core properties. Changes in the safety rod worth, as compared to a central pool location, are less than 5 percent and are accounted for in the determination of the shutdown margin required by TS 3.1.3.

After first passing through an adjacent graphite box, Beam Port 7 receives thermal neutrons from the D_2O thermal column. This box is pressurized with air to prevent the infiltration of pool water and has a receiver location for a pneumatic transfer system.

10.2.3 Vertical Irradiation Tubes

Section 10.2.3 of the PSBR SAR describes the design and construction of the vertical irradiation tubes. The PSBR facility has a number of vertical irradiation tubes that can be used for experiments that must remain dry. These tubes are constructed of aluminum and are available in a number of sizes, with an inner diameter of up to 15 cm (6 in.). Additional tubes are available with a rectangular cross-section. The tubes are weighted to ensure they do not float, and some are designed to attach to the reactor support plate or the instrument bridge. The tubes have bends, plugs, or both, to minimize radiation streaming. Two tubes are permanently mounted in the reactor core fuel region. A jib crane mounted on the reactor bridge supports the shielding plugs used in the vertical tubes.

10.2.4 Fast Flux Tube and Fast Neutron Irradiator

Section 10.2.4 of the PSBR SAR describes the design and construction of the FFT and the FNI. Both the FFT and the FNI are located at the north end of the reactor pool, and procedures control the use of either device.

The FFT is a fixed tube with lead, boron, and cadmium shielding around a 15-cm (6-in.) inner diameter irradiation space. The annular shielding material is designed to minimize the gamma and thermal neutron component of the incoming radiation. The FFT contains a shield plug to minimize radiation streaming to the surface of the pool and also inhibits air movement, minimizing the production of Ar-41.

The FNI was designed to irradiate large silicon wafers. It has an inner diameter of 25.4 cm (10 in.). However, the surrounding boron and lead shielding are rectangular in cross section, allowing a closer fit with the edge of the reactor. The FNI has asymmetrical shielding with 5 cm (2 in.) of lead on one side and 10 cm (4 in.) on the other, providing more options for irradiation.

10.2.5 Central Thimble

Section 10.2.5 of the PSBR SAR describes the design and construction of the central thimble, located in the core position with maximum neutron flux. It is an aluminum tube in the center of the core that extends up to the reactor bridge. The thimble has several holes that allow water to fill the length of the tube to minimize radiation streaming to the reactor bridge. A cutout section allows experiments to be removed from the thimble below the water level to reduce radiation doses to personnel.

10.2.6 Pneumatic Tube System

Section 10.2.6 of the PSBR SAR describes the design and construction of the pneumatic tube system. This pneumatic transfer system is available to insert samples into the reactor. The sample containers, commonly known as "rabbits," are moved into and out of the core using carbon dioxide gas. Samples can be sent into the reactor for a preset time or by manual control. The pneumatic tubes send and receive rabbits from the laboratory wing of the facility.

10.2.7 Instrument Bridge

Section 10.2.7 of the PSBR SAR describes the design and construction of the instrument bridge. The instrument bridge is an external mast that allows the positioning of experiments in the vicinity of the reactor core. The bridge is constructed of steel beams supported by the pool wall. An aluminum mast is suspended from the bridge into the pool. The overhead crane can removed the instrument bridge.

10.2.8 Hot Cells

Section 10.2.8 of the PSBR SAR describes the design and construction of the two hot cells at the PSBR. These cells are constructed of high-density concrete, and each has an inner volume that is 1.5 m (5 ft) by 2.3 m (7.5 ft) by 4 m (13 ft). Each cell has a lead glass window and can accommodate a 100-curie Co-60 source or equivalent. Master-slave manipulators are used for work in the cells, and additional cranes are available. Access to the hot cells is through locked concrete doors or through shield plugs that require the use of cranes to remove.

10.2.9 Argon-41 Production Facility

Section 10.2.9 of the PSBR SAR describes the design and construction of the Ar-41 production facility. This facility is used to produce quantities of Ar-41 gas for commercial customers. An irradiation chamber adjacent to the core can be filled through a manifold with pressurized Ar-40 gas. The activation product, Ar-41, can then be transferred to a shielded shipping container for transport.

10.2.10 Nonreactor Irradiation Facilities

Sections 10.2.10 and 10.2.11 of the PSBR SAR describe the design and construction of nonreactor irradiation facilities at the PSBR site. The first facility uses Co-60 in the form of aluminum-clad cylinders stored in a water pool. Materials to be irradiated are lowered into the pool. Vertical tubes (similar to the vertical irradiation tubes in the reactor pool) are available for dry irradiations. The second facility is a GammaCell 200 Excel Irradiator, and it is also available for material irradiations.

10.3 Experiment Review

Section 10.3 of the PSBR SAR describes the process for reviewing experiments. TS 4.7, "Experiments," requires authorized reviewers to ensure that the experiment limitations of TS 3.7 are addressed. Furthermore, TS 6.4, "Review and Approval of Experiments," requires that all new experiments be reviewed for TS compliance and compliance with the requirements of 10 CFR 50.59. Level 2 management or a designated alternate must also approve new experiments.

Facility procedures outline the steps required to ensure that all safety requirements are met before irradiation. Areas of experiment review include Ar-41 production, effluent release considerations, and ALARA requirements. Also, experiments are subject to review by the Penn State Reactor Safeguards Committee (PSRSC). TS 6.2.1, "Safeguards Committee Composition," specifies the composition of the PSRSC and requires that PSRSC members have broad expertise in reactor technology and other sciences. At least one member must have health physics experience.

TS 3.7 also includes specific criteria for the evaluation and approval of experiments at the PSBR. These include a reactivity limit for a single secured experiment or for the sum of all experiments of 2.45% $\Delta k/k$. Movable experiments have a reactivity limit of 1.4% $\Delta k/k$. TS 3.7 provides a specific list of failure mechanisms that must be considered in the experiment review process. The licensee evaluates all experiments to ensure that they will not result in the loss of reactor shutdown capability or in the release of fission products, under normal or accident conditions, that could result in offsite concentrations of radioactive material in excess of the limits specified in 10 CFR Part 20.

10.4 Conclusions

Based on the above discussions, the staff concludes that the experimental facility design features and administrative controls governing the PSBR experimental program will minimize the risk of unintended radiation exposure of experimenters and facility personnel. The NRC staff concludes that the review process for experiments and use of experimental facilities provides reasonable assurance that appropriate precautions are taken to minimize the risk to personnel from unintended radiation exposure. Furthermore, the NRC staff concludes that the review process will provide confidence that the use of experiments or experimental facilities will not damage the fuel, will keep releases of radioactive material below regulatory limits, and will not pose a significant risk to the health and safety of the public or facility personnel.

11 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

11.1 Radiation Protection

11.1.1 Radiation Sources

The primary source of radiation is the reactor itself. The reactor core is submerged below grade in the reactor pool, which shields personnel from the radiation. During normal operations, the reactor generates neutrons for a number of research purposes. Beam ports for experiments allow neutrons to pass from the reactor through a side of the reactor pool to experiment areas. A number of other irradiation methods use the radiation from the reactor core. The reactor also has a pneumatic transfer system for in-core experiments, which can create radioactive materials.

