Safety Evaluation Report Related to the Renewal of

Facility Operating License No. TR-5 for the

National Bureau of Standards Test Reactor,

National Institute of Standards and Technology

June 2009
Office of Nuclear Reactor Regulation

ABSTRACT

This safety evaluation report summarizes the findings of a safety review conducted by the NRC 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 National Institute of Standards and Technology (the licensee or NIST) for a 20-year renewal of Facility Operating License No. TR-5 to continue to operate the National Bureau of Standards test reactor (NBSR or the facility). The facility is located at the NIST campus in Gaithersburg, MD. 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 NIST can continue to operate the NBSR, in accordance with the renewed license, without posing a significant risk to the health and safety of the public, facility personnel, or the environment.

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LIST OF ABBREVIATIONS

Abbreviation	Definition
ac	alternating current
ACRS	Advisory Committee on Reactor Safeguards
ADAMS	Agencywide Documents Access and Management System
AEA	Atomic Energy Act of 1954, as amended
AEC	Atomic Energy Commission
ALARA	as low as reasonably achievable
ALI	annual limit on intake
AmBe	americium beryllium
ANSI/ANS	American National Standards Institute/American Nuclear Society
Ar	argon
ARM	area radiation monitor
ASME	American Society of Mechanical Engineers
BOC	beginning of cycle
C	Celsius
CDE	committed dose equivalent
CFR	*Code of Federal Regulations*
CHFR	critical heat flux ratio
Ci	curie
cm	centimeter
CP	construction permit
CP-5	Chicago Pile-Five
CsI	cesium iodide
D_2	deuterium
DAC	derived air concentration
dc	direct current
DNB	departure from nucleate boiling
DOE	U.S. Department of Energy
ECC	extended continental crust
ECS	emergency cooling system
EOC	end of cycle
EP	emergency plan
EPZ	emergency planning zone
ESF	engineered safety feature
F	Fahrenheit
ft	foot
ft^3	cubic foot
FY	fiscal year
g	acceleration due to gravity
gal	gallon
gpm	gallon per minute

LIST OF ABBREVIATIONS

Abbreviation	Definition
^3H	tritium
HE	heat exchanger
HEPA	high-efficiency particulate air
hr	hour
HVAC	heating, ventilation, and air conditioning
I	iodine
I&C	instrumentation and controls
I-270	Interstate 270
in.	inch
IRM	Iapetan Rifted Margin
km	kilometer
kPa	kilopascal
kV	kilovolt
kW	kilowatt
L	liter
L/s	liter per second
LCO	limiting condition for operation
LSSS	limiting safety system setting
m	meter
m^3	cubic meter
MCHFR	minimum critical heat flux ratio
MCNP	Monte Carlo N-Particle
MHA	maximum hypothetical accident
MPa	megapascal
mi	mile
mm	millimeter
mrem	millirem
msec	millisecond
mSv	millisievert
MTR	materials testing reactor
MWD	megawatt-day
MW(t)	megawatt thermal
N	nitrogen
n/cm^2	neutrons per square centimeter
NBSR	National Bureau of Standards test reactor
NCNR	NIST Center for Neutron Research
NIST	National Institute of Standards and Technology
NRC	U.S. Nuclear Regulatory Commission
OFI	onset of flow instability
psf	pounds per square foot

LIST OF ABBREVIATIONS

Abbreviation	Definition
psi	pounds per square inch
psig	pounds per square inch gauge
RCS	reactor control system
RPS	reactor protection system
s	second
SAC	Safety Assessment Committee
SAR	safety analysis report
SEC	Safety Evaluation Committee
SER	safety evaluation report
SOI	statement of intent
SSC	structures, systems, and components
SU	startup core configuration
TEDE	total effective dose equivalent
TS	technical specification(s)
U	uranium
UPS	uninterruptible power supply
USGS	U.S. Geological Survey
yr	year
μs	microsecond
μS	microsiemens

1 INTRODUCTION

1.1 Overview

By letter (and supporting documentation) dated April 9, 2004, as supplemented by letters dated October 2, 2006, May 30 and August 14, 2007, September 16, October 21, and December 8, 2008, and March 3, March 19, and April 22, 2009, the National Institute of Standards and Technology (NIST, 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. TR-5 (NRC Docket No. 50-184). The renewed license would authorize continued operation of the National Bureau of Standards test reactor (NBSR) at the NIST Center for Neutron Research (NCNR) located on the NIST campus in Gaithersburg, MD. In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR), Section 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), an environmental report, 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, and/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. If you do not have access to ADAMS or have problems accessing the documents located in ADAMS, or if you want access to documents dated before November 24, 1999, contact the NRC Public Document Room Reference staff 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 NUREG-1873, "Environmental Impact Statement for License Renewal of the National Bureau of Standards Reactor," issued December 2007 (ADAMS Accession No. ML072970861), will serve as the basis for issuance of a renewed license authorizing operation of the NBSR at power levels up to 20 megawatts thermal (MW(t)). 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, 73, and 100; 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.

Mr. 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 Mr. Marvin M. Mendonca, Mr. William C. Schuster, Mr. Michael B. Norris, Mr. Paul V. Doyle, Mr. Ronald B. Uleck, and Ms. JoAnn Simpson of the NRC staff.

Under contract to the NRC, Mr. William Watkins, Mr. James Willison, and Mr. 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 NBSR 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 is acceptable to maintain and safely operate the reactor. The licensee's management organization, training and research activities, and security measures continue to be acceptable to ensure safe operation of the facility and the protection of its special nuclear material.

- The licensee and NRC staff have conservatively 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 are less than the doses specified in 10 CFR Part 20, "Standards for Protection Against Radiation," for facility personnel and members of the general public and satisfy the regulatory dose requirements of 10 CFR Part 100, "Reactor Site Criteria."

- Exposures from and releases of radioactive effluents and wastes from the facility are not expected 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 are consistent with as-low-as-reasonably-achievable (ALARA) principles.

- The renewed facility operating license and TS, which state limits controlling 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 NBSR 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 NIST can continue to operate the NBSR in accordance with the Atomic Energy Act of 1954, as amended (AEA, the Act), NRC regulations, and Facility Operating License TR-5 without endangering the health and safety of the public.

1.3 General Facility Description

The NBSR is a heavy-water (D_2O)-moderated-and-cooled, enriched-fuel, tank-type reactor designed to operate at 20 MW(t) power. It is a custom-designed variation of the Argonne Chicago Pile-Five (CP-5) class reactor. It differs from the CP-5 in its power rating, core configuration, and cold neutron source, but retains the proven technology. The three most notable modifications to this basic design are (1) an 18-centimeter (cm) (7-inch (in.)) gap between the upper and lower fuel regions in each fuel element to reduce the fast neutron background in the neutron beams, (2) a double plenum at the bottom of the vessel to provide optimized cooling to the core, and (3) the method for remote handling of fuel elements during refueling. This type of reactor was chosen because of its well-thermalized neutron spectrum, high neutron flux, flexibility for research, and inherent safety.

The NCNR has a wide range of research capabilities. The liquid hydrogen cold source provides cold neutrons (neutrons slowed to speeds of 1,000 meters per second (m/s) (2,200 miles per hour (mi/hr)) or less) directly to experiments in the confinement building and, through a network of seven neutron guides, to experiments located in the cold neutron guide hall. Beam tubes provide thermal neutrons for experiments located within the area immediately adjacent to the reactor. A pneumatic sample transfer system gives researchers the ability to automatically inject samples into the core region of the reactor, while vertical thimbles provide locations for manual loading. The reactor uses U_3O_8 aluminum dispersion fuel clad in aluminum. Heavy water provides neutron moderation and core cooling. The closed primary coolant system circulates heavy water in an aluminum and stainless steel system. The heavy water pumped through heat exchangers transfers its heat to light water (H_2O) before reentering the core and returning to the pumps. The light-water secondary cooling system transfers its heat to the atmosphere by means of evaporation from a cooling tower located outside of the confinement building.

1.4 Shared Facilities and Equipment

The confinement building, constructed of reinforced concrete and situated partially below grade, adjoins a laboratory complex dedicated primarily to research related to nuclear science and other reactor support functions. The confinement building has an independent ventilation control system, capable of either an isolation mode of operation, or a dilution mode of operation when air is exhausted to the atmosphere through an elevated stack located adjacent to the building. Local utilities provide municipal water and sewerage, natural gas, and electricity to the NCNR.

1.5 Comparison with Similar Facilities

The NBSR design is a variation of the Argonne CP-5 class reactor. The NBSR uses materials testing reactor (MTR)–type plate fuel, which is also in use at the Massachusetts Institute of Technology reactor, a 5-MW(t) tank-type research reactor. The Brookhaven National Laboratory and the Savannah River Site reactors also used similar fuel. The NBSR does not have any unique features that would preclude applying general knowledge and experience gained in the operation of these other reactors to operation of the NBSR.

1.6 Summary of Operation

The licensee has operated the NBSR in accordance with Facility Operating License No. TR-5 and established procedures to provide research services to government and commercial scientists. Research activities primarily take advantage of the cold neutron source which provides a unique resource for investigations of materials science and neutron properties. The reactor typically operates at 20 MW(t) 24-hours-a-day, 7-days-a-week, with a routine shutdown every 5-½ weeks. During the routine shutdown, which lasts approximately 10 days, the licensee refuels the reactor and performs maintenance. According to annual reports submitted by the licensee, the NBSR operated for a total of 254 full-power days in fiscal year (FY) 2006 and 268 full-power days in FY 2007. 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 302(b)(1)(B) of the Nuclear Waste Policy Act of 1982 provides that the NRC may require, as a precondition to issuing or renewing an operating license for a research or test reactor, that the applicant shall have entered into an agreement with the U.S. Department of Energy (DOE) for the disposal of high-level radioactive waste and spent nuclear fuel. DOE (represented by R.L. Morgan) has informed the NRC (represented by H. Denton) by letter dated May 3, 1983, that it has determined that universities and other government agencies operating nonpower reactors have entered into contracts with DOE that provide that DOE retains title to the fuel and is obligated to take the spent fuel and/or high-level waste for storage or reprocessing. The NBSR is a federally owned and operated nonpower reactor that is part of the NIST. The NBSR and DOE have a contractual arrangement whereby DOE retains title to the fuel and is obligated to take the spent fuel and/or high-level waste for storage and reprocessing. NIST has returned all of the spent NBSR fuel to DOE pursuant to this arrangement. Accordingly, NIST is in conformance with the Nuclear Waste Policy Act of 1982.

1.8 Facility Modifications and History

The Atomic Energy Commission (AEC) received the application for the NBSR construction permit (CP) and operating license on February 1, 1961. Construction began in 1963 when the AEC issued the CP. The CP was converted to Provisional Facility Operating License TR-5 in 1967. The reactor achieved initial criticality on December 7, 1967, and began full-power operation at 10 MW(t) on February 9, 1969. On May 21, 1970, the AEC issued Facility Operating License TR-5. On December 2, 1980, the NBSR requested an increase in the maximum licensed power from 10 MW(t) to 20 MW(t) and a 20-year renewal of the license. The NRC issued the license renewal, including authorization to operate at the increased power level, on May 16, 1984. Since the license renewal, the licensee has made no significant changes to the facility or facility operations.

2 SITE CHARACTERISTICS

2.1 Geography and Demography

2.1.1 Geography

The NBSR is located at latitude 39º7′34″ north and longitude 77º13′6″ west. The corresponding Universal Transverse Mercator coordinates are Zone Number 18, Northing 4333105 meters (m), and Easting 308252 m. The NCNR facility is located on the southern portion of the 2.33-square kilometer (575-acre) NIST campus in Gaithersburg, Montgomery County, MD, approximately 32 kilometers (km) (20 miles (mi)) northwest of Washington, DC. There are no prominent natural features in the immediate vicinity of the reactor, and the most prominent manmade feature is Interstate 270 (I-270) adjacent to the eastern boundary of the NIST campus. NIST is a nonregulatory Federal agency of the U.S. Commerce Department within the Technology Administration.

The NIST campus is located on the Maryland Piedmont Plateau approximately 48 km (30 mi) southeast of the Blue Ridge Mountains. The Blue Ridge Range rises to about 580 m (1,900 feet (ft)) above mean sea level compared to the NIST elevation of 128 meters (m) (420 ft) above mean sea level. The general area within an 8-km (5-mi) radius of the NBSR has a gently rolling topography with no geographic features that could significantly affect the diffusion and dispersion of airborne effluents. Few buildings are over three stories high within the immediate area of the NBSR.

2.1.2 Demography

Montgomery County is the most populous county in the State of Maryland. Between 1990 and 2000, the county‘s population grew 15.4 percent, with much of the growth occurring in the southern half of the county. The population of the county is expected to increase by an additional 22 percent by 2025.

The NIST campus is surrounded by the city of Gaithersburg, MD. The cities of Gaithersburg, Washington Grove, and Rockville and the unincorporated areas of Germantown, Montgomery Village, Darnestown, and North Potomac all lie within 8 km (5 mi) of the NBSR. According to the 2000 census, more than 223,000 people live within this 8-km (5-mi) circle.

The facility emergency planning zone (EPZ), a 400-m (1,300-ft) circle centered on the NBSR ventilation stack, lies entirely within the NIST campus. NIST security controls access to the campus. Access is limited to employees, contractors, and individuals who have business on the site. The area within the EPZ has laboratories and office buildings but no residential buildings. There is no part-time, transient, or seasonal occupation of any of the campus buildings. Thus, the effect of county population growth on potential public exposure to radiation from accidents at the NBSR is not expected to be significant. The closest permanent residences are more than 400 m (1,300 ft) directly to the east and directly to the west of the NBSR.

2.2 Nearby Industrial, Transportation, and Military Facilities

The I-270 Technology Corridor is a major research and development center. Some manufacturing occurs in the area, but no significant manufacturing plants are located near the NBSR. No chemical plants or refineries are in the immediate area surrounding the NBSR. Mining and quarrying operations in the vicinity are limited to those associated with constructing new office buildings. A natural gas pipeline lies 3 km (2 mi) to the south of the reactor, and a liquid petroleum/gas pipeline is located 1.6 km (1 mi) to the north.

I-270 forms the northeast boundary of the NIST campus and is a major commuter and truck route serving the area. Three arterial and collector roads that abut the NIST campus serve the Gaithersburg area surrounding the NIST campus, as they provide truck routes serving the local economy. A rail line, which parallels the northeast boundary approximately 2 km (1.25 mi) from the NIST campus, carries goods through the region. The nearest waterway is the Potomac River. Its nearest point is 10 km (6.4 mi) from the reactor.

The closest commercial airport to the reactor is Dulles International, 29 km (18 mi) from the reactor. Located 7.2 km (4 5 mi) from the reactor is the Montgomery County Airpark, a general aviation airport. Based on a review of airport maps, the runways and approach traffic for either airport are not in line with the NBSR facility.

Andrews Air Force Base, the nearest military base, is approximately 52 km (32 mi) away. Because of the distance between the two, operations at Andrews Air Force Base facility are not expected to affect the safe operation of the NBSR.

Based on the historical lack of serious transportation accidents in the vicinity and the distances involved to the NBSR, the NRC staff concludes that neither local industry, transportation, nor government facilities pose a significant risk to the continued safe operation of the NBSR.

2.3 Meteorology

The National Oceanic and Atmospheric Administration maintains historical meteorological data sufficient to characterize the NBSR site. Wind data are available from the National Weather Service stations at Dulles International Airport (29 km (18 mi) to the southwest) and at Reagan National Airport (40 km (25 mi) to the south-southeast). NIST has meteorological instrumentation located on site. Although this instrumentation has not been in place long enough to provide useful site characterization data when compared to the National Weather Service locations, it could be used for atmospheric dispersion calculations in the event of a release of radioactive material.

According to the NBSR SAR, monthly average temperatures range from a low of -4.6 degrees Celsius (C) (23.8 degrees Fahrenheit (F)) in January to a high of 29 degrees C (85 degrees F) in July. The annual mean humidity is 80 percent in the morning and 55 percent in the afternoon. Precipitation is evenly distributed throughout the year with an annual average of 81 cm (32 in.). The maximum recorded precipitation in a 24-hour period was 20 cm (7.9 in.) in June 1972. Snowfall occurs typically from November to March and averages 48 cm (19 in.) annually. Snowfalls of several inches are typical and remain on the ground for several days before melting. Over the past 11 years, Montgomery County has experienced an annual frequency of 4.3 winter weather events, of which 2.3 per year were mixed rain and snow or ice and snow. The maximum recorded 2-day snowfall total was 65.3 cm (25.7 in.) in 1996. The 100-year return period snowpack including 2-day probable maximum precipitation was calculated for Montgomery County as 1.3 kilopascal (kPa) (27.2 pounds per square foot (psf)). The NRC staff independently reviewed the licensee's snowpack calculations and found them to be reasonable.