The airborne radioactive materials of principal concern generated during reactor operation are Ar-41 and N-16. Argon is a natural component of the atmosphere and becomes activated to Ar-41 upon neutron bombardment. Minimization of Ar-41 production is accomplished by minimizing the circulation of ambient air into areas subjected to neutron bombardment. N-16 is generated by the fast neutron bombardment of oxygen in the reactor pool water. The nitrogen forms bubbles that rise to the surface of the pool. A diffuser is used over the top of the reactor core that delays the rise of any formed bubbles to the pool surface, allowing for decay of the short-lived N-16 before it can expose individuals. While not as volatile as Ar-41 and N-16, tritium is also produced by neutron activation of natural deuterium in the pool water. The tritium in the reactor pool may then evaporate into the reactor bay air, but the dose consequence of tritium produced in this manner will be a small fraction of that of Ar-41 and N-16.

According to Chapter 11 of the PSBR SAR, airborne concentrations of Ar-41 are significantly less than 1 DAC in restricted areas, and effluent concentrations are similarly a small fraction of the air effluent concentration limits specified in 10 CFR Part 20, Appendix B. The PSBR SAR lists data from July 1, 2003, to June 30, 2004, from which the licensee calculated the Ar-41 concentration in the reactor bay at 1.4 percent of the restricted area DAC. For the same time period, the calculated Ar-41 concentration at the facility fence line was 2.7 percent of the air effluent concentration limit in 10 CFR Part 20, Appendix B,. The NRC staff performed independent calculations using the licensee's assumptions and data and found the licensee's calculated Ar-41 concentration at the facility fence to be reasonable.

Liquid radiation sources at the PSBR consist primarily of activation products of the coolant and impurities in the coolant. The principal isotope is N-16, and prolonged operation can result in small amounts of sodium-24 (Na-24). N-16 has a 7-second half-life and is only a radiation hazard during reactor operations or immediately after reactor shutdown. As discussed above, N-16 exposure is controlled by the use of a diffuser system that delays the rise of N-16 to the surface of the reactor pool. Although the reactor pool water is kept quite clean, it may occasionally contain activation products, such as Na-24, from contaminants in the water. A demineralizer maintains water purity. As mentioned above, tritium may be produced in the main pool in small quantities but is limited by the small natural abundance of deuterium in light water. Of more interest is the tritium generated by neutron activation of the deuterium in the D_2O thermal column. This is a sealed system, and the licensee performs periodic sampling and level monitoring to track the activity and inventory of the system. Floor drains from the reactor building drain to either of two storage tanks. These tanks can be processed by the evaporator.

PSBR operations generate solid radioactive materials. Chief among these are the spent fuel assemblies. After irradiation in the core, the spent assemblies are stored in fuel racks. Chapter 9 of this SER discusses spent fuel movement and storage. Other solid radioactive sources include demineralizer resins and filters, reactor components, experiment components from the high flux location, and activated samples. According to the licensee, it disposes of solid radioactive waste in accordance with appropriate NRC regulations and transfers it to organizations authorized to receive the material.

Based on the above discussion, the NRC staff concludes that the description and characterization of the radiation sources at the PSBR is reasonable for a nonpower reactor of this type and size and that this information is sufficient to evaluate the radiation protection program and controls described in the remainder of this chapter of the SER.

11.1.2 Radiation Protection Program

Section 11.1.2 of the PSBR SAR describes the radiation protection program required by 10 CFR 20.1101, “Radiation Protection Programs.” This program includes the stated policy to employ the ALARA concept in all operations at the PSBR.

The RPO conducts radiation protection activities at the PSBR. The RPO is part of the Department of Environmental Health and Safety and ultimately reports to the Vice President for Physical Plant and the Senior Vice President for Finance. This is a separate reporting chain than for reactor operations, which reports to the Vice President for Research and the Dean of the Graduate School. This separate reporting chain is consistent with the guidance in ANS/ANSI-15.1 and provides reasonable assurance that radiation safety decisions are independent of operational considerations.

The RPO consists of the manager of radiation protection, one associate health physicist, three health physics specialists, one assistant health physicist, and administrative support staff. The manager of radiation protection also acts as the university radiation safety officer. The TS require that a health physics representative serve on the PSRSC. The PSRSC reviews procedures for implementing the radiation protection program before their adoption.

All individuals who work with radioactive materials or use personnel monitoring devices receive radiation safety training. The training covers basic health physics principles, as well as the regulations of 10 CFR Parts 19, 20, and 21, and rules regarding the use of radioactive material at PSU. This training is commensurate with the level of activities of the individual and the potential for radiation exposure.

TS 6.7.3, “Records to be Retained for the Life of the Reactor Facility,” requires the licensee to retain records relating to personnel dosimetry or exposure investigations, as well as records of 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 6.7.1, “Records to be Retained for at Least Five Years,” 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; thus, the NRC staff finds this acceptable.

The NRC staff reviewed the structure and strategy of the radiation protection program for the PSBR and found it to be consistent with the guidance of ANSI/ANS-15.11, “Radiation Protection at Research Reactor Facilities,” issued 1993. Based on this finding, the NRC staff concludes that the PSBR 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.

11.1.3 ALARA Program

PSU has a defined ALARA policy for exposure to radiation, established by the UIC. The policy states that the release of radioactive material and the exposure of individuals to ionizing radiation are to be kept ALARA. TS 3.6.4 also requires an ALARA program.

The main tenant of the policy is that releases and individual exposures may not exceed applicable regulatory limits. Furthermore, any releases or unplanned exposures in excess of 10 percent of those limits are to be investigated with an aim of determining actions that could reduce these releases or exposures in the future. Also, any planned release or exposure over 10 percent of appropriate limits requires a prior ALARA review by the RPO. Exposures or releases below 10 percent of applicable limits require no additional ALARA considerations.

These methods are typical for ALARA programs, and the NRC staff concludes that they provide reasonable assurance that the licensee will minimize personnel exposure to radiation and releases of radioactive material.

11.1.4 Radiation Monitoring and Surveying

The RPO maintains numerous fixed and portable radiation detection instruments throughout the PSBR facility. Eight fixed-area radiation monitors are located throughout the reactor bay, the Co-60 bay, and the neutron beam laboratory to alert facility personnel and experimenters to changing radiation conditions. All of these instruments read out in the control room. The monitors have local alarm lights. An alarm on any of the building monitors, except for the monitor on the reactor bridge, results in an automatic evacuation initiation by the control system.

TS 3.6.1 requires that at least two radiation monitors (area radiation and continuous air monitor) in the reactor bay be operable before reactor operations begin. The continuous air monitor measures particulates in the reactor bay air, while the area radiation monitor measures gamma radiation levels. If the neutron beam laboratory is in use, TS 3.6.1 requires an operable radiation monitor at that location as well. A Geiger-Mueller detector on the stack monitors Ar-41 in the effluent. Additional monitoring on an as-needed basis supports nonroutine activities.

Fixed radiation monitors are used in the facility to detect personnel contamination. These contamination monitors are located at the main doors to the reactor building and other locations, as needed. Portable instrumentation is available to survey areas in the PSBR facility for all types of radiation and radioactive contamination that may be present from facility operations. This includes ion chambers and friskers. According to the licensee, it nominally conducts facility surveys on a weekly basis, with more frequent surveys based on work levels and types. Surveys will be performed before allowing activities in those areas not normally in use. This includes the currently unused beam ports in the neutron beam laboratory.

As noted in Section 11.1.1 of this SER, neutron bombardment of deuterium in the D_2O tank generates tritium. The licensee takes samples monthly from the D_2O tank and analyzes them for tritium using liquid scintillation analysis.

In addition to the fixed and portable radiation detection equipment, additional laboratory monitoring equipment is available to support analyses. This includes a National Institute of Standards and Technology-traceable calibration facility that is used for calibrations of area, air, and portable survey instruments. Approved facility procedures control calibration activities.

Based on its review, the NRC staff concludes that the installed and available radiation detection equipment is of the proper type, range, and sensitivity to detect and quantify the types of radiation at PSBR. Furthermore, 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.

11.1.5 Radiation Exposure Control and Dosimetry

The PSBR is located in a fenced area with limited access. The facility fence meets the definition of a controlled area in 10 CFR 20.1003, “Definitions.” This fence is normally locked and access controlled by PSU security personnel. Access to the building requires training appropriate to the expected level of access to radioactive materials.

According to the PSBR SAR, and confirmed by firsthand observations by the NRC staff, all personnel entering the areas where radiation and radioactive material could be present use individual dosimetry. The PSBR uses a contract dosimetry supplier who is certified by the National Voluntary Laboratory Accreditation Program, as 10 CFR 20.1501(c) requires. Dosimeter results are corrected by the vendor for background radiation using control badges. Supplementary dosimetry is available as needed. Procedures require the use of extremity dosimetry for individuals who could exceed 10 percent of the annual extremity limit, in accordance with 10 CFR 20.1501(a). According to the licensee, typical annual radiation doses for individual facility personnel are approximately 0.5 mSv (50 mrem). This is well below the regulatory limit of 50 mSv (5000 mrem) specified by 10 CFR 20.1201.

The licensee does not normally require internal monitoring at the PSBR, but bioassay measurements can be made, if necessary. The PSBR does not use respiratory protection devices to limit radiological uptakes, as engineering controls are sufficient to maintain airborne radioactive material below levels that would warrant their use. This is consistent with Subpart H, “Respiratory Protection and Controls To Restrict Internal Exposure in Restricted Areas,” of 10 CFR Part 20.

Locks control the entry points to high-radiation areas, and the licensee controls access to the keys administratively. Direct surveillance or warning devices may supplement controls for short-term experiments. This meets the requirements of 10 CFR 20.1601, “Control of Access to High Radiation Areas.”

Based on the above discussion and firsthand observations, the NRC staff finds that the exposure control and dosimetry program at the PSBR is adequate to monitor and control exposures to personnel below the limits in 10 CFR Part 20, Subpart C, “Occupational Dose Limits,” and is consistent with the guidance in ANSI/ANS-15.11.

11.1.6 Contamination Control

The PSBR uses staff training and surveys, as needed, to detect contamination. Survey equipment is available for personnel to monitor for contamination. Posting materials are available to control potentially contaminated areas. Decontamination supplies are available to clean up any spilled material. According to the licensee, it nominally conducts routine surveys of the facility on a weekly basis, supplemented as needed, based on the type of activity taking place. The experiment review process identifies experiments likely to generate significant contamination, and the design includes appropriate controls. These types of contamination control measures are common for research reactors similar to the PSBR, and the NRC staff finds them acceptable.

11.1.7 Environmental Monitoring

A commercial vendor provides environmental monitoring for the PSBR. According to the PSBR SAR, environmental monitors are at seven locations around the reactor facility; four at the facility, one at the nearby child care center (155 m (490 ft) west), and two at control locations. The licensee presented quarterly averages for each of 5 years preceding the license submittal. The quarterly doses presented are a fraction of the dose limits for members of the public specified in 10 CFR 20.1301, and fluctuations in the quarterly doses do not correlate with the power history of the PSBR during the listed years.

Based on its review, the NRC staff concludes that the environmental monitoring locations are sufficient to properly characterize the potential public dose from PSBR operations and to demonstrate compliance with the dose limits of 10 CFR 20.1301. Furthermore, the dose rates are consistent with background radiation levels in the Appalachian areas of the United States.

11.2 Radioactive Waste Management

11.2.1 Radioactive Waste Management Program

Section 11.2 of the PSBR SAR describes the radioactive waste management program. All individuals who work with radioactive materials at the PSBR are required to have training approved by the RPO. This training includes instruction on dealing with radioactive waste. Implementation of the ALARA principle, as described in Section 11.1.3 of this SER, also includes minimizing the generation of radioactive waste. The licensee incorporates ALARA reviews into the design of experiments to minimize the unnecessary generation of radioactive material. The RPO oversees the radioactive waste management program. As much radioactive material is stored for decay as practical. The licensee uses waste minimization practices throughout the facility to minimize disposal costs. These practices include the use of materials with low neutron activation potential. The NRC staff reviewed the radioactive waste management program presented in the PSBR SAR and found it to be comparable in scope to programs successfully implemented at similar research reactors. Based on its review, the NRC staff concludes that the program contains appropriate provisions for training, review, and oversight that are commensurate with the types and quantities of radioactive wastes expected as a result of facility operations.

11.2.2 Radioactive Waste Controls

The licensee controls gaseous radioactive wastes (gaseous effluents) by minimizing the production of the waste products and discharging the effluents to the unrestricted area. The only gaseous waste of concern is Ar-41. Chapter 11 of the PSBR SAR analyzes the production of Ar-41 in the reactor pool water and experimental facilities. This analysis includes measurements of Ar-41 production for both low-power and high-power reactor operation. The licensee performs calculations based on these measurements to track the production of Ar-41 within the reactor building and the radiological impact of releasing the effluent. Section 11.2.3 below evaluates the radiological impact of releasing the effluent. The staff evaluated the analysis in the PSBR SAR and found it to be acceptable. The staff found that the licensee's characterization of Ar-41 sources and its method of calculating Ar-41 production provide reasonable assurance that gaseous waste will not pose an undue risk to facility personnel or result in effluent releases above the limits specified in Appendix B to 10 CFR Part 20.

The licensee controls liquid radioactive waste by using the primary coolant cleanup system, described in Section 5.4 of this SER, and sampling the primary coolant for radioactivity. The cleanup system removes activation products, primarily Na-24, from the primary coolant by passing the coolant through demineralizer resins and filters. The licensee restricts access to the demineralizer area to minimize the radiation dose from the activation products retained in the system. The licensee performs periodic sampling of the primary coolant in accordance with PSBR operating procedures. TS 3.3.5 limits the conductivity of the primary coolant during reactor operation to minimize the production of activation products in the coolant. TS 4.3.3 contains surveillance requirements for pool water conductivity. The licensee periodically samples the D_2O thermal column tank to characterize the tritium content of the D_2O.

Solid radioactive waste generated at the PSBR includes spent demineralizer resins and filters, gloves, pads, decontamination materials, and various activation products from the experimental facilities. The licensee collects solid waste materials at the point of generation in marked waste containers. Handling of higher activity material may require the use of radiation work permits and additional review by the RPO. The waste materials are then consolidated with other university radioactive material for final disposition under PSU's broad byproduct material license. Operation of the PSBR also produces high-level radioactive waste in the form of spent fuel. As discussed in Section 1.7 of this SER, PSU maintains a contract with DOE for return of the spent fuel. The staff compared the solid radioactive waste produced at the PSBR to that of similar research reactors and found that the types of waste produced at the PSBR are common for research reactors and can be safely stored and disposed of in accordance with applicable regulations.