The average windspeeds at Dulles International Airport and Reagan National Airport are 3.3 m/s (7.4 mi/hr) and 4.2 m/s (9.4 mi/hr), respectively. The most common wind direction recorded is generally from the northwest, with more frequent winds from the south during the summer months. Since 1950, only Hurricane Fran (1996) has affected the NIST site, although remnants of tropical storms have periodically had an impact on the site. Tornadoes occur relatively infrequently in Maryland, with an annual frequency in Montgomery County of less than 0.22 events per year. The tornadoes in Montgomery County were rated as F0 (40–72 mi/hr) and F1 (73–112 mi/hr), which represent the lowest categories on the Fujita Tornado Scale. The licensee calculated the 100-year return windspeed as between 35.7 m/s (79.8 mi/hr) and 45.8 m/s (102.5 mi/hr) for Montgomery County.

The frequency of hail events in Montgomery County has been 2.1 events per year. The frequency of lightning events in Montgomery County causing property damage is 1.8 events per year. Neither hail nor lightning has had any significant impact on the operation of NBSR.

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

2.4 Hydrology

The topography in the vicinity of the NBSR is undulating and the relief is moderate. The confinement building and cooling towers are at an elevation of approximately 128 m (420 ft). The elevation of the surface-water body at the nearest point to the confinement building is approximately 116 m (380 ft). The nearest naturally occurring surface water body to the site is a tributary of Muddy Branch, approximately 300 m (1,000 ft) to the west-northwest of the NBSR. Surface water drainage at the site flows southwest to the Muddy Branch stream and its tributaries. Muddy Branch discharges to the Potomac River. The site is more than 16 river-kilometers (10 river-miles) from the Potomac River.

The nearest mapped flood zone to the reactor, the Muddy Branch floodplain, is located approximately 610 m (2,000 ft) south of the site. The highest 100-year flood elevation for this floodplain is 115 m (376 ft) at the confluence of the Muddy Branch tributary, approximately 580 m (1,900 ft) southeast of the site. A topographical rise separates the site from the 100-year flood zone for the Muddy Branch tributary.

The Federal Emergency Management Agency's floodplain mapping and the topography of the site show that the site lies outside the 100-year and 500-year flood zones of the nearest surface water bodies. There is no documented history of flooding occurring at the site either before or after construction of the NBSR. The 100-year flood event for Muddy Branch and its tributaries is considered the controlling event for determining appropriate measures for flood protection. All the existing safety-related structures for the site are protected against that event.

As the confinement building and support structures are outside the 100-year and 500-year flood zones of these water bodies, there are no additional flood-design considerations. There are sufficient surface water drainage systems at the site to convey the precipitation from local, intense events.

This site has no probable maximum surge and seiche flooding considerations, as no large bodies of water where significant storm surges and seiches can form are near the site. As the site is not adjacent to a coastal area, tsunami flooding is not considered credible. There are no existing or proposed dams on Muddy Branch or its tributaries upstream of the site; thus, there are no seismically induced potential dam failure considerations for the site.

There is minimal potential for discharges of contaminated ground water to surface waters from accidental releases at the site. The licensee conducts routine environmental sampling of grass, soil, and water in nearby streams and ponds. Because of the proximity of the groundwater to the surface and direction of groundwater flow (southwest from the site), sampling of a pond located approximately 300 m (1,000 ft) to the west south-west of the reactor building provides acceptable means for monitoring groundwater at the site. All environmental samples are analyzed for possible neutron activation nuclides and fission product nuclides. Water samples are also assayed for tritium (^{3}H). The licensee reports the results to the NRC in the NBSR Annual Report. The NRC staff reviewed the licensee's annual reports from 1999 to 2007 and found that the licensee had detected no contamination.

The Potomac River is the source for the majority of the potable water distributed to Montgomery County. According to the licensee's environmental report submitted as part of the renewal application, the Washington Suburban Sanitary Commission draws water from the Potomac River more than 19 river-kilometers (12 river-miles) from the reactor site. The environmental report also states that the Montgomery County Health Department has a permit on record for only one groundwater well within a 1.6 km (1 mi) radius. Based on the licensee's groundwater monitoring program and the distances between the reactor site and the water supply extraction points, the NRC staff concludes that continued operation of the NBSR poses no significant risk of contamination of the local water supply.

Based on the above information, the NRC staff concludes that the local hydrology does not pose a significant risk to the continued safe operation of the NBSR. The NRC staff also concludes that the licensee's groundwater monitoring program and the distances between the reactor site and the water supply extraction points provide reasonable assurance that any releases of radioactive material to the groundwater will be detected in a timely manner and not lead to contamination of the local water supply.

2.5 Geology, Seismology, and Geotechnical Engineering

The NBSR is located in the southwestern portion of the city of Gaithersburg, MD. The reactor site lies within the Piedmont physiographic province. The eastern Piedmont is characterized by gently sloping upland areas and broad, relatively shallow valleys. The Fall Line, which is the physiographic and tectonic boundary between the Coastal Plain and Piedmont provinces, is approximately 26.5 km (16.5 mi) to the southeast of the NBSR site. The eastern margin of the Blue Ridge Province, the Catoctin Mountains, is approximately 32 km (20 mi) west of the site.

No large-scale geologic structures have been mapped near the NBSR site. The deformations that have been mapped occurred more than 400 million years ago and do not affect the site's safety. The only seismic source zones of concern are the Extended Continental Crust (ECC) and the Iapetan Rifted Margin (IRM). The NIST site lies in the western portion of Zone ECC, approximately 32 km (20 mi) from the eastern margin of Zone IRM.

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 employed a probabilistic methodology that uses a combination of gridded, spatially smoothed seismicity, large background zones, and specific fault sources to calculate hazard curves for a grid of sites throughout the country. The USGS probabilistic analysis results show relatively low ground-motion risk for a broad area surrounding the NBSR site.

The NBSR site is located in an area that has experienced only minor earthquake activity. No earthquakes with magnitudes greater than those considered for earlier licensing actions have occurred within the ECC and the IRM since publication of NUREG-1007, "Safety Evaluation Report Related to the License Renewal and Power Increase for the National Bureau of Standards Reactor," in September 1983. The licensee's seismic analysis concluded that the maximum potential earthquake for the area would generate a maximum peak horizontal ground-acceleration at the site of 0.07 to 0.1 times the acceleration due to gravity (g). A 2008 National Seismic Hazard Map produced by the USGS shows only a 2-percent probability that in 50 years peak lateral ground acceleration will exceed 0.06 g.

No surface faulting has been documented for any earthquakes occurring in the ECC or the IRM. The only faults mapped within 8 km (5 mi) of the site experienced deformation in the Paleozoic era. The potential for surface faulting at the site is negligible. The characteristics of the underlying soils and low level of seismicity indicate that the potential for liquefaction is practically nonexistent.

The NBSR buildings and facilities are supported on shallow and deep foundations. The shallow foundations are believed to rest on competent residual soils and/or transition materials. Similarly, deep foundations are believed to be bearing into transition materials, bedrock, or both. This conclusion is supported by the successful performance of the buildings' and facilities' foundations since their original construction, and the lack of any reported signs of distress or movement in the foundations. The foundations are expected to continue to perform satisfactorily as long as the geologic, hydrogeologic, and superimposed loads continue to remain consistent with those adopted in designing the facilities.

Based on the above information, the NRC staff concludes that the geology of the NBSR site is suitable for supporting the reactor building, structure, and systems and potentially damaging seismic events are unlikely to occur during the period of the renewed license.

2.6 Conclusions

The NRC staff concludes that the reactor site has experienced no significant geographical, meteorological, or geological change, and therefore the site remains suitable for continued operation of the NBSR. 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 NBSR. The demographics of the area surrounding the reactor have not changed and are not expected to change in any way that discernibly increases the risk to public health and safety from continued operation of the NBSR.

3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.1 Design Criteria

The design criteria for SSCs provide that SSCs related to safe operation and shutdown of the reactor must be able to perform their intended functions as described in the NBSR SAR. The principal safety-related SSCs are the fuel, core support structure, reactor protection system (RPS), reactor coolant systems, and the confinement 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 preclude the release of fission products.
- The core support structure must maintain its orientation, geometry, and structural integrity.
- The reactor safety system must be able to shut down the reactor.
- The reactor coolant systems must be able to remove heat from the reactor core and keep fuel elements below temperatures that could result in cladding damage.
- The confinement building must isolate under abnormal conditions and prevent the uncontrolled release of radioactive materials to the environment.

The SSCs mentioned above were designed and constructed in accordance with the construction permit issued by the AEC in 1963. Section 4.2 of this SER discusses the design of the fuel, control elements, and the core support structure; Section 7.4 discusses the design of the RPS; and Chapter 5 discusses the design of the reactor coolant systems. Design of the confinement building is discussed in Section 6.2 of this SER. TS 4.1, "Reactor Core Parameters," requires routine verification of the confinement function of the confinement building and functionality of the instrumentation that provides the confinement closure signal.

These SSCs have since been maintained and/or changed using license amendments or licensee review processes, including 10 CFR 50.59, "Changes, Tests, and Experiments," and maintenance, and special procedures, as appropriate, in accordance with the Commission's rules and regulations and Facility Operating License TR-5, as amended. The NRC staff previously evaluated all amendments to the facility license, and the NRC inspection program verified that the licensee had conducted the proper reviews. In its safety evaluation supporting issuance of Amendment No. 5 dated May 16, 1984 (NUREG-1007), the NRC staff evaluated the licensee's request for license renewal and an increase in the maximum reactor power level from 10 MW(t) to 20 MW(t). The NRC staff concluded that the SSCs in place were adequate to support safe operation during the period of the license renewal at the increased power level. Experience accumulated over the past 24 years of reactor operation supports the NRC staff's conclusion. Chapter 16 of this SER discusses age-related issues. Based on the above, the NRC staff concludes that the design and construction of safety-related SSCs provide reasonable assurance that SSCs will continue to meet the design criteria.

3.2 Meteorological Damage

Section 2.3 of this SER presents the meteorology of the NBSR site. While severe storms or tornadoes are possible at the NBSR site, the reactor and associated safety systems are housed in a reinforced concrete structure which provides considerable protection. The reactor and associated shielding are supported by a central concrete column. Shielding walls are nominally 3- to 5-ft thick and are an integral part of the building foundation. The confinement structure is designed for a 100-mi/hr wind load which is greater than the 100-year return period wind load of 96 mi/hr given in the applicable American Society of Civil Engineers standard (ASCE 7-05, “Minimum Design Loads for Buildings and Other Structures”). The NRC staff reviewed the licensee’s method for computing the 100-year maximum wind speed and found it to be conservative. The roof of the confinement structure is designed to withstand a snow load of 25 psf. The licensee calculated a 100-year return period ground snowpack of 27.2 psf, corresponding to a 22.8 psf confinement building roof snow load. The NRC staff reviewed the licensee’s calculation and found it to be reasonable and more conservative than the value given in ASCE 7-05. The NRC staff concludes that the design of the confinement building is sufficient to withstand likely meteorological conditions during the period of the renewed license. As discussed in Section 8.2 of this SER, there is reasonable assurance that failure of offsite power because of meteorological phenomena will not cause damage to the reactor or preclude safe shutdown.

3.3 Water Damage

Section 2.4 of this SER presents the hydrology of the NBSR vicinity. There are no bodies of water in the immediate vicinity that could flood the NBSR site. Further, the licensee indicated that surface drainage measures and the watertight confinement building would prevent heavy rainfall from impacting the reliable operation of safety-related equipment. Therefore, the NRC staff concludes that water damage poses no significant risk to safe operation or shutdown of the reactor.

3.4 Seismic Damage

Section 2.5 of this SER describes the seismicity of the NBSR vicinity. The NBSR is located in a zone of low seismic activity. The confinement building and reactor systems have been analyzed and shown to be able to withstand the stresses generated by peak lateral ground acceleration of 0.1 g. The probability of an earthquake resulting in peak lateral ground acceleration larger than 0.06 g is less than 2 percent in 50 years. The confinement building was designed in accordance with Building Officials and Codes Administrators Codes for seismic design. The reactor vessel was designed in accordance with American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section VIII, as well as other applicable codes and standards. The combined stress levels resulting from seismic loading plus all other design loads are well within the allowable limits for the various vessel sections. The NBSR reactor core and core components, including control rods, are designed to fail-safe. On receipt of a scram signal or a loss of power, all four shim safety arm clutches are deenergized, and the corresponding shim safety arms are inserted into the core. The shutdown-margin criterion for safe shutdown (TS Definition 1.3.30, “Shutdown Margin,” and TS 3.1.2, “Reactivity Limitations”) ensures that the reactor can be shut down even if the most reactive of the four shim safety arms fails to insert. The NBSR accident analyses, discussed in Chapter 13 of this SER, include a seismic event. As shown by the licensee’s analysis, the postulated seismic event will not pose a significant risk to the health and safety of the public.

3.5 Systems and Components

The systems and components most important to safety are the RPS and the fuel cladding. The RPS consists of the control devices and safety-related instrumentation and controls (I&C). Section 4.2.2 of this SER discusses the design requirements of the control devices. Chapter 7 of this SER discusses the design requirements of the safety-related I&C. Section 16.1.2.2 of this SER considers aging issues associated with control devices. These discussions show that the reactor safety system design bases and related TS provide reasonable assurance that the RPS will function as designed to ensure safe operation and safe shutdown of the reactor.

Section 4.2.1 of this SER discusses the fuel cladding design requirements, Chapter 13 examines accident scenarios, and Section 16.1.2.1 addresses 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.

3.6 Conclusions

On the basis of the above considerations, the NRC staff concludes that the design and construction of the NBSR are adequate to withstand and ensure safe shutdown as a result of all credible and likely wind, water, and seismic events associated with the site. Safe operation during the period of the current license and routine NRC inspections have verified the design and 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, and the environment.

4 REACTOR DESCRIPTION

4.1 Summary Description

The NBSR is a D_2O-moderated and -cooled, tank-type reactor designed to operate at a power level of 20 MW(t). The core is located in the lower section of an aluminum tank. The core contains 37 fuel element locations and 4 semipermanent irradiation thimble tubes. Of the fuel element locations, 7 are specially adapted for thimble tubes, leaving only 30 positions available for fuel element assemblies. The NBSR fuel elements are plate-type elements consisting of U_3O_8 mixed with aluminum powder contained in aluminum clad plates. Each fuel element contains an upper and lower fuel section separated by a gap, resulting in a “split core” design. This “split-core” design, with uranium fuel placed above and below the midplane of the reactor core, results in the thermal neutron flux reaching a peak in the center of the gap.

4.2 Reactor Core

The reactor core is located in the lower section of an aluminum tank, 2 m (7 ft) in diameter by 5 m (16 ft) in height. Upper and lower grid plates hold the fuel elements in place. The grid plates contain 37 fuel element positions placed in seven rows to form a hexagonal pattern with the rows oriented east to west in the core. The fuel is arranged in three rings within this hexagonal pattern, with the inner two rings having 6 fuel elements each and the outer ring having the remaining 18 fuel elements. The seven experimental thimble positions form a circular pattern about the center location. The overall length of the fuel element assembly is approximately 175 cm (68.8 in.). Each fuel element fits into fixed openings in the grid plates.

Four semaphore-type shim safety arms and one automatic regulating rod provide control of the reactor. The four shim arms provide primary control of the reactor. They 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 xenon, temperature, and fuel burnup. The four shim arms are mounted on hanger brackets just under the upper grid plate. The regulating rod provides fine control of the reactor. The regulating rod is located in an 8.9-cm (3.5-in.) vertical thimble.

4.2.1 Reactor Fuel

The NBSR reactor uses only the MTR-plate-type fuel element consisting of enriched U_3O_8 mixed with aluminum powder contained in aluminum-clad plates. Extensive testing of fuel plates to determine the limits on fission density as a function of fuel loading has been performed.[1] NBSR fuel is moderately loaded at 18-percent volume fraction with a maximum possible fission density of 2.6×10^{27} fissions per cubic meter (fissions/m^3). With a burnup of 73 percent in the eight-cycle fuel elements, the typical average fission density is 1.9×10^{27} fissions/m^3. This is within the 2.0×10^{27} fissions/m^3 limit specified in TS 3.1.4, “Fuel Burn-up,” and within acceptable levels for the prevention of unacceptable swelling reported in the literature.[2]

1 G.L. Hofman, J. Rest, and J.L. Snelgrove, “Irradiation Behavior of Uranium Dioxide—Aluminum Dispersion Fuel,” Argonne National Laboratory, Argonne, IL, October 1996.

2 J.L. Snelgrove and G.L. Hofman, “Dispersion Fuels,” in *Materials Science and Technology: A Comprehensive Treatment,* Volume 10A: *Nuclear Materials*, Part I, R.W. Cahn, ed., New York, 1994.

High-temperature testing of irradiated MTR fuel plates has been performed to determine limits for fission product release. Blistering of the U_3O_8 plates occurs between 450 degrees C (840 degrees F) and 550 degrees C (1,020 degrees F).[2] To preclude any blistering of the fuel cladding, the NBSR maximum allowable fuel clad temperature is 450 degrees C (840 degrees F), as specified by TS 2.1, "Safety Limit." Additionally, TS 3.9.2.2, "All Other Conditions," requires that fuel remain submerged in water after the reactor is shut down for at least one hour per megawatt of full power operation. This TS ensures that the fuel cladding temperature will remain below the safety limit during fuel transfer to the spent fuel pool.

Fabrication of NBSR fuel elements is in accordance with standard industry techniques for the manufacture of MTR plate-type fuel elements and the NIST specification for aluminum clad fuel elements including TS 5.3, "Reactor Core and Fuel." According to the licensee, new fuel element assemblies are subjected to stringent quality assurance before insertion into the core. The manufacturer inspects the fuel assemblies in accordance with DOE requirements before shipment to the licensee.

The corrosion history of aluminum MTR-type fuel elements has been studied extensively. Fuel plates of the same basic configuration and the same material as those used in the NBSR fuel elements have been operated at higher coolant flow rates, higher temperatures, and much higher heat fluxes than are achieved in the NBSR. All of these factors generally increase the corrosion rate of the fuel, yet corrosion of fuel elements during lifetimes comparable to those in the NBSR has not been a problem from the standpoint of structural integrity. According to the licensee, no NBSR fuel element has exhibited significant signs of corrosion or symptoms of corrosion damage. Based on the licensee's fuel utilization program and the limit imposed by TS 3.1.4, the lifetime of an NBSR fuel element is typically 1 year.

The outer shell of the NBSR fuel element represents the only major variation from the classic MTR plate-type fuel element. Since this outer shell controls the establishment of the proper hydraulic regime for heat transfer purposes, confirmation of the structural and hydraulic design objectives was accomplished on a hydraulic stand, using a fuel element assembly fitted with dummy plates. In these tests performed by the licensee, flow rates over a wide range of velocities were employed to measure flow conditions in each channel and across typical channels as well as the total pressure drop, drag forces, bypass flow around the lower nozzle, and the vibration characteristics of the spring-loaded element lock. The tests confirmed the predicted performance of the NBSR fuel element design.

The NRC staff reviewed the NBSR safety analysis and, based on the information above, determined that the NBSR fuel design is adequately supported by the MTR fuel development program. The NRC staff also concludes that the NBSR limits on fuel temperature (TS 2.1) and fuel burnup (TS 3.1.4) are supported by research and testing performed on similar U_3O_8 dispersion fuels. The historical evidence of the fuel integrity and corrosion-resistance performance of the NBSR fuel, as described by the licensee in the SAR, offers additional support for the adequacy of the NBSR fuel design. Therefore, the NRC staff concludes that continued operation as limited by the TS offers reasonable assurance that the fabricated fuel can meet the design objective of maintaining fuel integrity and thereby function safely in the reactor without undue risk to the health and safety of the public or the environment.

4.2.2 Control Rods

The three NBSR reactivity control systems consist of four shim safety arms, a single regulating rod, and a moderator dump system. The shim safety arms provide the primary means of reactivity control and shutdown capability. Each shim arm is a 2.54-cm-thick by 12.7-cm-wide by 132-cm-long (1-in. by 5-in. by 52-in.) poison tube with a hollow interior filled with helium. The poison material is a 0.102-cm-thick (0.040-in.) cadmium plate clad with 1100 series aluminum. The total reactivity worth of the four shim safety arms is approximately 26.4 $\%\Delta\rho$ at the end of an operating cycle (end of cycle (EOC)). Reactor shutdown capability is maintained from the most reactive state with the most reactive shim safety arm stuck in the fully withdrawn position, as required by TS 3.1.2. This requirement satisfies the “stuck rod” criterion found in the guidance in NUREG-1537 and ANSI/ANS-15.1, “The Development of Technical Specifications for Research Reactors.”

The four separate shim safety arms, as well as the independent moderator dump system required for reactor operation by TS 3.3.3, “Moderator Dump System,” provide redundancy. The NBSR shim safety arm control system is designed to ensure safe reactor control and shutdown under all operating conditions. This is achieved by using a design that relies on a passive feature (gravity) to achieve the safety function. All four shim safety arms are coupled to their drive motors by electromagnetic clutches. Thus, the only action required to effect a safe and rapid shutdown is to deenergize the electromagnetic clutches. The system is fail-safe because of the following features:

- No power source is required to initiate a shutdown.
- Loss of electrical power automatically results in a shutdown.
- No mechanical action, such as the release of a latch, is required in order to insert a shim arm.
- Insertion of any three of the four shim safety arms will result in a reactor shutdown under the most reactive core conditions.

According to the licensee, the shim safety blades are subjected to fabrication quality control inspections and radiography to ensure that all design requirements are met. The system design includes mechanical shock absorbers to ensure that the shim safety arms can withstand all anticipated stresses during operation. Mechanical stops prevent over-travel of the shim safety arms in the event of a mechanical failure in the blade drive system.

Poison burnup, corrosion, and radiation damage affect the lifetime of the shim arms. According to the licensee, under normal operating conditions, the shim safety arms have a lifetime of approximately 21,000 megawatt-days (MWD). The use of similar shim arms in the CP-5 reactor for a period of 8 years demonstrated that poison burnup, and not corrosion damage, limits the useful life of the shim arms. Over 30 years of NBSR operation have shown this finding to also be true for the NBSR shim arms. TS 4.1.2, “Reactivity Limitations,” requires annual determination of the reactivity worth of each shim arm. According to the licensee, radiation damage to structural materials of the shim safety arms is not significant during reactor operation because the shim arms are in the top reflector above the core where the fast neutron flux is relatively low. Shim arm sets have been replaced at the NBSR reactor five times since the previous license renewal, with no apparent radiation damage in the shim arms removed. Replacement is determined by acceptable reactivity margins necessary to maintain the shutdown capability required by TS 3.1.2.

The shim safety arm design provides digital position indication on the reactor control panel by a potentiometer coupled to the shim safety arm drive shaft. An electric motor drives each shim arm drive, through a high-ratio gear case and finally through an electromagnetic clutch. The shim arms may be controlled individually or as a bank. The four independent shim arm drives and control systems prevent a common mode malfunction.

The shim arm drives are constant speed mechanical devices with a drive speed of 0.0445 degrees per second, consistent with the safety analysis for the insertion of excess reactivity accident and TS 3.2.1, “Shim Arms.” Scram is aided by a spring that opposes drive motion during arm withdrawal. The shim safety arms are considered operable for a scram if they drop the top 5 degrees of travel within 240 milliseconds (msec), as required by TS 3.2.1. Withdrawal and insertion speeds or scram time should not vary except as a result of mechanical wear. The withdrawal and insertion speeds of each shim arm are determined at least semiannually, and scram times of each shim arm drive are measured at least semiannually, as required by TS 4.2.1, “Shim Arms.” These surveillance requirements, chosen to provide a significant margin over the expected failure or wear rates of these devices, meet or exceed the guidance for control rods in ANSI/ANS-15.1.

The regulating rod provides for fine control of core reactivity. The regulating rod consists of a solid aluminum cylinder, 6.35 cm (2.5 in.) in diameter by 74 cm (29 in.) long, located in a vertical thimble 8.9 cm (5.5 in.) in diameter. The regulating rod operates in a shroud which has the same configuration as the experimental thimbles. A fixed orifice in the nozzle of the shroud provides cooling water flow for the regulating rod. The regulating rod is designed with a reactivity worth of approximately 0.58% $\Delta\rho$, and because the regulating rod is aluminum, poison burnup is insignificant. Corrosion damage is similar to that of other core components and is minimal. A standard commercial design vertical drive mechanism mounted in the top plug drives the regulating rod. The regulating rod automatically drives into the core to the “full in” position upon receipt of a scram signal.

The moderator dump system provides a redundant shutdown capability. Moderator dump is initiated by manually actuating an air-operated diaphragm valve which allows the moderator level to drop to 2.54 cm (1 in.) above the top of the core. Although the reactivity worth of the moderator dump is a function of the position of the shim arms, the moderator dump will make the reactor subcritical for any core configuration allowed by the TS. TS 3.3.3 requires the moderator dump system to be operable when the reactor is operated, and TS 4.3.3, “Moderator Dump System,” requires annual cycling of the moderator dump valve.

TS 3.1.2, 3.2.1, 3.3.3, 4.1.2, 4.2.1, and 4.3.3 contain appropriate limiting conditions for operation (LCOs) and surveillance requirements necessary to ensure proper operation of the reactor control and safety systems, including the shim safety arms, regulating rod, and moderator dump system.

The NRC staff reviewed the NBSR safety analysis and determined that the control rod section adequately describes the reactivity control and shutdown systems of the NBSR. The descriptions include design, fabrication, acceptance testing, and reactivity worths of the systems. The analyses presented in the NBSR SAR (including accident analyses in Chapter 13) demonstrate sufficient reactivity worth for control of excess reactivity allowed by TS 3.1.2, adequate shutdown margin (TS 3.1.2), and acceptable control rod dynamic characteristics (TS 3.2.1) for both normal and accident conditions. Based on these considerations, 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 NBSR.

4.2.3 Neutron Moderator and Reflector

The tank-type design of the NBSR uses D_2O as a moderator, reflector, and coolant. The core is immersed in D_2O to thermalize fast neutrons to sustain the nuclear chain reaction, to remove heat created by the reaction, and to serve as the first stage of shielding. No other material is used within or in the area immediately surrounding the core region to moderate the fast neutrons created by the fission process.

The side reflector is 50.8 cm (20 in.) thick, and the top reflector thickness is normally maintained at 300 cm (118 in.). The top reflector level can be manually lowered to approximately 2.54 cm (1 in.) above the core to effect an emergency shutdown of the reactor, as discussed in the previous section.

The primary coolant purification system maintains D_2O chemistry to limit corrosion of the fuel elements and other materials in the reactor vessel and the primary coolant system. The purification system removes suspended particles and maintains the pH and conductivity by continuous flow of the D_2O through filters and ion exchangers. TS 3.3.1, "Primary and Secondary," requires all materials in contact with the primary coolant to be compatible with the D_2O environment, thereby limiting degradation of the primary system chemistry. Evaporative losses and the products of radiolytic degradation are reduced by the helium sweep system which maintains a blanket of helium on the D_2O in the reactor coolant systems. TS 3.3.1 limits D_2 concentration to a maximum of 4 percent in the helium sweep system to ensure a substantial margin below the explosive value.

Thermal expansion of the D_2O is accommodated by the D_2O storage tank, which allows the primary coolant to expand and contract with variations in coolant temperature. The capacity of the tank is sufficient to hold the entire coolant inventory. D_2O makeup is added directly from 55-gallon drums on an "as needed" basis.

Activation of the oxygen in the D_2O produces a potential ^{16}N radiation hazard. Passive shielding of the primary coolant system and access control at the NBSR limit worker exposures to radiation from ^{16}N without the need for a dedicated active ^{16}N control system.

The NRC staff reviewed the NBSR safety analysis and determined that the moderator and reflector section adequately describes the moderator and reflector design and safety considerations. The description includes material compatibility with respect to chemical, thermal, and radiation environmental performance. Based on its review, the NRC staff concludes that the NBSR moderator and reflector design adequately accounts for radiological degradation and the physical and chemical environment for the system. Accordingly, continued operation as limited by the TS offers reasonable assurance that the D_2O moderator and reflector can continue to perform as designed and will not pose a significant risk to the NBSR or the health and safety of licensee personnel during the period of the renewed license.

4.2.4 Neutron Startup Source

The normal power history of the NBSR produces a sufficiently strong photoneutron source for reactor startup, thereby obviating the routine use of an external neutron source. After extended shutdown, an encapsulated 1.9-curie (Ci) AmBe neutron source (half-life of 458 years), with a neutron yield of 2.2×10^6 neutrons per second per Ci, can be inserted into one of the vertical experimental thimbles to provide sufficient source neutrons for startup. The source is cooled directly by the D_2O flowing through the experimental thimble. Once the reactor is critical and before the power level is raised to 20 MW(t), the startup source is removed from the reactor and placed in its shielded storage container.

To prevent the reactor operator from inserting positive reactivity into the core without having a visible indication of the power level in the reactor, shim blade withdrawal is blocked if the source range nuclear instruments indicate less than 2 counts per second.

The NRC staff reviewed the documentation provided by the licensee on the neutron startup source and concludes that the design is adequate to provide sufficient startup neutrons and source range indication for reactor startup.

4.2.5 Core Support Structure

The core support structure is designed to ensure that all fuel elements, reactivity control devices, and in-core experimental facilities are properly secured against all anticipated loads including both the buoyant force of the coolant and the hydraulic forces associated with the primary coolant flow. The principal support features are the upper and lower grid plates. The upper grid plate is attached to four mounting brackets welded to the vessel wall. The lower grid plate is supported by the outer plenum flange plate which is welded to the outer plenum. The outer plenum is welded to the vessel bottom, so the vessel supports the load of the lower grid plate.

The grid structure in the upper and lower grid plates maintains the position of the internal core components. The fixed pattern in the grid plates aids in maintaining accurate positioning of the fuel elements, the reactivity control devices, and the experimental thimbles. This grid is designed to lock the fuel and other core components in place during reactor operation and to prevent movement of the core components by hydraulic forces. TS 3.1.3, “Core Configuration,” requires that core grid positions are filled with full-length fuel elements or thimbles.

The selection of 6061-T6 aluminum alloy for the upper and lower grid plates makes them compatible with the materials of the vessel and primary piping. Aluminum is chemically compatible with the D_2O coolant and exhibits excellent resistance to corrosion and erosion. It has low induced radioactivity and is resistant to radiation damage.

The NRC staff reviewed the NBSR safety analysis and determined that the core support structure section adequately describes the design for providing structural support for the core, accurate positioning of the fuel elements, and acceptable guides for other essential core components, such as shim safety arms and experimental thimbles. Based on its review, the NRC staff concludes that the core support structure is conducive to sufficient coolant flow, as well as being compatible with the coolant and radiation environment. The NRC staff therefore concludes that the core support structure is adequate for continued safe operation of the NBSR.

4.3 Reactor Tank

The aluminum-alloy reactor vessel is 2.1 m (7 ft) in diameter and 4.9 m (16 ft) in height. The vessel is a vertical cylinder with an elliptical bottom and a flange top. The upper girth of the reactor vessel is made of 6061-T6 aluminum alloy with a thickness of 1.27 cm (0.50 in.), which extends down approximately 294 cm (116 in.) below the surface of the reactor vessel flange. The lower girth and the bottom are made of aluminum alloy 5052 with a thickness of 2.22 cm (0.875 in.). The lower girth extends down from the upper girth approximately 422 cm (166 in.) below the reactor vessel flange. The beam ports, through tubes, cold source, and rabbit tubes all attach to the reactor vessel in the lower girth. The reactor vessel is supported by the vessel flange which is bolted to the top of the thermal shield shim ring by twenty-four 2.54-cm (1-in.) bolts. The vessel was designed in accordance with the ASME Boiler and Pressure Vessel Code for Unfired Pressure Vessels, ASME Code Cases 1270N and 1273N, the American Standards Association, the American Society for Testing Materials, the Aluminum Association, and the American Welding Society.

The design temperature for the reactor vessel is 121 degrees C (250 degrees F), and the design pressure is 345 kPa (50 pounds per square inch gauge (psig)). The normal reactor outlet temperature is 45.5 degrees C (114 degrees F), and the normal operating pressure is atmospheric. The maximum hydrostatic pressure, which occurs at the bottom of the reactor vessel, is 50 kPa (7.2 psig). The design of the vessel includes consideration of loads from constraining forces, along with those from steady-state and transient thermal conditions. The vessel's low heating rates and the excellent thermal conductivity of the aluminum combine to yield negligible stresses from internal temperature gradients. The very small temperature differentials between the coolant and the vessel components generate insignificant thermal transient loads. The NBSR vessel is fabricated entirely of aluminum alloys. Therefore, stresses resulting from differential expansion between dissimilar materials are negligible. The reactor vessel and its associated piping move freely under the influence of thermal expansion. Sliding pad-type pipe supports minimize reaction loads on the vessel, and in conjunction with all other loadings, do not cause any stress levels above the maximum allowable working stress for various reactor sections.

No impact loads are transmitted to the vessel. Pressure surges that might be generated in the vessel by reactor power transients are small and would not cause the vessel to exceed the design pressure. The licensee's analysis of seismic forces from horizontal accelerations of 0.1 g resulted in combined stress levels from this loading plus all other design loads that were well within the allowable limits for the various sections of the vessel.

A 3-millimeter (mm) (0.125-in.) corrosion allowance on all of the vessel's pressure containment surfaces was incorporated into the design. Conservative estimates by the licensee predict a minimal corrosion rate. This was corroborated by a visual inspection of the vessel's internal components that revealed little corrosion. The licensee performed the inspection in 1994.