11.2.3 Release of Radioactive Waste

Radiation detectors in the reactor bay monitor the gaseous radioactive effluent. TS 3.6.1 requires a continuous air monitor in the reactor bay during operations. TS 3.6.3, "Argon-41 Discharge Limit," and TS 4.6.2, "Argon-41," limit the release of Ar-41 in airborne effluents and require the periodic calculation of the dose from Ar-41 releases. Regulations require this dose calculation for each experiment that may result in a dose greater than 0.01 mSv (1 mrem) at the facility boundary. These calculations are necessary to confirm compliance with 10 CFR Part 20 dose and concentration limits for airborne radioactive materials. The NRC staff independently calculated the Ar-41 releases, based on the licensee's assumptions and data contained in annual reports submitted to the NRC. The NRC staff obtained similar results to those of the licensee. Based on the independent calculations and appropriate requirements in the TS, the NRC staff finds that the release of Ar-41 from the PSBR as an air effluent will be in accordance with the requirements of 10 CFR Part 20.

As discussed in Chapter 9 of this SER, liquid radioactive waste can be disposed of using an evaporator. Disposal by evaporation results in demineralized water and traces of solid radioactive waste that are left in the evaporator tank for decay. The licensee samples and analyzes liquid radioactive waste disposed of by release to the sanitary sewer to confirm that the waste meets 10 CFR 20.2003 requirements for solubility and concentration. These releases must also meet the PSU ALARA policy. Releases to the sanitary sewer require prior RPO approval. Liquid radioactive waste may also be transferred to PSU's broad byproduct material license for final disposition. These methods of disposal of liquid radioactive wastes are common for research reactors similar to the PSBR, and the NRC staff finds them acceptable.

PSU transfers all solid low-level radioactive waste to its broad byproduct material license and returns high-level waste to DOE. PSU packages and transports all waste according to appropriate NRC regulations and the applicable State license of the recipient.

Based on the findings discussed above, the NRC staff concludes that administrative controls, radioactive release methods, and TS are adequate to provide reasonable assurance that the release of radioactive waste from the PSBR will meet the requirements of 10 CFR Part 20 and therefore will not pose an undue risk to the health and safety of the public or the environment.

11.3 Conclusions

The staff concludes that the PSBR radiation protection and ALARA programs, radiation monitoring and surveying, and exposure control and dosimetry are adequate to provide reasonable assurance that doses to facility personnel will be maintained below the regulatory limit and ALARA. The staff concludes that the licensee's environmental monitoring program and radioactive waste disposal methods 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 is adequate to provide 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.

12 CONDUCT OF OPERATIONS

The conduct of operations involves the administrative aspects of facility operation, the facility EP, and facility security. The administrative aspects of facility operations are the facility organization, training, operational review and audits, procedures, required actions, and records and reports.

12.1 Organization

The Director of the PSBR reports to the Dean of the College of Engineering. For matters related to reactor operations and Facility Operating License No. R-2 license, the Dean reports to the Senior Vice President for Research and Dean of the Graduate School, who is the individual responsible for the license. The directorship of the PSBR is one of the broader responsibilities assigned to the Director of the Radiation Science and Engineering Center. TS 6.1, “Organization,” contains an organization chart. TS 6.6.2, “Special Reports,” requires a written report to the NRC within 30 days for changes in the facility organization involving level 1 and level 2 personnel. The Associate Director for Operations reports to the Director of the Penn State Breazeale Reactor. The licensed senior reactor operators and reactor operators report to the Associate Director for Operations for matters pertaining to reactor operations.

The RPO has a reporting chain independent of the reporting chain for PSBR operations. The Manager of Radiation Protection reports to the Director of Environmental Health and Safety, who in turn reports to the Vice President for Physical Plant, who ultimately reports to the Senior Vice President for Finance and Business/Treasurer.

The NRC staff reviewed the organization and finds that it adequately reflects the activities of the entire facility. The radiation protection organization reports separately from the operating organization and has the authority to interdict or terminate activities for reasons related to radiation safety. Responsibilities for the management levels reflect the operation and policies of the reactor facility. TS 6.1.3 adequately identifies the minimum qualified staff to safely operate and shut down the reactor. As required by TS 6.1.4, “Selection and Training of Personnel,” the selection and training of the operating staff is in accordance with ANSI/ANS-15.4, “Selection and Training of Personnel for Research Reactors,” issued 1988.

12.2 Review and Audit Activities

The PSRSC performs the independent review and audit functions. TS 6.2.1 specifies that the PSRSC is to be staffed with technically experienced members from both within and outside PSU and is to advise the Director on all matters of policy pertaining to safety. Members are appointed by the Dean of the College of Engineering, acting for the Senior Vice President for Research and Dean of the Graduate School. The PSRSC Charter and Operating Procedure outlines the composition and qualifications of the committee members. TS 6.2.1 also includes meeting frequencies, quorums, and frequencies of audits, and lists audit activities.

The NRC staff reviewed the structure for the conduct of review and audit activities at the PSBR and found they were consistent with the guidance of ANSI/ANS-15.1. Based on this finding, the NRC staff concludes that the review and audit functions are sufficient to ensure that safety-related matters will be appropriately reviewed.

12.3 Procedures

Written approved procedures govern all aspects of PSBR operation and use. The existence and scope of these procedures, which TS 6.3 specifies, encompass, but are not limited to, the following areas:

- startup, operation, and shutdown of the reactor
- core loading, unloading, and fuel handling
- routine maintenance of major components of systems that could affect reactor safety
- surveillance test or calibrations required by TS
- emergency procedures
- experiment evaluation and authorization
- loss of pool water

The Director approves all procedures. TS 6.2.3 requires that the scope of the PSRSC review include all new procedures or major revisions to procedures having a significant effect on safety. PSU periodically reviews procedures to ensure that their wording is clear and concise and that they remain technically adequate.

The NRC staff finds that TS 6.3 addresses the procedure topics provided in ANSI/ANS-15.1, in addition to other applicable procedures related to operation of the reactor facility. Based on this finding, the NRC staff concludes that the process and methodology described in the PSBR SAR and required by the TS provide reasonable assurance that procedures will be properly controlled and reviewed.

12.4 Required Actions

TS 6.5, “Required Actions,” specifies actions to be performed by the licensee if there is a reportable occurrence or if a safety limit is exceeded. As discussed in other chapters of this SER, the licensee provided information and analyses demonstrating that no credible failure could lead to a fuel temperature equal to the SL specified in TS 2.1. Reportable events are listed in TS 1.1.34, “Reportable Occurrence.” The required actions specified by TS 6.5 include actions by the reactor operator to ensure that the reactor is in a safe condition, as well as reporting and notification and review of the occurrence by the PSRSC. PSBR administrative procedures provide additional guidance on reportable events. These procedures also cover events not listed in TS 1.1.34 but that may be reportable to the NRC, such as a regulatory violation of the limits in 10 CFR Part 20, or the discovery of an operation or system failure reportable under 10 CFR Part 21, “Reporting of Defects and Noncompliance.” The staff evaluated the requirements of TS 6.5 and found that they are consistent with ANSI/ANS-15.1; satisfy the requirements of 10 CFR 50.36, “Technical Specifications”; and provide reasonable assurance that the facility will respond to unanticipated occurrences in a manner that emphasizes reactor safety and the protection of public health and safety.