The NBSR vessel is fabricated from aluminum alloys 5052 and 6061. Analysis of heavily irradiated (4.2×10^{23} thermal, 2.0×10^{22} fast neutrons per square centimeter (n/cm^2)) samples of the 6061-T6 alloy found in the literature indicates that the ductility retains approximately 70 percent of the original value, and the Charpy energy has dropped by over a factor of 6.[3] The most heavily irradiated portions of the NBSR vessel, the tips of the beam tubes, will have accumulated less than 2×10^{23} n/cm^2 thermal neutron fluence by the year 2024. Stress analysis performed by the licensee incorporating the reduction in Charpy energy and ductility indicates insignificant reductions in design safety margins.

Four D_2O inlet and outlet pipes penetrate the reactor vessel. The outer plenum is welded in the center of the vessel bottom while the inner plenum is located within and concentric to the outer plenum. The two outlet pipes are welded to the bottom on either side of the outer plenum pipe. The lower grid plate is bolted to both the inner and the outer plenums forming a watertight seal.

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3 J.R. Weeks, C.J. Czajkowski, and K. Farrel, "Effects of High Thermal Neutron Fluences on Type 6061 Aluminum," *Effects of Radiation on Materials: 16th International Symposium*, A.S. Kumar, et al., eds., ASTM Publication Code Number 04-011750-35, Philadelphia, 1993.

The D_2O holdup pan surrounds the core to a height just above the lower fuel section of the core and is attached to the lower grid plate, thereby trapping an inventory of cooling water during a loss-of-coolant accident.

The reactor vessel design includes a 51-cm-thick (20-in.-thick) side reflector and a top reflector normally maintained at 300 cm (118 in.). This D_2O reflector surrounding the core serves as the first stage of shielding, followed by the thermal and biological shields.

TS 5.2, "Reactor Coolant System," specifies the design features of the reactor vessel." TS 3.3.1 ensures the material compatibility of all reactor vessel components with the D_2O environment.

The NRC staff reviewed the NBSR safety analysis section on the reactor tank and determined that the licensee has adequately demonstrated the design as capable of withstanding all anticipated mechanical and hydraulic forces to prevent loss of integrity which could lead to a loss-of-coolant or other malfunction. The licensee has demonstrated by analysis and inspection that the design is capable of withstanding the corrosion and radiation environment in the tank for the period of the renewed license. The licensee also described reactor penetrations and provisions for maintaining core coolant coverage in the case of a loss-of-coolant accident (discussed in Chapter 13 of this SER). Based on the above considerations, the NRC staff concludes that the reactor vessel design is adequate for continued safe operation of the NBSR.

4.4 Biological Shield

The NBSR is enclosed in a shielding system consisting of the thermal and biological shields. The combined effects of this system are a gamma-ray dose rate of approximately 0.02 millisievert per hour (mSv/hr) (2 millirem per hour (mrem/hr)) at the outer face of the biological shield and a negligible dose rate from neutrons. The gamma dose rate at the top of the center plug is about 0.001 mSv/hr (0.1 mrem/hr) and less than 0.005 mSv/hr (0.5 mrem/hr) in the immediate vicinity of a pickup tool holding an element during fuel movement. All of these dose rates are well within the requirements of 10 CFR Part 20 and the guidelines of the facility ALARA program.

The thermal shield consists of 5 cm (2 in.) of lead and 20 cm (8 in.) of steel and, except for the top of the vessel and experimental ports, nearly surrounds the reactor vessel. The thermal shield is cooled by H_2O to remove the energy deposited by captured radiation. At full power, about 350 kilowatts (kW) are deposited in the thermal shield, preventing the concrete in the biological shield from excessive heating.

The design of the shield minimizes the effects of voids, necessary because of some structural features such as a pipe or shutter well, by adding enough lead to compensate for the gamma-stopping power of the concrete that was removed. Experimental beam holes designed to extract intense radiation beams from the reactor require extensive individual shielding. Each beam line was specifically reviewed by the licensee at the design stage, checked upon installation, and verified to have acceptable radiation dose rates during operation.

The NRC staff reviewed the NBSR biological shield design and determined that the biological shield section adequately describes the design and offers reasonable assurance that the shield design will limit exposures so as not to exceed the limits of 10 CFR Part 20 and the guidelines of the NBSR ALARA program.

4.5 Nuclear Design

4.5.1 Normal Operating Conditions

The normal operating cycle for the NBSR is a 38-day cycle beginning with less than 15% $\Delta\rho$ excess reactivity (TS 3.1.2) and ending with no excess reactivity and all control elements fully withdrawn. An example calculation in Section 4.5.1.3.2 of the NBSR SAR provides a typical startup core excess reactivity of 6.57% $\Delta\rho$. The shim safety arms and the regulating rod are cooperatively utilized during the cycle to manage the reactivity changes resulting from temperature changes, fission product poisons, and burnup.

A typical shutdown margin of 9.4% $\Delta\rho$ is provided by example calculation in Section 4.5.1.3.2 of the NBSR SAR with the highest worth shim safety arm fully withdrawn for the beginning of cycle (BOC) startup core. The moderator dump system provides a redundant minimum shutdown margin of 7.4% $\Delta\rho$ as demonstrated by example calculation in the SAR. These shutdown margins exceed the requirements of TS 3.1.2. Individual shim safety arm and regulating rod reactivity worths are determined annually or following any significant change in core or shim arm configuration, as required by TS 4.1.2.

The NBSR utilizes a combination of seven-cycle and eight-cycle fuel elements in a standardized fuel management configuration that results in 38 days of full-power operation consuming approximately 970 grams ^{235}U. The average burnup in the seven- and eight-cycle fuel elements is 66 percent and 73 percent, respectively. Because of the enrichment of the fuel, the total plutonium inventory at EOC has a negligible effect on reactivity. At the beginning of each cycle, four fuel elements are replaced with fresh fuel elements, and the remainder rearranged according to the standardized fuel management scheme. TS 3.1.4 limits maximum fuel burnup to 2.0×10^{27} fissions/m^3.

The experimental facilities in the NBSR represent large voids in the core reflector which may insert positive reactivity if flooded by a crack in the beam tubes or failure of a D_2O-cooled experiment. The largest single reactivity insertion from such a failure would be flooding of the cold neutron source, resulting in a reactivity addition of 0.49% $\Delta\rho$. This is below the TS reactivity limit of 0.5% $\Delta\rho$ for a single experiment, as specified in TS 3.8.1, “Reactivity Limits.” An analysis of a ramp reactivity insertion of 0.5% $\Delta\rho$ in 0.5 seconds in Chapter 13 of the NBSR SAR demonstrated no loss of fuel integrity from the accident.

TS are provided to restrict the reactor core parameters to acceptable fuel loading, configuration, and burnup (TS 3.1.2, 3.1.3, 3.1.4), ensure adequate reactivity control (TS 3.1.2, 3.2.1), and limit the reactivity associated with experimental facilities (TS 3.8.1).

From its evaluation of the normal operating conditions for the NBSR, the NRC staff concludes that the TS provide reasonable assurance that continued operation of the NBSR will not pose an undue risk to the health and safety of the public or the environment.

4.5.2 Reactor Core Physics Parameters

The reactor physics parameters for the NBSR are determined primarily by calculations using the Monte Carlo N-Particle (MCNP) computer code. MCNP is a standard industry nuclear physics tool developed at Los Alamos National Laboratory. The code has been benchmarked against critical experiments and power reactor applications. The licensee benchmarked its MCNP results against measured values for the regulating rod and shim bank reactivity worths.

Delayed neutron groups are presented in the 1980 Addendum to the original NBSR SAR and were taken from the literature.[4,5] At a steady reactor power, the fraction of all the neutrons that are delayed is β_{eff} = 0.007574.

D_2O-moderated and -reflected research reactors have prompt neutron lifetimes reported to be on the order of 700 microseconds (µs) (Georgia Tech Research Reactor: 770 µs; expanded Special Excursion Reactor Test II core: 750 µs; High Flux Beam Reactor: 672 µs).[6,7,8] The licensee calculated values of the prompt neutron lifetime using two-group diffusion theory and a homogenized core model of 500 µs to 800 µs, depending on the core volume and the fuel loading. Prompt neutron lifetimes calculated at Brookhaven National Laboratory (Hanson et al., 2004) using MCNP simulations of a pulsed neutron source in the subcritical NBSR-produced neutron lifetimes of 774 ±35 µs (startup core configuration (SU)) and 819 ±48 µs (EOC).[9]

Although the fuel elements in the reactor core are widely spaced, the NBSR is an undermoderated reactor. The bulk temperature coefficient for the D_2O moderator for the SU and EOC cores is -0.017%$\Delta\rho$/ºF and -0.014%$\Delta\rho$/ºF, respectively. The magnitude of the void coefficient depends on the location of the void. The values calculated for the moderator between the fuel elements are -0.043%$\Delta\rho$/liter (SU) and -0.030%$\Delta\rho$/liter (EOC). Therefore, a decrease in the D_2O density anywhere in the reflector, the moderator, or the coolant inside the fuel elements results in a negative reactivity insertion, which contributes to the inherent safety of the core design.

The licensee calculated axial and radial flux distributions by using MCNP for SU and EOC cores. The maximum thermal flux, 3.5×10^{14} n/cm^2/s, occurs very close to the core midplane in the unfueled region between the upper and lower cores. The calculated values are in good agreement with the measured peak thermal neutron flux in the central thimble of 3.5×10^{14} n/cm^2/s at 20 MW(t). Calculated distribution patterns agree favorably with expected results based on fuel arrangement, the location of the control elements, and the location of the unfueled region in the core central plane.

For use in thermal-hydraulic analyses, the licensee calculated the peak-to-average heat generation factors by using the MCNP results to determine the energy deposited in all 1,080 coolant channels for the SU and EOC cores. Fuel element, axial, and lateral peaking factors are determined and then corrected for uneven burnup to determine the relative power factors for the limiting thermal-hydraulic analyses. These factors were determined for both SU and EOC cores.

The NRC staff has evaluated the reactor core physics parameters for the NBSR and concludes that the values and calculation methods are appropriate and consistent with methods for similar

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4 R.J. Tuttle, "Delayed Neutron Data for Reactor Physics Analysis," *Nuclear Science and Engineering*, 56:37, January 1975.

5 M.W. Johns and B.W. Sargent, *Canadian Journal of Physics*, 32:136, 1954.

6 M.M. Bretscher, "Perturbation-Independent Methods for Calculating Research Reactor Kinetic Parameters," ANL/RERTR/TM-30, Argonne National Laboratory, Argonne, IL, 1997.

7 J.E. Grund, "Self-Limiting Excursion Tests of a Highly Enriched Plate Type D_2O Moderated Reactor: Part I, Initial Test Series," IDO-16891, Phillips Petroleum Company, Idaho Falls, ID, 1963.

8 J.M. Hendrie, "Final Safety Analysis Report on the Brookhaven High Flux Beam Reactor," BNL-7661, Brookhaven National Laboratory, Upton, NY, 1964.

9 A.L. Hanson, H. Ludewig, and D. Diamond, "Calculation of the Prompt Neutron Lifetime in the NBSR," *Nuclear Science and Engineering*, Vol. 153, 2006.

reactors. Temperature and void coefficients are negative everywhere in the core and provide inherent safety characteristics during normal operation and transients. The NRC staff concludes that flux distribution methods are adequate and the selection of peaking factors is acceptable for use in normal operating limit and accident analysis calculations.

4.5.3 Operating Limits

TS 3.1.3 limits the core configuration to the 30 fuel element loading as analyzed in Chapter 13 of the NBSR SAR. The excess reactivity is limited to a maximum of 15% $\Delta\rho$, and the shutdown margin is required to be greater than $1.00 for any core condition, with all movable experiments in their most reactive condition. The reactor is also required to remain subcritical with the highest worth shim arm fully withdrawn. TS 3.1.2 specifies these restrictions on excess reactivity and shutdown margin. The NRC staff reviewed the calculations provided in the NBSR SAR and determined that they demonstrate the adequacy of the shutdown margin for all core configurations.

TS 3.2.1 and 3.3.3 require all four shim safety arms and the moderator dump system to be operable for reactor operation. TS 3.2.1 also limits the maximum reactivity insertion rate to 5×10^{-4} $\Delta\rho$/s. These limits conservatively ensure that adequate shutdown margin is maintained, since only three of the four shim safety arms are required for shutdown of the most reactive core. In addition, the moderator dump system, which would provide adequate shutdown margin with all shim safety arms withdrawn, is required to be operable during reactor operation to ensure redundant shutdown margin capability. The maximum insertion rate limit ensures no fuel damage from reactivity insertion excursions. Chapter 13 of the NBSR SAR presents an analysis of this insertion rate for a rod withdrawal accident using the maximum beginning-of-life rod worths with the rods operating at the design speed of their constant speed mechanisms. The analysis shows that the most severe accident, a startup withdrawal accident from source level, is bounded by the maximum reactivity insertion accident and will not result in core damage.

TS 3.8.1 limits the reactivity associated with experiments in the NBSR to a maximum of 0.5% $\Delta\rho$ for a single experiment, and all experiments are limited to a total of 2.6% $\Delta\rho$. Analysis in Chapter 13 of the NBSR SAR shows that, if the most reactive single experiment were removed in 0.5 seconds, this ramp insertion into the NBSR operating at 20 MW(t) would not result in any fuel failure.

The NRC staff has reviewed the NBSR 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. Reactor core physics parameters are determined by acceptable analytical methods, and the TS require operating limits that will ensure fuel integrity. The NRC staff therefore concludes that the nuclear design, as limited by the TS, is adequate for continued safe operation of the NBSR.

4.6 Thermal-Hydraulic Design

The thermal-hydraulic design basis for the NBSR is that there shall be no fuel damage resulting in the release of fission products during normal operation or from any credible accident. As previously discussed, the fuel cladding may begin to blister at 450 degrees C (840 degrees F), and during the blistering process, cracks will develop that can release gaseous fission products. The criteria for ensuring that this limiting fuel clad temperature is not exceeded are that no departure from nucleate boiling (DNB) or onset of flow instability (OFI) conditions occur in the coolant.

The licensee selected nominal settings of flow and inlet temperature by the requirement that there be no nucleate boiling at the hottest spot on the fuel cladding. This was conservatively estimated by the requirement that the fuel clad temperature at the hot spot remain below the saturation temperature for D_2O at the appropriate pressure. The licensee calculated the clad temperature with a heat transfer coefficient determined using the Dittus-Boelter correlation.

Having determined nominal flows, the limits of safe operation were determined using the most limiting criteria of DNB and OFI. For forced circulation, the licensee chose the Mirshak correlation to test for DNB, based on the close similarity between the conditions for which the correlation was determined and the conditions in the NBSR core. The Saha-Zuber correlation was used to test for OFI.

For natural convection, the licensee used the Sudo-Kaminaga and Oh/Chapman correlations to check for DNB and OFI, respectively. These calculations were performed for both the inner and outer plenums, at both the hot spot and the exit of the hottest fuel channel of the upper fuel section. Critical heat flux and OFI ratios calculated by the licensee show ample thermal safety margins (greater than a factor of 2) for steady-state operating conditions. The licensee used the RELAP5 code to analyze abnormal transients during 500-kW power operation with natural convection. These analyses show peak clad temperatures lower than the blistering temperature, thereby demonstrating the acceptability of the limiting safety system settings (LSSSs) required by TS 2.2, “Limiting Safety System Settings,” for operation with natural convection. Although the licensee performed calculations for operation at 500 kW, TS 2.2 requires that the reactor power be limited to 10 kW during operation with natural circulation. This requirement provides a large safety margin for operation with natural circulation.

The licensee demonstrated that shutdown cooling will provide ample cooling for all shutdown conditions, including loss of offsite power followed by failure of both shutdown pumps. This scenario, as analyzed in Chapter 13 of the NBSR SAR, results in no damage to the fuel, thereby showing that natural convection cooling is adequate to provide cooling of the fuel, even immediately following a scram resulting from loss of all primary pumps.

The reactor safety limit (TS 2.1) states that the fuel cladding temperature shall not exceed 450 degrees C (840 degrees F) for any operating conditions of power and flow. Maintaining the cladding temperature below this limit will prevent blistering and thereby ensure fuel integrity. The LSSS (TS 2.2) ensures that the safety limit is not violated. Conservative calculations performed by the licensee in Chapter 13 of the NBSR SAR have shown that the LSSS limiting combinations on reactor power (130 percent), coolant outlet temperature (147 degrees F), and coolant flow (not less than 60 gpm/MW(t) for the inner plenum and not less than 235 gpm/MW(t) for the outer plenum) will ensure that any abnormal reactor condition caused by equipment malfunction or operator error will be terminated well before the safety limit is reached; the calculations include allowances for uncertainties in process instrumentation.

The NRC staff has reviewed the NBSR safety analysis and determined that the thermal-hydraulic design section adequately demonstrates the thermal-hydraulic characteristics necessary to provide the limits on cooling conditions that ensure fuel integrity will not be lost under any reactor conditions, including accident conditions. Thermal-hydraulic parameters are determined by acceptable analytical methods, and safety limits and LSSS are specified in the TS that will ensure compliance with the design criteria of no DNB or OFI. Therefore, the NRC staff concludes that the thermal-hydraulic design, as limited by the TS, is adequate for continued safe operation of the NBSR.

4.7 Conclusions

The NRC staff concludes that the licensee has adequately described the bases and functions of the reactor design and demonstrated that the NBSR 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. Nuclear and thermal-hydraulic design and operating limits as established by the TS adequately provide for the protection of fuel integrity. Therefore, the NRC staff concludes that continued operation of the NBSR within the limits of the TS and facility license will not result in undue risk to the health and safety of facility personnel, the public, or the environment.

5 REACTOR COOLANT SYSTEMS

5.1 Summary Description

The reactor coolant systems at the NBSR facility include the following: primary coolant system, secondary coolant system, primary coolant purification system, primary coolant makeup, ^{16}N control, and D_2O experimental cooling system. The primary purposes of the reactor coolant systems are to remove the fission and decay heat generated in the core, to dissipate the decay heat to the environment, and to serve as one of the barriers to prevent fission product release to the environment. The primary coolant is D_2O, and the secondary coolant is H_2O.

5.2 Primary Coolant System

The primary coolant system is designed to transfer 20 MW(t) of heat from the core to the secondary coolant system with nominal operating values of 568 liters per second (L/s) (9,000 gallons per minute (gpm)) flow, 38.8 degrees C (100 degrees F) reactor inlet temperature, and 45.6 degrees C (114 degrees F) reactor outlet temperature. The reactor coolant systems at the NBSR facility are designed to remove sufficient heat to support continuous full-power operation at a power level of 20 MW(t) and remove the decay heat generated after shutdown from extended full-power operation.

The primary coolant system, consisting of pumps, heat exchangers, piping, and valves, is located entirely within the reactor confinement building. While this system is not pressurized, it is closed to the atmosphere. Therefore, it serves as one of the three barriers to fission product release, the other two being the fuel cladding and the reactor confinement building.

The primary coolant system normally operates under conditions of forced flow in which primary coolant enters the bottom of the reactor vessel through the inner and outer plenums. The inner plenum feeds primary coolant to the center 6 fuel assemblies, while the outer plenum feeds the remaining 24 fuel assemblies. The coolant flows up through the fuel and removes the heat generated by fission, before exiting from the bottom of the vessel through two outlet pipes. Then, the primary coolant flows through the D_2O main circulating pumps to plate-type main heat exchangers, where the heat from fission in the core is transferred to the secondary coolant. The primary coolant passes through a strainer before returning to the reactor vessel. A shutdown cooling system is provided to remove decay heat, although analyses in Chapter 13 of the NBSR SAR demonstrate that natural circulation cooling alone is adequate to protect the integrity of the fuel.

The process room contains the piping, strainers, D_2O main circulating pumps, D_2O shutdown pumps, main heat exchangers, control valves, and instrumentation associated with the primary coolant system. A curb captures any primary coolant that may leak from the system and collects it in a sump. A sump pump, which returns spilled coolant to the overhead storage tank as part of the emergency core cooling system, is required for reactor operation by TS 3.3.2, “Emergency Core Cooling.” Vent and drain lines are equipped with manual valves and flanges with quick-disconnect fittings to provide two barriers between the primary coolant and the atmosphere.

Detectors located in the secondary system monitor potential leakage from the primary to secondary systems. If a detector alarms, the secondary water is sampled for ^{3}H. In addition, a leak into the secondary system can be detected by a change in the level of the D_2O storage tank and by periodic sampling of the secondary water for ^{3}H. These methods are sensitive enough to detect a leak of above 135 liters (L) (36 gallons (gal)) in 1 day or 190 L (50 gal) in 1 week.

TS 3.7.1, “Monitoring Systems, and Effluent Limits,” requires either a secondary cooling water activity monitor or a D_2O storage tank level monitor for reactor operation. TS 3.7.1 restricts primary coolant ^3H activity to 5 Ci/L or less to ensure that effluent ^3H concentrations are below regulatory limits. Deuterium gas (D_2) in the helium cover gas system is monitored and recombined in a catalyst bed to prevent concentrations from exceeding the 4-percent limit (TS 3.3.1). This limit on D_2 gas ensures a substantial margin below the lowest potentially explosive concentration.

Primary system instrumentation provides information on reactor inlet flow to each plenum, reactor outlet flow, primary coolant temperature change through the core, reactor vessel level, reactor overflow, and primary-to-secondary pressure differential in the main heat exchangers to the control room. This arrangement, including alarm panels to alert the operator to changing conditions, gives the reactor operator a single convenient location from which to monitor and operate the reactor. Low-level scram and low reactor outlet flow channels are required for reactor operation, as specified in TS 3.2.2, “Reactor Safety System Channels.”

The NRC staff has reviewed the primary coolant system design and TS described in the NBSR SAR. TS ensure that necessary coolant system equipment and instrumentation are provided for reactor operation. Primary coolant system integrity is assured by TS that require monitoring for system leakage, D_2 gas accumulation, and pressure relief system functionality. TS 4.2.2, “Reactor Safety System Channels,” 4.3.1, “Primary and Secondary,” and 4.7.1, “Monitoring System,” provide surveillance requirements for equipment operability and sampling frequency.

Based on its review, the NRC staff concludes that the primary system design and TS provide reasonable assurance of necessary primary coolant system operability for reactor operations as analyzed in the NBSR SAR while posing no undue risk to the health and safety of the public, licensee personnel, or the environment.

5.3 Secondary Coolant System

The secondary coolant system is designed to transfer heat from the following heat exchangers associated with the reactor coolant and auxiliary support systems: main heat exchangers (HE-1A, 1B, 1C), D_2O purification heat exchanger (HE-2), thermal shield heat exchanger (HE-6), thermal column heat exchanger, experimental demineralizer heat exchanger (HE-7), and the helium compressor secondary cooling heat exchanger. The heat load in the secondary coolant from these heat exchangers is transferred to the atmosphere via a 22-MW(t) hybrid wet/dry, plume abatement cooling tower. The heat removal capacity of the system exceeds the total heat generated in all of these individual systems.

The secondary coolant system circulates H_2O through two 10-MW(t) main heat exchangers to remove heat from the primary coolant that is generated by fission in the core. There are six parallel main secondary coolant pumps, arranged in two sets of three parallel pumps. When the reactor is shut down, a single smaller pump provides secondary flow. An installed flow element measures secondary system flow. The operator maintains a constant reactor inlet temperature by regulating the amount of secondary coolant bypassing the cooling tower.

Upon leaving the main heat exchangers, part of the secondary coolant passes through a radiation detector and a test-coupon station. The radiation detectors monitor the secondary water for the presence of ^{16}N, an indicator of a primary-to-secondary leak (TS 3.7.1). The test coupons monitor for any long-term effects that the secondary coolant might have on the secondary piping.

A chemical addition system regulates corrosion and biological growth in the secondary system. Water is continuously blown down to the sewer system to remove concentrated solids and to maintain a low concentration of dissolved solids.

Two secondary auxiliary booster pumps supply water from the discharge header of the main secondary coolant pumps to the D_2O purification heat exchanger, the thermal shield heat exchanger, and the thermal column heat exchanger. Normally, one of these pumps is operating while the other remains in standby.

The helium compressor secondary cooling pumps supply water from the suction header of the main secondary coolant pumps to the helium compressor secondary cooling heat exchanger. This removes the heat generated in the cold source refrigerator. Normally, one of these pumps is operating while the other remains in standby.

Based on its review of the information presented in the licensee's SAR and the above discussions, the NRC staff concludes that the secondary cooling system design and operation provide sufficient cooling capacity to maintain primary cooling system temperatures within the limits analyzed in the NBSR SAR. As discussed in Chapter 11 of this SER, TS 3.7.1 and 4.7.1 ensure adequate monitoring for radioactivity in the secondary cooling system to maintain ^{3}H releases below regulatory limits. The NRC staff therefore concludes that the requirements of these TS provide reasonable assurance that any primary-to-secondary coolant leaks will be readily detected and not result in any significant risk to public health and safety or the environment.

5.4 Primary Coolant Cleanup System

The primary coolant purification system is designed to maintain the chemistry and purity of the primary coolant by removing both soluble and insoluble corrosion products and other foreign materials. The chemistry must be properly controlled to ensure that the components in contact with the primary coolant are not degraded over the life of the plant. The purification system maintains the primary coolant pH between 5.0 and 6.0 and the conductivity less than 1 microSiemens (μS).

Mechanical filtration of the primary coolant removes particles 5 microns and larger. Purity of the primary coolant is essential to minimize the contaminants that might be exposed to the neutron flux, thereby decreasing personnel radiation exposure and limiting production of radioactive waste.

The NRC staff compared the primary coolant cleanup system design and operation with recommended operational parameters for aluminum-water systems found in the literature and with systems for similar licensed nonpower reactors and found them to be consistent.[10] The NRC staff therefore concludes that continued operation within the parameters described in the NBSR SAR provides reasonable assurance that corrosion of the system components and primary coolant impurity levels will be acceptable.

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10 U.S. Department of Energy, *DOE Fundamentals Handbook, Chemistry,* Vol. 1 and 2, DOE-HDBK-1015, Washington, DC, January 1993.

5.5 Primary Coolant Makeup Water System

The NBSR primary coolant system is a closed system and experiences no significant evaporative losses. In additional, the D_2O storage tank has sufficient capacity to hold the entire coolant inventory of the primary coolant system and its associated D_2O systems. As a result, the primary coolant inventory is maintained nearly constant during normal reactor operation and maintenance or other activities that require draining of the reactor vessel or primary system piping. Accordingly, the NBSR reactor does not have a dedicated system for adding makeup water to the system. Makeup D_2O is added directly from 210-L (55-gal) drums on an "as needed" basis. Since the makeup water is added through existing primary system lines and the primary system is not connected to any potable water supply lines, the NRC staff concludes that the addition of makeup D_2O will not result in any loss of primary coolant and cannot contaminate any potable water supply.

5.6 D_2O Experimental Cooling System

The D_2O experimental cooling system distributes D_2O from the primary coolant purification system to cool the cold neutron source; pneumatic sample tubes RT-1, RT-2, and RT-4; and other experimental facilities. The system uses cooled primary coolant from the discharge of the D_2O purification heat exchanger and returns the heated coolant to the D_2O storage tank.

In an emergency, D_2O is available from the D_2O emergency cooling tank to cool the system components. An emergency backup supply of cooling water is also available from the domestic water system.

5.7 Conclusions

The design of the NBSR cooling systems, as described in the NBSR 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 systems contain sufficient features to protect personnel from excessive radiation hazards, minimize corrosion of system components and fuel, prevent or detect losses of coolant, and provide one of the barriers to prevent the release of fission products to the environment. The NRC staff concludes that the coolant systems of the NBSR are sufficient for continued safe reactor operation within the related limits of the facility license and TS.

6 ENGINEERED SAFETY FEATURES

6.1 Summary Description

Engineered safety features (ESFs) are designed to prevent or mitigate accidents by controlling the release of radioactive materials to the environment. The ESFs at the NBSR include the emergency cooling system, the confinement building, and the ventilation systems. ESFs can be actuated automatically by the protection instrumentation that monitors various parameters during reactor operation or manually by the reactor operator. The ESFs provide protection against (1) overheating of the core should forced flow of primary coolant be unavailable and (2) uncontrolled release of radioactive material to the surrounding environment.

6.2 Detailed Descriptions

6.2.1 Confinement

The confinement building, a three-level structure with a volume of approximately 16,000 cubic meters (m^3) (600,000 cubic feet (ft^3)), was designed and constructed to ensure minimum air leakage. The ventilation systems allow the building atmosphere to be maintained at a slight underpressure during both normal and emergency conditions, assuring that any leakage is into the confinement building. All ventilation ductwork that penetrates the reactor building has automatically sealing closure valves or dampers. Personnel access to the confinement building is through entrances equipped with a sliding steel door with inflatable gaskets. During normal operations, these sliding doors are fully open. In an emergency, the sliding steel doors automatically close, and the gaskets automatically inflate to seal the entrances to the building. Because the building will be sealed during emergency conditions, a vacuum relief valve is incorporated to prevent any detrimental pressure differential from developing across the building walls or roof.

TS 3.4.1, “Operations that Require Confinement,” specifies the operating conditions that require confinement, and TS 3.4.2, “Equipment to Achieve Confinement,” prescribes the equipment operability requirements to establish confinement. Surveillance tests required by TS 4.4, “Confinement System,” periodically check the operability of the trip features of the confinement closure system and require an annual integrated leakage rate test to ensure that the leakage rate remains within acceptable limits. These TS ensure that the confinement system is available in the event of a significant radiological release and that the release rate will be within that analyzed in an MHA. The MHA analyzed in Chapter 13 of the NBSR SAR demonstrates for a worst-case accident scenario that the confinement system adequately controls and mitigates the release of radiological material from the confinement system such that radiological doses to members of the public are within the regulatory requirements of 10 CFR Part 20 and 10 CFR Part 100.

6.2.2 Ventilation

The intake air supply for the normal ventilation system includes both fresh air from outside and recirculated air within the building. All effluent air exhausted from the confinement building is monitored for radioactivity. Each of the effluent pathways has filter banks which are monitored for particulate activity, air samples are withdrawn for counting of gaseous activity, and a monitor measures activity in the stack at the point of release to the environment. If high radiation levels are detected, the normal ventilation system will be shut down, all building closure devices will be sealed, and the emergency ventilation system will be activated.

Under emergency conditions, the air inside the confinement building can be recirculated after being filtered through a system consisting of both a high-efficiency particulate air (HEPA) and a charcoal filtering system. If the inside air warrants cooling and humidity control, the normal air conditioning system could be used to condition the inside air as appropriate during an emergency. In addition, the emergency exhaust system is designed to draw air from the building at such a rate that a pressure differential can be established across the building structure to ensure that any leakage is into the building regardless of likely outside pressure variations resulting from wind or barometric changes. The emergency exhaust system consists of two redundant subsystems each of which could draw air from the normal exhaust system ductwork. Air exhausted from the building in each subsystem passes through a filtering system consisting of both HEPA and charcoal filters before being released through the stack.

TS 3.5, "Ventilation System," specifies ventilation system requirements necessary for reactor operation. Surveillance requirements in TS 4.5, "Ventilation System," provide for periodic tests of valves, controllers, instruments, and particulate and charcoal filters to ensure operability of the ventilation systems. The requirements of these TS ensure that the emergency functions of the ventilation system will be available and functional to provide for a controlled, mitigated release to the environment in the event of a radiological accident.

6.2.3 Emergency Core Cooling System

The emergency cooling system (ECS) provides cooling for the reactor core in case primary coolant is lost through leakage from the primary coolant system. An inner reserve tank located within the reactor vessel contains approximately 3,028 L (800 gal) and would provide a minimum of 28 minutes of coolant flow to the core with no operator action. By operator manipulation of control valves from the emergency cooling tank, the D_2O emergency cooling tank located above the reactor vessel can provide an additional 11,360 L (3,000 gal) of D_2O to the core top or to each of the reactor inlet plenums. In either case, emergency coolant flows through the reactor vessel and primary coolant system to the pipe rupture location where it drains onto the process room floor. The concrete curb and floor drains in the process room direct the D_2O to the emergency sump where it is pumped back either to the D_2O emergency cooling tank or to the D_2O storage tank located in the confinement building. There is sufficient D_2O in the inner reserve tank and emergency cooling tank to provide 2.5 hours of cooling on a once-through basis. The ECS also has the capability to add domestic H_2O to the D_2O emergency cooling tank through a spool-piece and double manual isolation valves.

TS 3.3.2 requires the ECS to be operable for reactor operation and requires a source of makeup water for the system. Surveillance tests of the valves and pump required by TS 4.3.2, "Emergency Core Cooling System," check the proper operation of the ECS components. Analysis in the NBSR SAR shows that the ECS provides adequate cooling during a loss-of-coolant accident. Therefore, the availability and operability of the ECS, as provided by these TS, and the NBSR loss-of-coolant accident analysis provide reasonable assurance that the ECS can continue to adequately protect against melting of the core and the associated release of fission products.

6.3 <u>Conclusions</u>

Based on the above discussions and evaluations performed in the referenced sections of this SER, the NRC staff concludes that the design of the ESFs, as described in the NBSR SAR, provides adequate protection to prevent the loss of fuel integrity from a loss-of-coolant accident and mitigate potential releases of radioactive material to the surrounding environment to acceptable levels, as demonstrated by the MHA analysis in Chapter 13 of the NBSR SAR.

7 INSTRUMENTATION AND CONTROL SYSTEMS

7.1 Summary Description

The NBSR I&C system consists of five major subsystems: the reactor control system (RCS), the RPS, the ESF, the main control panel, and the radiation monitoring system. These subsystems consist of instrumentation, controls, and annunciators, most of which have control or indication devices at the main control panel to allow for remote operation and monitoring by the reactor operator.