12.5 Reports

TS 6.6, “Reports” specifies reports that the NRC requires from the licensee. These include an annual operating report and special reports. TS 6.6.1, “Operating Reports,” lists the required contents of the annual operating report, including operational history, major maintenance performed, approved major changes to the facility, and radioactive effluents. TS 6.6.2.a, "Special Reports," discusses how to file special reports for a violation of safety limits, a release of radioactivity from the site above allowed limits, and a reportable occurrence. TS 6.6.2.b requires written special reports for permanent changes in facility management at levels 1 and 2, and for significant changes in the transient or accident analysis, as described in the SAR. The NRC staff reviewed these reporting requirements and found they were consistent with ANSI/ANS-15.1 and provide reasonable assurance that the licensee will report, in a timely manner, appropriate information to the NRC regarding routine operation, nonroutine occurrences, and changes to the facility and personnel.

12.6 Records

TS 6.7, "Records," specifies records that the PSBR is required to maintain in a manner that facilitates convenient review. The licensee specified the types of records that the facility will retain and the period of retention, to ensure that important records will be retained for an appropriate time. The staff evaluated the requirements of TS 6.7 and found that they are consistent with ANSI/ANS-15.1 and provide reasonable assurance that the facility will maintain appropriate records to facilitate NRC inspection, including an adequate history of the facility.

12.7 Emergency Planning

The licensee requested that Revision 4 of the PSBR EP be considered as part of the license renewal application. The NRC staff reviewed the EP 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; 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; NRC Information Notice 92-79, "Non-power Reactor Emergency Event Response," issued December 1992; and the requirements of 10 CFR 50.34(b)(6)(v). The NRC staff concluded that the PSBR EP is in accordance with the guidance and regulations. The licensee has demonstrated the ability to make changes to the EP in accordance with 10 CFR 50.54(q). Accordingly, the NRC staff concludes that the PSBR EP provides reasonable assurance that the licensee can respond appropriately to a variety of emergency situations and that the PSBR EP will be adequately maintained during the period of the renewed license.

12.8 Security Planning

The NRC staff reviewed the licensee's measures for physical security and protection of SNM against the applicable requirements of 10 CFR Part 73, "Physical Protection of Plants and Materials." Additionally, the staff reviewed the licensee's measures using the guidance contained in Regulatory Guide 5.59, "Standard Format and Content for a Licensee Physical Security Plan for the Protection of Special Nuclear Material of Moderate or Low Strategic Significance," issued February 1983. The NRC staff found that the licensee's measures for physical security and protection of SNM meet the intent of the guidance and satisfy the applicable regulatory requirements. Additionally, the NRC routinely inspects the licensee's measures for physical security and protection of SNM to verify that they continue to satisfy the applicable regulatory requirements. In 2007, an NRC inspection found that the licensee's security measures continued to satisfy applicable regulatory requirements and were acceptable. Based on the NRC staff review and routine inspection program, the NRC staff concludes that the licensee's measures for physical security and protection of SNM provide reasonable assurance that continued possession and use of licensed SNM at the facility will not pose an undue risk to public health and safety and will not be inimical to the common defense and security.

12.9 Quality Assurance

The PSBR SAR states that the licensee will maintain procedures and administrative controls that incorporate principles of the guidance found in ANSI/ANS-15.8, "Quality Assurance Program Requirements for Research Reactors," issued in 1995, as they relate to an as-built facility. Section 12.9 of the PSBR SAR contains brief descriptions that address many areas of the guidance. The NRC staff did not perform a detailed review of these descriptions but did review the TS referenced in the descriptions. This chapter and Chapter 14 of this SER primarily describe these reviews.

12.10 Operator Training and Requalification Program

Responsibility for administering the requalification program rests with the Associate Director for Operations or a designate appointed by the Director of the PSBR. The licensee's training and requalification program, required by TS 6.1.4, provides reasonable assurance that the licensee will have technically qualified reactor operators. The licensee included the operator requalification program approved by the NRC in 1997 in the application for license renewal. The NRC staff reviewed the program and found that it meets all applicable regulations (10 CFR 50.54(i–l) and 10 CFR Part 55, "Operators' Licenses") and is consistent with the guidance in ANSI/ANS-15.4.

12.11 Conclusions

Based on the above discussions, the NRC staff concludes that the licensee has the appropriate organization, experience levels, and adequate controls through the TS to provide reasonable assurance that the PSBR is managed and operated in a manner that will not cause significant radiological risk to facility personnel, the public, or the environment.

13 ACCIDENT ANALYSES

13.1 Maximum Hypothetical Accident

The MHA for the PSBR is the rupture of the cladding of an air-cooled fuel element, resulting in the release of volatile fission products into the reactor bay. The fuel element is conservatively assumed to be a 12 wt% U-ZrH fuel element operated in a core position of highest power density with a power history of continuous 1 MW(t) operation. These assumptions will result in the worst case fission product inventory, source term, and potential dose consequences to facility personnel and members of public for the MHA. Radiation and air monitors in the reactor bay would detect the released activity and actuate an evacuation alarm, resulting in isolation of the FES and actuation of the EES, as described in Section 6.1 of this SER. The released activity would be removed from the reactor bay through a pre-filter, absolute filter, and carbon filter before being exhausted through the stack. TS 3.3.3 and TS 3.6.1 require the radiation and air monitors. TS 3.6.2 requires the evacuation alarm.

The licensee calculated the maximum effective doses, assuming a conservative maximum measured fuel temperature of 650 degrees C (1,200 degrees F) and using a release fraction equation based on experimental data from GA, the TRIGA reactor vendor. The dose calculations account for radioactive decay and assume an immediate and complete mixing of the fission product release in the reactor bay and no mitigating plating-out on surfaces or retention by pool water. The dose to the staff for a 1-minute evacuation time is about 10.38 mSv (1,038 mrem) TEDE and 64.15 mSv (6,415 mrem) thyroid committed dose equivalent. The dose calculations for the unrestricted area include a rapid removal of the activity from the reactor bay by the EES, radioactive decay, and a dilution factor to account for the dispersion of the plume. The TEDE dose in the unrestricted area received in the first hour is approximately 0.24 mSv (24 mrem) and the 24-hour TEDE dose to the unrestricted area is about 0.26 mSv (26 mrem). The first hour and 24-hour thyroid dose estimates are 6.96 mSv (696 mrem) and 7.44 mSv (744 mrem), respectively. The dose to the unrestricted area is calculated without credit for the carbon filter in the EES; therefore, the actual dose would be even less. The NRC staff performed independent calculations to confirm that the PSBR doses represent conservative estimates for the MHA. The NRC staff confirmed that the results of the licensee's calculations represent dose estimates for the MHA that are within the occupational dose limits of 10 CFR 20.1201 (5,000 mrem) for the PSBR staff (1-minute evacuation) and the public dose limits of 10 CFR 20.1301 (100 mrem).

The MHA for the PSBR is considered a highly unlikely scenario. Nevertheless, conservative estimates of fission product release and dose estimates for members of the public in the unrestricted area are determined and found to be below regulatory limits. Therefore, even if the MHA should occur, with consequences that bound all credible accidents at the facility, the health and safety of the public are protected.

13.2 Insertion of Excess Reactivity

13.2.1 Step Reactivity Insertion Accident

TS 3.1.4 requires a maximum allowed pulse reactivity of $3.50 to prevent the fuel temperature from exceeding the SL of 1,150 degrees C (2,100 degrees F). This limit is based on the maximum NP of 2.2 and for pulses initiated below 1 kW reactor power. The reactivity limits of TS 3.1.4 and the core configuration limits of TS 3.1.5 ensure that a pulse below the limit will not exceed the fuel temperature SL for any core configuration.

The step reactivity insertion accident for the PSBR assumes an insertion of reactivity of $2.25 from the transient rod at an initial reactor power level of 1.15 MW(t). This is the maximum reactivity remainder available at the initial conditions for the Core Loading # 47 data used in the analysis. According to the licensee, the measured fuel temperature for a B-ring, 12 wt% U fuel element would increase from 650 degrees C (1,200 degrees F) to 1,030 degrees C (1,890 degrees F), as calculated for instrumented fuel element I-15 using historical data. The measured fuel temperature is a reasonable estimate of the maximum fuel temperature for the high-power initial conditions, as discussed in the licensee's analysis. This peak temperature of 1,030 degrees C (1,890 degrees F) is, according to the licensee, for a bounding core loading (Core Loading # 47) and is well below the SL of 1,150 degrees C (2,100 degrees F) specified in TS 2.1.

This accident is further prevented by: (1) PSBR standard operating procedures that require all four control rods to be balanced above the 900 kW(t) power level; therefore, the full worth of the transient rod would not be available; (2) reactor power is limited to a maximum 1.1 MW(t) by TS 3.1.1, "Non-Pulse Mode Operation"; (3) pulse operation is not allowed above 1 kW(t) by TS 3.1.4; (4) high-power scrams required by TS 3.2.4 would prevent operation above 1.1 MW(t); and (5) the pulse mode inhibit interlock required by TS 3.2.4 would prevent pulsing from power levels above 1 kW(t). Nevertheless, the results of the analysis of this accident demonstrate peak fuel temperatures below the SL of 1,150 degrees C (2,100 degrees F), and therefore no failure of the fuel element cladding or fission product release would be expected.

13.2.2 Ramp Reactivity Insertion Accident

The licensee offers a referenced analysis of a ramp reactivity insertion accident for a $5.00 ramp over a 2-second interval performed by GA for the Armed Forces Radiobiological Research Institute (AFRRI). The withdrawal accident is initiated from a subcritical condition, with the accident terminated by a high-power safety system scram at 110 percent of maximum power, or 1.1 MW(t). The prompt temperature coefficient used in the analysis is 1×10^{-4} Δk/k-°C, and the coolant temperature coefficient is assumed to be zero. The rod drop time assumed in the analysis is 1 second from full out to full in, with a delay time of 0.015 seconds to allow the magnetic field coupling the rods to decay. The analysis indicated that a reactivity of $1.86 was added after criticality was achieved and before the scram occurred.

The AFRRI transient analysis uses the following parameters similar to those that would be applicable to the PSBR:

- The scram setpoint is 110 percent of full power (1.1 MW(t)); TS 3.2.4 requires the scram setpoint to be less than 1.1 MW(t).
- The scram time is less than 1 second; TS 3.2.6 requires the scram time to be less than 1 second.
- The prompt temperature coefficient is -1×10^{-4} Δk/k-°C; the PSBR temperature coefficient is -1×10^{-4} Δk/k-°C.
- The prompt neutron lifetime is 39 microseconds; the PSBR prompt neutron lifetime is a more limiting 38 microseconds.

The maximum fuel temperature for this accident in the AFRRI reactor was determined to be 330 degrees C (626 degrees F). Since the $1.86 of reactivity inserted in this accident scenario is significantly below the allowable step insertion limit of $3.50 (TS 3.1.4), there is reasonable assurance that the SL fuel temperature of 1,150 degrees C (2,100 degrees F) would not be exceeded, and therefore, no failure of the fuel element cladding or fission product release would be expected. In addition, TS 3.2.2 restricts the rate of reactivity insertion for a single control rod or a combination of more than one rod to ~$0.90/second when averaged over full rod travel, representing a significantly smaller reactivity addition rate than analyzed in the accident analysis.

13.3 Loss-of-Coolant Accident

The largest conceivable LOCA scenario for the PSBR results from a malfunction in the pool barrier system. Alarm systems would alert the operator to shut down the reactor and initiate corrective action. TS 3.3.2 requires the pool level alarm. The licensee determined the time for the water loss from the pool, and the NRC staff confirmed this estimate by independent calculations. Using data from a set of LOCA experiments conducted for TRIGA reactors by GA, the licensee evaluated the maximum fuel temperatures from air cooling of the core after the pool water loss. Using the conservative operating power history of continuous operation for 1 week at 1 MW(t) before the LOCA, and a maximum power density fuel element of 24.7 kW (TS 3.1.5), the maximum calculated temperature is 468 degrees C (874 degrees F). Therefore, for the most severe LOCA conceivable for the PSBR, the maximum fuel temperature remains below the air-cooled temperature limit of 950 degrees C (1,740 degrees F), and there is reasonable assurance that no failure of the fuel element cladding or fission product release would result. Further analysis provided by the licensee indicates that, even in the event of this worst case LOCA, radiation dose rates at the closest uncontrolled area for a conservative operating history are below 0.10 mSv (10 mrem) per hour. Therefore, sufficient time would be available to evacuate the area in accordance with the EP before unacceptable exposure levels were reached.

13.4 Loss-of-Flow Accident

Since the PSBR is a system cooled by natural convection, this accident is not applicable to the PSBR.

13.5 Mishandling or Malfunction of Fuel

According to the licensee, no fuel elements at the PSBR have leaked fission products. Any accident involving damage to a fuel element would be bounded by the air-cooled rupture of a fuel element analyzed in the MHA.

Criticality safety is controlled by restricting all fuel storage such that the k_{eff} shall be less than 0.80, as required by TS 5.4. This TS also requires storage conditions such that natural convection cooling will prevent the fuel element temperature from reaching the SL.
Based on a review of the TS, the accident analysis, and other information presented in the PSBR SAR, the NRC staff concludes that continued operation within the TS provides reasonable assurance that mishandling or malfunction of the fuel will not result in radiological consequences in excess of those analyzed in the MHA.

13.6 Experiment Malfunction

In Table 4-4 of the PSBR SAR, the licensee provided the reactivity worths of experimental facilities for a typical core loading. In addition, all experiments are subject to the restrictions of TS 3.7, which limit the reactivity worths of all experiments, single experiments, and movable experiments. These limitations preclude a credible accident exceeding the previously analyzed reactivity addition accidents. Further restrictions in TS 3.7 ensure that experiments will have the proper review to meet the requirements of 10 CFR 50.59, including failure mechanisms involving corrosion, overheating, impact from projectiles, and chemical or mechanical explosions. TS 3.7 also contains requirements for experiments that could off-gas, which restrict the airborne concentration from potential accidents averaged over a year to the limit in Appendix B to 10 CFR Part 20. Further restrictions are made for experiments that could potentially contain iodine isotopes to ensure that the consequences of a potential experiment failure would be bounded by the MHA.

The NRC staff evaluated the experimental facilities and provisions for experiment review for the PSBR. Based on its review of the requirements in the TS, the NRC staff concludes that the performance of experiments within the restrictions of the TS provides reasonable assurance that the potential consequences of experiment malfunctions would be less severe than those already evaluated in the MHA and accidents involving the insertion of excess reactivity.

13.7 Loss of Normal Electric Power

The loss of normal power is an anticipated event for the PSBR and would not be expected to cause an accident. Reactor shutdown is passive and fail safe in that if normal, UPS, and diesel generator power are all lost, the control rods automatically fall into the core by gravity, thereby shutting down the reactor. The major methods of adding water to the pool in case of a leak are not dependent on reactor building power. No TS require building power, UPS power, or diesel generator 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, a loss of normal electrical power would not pose a significant risk to the health and safety of the public.