7.2 Design of Instrumentation and Control Systems

The NBSR I&C system design criteria consist of the following elements:

- redundancy and diversity
- automatic initiation to mitigate the consequences of abnormal conditions
- fail-safe capability
- single failure will not prevent a safe shutdown of the reactor
- emergency power supply to instruments required for the reactor to be placed in a shutdown condition
- redundant or diverse instruments important to reactor safety and signal cables routed in separate cable trays and cable chases to prevent common-mode failures

The following sections discuss the specific design elements of the separate I&C subsystems.

7.3 Reactor Control System

The RCS provides for the withdrawal and insertion of the four shim safety arms and the regulating rod. The system consists of four individual shim safety arm withdraw/insert circuits, one regulating rod withdraw/insert circuit, their associated interlocks, and the automatic control circuit. The main control panel displays the operating controls and positioning information. The system is enabled by the startup prohibit and withdraw prohibit circuits. In the presence of a prohibit signal from either circuit, the withdrawal of any reactivity control device is prevented, regardless of whether the reactor is being operated in manual or automatic mode. The rundown function consists of selected plant parameters; when there is an indication of an abnormal condition in any of these parameters, the four shim safety arms and the regulating rod will be driven into the reactor core to automatically reduce reactor power. A reactor rundown, once initiated, continues until the condition clears or until the reactivity control devices are fully inserted. TS 3.2.1 requires operability of the shim arms, and TS 4.2.1 requires periodic surveillance tests.

Eight separate channels of nuclear instrumentation monitor the reactor power level and period continuously from shutdown to full power. All eight channels are displayed and recorded on the main control panel for use by the reactor operator. The detectors are located in instrument wells located in the biological shield; hence, the detectors measure only the leakage flux from the core.

Source-range channels NC-1 and NC-2 monitor the reactor power in the source range. Wide-range channels NC-3 and NC-4 provide reactor period trip signals to the RPS. Linear power channel NC-5 provides a reactor power level signal to the automatic flux controller. Wide-range channel NC-6 provides a reactor power trip signal to the RPS. These units are designed to be fail-safe, such that a loss of power will cause a reactor scram. A minimum of one decade of overlap is designed into the transition between source-range and intermediate-range nuclear instrumentation and between intermediate-range and power-range nuclear instrumentation. Normally, the photoneutron source is sufficiently strong that the two source-range channels are not necessary. As a result, these channels are normally deenergized and their detectors removed from the instrument wells.

7.4 Reactor Protection System

The RPS includes the scram, major scram, and moderator dump methods for rapid shutdown of the reactor. A trip of any of the scram relays will cause a loss of power to all of the shim safety arm magnets, thereby decoupling the arms from their respective drive motors and dropping the arms into the core by gravity and compressed spring force. The scram functions are controlled by logic circuits containing relay contacts from the various nuclear and process instrumentation circuits necessary for safe operation of the reactor. If any of the contacts in the scram logic strings are open, indicating that the plant parameters associated with these contacts are not in their normal range, then power is removed from the scram relays resulting in a reactor scram. TS 3.2.2 specifies the instrumentation and scram channel requirements during reactor operation, and TS 4.2.2 prescribes the surveillance requirements for tests and calibrations to check operability of the system. These requirements provide protective action for nuclear and process variables to ensure that the LSSS values are not exceeded.

The normal air monitor channel, irradiated air monitor channel, and stack monitor channel control the relays in the major scram circuit. When any of the three channels detects an excessive activity level, the major scram relays open contacts in the scram logic string and initiate confinement building isolation. The relays also shut the doors at the entrances to the confinement building by tripping the door scram relays, shift the ventilation lineup to recirculation mode by tripping the fan scram relays, and close the neutron guide isolation valves. TS 3.7.1 requires two of the three gaseous effluent monitors (normal air, irradiated air, and stack air) to be operable during reactor operation.

The moderator dump function is also part of the RPS. Negative reactivity can be inserted by removing the moderator from the area immediately above the reactor core. The moderator dump switch on the main control panel provides a backup shutdown option to the reactor operator. Actuating the dump switch opens contacts in the scram logic string which initiates a reactor scram. Additional contacts in this switch open the moderator dump valve DWV-9, which dumps primary coolant above the core into to the D_2O storage tank. These contacts also open the circuit breakers supplying power to the main coolant pumps. TS 3.3.3 requires the moderator dump system to be operable during reactor operation.

7.5 Engineered Safety Feature Actuation Systems

ESFs are designed to mitigate accidents and control the release of radioactive materials to the environment should an accident occur. The NBSR ESFs include the ECS, confinement building, and the ventilation systems. ESFs may be automatically actuated by the protection instrumentation that monitors various parameters during the reactor’s operation or manually actuated by the reactor operator. For example, in addition to shutting down the reactor, a major scram signal will actuate the confinement building isolation system to prevent the release of radioactivity to the environment. This signal may be generated manually by the reactor operator or automatically in response to high radiation levels detected by radiation monitors in the ventilation system.

7.6 Control and Console Display Instruments

The NBSR main control panel in the control room provides all of the information and controls needed by the operator to safely operate the reactor from a centralized location. The center, angled portion of the console provides all of the instrumentation and controls associated with the reactor. Nuclear instrumentation provides overlapping indication of reactor power level from startup to full power, as well as indication of reactor period. Controls and indications are provided on this section of the console to control the shim safety arms, the automatic regulating rod, the primary and secondary cooling pumps, and the operation of the cooling tower. Three annunciator panels, located at the top of the vertical backboard of the console, alert the reactor operator to off-normal conditions in these systems and indicate the source of scrams and rundowns. An additional, three-window annunciator panel located just below the center annunciator panel alerts the operator to scram, rundown, and withdraw prohibit conditions.

The console to the left of the reactor controls provides I&C associated with the auxiliary systems, experimental facilities, and radiation monitoring equipment. Two annunciator panels on this portion of the console alert the reactor operator to off-normal conditions in these systems. Each alarm is labeled with the underlying cause (e.g., reactor D_2O level high). In addition, the number of the annunciator panel and the individual alarm window (e.g., AN3-14) corresponds to the associated annunciator procedure number, thereby assisting the operator in quickly locating the appropriate alarm response procedure to take corrective action for each alarm condition.

The console to the right of the reactor includes the nuclear instrumentation cabinet and additional I&C associated with nuclear instrumentation, experimental facilities, and radiation monitoring equipment.

A second function of the display system is to provide essential information to the emergency control station located outside of the confinement building in the basement level B-2 of the NBSR office building, which is adjacent to the confinement building. Information is available at this location for use during emergencies that result in the confinement building becoming inaccessible.

7.7 Radiation Monitoring Systems

The radiation monitoring system consists of both area and effluent monitors. The area monitors provide indication of radiation levels throughout the containment building. These have been positioned at locations where either experimental work is performed or work involving radioactive material is likely to occur. As a result, the reactor operator can observe radiation levels and warn both experimenters and operations personnel of any unanticipated changes or hazards. The effluent monitors indicate the radioactivity of the air and water that leave the building. All effluent paths are monitored. The monitored paths are exhaust air (gaseous and particulate), secondary coolant (beta-gamma), and sewer discharge (beta-gamma).

All area and effluent radiation monitors alarm in the control room. The effluent monitors can initiate a confinement building isolation automatically in response to high radiation levels detected by the radiation monitors in the ventilation system. TS 3.7.1 specifies the criteria for facility area radiation monitoring and secondary coolant activity monitor requirements. TS 4.7.1 provides periodic testing and calibration requirements for the area radiation monitors (ARMs) and secondary cooling water activity monitor. Exhaust air monitor requirements are specified as part of the reactor safety system channel in TS 3.2.2 and TS 4.2.2, and operability tests of the confinement enclosure system prescribed in TS 4.4 include annual testing of the exhaust radiation monitors with a radiation source.

7.8 Conclusions

The NRC staff has reviewed the design of the I&C systems, as described in the NBSR SAR, and concludes that the systems and TS are adequate to support normal reactor operation and to achieve safe reactor shutdown upon detection of abnormal conditions. The RCS and the nuclear and process instrumentation are sufficient to provide for safe control of reactor power and monitoring of reactor safety parameters. The RPS is adequate for maintaining operation within the LSSS, and the ESF actuation systems are sufficient to respond to abnormal conditions for mitigation of the consequences of postulated accidents. The licensee has shown that all nuclear and process parameters important to safe and effective operation are adequately displayed at the main control panel, and sufficient radiation monitoring is provided to detect abnormal radiation levels and prevent excessive radiation exposure of personnel or release to the environment.

8 ELECTRICAL POWER SYSTEMS

8.1 Normal Electrical Power Systems

The NBSR electrical distribution system is designed to supply the electrical power necessary to operate the NBSR during both normal and shutdown conditions by providing adequate sources of power to all of the equipment and instrumentation necessary for reactor operation. Three independent, underground 13.8-kilovolt (kV) primary feeders supply electrical power to the NBSR. Loss of any single feeder will not interrupt operation of the NBSR. Only one input feeder is required to support a normal shutdown of the reactor. Power from the primary feeders is not required to achieve and maintain safe-shutdown conditions.

The redundancy and the protective scheme of the electrical distribution system prevent any single failure from causing loss of offsite power. The input switchgear is arranged in a linear configuration and is divided into three sections. Each section can be separately powered from its input transformer. The normal alignment has the three sections cross-connected through electrically operated tie-breakers.

Each of the electrical loads associated with the normal and emergency operation of the reactor is powered from one of the two parallel sets of motor control centers. The reactor does not require electrical power to shut down, and a loss of electrical power will cause an automatic shutdown of the reactor.

8.2 Emergency Electrical Power Systems

The NBSR emergency electrical power distribution system is designed to provide emergency power should a complete loss of offsite power occur.

Each of the two 150-kW diesel-powered alternating current (ac) generators is capable of supplying emergency power to all necessary emergency equipment. Each diesel engine set is connected to a single-bearing ac generator rated at 150-kW, 226 amperes with a 0.8 power factor, 480 volts, 3 phases, and 60 cycles. Each control panel is equipped with an automatic voltage regulator, voltage regulator switch, manual field rheostat control, generator voltmeter, generator ammeter, frequency meter, and an AUTO/STANDBY selector switch. TS 3.6, “Emergency Power System,” requires that at least one diesel generator be operable during reactor operation. TS 4.6, “Emergency Power System,” requires that each diesel generator be tested for automatic starting and operation at least quarterly and under a simulated complete loss of offsite power at least annually. Should one of the diesel generators become inoperative, the second diesel generator is start tested at least monthly.

The 125-volt direct current (dc) station battery is capable of independently supplying the emergency loads. The station battery consists of sixty 2-volt, lead-acid-type batteries with a capacity of 880 ampere-hours. This capacity allows the supply of the dc bus loads, which total approximately 100 amperes, for approximately 8 hours. If ac power is lost to the input of the online uninterruptible power supply (UPS), the trickle charge of the station battery ceases, and the battery automatically supplies the loads on the dc distribution panel directly and the critical power panel loads indirectly through the inverter of the UPS. When ac power is restored, either from the diesel generator or from another source, the UPS rectifier automatically resumes charging the battery, and the UPS automatically resumes supplying power to the critical power panel and the dc distribution panel. To ensure a safe-shutdown condition and to give an adequate response in emergency situations, the following essential loads are powered by the dc bus:

- dc-powered shutdown cooling pumps
- dc-powered emergency ventilation system fans and controls
- valve control power
- emergency lighting
- UPS inverter
- reactor rod control
- reactor process instrumentation
- nuclear instrumentation

TS 3.6 requires that the station battery be operable during reactor operation. TS 4.6 requires that the voltage and specific gravity of each cell of the station battery be tested annually and a discharge test of the entire battery be performed once every 5 years.

8.3 Conclusions

The NRC staff reviewed the design of the electrical power system, as described in the NBSR SAR, and concludes that the systems and TS are adequate to support normal reactor operation and to achieve and maintain safe reactor shutdown under all abnormal operating conditions. The electrical power system is sufficient to provide power to all equipment loads required for reactor operation and instrumentation needed for safe control of reactor power and monitoring of reactor safety parameters.

9 AUXILIARY SYSTEMS

9.1 Heating, Ventilation, and Air Conditioning Systems

Section 9.1 of the NBSR SAR describes heating, ventilation, and air conditioning (HVAC) systems. The intake air supply for the normal ventilation system includes both fresh air from outside and recirculated air within the building, while for the emergency ventilation system, the air inside the confinement building is recirculated after being appropriately filtered. All effluent air exhausted from the confinement building is monitored for radioactivity. Section 7.7 and Section 11.1.4 of this SER discuss the monitoring of radioactive effluents. Each of the effluent pathways has filter banks which are monitored for particulate activity, air samples are withdrawn for counting of gaseous activity, and a monitor measures activity in the stack at the point of release to the environment. If high radiation levels are detected, the normal ventilation system will be shut down, all building closure devices will be sealed, and the emergency ventilation system will be activated.

Three separate exhaust systems operate during normal operation of the reactor. The normal exhaust system takes air from those areas supplied by conditioned air and combines with the exhaust air from fume hoods. The reactor basement exhaust system draws air from the process equipment area. Finally, the irradiated air exhaust system takes air from potentially contaminated areas. Air from each of these systems passes through similarly designed HEPA-type filtering systems. The air is then released through the stack after being appropriately diluted and monitored for an acceptable level of radioactivity.

The NBSR contains several other HVAC systems that are routinely in operation to provide a comfortable environment for personnel and equipment. These include the pump room HVAC system, the cold laboratory HVAC systems, the warm laboratory HVAC systems, and other miscellaneous office and laboratory ventilation systems. Each of these HVAC systems brings in fresh air from outside the building and has associated exhaust fans. The individual systems vary the amount of outside air based on environmental conditions and time of year. The failure of any of these systems will not affect the emergency safety features of the NBSR. TS 3.5 requires that the emergency recirculation system and emergency exhaust systems are both operable before reactor operations begin.

Based on the above discussion, the NRC staff concludes that the HVAC systems are adequate to maintain conditions conducive to reliable reactor operation, including instrumentation and equipment temperature control and operator comfort. Additionally, the NRC staff concludes that the ventilation system design and controls are adequate to control the release of radioactive materials during normal reactor operation and abnormal facility conditions.

9.2 Handling and Storage of Reactor Fuel

Section 9.2 of the NBSR SAR describes handling and storage of reactor fuel. The fuel handling system for the NBSR allows used fuel assemblies to be removed from the core and moved to the fuel storage pool. It also allows fuel assemblies to be moved to any fuel location in the reactor and is used to add new fuel assemblies to the reactor.

Movement of fuel assemblies occurs entirely beneath the top shielding plug. Each fuel assembly location is located beneath a pickup head. The pickup heads can extend down and engage the top of the corresponding fuel assembly. Locking slots in the pickup heads engage pins in the top of the fuel assembly and ensure that the fuel assembly will not be dropped. The pickup head is then raised, which removes the fuel assembly completely from the core. The fuel assembly can then be engaged by 1 of 10 fuel transfer arms. Once a fuel assembly has been latched by one of the transfer arms, the pickup head can be released. The fuel assembly can then be moved over any fuel location in the core by transferring the attached fuel from one transfer arm to another. Once over the desired fuel location, the appropriate pickup head is engaged, the transfer arm is moved out of the way, and the pickup head can lower the assembly into the core. TS 3.9.2, "Fuel Handling," requires that fuel element latching is verified before further fuel movement. TS 3.9.2 also requires that a cooling time in hours equal to the operating power level in megawatts elapse before movement of fuel from the reactor. TS 6.1.3, "Staffing," requires the presence of a senior reactor operator in the facility whenever fuel is being moved within the reactor vessel.

One of the locations that the fuel transfer arms can reach is the fuel transfer chute. This chute extends upward to the top of the reactor shielding and is the pathway for new fuel assemblies to enter the core. The shielding plug over the transfer chute is removed, and new fuel assemblies are lowered until they can be engaged by the fuel transfer arms. A cylinder extends into the D_2O moderator to minimize the surface area of D_2O moderator that is available for evaporation during new fuel additions.

The fuel transfer chute also extends down to the lower levels of the reactor building where the spent fuel pool is located. To lower a fuel assembly, a hydraulic receiver is raised from the bottom of the transfer chute to a position where it can accept a spent fuel assembly. It is then lowered to the level of the spent fuel pool and rotated to a horizontal orientation for entry into the spent fuel pool. The spent fuel assembly will heat up once removed from the moderator and stabilize at a temperature that depends on the operating history of the reactor and the amount of time since the reactor was shut down. TS 3.9.2 requires a time limit of 1 hour for every megawatt of final power level before shutdown is required to limit the temperature that a fuel assembly may reach during transfer to the spent fuel pool. As the spent fuel pool contains H_2O, the assembly is allowed to dry during the passage through the fuel transfer chute. As an additional precaution, the chute is isolated from the reactor before its entry into the H_2O of the spent fuel pool, as this will generate steam. Auxiliary cooling is available should a fuel assembly hang up in the transfer chute.