13.8 External Event

The PSBR SAR identifies no external events that could cause damage to the reactor core. Chapter 2 of the PSBR SAR discusses the seismic history of the region surrounding the PSBR and shows that the maximum earthquake potential for the PSBR site is well below the level that would cause damage to the facility. Chapter 2 and Chapter 3 of this SER discuss the meteorological hazards of the reactor site and the design criteria of the reactor SSCs. These discussions show that meteorologically induced damage to the reactor building is very unlikely. Furthermore, as described in Section 13.7 of this SER, a complete loss of electrical power to the reactor building will not prevent safe shutdown of the reactor. No other naturally-occurring external events are associated with the site that could prevent safe shutdown of the reactor or damage to the reactor. Therefore, there is reasonable assurance that no external event would pose an unacceptable risk to the health and safety of the public.

13.9 Mishandling or Malfunction of Equipment

No accident scenarios involving mishandling or malfunction of equipment were identified that could cause consequences not bounded by the previously analyzed accidents.

13.10 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 staff evaluated the licensee's assumptions and methods of calculating doses and found 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 staff evaluated the accident scenarios and assumptions and concludes that the licensee has analyzed an appropriate spectrum of credible accidents for the PSBR and that the MHA bounds the consequences of the credible accidents. The licensee has shown that credible accidents at the PSBR do not have any significant offsite radiological consequences. Accordingly, the staff concludes that accidents at the PSBR will not pose a significant risk to the health and safety of the public, facility personnel, or the environment.

14 TECHNICAL SPECIFICATIONS

The NRC staff evaluated the TS as part of its review of the application for renewal of Facility Operating License No. R-2. The TS define certain features, characteristics, and conditions governing the operation of the PSBR. The TS are explicitly included in the renewed license 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. Other chapters of this SER discuss the evaluations of individual TS. The NRC staff specifically evaluated the content of the TS to determine if the TS meet the requirements in 10 CFR 50.36, "Technical Specifications." The NRC staff concluded that the PSBR TS meet the requirements of the regulations. The NRC staff based this conclusion on the following findings:

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

- The PSBR 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 PSBR SAR.

- To satisfy the requirements of 10 CFR 50.36(c)(1), the licensee provided TS specifying a safety limit on the fuel cladding temperature and limited safety system settings for the RSS to preclude them from reaching the SL.

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

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

- The TS contain design features that satisfy the requirements of 10 CFR 50.36(c)(4). As part of the license renewal, the licensee proposed bases for the TS related to design features. The NRC staff reviewed the proposed bases and found them acceptable and based on analyses contained in the PSBR SAR.

- 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)(1,2,7,8).

The NRC staff finds the TS to be acceptable and concludes that normal operation of the PSBR within the limits of the TS will not result in radiation exposures in excess of the limits specified in 10 CFR Part 20 for members of the general public or for occupational exposures. The NRC staff also finds that the TS provide reasonable assurance that the facility will be operated as analyzed in the PSBR SAR and adherence to the TS will limit the likelihood of malfunctions and the potential accident scenarios discussed in Chapter 13 of this SER.

Based on discussions between the NRC project manager, the facility Director, and the facility Associate Director for Operations, the NRC staff modified several of the proposed TS submitted by the licensee. The modifications included editorial changes for clarity and consistency with the regulations. The NRC project manager explained the modifications to the licensee, and the facility Director and the facility Associate Director for Operations agreed to the modifications. The following list describes the modifications.

- The NRC staff modified the basis of TS 3.5 to clarify that operation of the emergency exhaust system is not necessary to maintain doses to individual members of the public calculated in the analysis of the MHA below the limits of 10 CFR 20.1301.
- The NRC staff removed the term "unreviewed safety question," from TS 3.7, TS 6.6, and TS 6.7, and replaced it with language referring to the requirements in 10 CFR 50.59 for determining whether a change, test, or experiment requires a license amendment. The NRC staff considers these modifications necessary because 10 CFR 50.59 no longer uses or defines the term "unreviewed safety question."
- The NRC staff revised TS 4.2.6 to separate the transient rod system surveillance requirements into three individual specifications. The licensee requested this change to better clarify that TS 4.2.6 contains three distinct surveillance requirements.
- The NRC staff modified TS 4.2.6a to clarify that the transient rod system shall be verified operable on each day the reactor is pulsed. The previous specification was ambiguous because it contained the phrase "pulse mode operation of the reactor is planned." TS 4.2.6a did not distinguish between planned and unplanned pulsing of the reactor and the NRC staff considered it necessary to eliminate the ambiguity.
- The NRC staff modified TS 4.2.6a to replace the phrase "functional performance check," with the phrase "verified operable." This modification provides consistency with the definitions in TS 1.1.
- The NRC staff modified the operating reports requirement in TS 6.6.1e to be consistent with the 0.10-mSv (10-mrem) ALARA dose constraint on air emissions of radioactive material specified in 10 CFR 20.1101(d). Specifically, TS 6.6.1e did not require the licensee to provide details of air emissions of radioactive material unless the concentration of the radioactive material was greater than or equal to 25 percent of the applicable limit specified in Appendix B to 10 CFR Part 20. An annually-averaged concentration of 25 percent of the applicable regulatory limit corresponds to a potential maximum annual dose to a member of the public of 0.125 mSv (12.5 mrem), which is greater than the dose constraint specified in 10 CFR 20.1101(d). The NRC staff changed the air emission concentration threshold in TS 6.6.1e from 25 percent to 20 percent, which corresponds to a potential maximum annual dose to a member of the public of 0.10 mSv (10 mrem) and is consistent with the ALARA dose constraint on air emissions of radioactive material specified in 10 CFR 20.1101(d).
- The NRC staff added TS 6.7.3e to emphasize the records retention requirements in 10 CFR 50.36(c)(1) and (2). The regulations require retention of records of reviews of violations of safety limits, limiting safety system settings, and limiting conditions for operation for the life of the facility. The licensee categorizes violations of limiting safety system settings and limiting conditions for operation as reportable occurrences, and TS 6.7.1f requires retention of records of reportable occurrences for a period of five years. The NRC staff was concerned that without the explicit requirements of TS 6.7.3e, the license might prematurely dispose of records required to be retained for the life of the facility.

15 FINANCIAL QUALIFICATIONS

15.1 Financial Ability to Operate the Facility

In 10 CFR 50.33(f), the following statement is made:

> Except for an electric utility applicant for a license to operate a utilization facility of the type described in §50.21(b) or §50.22, [an application shall state] information sufficient to demonstrate to the Commission the financial qualification of the applicant to carry out, in accordance with regulations of this chapter, the activities for which the permit or license is sought.

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

According to the application, the PSBR is the principal facility within the RSEC and is administered for PSU by the PSU College of Engineering (COE). Operating costs for the PSBR are estimated to range from $1.47 million in fiscal year 2009 (FY09) to $1.79 million in FY13. According to PSU, the projected sources of funds to cover operating costs will be provided by the COE, service activities, and research activities. The NRC staff reviewed PSU's projected operating costs and projected sources of funds to cover these costs and found them to be reasonable.

Based on its review, the NRC staff finds that PSU has demonstrated reasonable assurance of obtaining the funds necessary to cover the estimated operating costs for the PSBR for the period of the license. Accordingly, the NRC staff has determined that PSU has met the financial qualification requirements in 10 CFR 50.33(f) and is financially qualified to engage in the proposed PSBR activities.

15.2 Financial Ability to Decommission the Facility

The NRC has determined that the requirements to provide reasonable assurance of decommissioning funding are necessary to ensure the adequate protection of public health and safety. The regulation in 10 CFR 50.33(k) requires that an application for an operating license for a utilization facility contain information to demonstrate how reasonable assurance will be provided that funds will be available to decommission the facility. The regulation in 10 CFR 50.75(d) requires each nonpower reactor applicant for or holder of an operating license to submit a decommissioning report that contains: (1) a cost estimate for decommissioning the facility, (2) an indication of the funding method(s) to be used to provide funding assurance for decommissioning, and (3) a description of the means of adjusting the cost estimate and associated funding level periodically over the life of the facility. The acceptable methods for providing financial assurance for decommissioning are specified in 10 CFR 50.75(e)(1).

The application, dated December 6, 2005, included a decommissioning cost estimate based on actual detailed costs to decommission the 5 MW Georgia Institute of Technology reactor and estimates to decommission the University of Virginia reactor. The estimated cost to decommission the PSBR was $7,141,464 in 2006 dollars. In a supplement to the application dated October 31, 2008, PSU increased the decommissioning cost estimate to $12,560,130 in FY08 dollars. The decommissioning cost estimate shows costs broken down by labor, waste disposal, and other (equipment and supplies), and includes a 25 percent contingency factor. In reviewing the decommissioning cost estimate by PSU for the PSBR (~12.56 million in FY08 dollars), the NRC staff took into consideration experience at other facilities with similar construction and operational history and concludes that the decommissioning cost estimate for the PSBR is reasonable.

PSU is currently using a self-guarantee to provide financial assurance for decommissioning, as allowed by 10 CFR 50.75(e)(1)(iii) for nonprofit entities such as universities. The regulation states that "…a guarantee of funds by the applicant or licensee may be used if the guarantee and test are as contained in Appendix E to 10 CFR Part 30." PSU submitted a self-guarantee agreement and information showing that the guarantor meets or exceeds the financial test criteria for a nonprofit university that issues bonds.

The 2005 application states that PSU will review and update the Decommissioning Funding Plan cost estimate at least every 3 years to account for changes in waste profiles and inflationary pressures. In the October 31, 2008, submittal, PSU states that it will adjust the decommissioning cost estimate periodically over the life of the facility by adding an annual 5 percent inflation adjustment to the current cost estimate.

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

15.3 <u>Foreign Ownership, Control, or Domination</u>

Section 104d of the AEA prohibits the NRC from issuing a license under Section 104 of the AEA to "any corporation or 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 in 10 CFR 50.38, "Ineligibility of Certain Applicants," contains language to implement this prohibition. According to the application, and supplements to the application, PSU is organized as a nonprofit corporation under the laws of the Commonwealth of Pennsylvania, principally doing business in Pennsylvania. The application includes the names, addresses, and citizenship of PSU's Board of Trustees, all of whom are U.S. citizens. According to the application, PSU is not owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government. The NRC staff does not know or have reason to believe otherwise.

15.4 <u>Nuclear Indemnity</u>

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

15.5 Conclusions

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 PSBR and, when necessary, to shut down the facility and carry out decommissioning activities. In addition, the NRC staff concludes there are no problematic foreign ownership or control issues or insurance issues that would prevent the issuance of a renewed license.

16 OTHER LICENSE CONSIDERATIONS

16.1 Prior Use of Reactor Components

As detailed in previous sections of this SER, the NRC staff concludes that the continued operation of the PSBR will not pose a significant radiological risk. The bases for this conclusion include the assumption that the facility systems and components are in good working condition. PSU must maintain or replace systems and components that perform safety-related functions to ensure that they continue to adequately protect against accidents. Such systems and components found at the PSBR include the fuel cladding and the RSS.

Section 4.2.1 of this SER describes the reactor fuel. As described in the literature, fuel growth and deformation can occur during normal operations. Swelling of the fuel is dependent on the amount of time the fuel spends over a temperature threshold. The threshold temperature for this phenomenon is about 750 degrees C (1,380 degrees F). Pulse mode operation of the reactor could result in fuel temperatures above the threshold temperature. Given that the duration of such temperatures would be very short, pulsing should not cause significant fuel swelling by this mechanism. In addition, TS 3.1.4 limits the maximum pulse reactivity addition to approximately $3.50. This limit is based on preventing the fuel temperature from reaching the SL. Further, TS 3.1.1 limits the nonpulse temperature to 650 degrees C (1,200 degrees F). TS 4.1.3 requires periodic inspections of the fuel elements for gross failure or significant deterioration. As stated in Chapter 4 of this SER, continued operation as limited by the TS provides reasonable assurance that the fabricated fuel can meet the design objective of maintaining fuel integrity and thereby function safely in the reactor without adversely affecting the health and safety of the public.

Additional considerations supporting the continued use of the fuel are described below.

- Natural convection cools the reactor fuel. Natural convection cooling does not generate coolant velocities or pressures that could erode the cladding.
- The design of the in-pool structures and components minimizes the chance for mechanical impact. Reactor components are contained between the top and bottom grid plates. Fuel handling requires specially designed tools that do not come into contact with the cladding. The top grid plate shields the fuel elements from tools and small objects, should they fall into the pool.
- TS 3.3.5 limits the conductivity of the coolant to 5 μS/cm. TS 4.3.3 requires that pool conductivity be measured daily during operations and monthly during shutdown periods. These surveillance requirements ensure that no significant corrosion of the cladding will occur.

The electrical design of the RSS (e.g., safety channel circuitry, control rod magnets) prevents accidents resulting from the failure of system components. As discussed in Chapter 7 of this SER, failure or removal for maintenance of safety-related instrumentation and control components causes a safe reactor shutdown. TS 4.2 specifies surveillance requirements for the reactor control and safety systems. These requirements are consistent with the guidance in ANSI/ANS-15.1 and ensure that gradual degradation of system components will be detected. Additionally, the PSBR staff performs regular preventive and corrective maintenance and replaces system components as necessary. Nevertheless, some equipment malfunctions have occurred at similar facilities. The NRC staff review indicates that most of these malfunctions were one-of-a-kind and typical of even industrial-quality electrical and mechanical instrumentation and components. There is no indication of significant degradation of the instrumentation and components, and there is strong evidence that the PSBR staff will remedy any future degradation with prompt corrective action. The staff did not consider prior use of other systems and components, because degradation would occur gradually, be readily detectable, or not affect the likelihood of accidents.

16.2 Conclusions

In addition to the considerations discussed above, the NRC staff reviewed licensee event reports and NRC inspection reports. On the basis of this review and the preceding considerations, the NRC staff concludes that there has been no significant degradation of facility systems or components. The NRC staff further concludes that the surveillance requirements in the TS provide reasonable assurance that the facility will continue to be adequately monitored for degradation of systems and components.

17 CONCLUSIONS

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

- The application for license renewal, dated December 6, 2005, as supplemented by letters dated October 31, 2008, and April 2, June 11, September 1, and October 21, 2009, complies with the standards and requirements of the AEA and the Commission's rules and regulations set forth in 10 CFR Chapter I, "Nuclear Regulatory Commission."

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

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

- As discussed in Chapters 4, 12, and 15 of this SER, the licensee is technically and financially qualified to engage in the activities authorized by the renewed license, in accordance with the rules and regulations of the Commission.

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

18 REFERENCES

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