Fuel in the spent fuel pool is stored in specified locations along the edge of the pool. Each full-length fuel assembly is hung by its pickup head. Boral is located along the back wall and extends out from the wall between each fuel element. Latching mechanisms preclude placing any fuel assembly closer than the 7.6-cm (3-in.) minimum separation required for criticality safety.

For shipping, fuel assemblies are cropped to remove nonfuel portions. The trimmed fuel lengths are temporarily stored in special racks that contain boral or similar materials and are configured to ensure criticality safety. TS 3.9.1, "Fuel Storage," requires that fuel elements be stored in conditions that will not exceed a k-effective of 0.90 under optimum conditions of moderation and reflection. TS 3.9.1 also specifies that water chemistry, level, and temperature in the spent fuel storage pool shall be maintained to ensure the integrity of the fuel elements.

Based on its review of the information presented in the NBSR SAR and the requirements of the NBSR TS, the NRC staff concludes that the fuel storage facility design, fuel handling procedures, and TS requirements provide adequate measures to preclude inadvertent criticality and unauthorized fuel movement and to minimize the risk of mechanical or chemical damage to the fuel during movement and storage.

9.3 Fire Protection Systems and Programs

The NBSR confinement building is constructed of steel and concrete. Most of the interior structures are made of fire-resistant materials, which limit the amount of combustible materials in the facility. Inventories of transient flammable materials are minimized. In the event of an upset condition caused by a fire, the shim arms would drop into the core by gravity with spring assist to shut down the reactor.

The building contains automatic fire detection systems supplemented by manual pull boxes throughout the facility, which the NRC staff observed during a site visit. Hose stations and fire extinguishers are also located throughout the facility. According to the NBSR SAR, all fire detection and alarm systems tie into the NIST fire system. This system is annunciated in the control room and includes visual and audible alarms throughout the reactor facility. The alarms are also sent to the NIST fire department which is on site and operational 24 hours a day, 7 days a week. The NIST fire department meets the National Fire Prevention Association requirements for firefighting training and also has cooperative agreements with local fire departments. The NIST fire department also periodically inspects fire extinguishers and hydrants at the reactor facility.

Based on these observations and the above discussion, the NRC staff concludes that adequate measures are in place to prevent and mitigate fire at the NBSR and that fire damage does not pose a significant threat to the safe operation or shutdown of the reactor.

9.4 Communication Systems

The NBSR contains multiple systems for communication between the control room and the rest of the facility. These include the NIST telephone system, the page phone system, and sound-powered phones. The page phone system is connected to a speaker system that allows announcements to the entire building and can be accessed through the NIST telephone system.

A separate microphone and speaker system is part of the refueling system that connects the refueling pool, the control room, and the reactor top. Also available are handheld radios for portable communication. NIST also has an emergency alarm system tied into speakers throughout the building.

Based on its review of the NBSR communications systems, the NRC staff concludes that adequate communication systems are in place at the NBSR to convey information between reactor operators and facility personnel during both normal operations and abnormal conditions.

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

Aside from the fuel assemblies, byproduct, source, and special nuclear material specifically associated with operation of the reactor, these types of materials at the NBSR are licensed under NRC License SNM-362 and not discussed further here. An AmBe neutron startup source of approximately 2 Ci is kept in a shielded container in the reactor source storage room.

9.6 Cover Gas Control in Closed Primary Systems

Section 9.5 of the NBSR SAR describes the cover gas control and processing system. The NBSR has a helium cover gas system for tanks and vessels that contain D_2O. These tanks are the reactor vessel, the D_2O storage tank, the emergency cooling tank, and the purge tank. The cover gas system includes chillers, gas pumps, a helium recombiner, and associated instrumentation. The system maintains appropriate pressure in the cover gas and contains charcoal cartridges that collect radioactive gases rather than allow their release to the process room. The helium cover gas system also contains a cold trap for the removal of water vapor. Six bulk helium tanks supply makeup helium. A separate backup supply of eight cylinders is available.

Based on its review of the cover gas control and processing system at the NBSR, the NRC staff concludes that the cover gas control and processing system is adequate to prevent the intrusion of ambient air into primary coolant systems and to minimize radiation doses to facility personnel from the generation of ^{41}Ar.

9.7 Other Auxiliary Systems

9.7.1 Carbon Dioxide System

The NBSR has a carbon dioxide gas system for purging air from the cavity between the reactor vessel and the thermal shield. The pneumatic transfer tube is also purged and filled from the carbon dioxide system. A bulk storage tank located outside the confinement building supplies the gas. Elimination of normal air from areas containing significant neutron flux minimizes the production of ^{41}Ar from reactor operation.

9.7.2 Instrument Air System

The NBSR has an instrument air system that supplies air at 689 kPa (100 psig) to loads within the facility. The system supplies air to pneumatic valve operators, ventilation control valves, air ejectors, and other loads within the confinement building, guide hall, pump house, and laboratory spaces. The instrument air system is required for proper operation of the confinement isolation doors, HVAC damper controls, and other systems throughout the facility.

The main NIST compressed air facility supplies instrument air. Air receiver tanks are used to ensure that required air is available when needed even if the main supply is interrupted. The actuators for the dampers and valves that are required for confinement isolation each have individual air receivers. Should the main supply of instrument air be interrupted, two air compressors are available in the facility that are powered from the diesel-backed emergency power buses. The pressure in the instrument air system is noted on a gauge in the control room and an annunciator is actuated if the air pressure drops below 590 kPa (85 psig). The two standby compressors will automatically start at 620 kPa (90 psig) and 550 kPa (80 psig), respectively.

9.7.3 Plant Chilled Water System

The plant chilled water system supplies water in support of experimental facilities and also to climate control equipment. The main NIST chilled water facility supplies chilled water to the NBSR.

9.7.4 Fuel Storage Pool Cooling System

The fuel storage pool cooling system is designed to remove heat from fuel assemblies stored in the pool and also to remove particulate matter from the water to maintain water clarity. The pool is nominally filled with 114,000 L (30,000 gal) of demineralized water, and the water treatment system supplies makeup water. The water is maintained at approximately 10 degrees C (50 degrees F).

The system has a maximum heat transfer capacity of 131.9 kW to the chilled water system. Water is circulated using two centrifugal pumps controlled from the control room. A mixed-bed ion exchanger maintains the purity and clarity of the pool water. A low water level in the pool will cause an alarm in the control room.

9.7.5 Thermal Shield Cooling System

The thermal shield cooling system is designed to remove heat from the thermal shield and also to maintain water clarity. Two centrifugal pumps are available to circulate the cooling water to a heat exchanger rated at 762 kW of heat removal. The system includes a storage tank that is used to maintain head for the pumps. Two mixed-bed ion exchangers maintain the purity and clarity of the cooling water. The system has extensive instrumentation including monitors for system temperature, flow, and storage tank level. Shim arm rundown is caused by low flow in the system.

9.7.6 Thermal Column Tank Cooling System

The thermal column tank cooling system removes heat from the bismuth shield and aids in the thermalization of neutrons in the thermal column. While the system uses 908 L (240 gal) of D_2O, it is completely independent of the primary D_2O system. Flow is provided by two pumps which circulate the D_2O through a heat exchanger and a filter system. Low-flow indication will result in a rundown of the reactor shim arms and cause an alarm. A helium bottle maintains a cover gas on the system.

9.7.7 Experimental Demineralized Water Cooling System

The experimental demineralized water cooling system supplies cooling water at each experimental facility. It also provides water pressure for the refueling cannon. The system contains a heat exchanger, an ion exchanger, and a storage tank of 9,085 L (2,400 gal).

9.8 Conclusions

Based on the above discussions, the NRC staff concludes that the auxiliary systems at the NBSR support safe operation of the facility and aid in the safe shutdown of the reactor. Further, the TS ensure that fuel elements are appropriately handled and that there will be 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 NBSR serves as a source of radiation for use in the research programs of the NCNR. It provides a unique radiation source that allows studies that are currently not possible at other reactor facilities in the United States. The experiment types range from fundamental research on particle properties to applied questions of technology development. Both university and commercial researchers perform experiments. The TS provide limitations for the effect on reactivity of all experiments and means for technical and safety review of experiments.

10.2 Experimental Facilities

Experimental facilities include:

- nine radial beam tubes
- two through tubes that pass completely through the reactor
- a cold neutron source that supplies seven neutron guides into an adjacent building
- a 137-cm (54-in.) by 132-cm (52-in.) thermal column
- four pneumatic transfer thimbles.
- 18 vertical thimble locations (both core and reflector locations)

10.2.1 Radial Beam Tubes

Section 10.2.1 of the NBSR SAR describes the design and construction of the nine radial beam tubes at the NBSR. The beam tubes range in size from 12.7 cm (5 in.) to 15.2 cm (6 in.) and have a thermal neutron flux at the core end of the beam holes of approximately 2.0×10^{14} n/cm^2-sec during 20-MW(t) power operations. An aluminum plate seals the region between the reactor and the thermal shield. Each beam port has an 45.7-cm-thick (18-in.-thick) shutter consisting mostly of lead that provides shielding during reactor shutdown for removal and insertion of plugs and collimators outside of the shutter.

10.2.2 Through Tubes

Section 10.2.2 of the NBSR SAR describes the design and construction of the two 10-cm (4-in.) tubes which pass completely through the reactor just below the core and off center. One end of each tube has a shutter, and the other end contains a plug.

10.2.3 Cold Neutron Source

Section 10.2.3 of the NBSR SAR describes the design and construction of the cold neutron source which consists of a sealed reservoir of liquid hydrogen next to the core that lowers the energy of emerging neutrons. These lower-energy or “cold” neutrons are then transported through guides to instruments in an adjacent building.

The heat absorbed by the hydrogen and its aluminum containment vessel is removed by the boiling of the liquid hydrogen. The hydrogen is then reliquefied by a closed-cycle helium gas refrigeration unit. The entire hydrogen system is designed to ASME code for pressure vessels with a maximum design stress of 41 megapascal (MPa) (6,000 psi), which is many times the maximum design working pressure.

Helium gas at higher than atmospheric pressure surrounds all hydrogen components to preclude the introduction of oxygen with the hydrogen.

10.2.4 Thermal Column

Section 10.2.4 of the NBSR SAR describes the design and construction of the thermal column which allows for irradiation of large specimens. Radiation from the core first passes through a bismuth gamma ray shield which is cooled by D_2O. The remaining neutrons then travel through a 94-cm (37-in.) thick collection of graphite blocks that has a total cross-section of 137-cm (54-in.) by 132 cm (52 in). Some of the graphite blocks are removable. Access to the graphite for irradiation is available through a vertical hole above the graphite or through the face of the graphite. A boral curtain acts as a thermal neutron beam shutter. When not being used, the thermal column is covered by a shield door.

10.2.5 Pneumatic Tube System

Section 10.2.5 of the NBSR SAR describes the design and construction of the pneumatic transfer system which is available to insert samples into four locations within the reactor vessel. The sample containers, commonly known as "rabbits," have an inner diameter of approximately 2.5 cm (1 in). Samples can be sent into the reactor for irradiations lasting from a few seconds to several days. The pneumatic tubes send and receive rabbits from radiological hoods in a radiological laboratory in the reactor basement.

10.2.6 Vertical Thimbles

Section 10.2.6 of the NBSR SAR describes the design and construction of vertical thimbles. A number of irradiation positions are available in both the core and in the reflector for experimental use. Seven 8.9-cm (3.5-in.) thimble locations are available, each of which has approximately 0.5 L/s (8 gpm) of coolant flow. One of the seven locations is used for the regulating control rod. At the 8.9-cm (3.5-in.) thimble locations, the fast flux is depressed and the thermal flux is enhanced. There are four 6.4-cm (2.5-in.) thimble locations in the core, each cooled by approximately 0.63 L/s (10 gpm) of coolant flow. The 6.4-cm (2.5-in.) thimble locations are closer to the fuel than the 8.9-cm (3.5-in.) thimble locations and have a higher fast flux. An additional seven thimble locations for experiments up to 8.9 cm (3.5 in.) are available in the reflector. While additional cooling can be provided for the in-core thimble locations, no supplementary cooling can be provided to the reflector locations.

10.3 Experiment Review

Proposals for the introduction of any new experimental instruments involving the NBSR must be submitted in writing to the NCNR Safety Evaluation Committee (SEC), in accordance with TS 6.2.3, "SEC Review Function." Among other information, the proposals are required to include instructions on how to shut down the instrument and place it in a safe configuration. These instructions must be posted by the instrument during use and also provided to the reactor operations staff.

TS 6.2, "Review and Audit," specifies the membership of the SEC, which must include at least two members of the NCNR and one member from the Health Physics Group. All members of the SEC are senior technical personnel.

The NCNR Safety Representative and the Hazard Review Committee review and approve the use of experimental facilities. The Hazard Review Committee is composed of members from the scientific, technical, and reactor operations staff and includes at least one representative from the Health Physics Group, one from the SEC, and one from reactor operations.

Use of experimental facilities requires a written proposal. The experiment proposals must include detailed descriptions of the experiment, the types of samples to be studied, and the potential risks. The Safety Representative may authorize the use of materials that are within the approved safety envelope. The Hazard Review Committee is responsible for review of all materials that are to be introduced that are outside of the existing safety envelope. No hazardous material may be introduced into the confinement building without the review and approval of the Director of the NCNR.

TS 3.8, "Experiments," and TS 4.8, "Experiments," also include specific criteria for evaluation and approval of experiments at the NBSR. These include a limit on the change in reactivity of 0.5 %$\Delta\rho$ for any experiment and a limit on the sum of the absolute values of all experiments of 2.6 %$\Delta\rho$. These reactivity values are within the reactivity insertion rate limit assuming at least 0.5 seconds for removal of experiments. As the time to remove experiments from the pneumatic tube is less than 0.5 seconds, the total reactivity for all pneumatic tubes is 0.2 %$\Delta\rho$. TS 3.8 also requires that experiment malfunctions must not cause malfunctions in other experiments and reactor transients must not cause experiments to fail so as to contribute to accidents. Containment measures are specified for explosive or metastable materials, as well as for experiments containing materials that are corrosive to reactor components or reactive with reactor coolants.

The specific review criteria, along with the experiment review process, provide confidence that any experiment performed at the NBSR can be performed safely and any unanticipated radioactive releases will be within the limits of Appendix B to 10 CFR Part 20.

10.4 Conclusions

The NRC staff reviewed the NBSR experiment program and associated requirements specified in the NBSR TS against the guidance found in NUREG-1537 and ANSI/ANS-15.1. Based on its review, the NRC staff concludes that the review process for experiments and use of experimental facilities provides high confidence that appropriate precautions are taken to minimize the risk to personnel from unintended radiation exposure. Further, the NRC staff concludes that the review process provides reasonable assurance that the use of experiments or experimental facilities will not damage the fuel and thus not pose a significant risk to the health and safety of the public, licensee personnel, or the environment.

11 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

11.1 Radiation Protection

11.1.1 Radiation Sources

The primary source of radiation at the NBSR is the reactor itself. The reactor core is surrounded by deuterium moderator and steel and concrete shielding to reduce direct radiation doses to surrounding areas.

During normal operations, the reactor generates neutrons for a number of research purposes. A thermal column is available for large cross-sectional exposures and, when used, is typically controlled as a high-radiation area. Beam ports for cold neutron experiments allow low-energy neutrons to pass from the reactor through shielded neutron guides and pass through a 25-ft-thick concrete wall into the guide hall. Each of the seven neutron guides has a keyed shutter along with status indication. The reactor also has a pneumatic transfer system for in-core experiments that can create radioactive materials.

As described in Chapter 11 of the SAR, the airborne radioactive materials generated during reactor operation of principal concern are ^{41}Ar and ^{3}H. Argon is a natural component of the atmosphere and becomes activated to ^{41}Ar upon neutron bombardment. ^{41}Ar production is minimized by surrounding the core structure with a carbon dioxide gas blanket and conducting activities such as maintenance in ways that minimize air intrusion into volumes subjected to neutron flux. ^{3}H is generated by the neutron bombardment of deuterium (^{2}H) in the reactor moderator and builds up during reactor operation. The primary moderator is periodically changed out to maintain the concentration of ^{3}H in the D_2O well below the limit specified by TS 3.7.1 of 5 Ci/L. TS 4.7.1 requires annual surveillance of the primary activity when the concentration is below 4 Ci/L and quarterly sampling otherwise. The action to keep primary coolant activity at 1 Ci/L or less reduces airborne ^{3}H levels from D_2O releases and is an ALARA measure.

According to Chapter 11 of the SAR, airborne concentrations of radioactive materials are significantly less than 1 derived air concentration (DAC), and calculated dose rates from ^{41}Ar for typical occupancy times result in an annual dose rate to facility personnel of 0.02 mSv per year (mSv/yr) (2 mrem/yr). ^{3}H doses to facility personnel average 0.4 mSv/yr (40 mrem/yr) or less for most personnel. Reactor operators involved in activities that involve exposure to D_2O, such as refueling, experience higher annual doses but normally not in excess of 1 mSv/yr (100 mrem/yr) and well within the 10 CFR 20.1201 limit of 50 mSv/yr (5,000 mrem/yr).

Liquid radiation sources at the NBSR consist primarily of activation products of the coolant and reactor components, principally ^{16}N and ^{3}H. ^{16}N has a 7-second half-life and is a radiation hazard only during reactor operations or immediately after reactor shutdown. The primary coolant piping that presents an external radiation hazard is identified and shielded. ^{3}H is a low-energy beta emitter and is not an external radiation hazard. Additional radionuclides are created in the primary coolant but, during operation, are less of a concern than the ^{16}N. Dose rates in the vicinity of primary coolant piping during full-power operation can be up to 60 rem/hr, though the dose rates drop to a few mrem/hr in shielded areas of the process room.

Radionuclides in liquids are also present, though at much lower activity levels than in the primary coolant, in the thermal column D_2O tank coolant and the thermal shield cooling system because of neutron activation of the coolant and structural materials. Small quantities of radionuclides in liquids may also be present in the reactor secondary coolant and the fuel storage pool from leakage or contamination from other systems. TS 4.7.1 requires monitoring of the secondary coolant system to detect any leakage through heat exchangers from the primary system.

NBSR operations generate solid radioactive materials. Chief among these are the spent fuel assemblies. After irradiation in the core, the spent assemblies are moved to the spent fuel storage pool. Section 9.2 of this SER discusses spent fuel movement and storage. Reactor shims are also activated after time in the reactor. These are also stored in the spent fuel storage pool and shipped off site with nonfuel element metal pieces. The shim bodies may be stored in shielded dry storage wall cavities. Other solid radioactive sources include reactor resins and filters, shielding plugs, neutron beam shields, experiment components from the high-flux location, and activated samples. Solid radioactive waste is disposed of in accordance with appropriate NRC regulations and is transferred to organizations authorized to receive the material.

Instrument check sources and calibration sources are also used at the reactor facility but are licensed under a separate byproducts materials license (SNM-362) and not under the reactor license.

TS 3.8.2, “Materials,” controls introduction of materials into the reactor as part of experiments and requires evaluation to ensure that appropriate safeguards are used for control of irradiated materials.

The NRC staff concludes that the description and characterization of the radiation sources at the NBSR are reasonable for a test reactor of this type and size and that this information provides sufficient information to evaluate the Radiation Protection Program and controls described in the remainder of this section of the SER.

11.1.2 Radiation Protection Program

Chapter 11.1.2 of the NBSR SAR summarizes 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 NBSR and to keep all radiation exposures well within regulations and guidelines.

The responsibility for administering the Radiation Protection Program at the NBSR belongs to the reactor health physics section as required by TS 6.3, “Radiation Safety.” This group is part of the NIST Health Physics Group and is in a separate reporting chain from the Director of the NCNR, which includes reactor operations. A reactor senior health physicist oversees the activities of the reactor health physics section and is responsible for implementing the Radiation Protection Program at the NBSR. Normal staffing for the section is two to four health physicists and two to four radiation protection technicians.

The responsibilities of the reactor health physics section include calibration of survey instruments, effluent monitoring, radiation and contamination surveys, training, sample analysis, and personnel monitoring. TS 6.2.1, “Composition and Qualifications,” requires that a health physics representative serve on the SEC to review new experiment equipment installations.

TS 6.4, “Procedures,” requires procedures for implementing the Radiation Protection Program. Both operations and health physics personnel must review the procedures before their adoption.

Plans and procedures not related to the NBSR are maintained under the NIST materials license (SNM-362).

According to the licensee, all individuals granted unescorted access to the NBSR facility receive radiation safety training. This training covers information required by 10 CFR Part 19, "Notices, Instructions and Reports to Workers: Inspection and Investigations," including basic radiation information, signage and alarm responses, and proper use of dosimetry. Individuals who have duties involving radioactive material or reactor experiments receive additional training related to their job duties and including proper use of survey instruments and specific regulations and procedures that apply to their work. The NBSR SAR indicates that refresher training is required every 24 months.

Records relating to personnel dosimetry or exposure investigations, as well as effluent records, are retained for the life of the facility as required by TS 6.8, "Records." The SAR indicates that other documents related to radiation protection be retained for a minimum of 10 years.

The NRC staff concludes that the structure and strategy of the Radiation Protection Program for the NBSR are consistent with the guidance of ANSI/ANS-15.11, "Radiation Protection at Research Reactor Facilities," and that the program as implemented is adequate to provide reasonable assurance that personnel are protected from radiation hazards.

11.1.3 ALARA Program

As described in Chapter 11.1.3 of the SAR, NIST has established an ALARA program for the NBSR. The program specifically emphasizes proposed experiments, planned activities that could result in significant exposures, and ambient radiation levels within NBSR. The program also includes ongoing supervisory reviews of worker and public doses to detect trends that may need corrective action.

The ALARA program includes extensive use of engineering controls to minimize facility dose rates. Shielding is used to reduce radiation levels in work areas. Activities above preset cumulative dose levels require a formal operating plan, pre-job meetings, identification of methods to reduce exposure, and specific oversight by the health physics staff. Consistent with the ALARA program, the ventilation systems go into a recirculation mode of operation during upset or abnormal operating conditions and a standby charcoal filter operates.

These methods are typical for ALARA programs, and the NRC staff concludes that they provide reasonable assurance that personnel exposure to radiation will be minimized.

11.1.4 Radiation Monitoring and Surveying

According to the licensee, the licensee's Health Physics Group maintains numerous fixed and portable radiation detection instruments throughout the NCNR facility. The NRC staff observed examples of such instruments during a facility visit. Backup instrumentation is available at the radiation physics building on the NIST site.

Section 7.7 of this SER discusses the ARMs used to alert staff and operators to changing radiation conditions. Ten fixed gamma ARMs are located throughout the confinement building. These monitors have local readouts, warning lights, and alarms to alert workers in the vicinity to increased radiation levels. These ARMs also have readouts and alarms in the control room. The monitors are nominally set for 0.05 mSv/hr (5 mrem/hr) and are adjusted as needed for nonroutine activities. One of the monitors is located in the spent fuel storage pool area and also serves as a criticality monitor, as does the monitor in the new fuel storage area. TS 3.7.1 requires at least two ARMS to be operable on floors C-100 and C-200 in order for the reactor to be operated.

Other fixed radiation monitors are used for detection of personnel contamination. These include hand and foot monitors and portal monitors. These contamination monitors are located at the entrance to the reactor building and other locations, as needed.

Radioactive effluents released through the plant stack are monitored by a Geiger-Mueller detector, as required by TS 3.7.1. The detector readings are calibrated by comparison with volumetric samples taken from the stack flow. ^{3}H effluent is monitored by the building ^{3}H monitoring system. The readings are compared with monthly samples. Additional monitoring is performed on an as-needed basis to support nonroutine activities.

In accordance with the requirements of 10 CFR 20.1501, "General," portable instrumentation is available to survey areas in the NBSR facility for all types of radiation and radioactive contamination that may result from facility operations. The different instruments available can measure alpha, beta, gamma, and neutron radiation. These detectors cover a range from $0.2x10^{-3}$ mSv/hr (0.02 mrem/hr) to 10 Sv/hr (1,000 rem/hr). Surveys using portable instrumentation are performed throughout the facility at least weekly. Survey frequency is adjusted during nonroutine activities to enhance detection of changing conditions. As required by TS 6.8.1, "Records to be Retained for a Period of at Least Five Years or for the Life of the Component Involved if Less than Five Years," records of radiation surveys are retained for a period of 5 years. This TS meets the requirements of 10 CFR 20.2103, "Records of surveys."

An installed gas-flow ion chamber system monitors the ventilation system in the reactor building for ^{3}H. Additional air monitoring equipment is available for use as needed in the facility. This includes both particulate and iodine monitoring as well as ^{3}H monitoring. Detection levels are a small fraction of applicable DAC levels.

The NRC staff concludes that the installed and available radiation detection equipment and the program to use the equipment satisfy the requirements of 10 CFR 20.1501(a)(b) and provide reasonable assurance that doses to personnel will be kept below the limits specified in 10 CFR 20.1201, "Occupational Dose Limits for Adults."

11.1.5 Radiation Exposure Control and Dosimetry

The NBSR is located on the NIST Gaithersburg site which is a controlled area as defined by 10 CFR 20.1003, "Definitions." The site is fenced and access is limited. The NCNR building, which houses the NBSR, is controlled as a restricted area. Access to the building requires training and a coded ID card for entry. Use of radioactive material within the facility is limited to designated areas.

According to Chapter 11 of the SAR, all personnel accessing the NBSR are monitored for external radiation. The facility staff is issued record dosimeters, while temporary personnel may be issued pocket ion chambers as an alternative. Typically, fewer than 20 personnel meet the criteria for required monitoring. Individuals who could exceed 5 mSv (500 mrem) annually are issued a pocket ion chamber in addition to their regular dosimeter. This practice allows for dose trending between quarterly processing of their regular dosimeter. Additional surveillance is performed for declared pregnant workers. Personal dosimeters worn by personnel at the NBSR are provided by a supplier certified by the National Voluntary Laboratory Accreditation Program, as required by 10 CFR 20.1501(c). Extremity dosimeters are available if needed. This monitoring program meets the requirements of 10 CFR 20.1502, “Conditions Requiring Individual Monitoring of External and Internal Occupational Dose,” and is consistent with the guidance of ANSI/ANS-15.11.

Internal monitoring is not normally required at the NBSR. ^3H monitoring for selected personnel is performed as an ALARA measure. A whole-body counting facility is available if needed for assessing potential internal uptakes. Respiratory protection devices are not used at the NBSR for limiting radiological uptakes. However, respirators are used for nonradiological purposes, such as protection during pipe welding and cutting.

The principal concern for exposure control is the neutron beams both in the vicinity of the reactor and in the guide hall. Of particular concern is the area around beam experiments. These experiments are reviewed before they are performed to minimize the radiation from activated materials and to minimize the radiation levels surrounding the equipment. The vicinity of many of the experiments is posted as a radiation area. Experiment design also includes ALARA considerations.

For routinely accessed areas at the NBSR, the design requirement for long-term experiment shielding is less than 0.05 mSv/hr (5 mrem/hr) at 1 ft, with an ALARA goal of 10 percent of this value. Radiation levels of up to 1 mSv/hr (100 mrem/hr) are shielded as practicable but allowed, provided that impact is minimal. Additional controls are used for high-radiation areas as required by 10 CFR 20.1601, “Control of Access to High Radiation Areas,” including additional postings, physical barriers, and warning devices.

Based on the above, the NRC staff concludes that the exposure control and dosimetry program at the NBSR is adequate to monitor and control exposures to personnel below the limits of 10 CFR Part 20, Subpart C, “Occupational Dose Limits.”

11.1.6 Contamination Control

The licensee states that contamination control at the NBSR is accomplished through engineered controls and an aggressive program to eliminate any contamination below detectable levels. Hoods and glove boxes are used to contain potential contamination. Area surveys for contamination are performed at least weekly. Any contamination found is cleaned to nondetectable levels, with the exception of contamination in nontraffic areas of higher external radiation where cleaning would be of minor benefit and inconsistent with the ALARA program.

11.1.7 Environmental Monitoring

The licensee states that it performs environmental monitoring to verify that doses to the public are below the limits specified in 10 CFR 20.1301, “Dose Limits for Individual Members of the Public.” Chapter 11.1.7 of the NBSR SAR describes an environmental monitoring program that includes radiation surveys, soil and vegetation sampling, and surface water sampling in the vicinity of the NBSR. This sampling program is required by TS 4.7.2, “Effluents.” Environmental samples are evaluated for activation products and fission products. Water samples are evaluated for ^3H. The NBSR SAR indicates that samples are taken quarterly at a minimum of four locations for each sample type. Environmental samples are further supplemented by radiation readings from thermoluminescent dosimeters or other radiation detection devices appropriately placed at the NIST site boundary. The NRC staff reviewed the methodology for the collection and analysis of environmental samples and found that the environmental monitoring program is appropriate for detecting the types of radioactive effluents that could be released to the environment by the NBSR at a sufficient variety of locations. Based on this finding, the NRC staff concludes that the environmental monitoring program is adequate for determining compliance with the requirements of 10 CFR 20.1301.

11.2 Radioactive Waste Management

11.2.1 Radioactive Waste Management Program

Chapter 11 of the NBSR SAR describes the NBSR Radioactive Waste Management Program. The program is assigned to the Health Physics Group, with a health physicist assigned primary oversight of the program. As described in the NBSR SAR, the program includes waste stream characterization.

The SAR describes a policy of radioactive waste minimization consistent with ALARA objectives. Waste minimization practices are used throughout the facility to minimize disposal costs. These practices include the use of materials with low neutron activation potential and the design of experiments to be reused and to minimize both activation and the potential for contamination. Efforts to minimize radioactive waste also include disassembling and segregation so that only activated or contaminated materials enter the waste stream.

The NBSR also has available 33 shielded concrete cavities for long-term storage of radioactive materials that have potential future use. These cavities are also used to allow radioactive decay of materials before disposal.

Sections 11.2.2 and 11.2.3 of this SER address the characterization, monitoring, and release of wastes. TS 6.8 requires that records of gaseous and liquid effluents released to the environment be retained for the life of the facility and that records of solid radioactive waste shipped off site be retained for a minimum of 5 years.

11.2.2 Radioactive Waste Controls

As described in the SAR, radioactive waste at the NBSR is first controlled through waste minimization efforts to limit initial generation of radioactive material. These efforts also are designed to preclude the production of mixed waste. Radioactive waste is identified either by process knowledge or by direct survey. For activation materials that are not easily surveyed, calculations are performed based on the materials present and the neutron irradiation characteristics.

Waste materials are collected at the point of generation in marked waste containers. The waste materials are then transported to another location of the NIST grounds for final disposition.

Surveys of potential waste material are performed in a low-background environment and use the appropriate detectors for the type of radiation emitted. A commercial HEPA-filtered waste compactor is used to minimize waste volume.

According to the licensee, liquid radioactive waste is sampled and analyzed to confirm that waste released to the sanitary sewer meets the requirements of 10 CFR 20.2003, “Disposal by Release into Sanitary Sewerage Requirements,” for solubility and concentration.

A Geiger-Mueller detector on the plant stack flow path monitors gaseous radioactive effluent. In the event of upset or abnormal conditions, the ventilation system can be set to recirculation and a standby charcoal filter placed into operation. TS 3.7.1 requires that effluent monitoring be operable so that the reactor can operate. TS 3.7.2, "Effluents," limits effluent releases such that the total exposure to any person at the site boundary shall not exceed 1 mSv (100 mrem) per calendar year, excluding external dose from the facility. These controls provide reasonable assurance that offsite doses to the public will be below the limits specified by 10 CFR Part 20.

11.2.3 Release of Radioactive Waste

Section 11.2.2 of this SER describes releases of liquid and gaseous waste. All solid radioactive waste is disposed of by transfer to licensed disposal sites or processing facilities. All waste is packaged and transported as required by appropriate NRC regulations and applicable State licenses of the recipient.

Based on the above information, the NRC staff concludes that the effluent radiation monitoring program at NIST is adequate to quantify and characterize the gaseous and liquid effluents released from the facility and keep effluent concentrations below the limits of 10 CFR Part 20, Appendix B. Further, the program is sufficient to provide reasonable assurance that doses to members of the public from effluents are well below the limits of 10 CFR 20.1301.

11.3 Conclusions

The NRC staff has reviewed the Radiation Protection and Radioactive Waste Management Programs, as described in the NBSR SAR, and concludes that the programs, systems, and TS are adequate to ensure that normal facility operation will not cause an undue risk to facility personnel, the public, or the environment from radiation or radioactive materials. The Radiation Protection Program has the organization, equipment, and personnel sufficient to maintain personnel doses below the limits specified in 10 CFR Part 20 and ALARA. The Radioactive Waste Management Program provides reasonable assurance that the licensee will properly handle and dispose of radioactive wastes generated by the operation of the NBSR.

12 CONDUCT OF OPERATIONS

The conduct of operations involves the administrative aspects of facility operation, the facility emergency plan, 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

Responsibility for the safe operation of the NBSR is vested within the chain of command shown in TS Figure 6.1. TS 6.1.1, “Structure,” and TS 6.1.2, “Responsibility,” provide details of the management requirements of the NBSR. The Chief, Reactor Operations and Engineering (Chief Nuclear Engineer) is delegated responsibility for overall facility operation. The Chief Nuclear Engineer is responsible to the Director, NCNR, for safe operation and maintenance of the reactor and its associated equipment. Individuals at the various management levels, in addition to having responsibility for the policies and operation of the reactor facility, are responsible for safeguarding the public and facility personnel from undue radiation exposure and for adhering to all requirements of the operating license and TS. The Chief, Reactor Operations and Engineering delegates the succession of this responsibility during his absence.

TS 6.1.3 contains the minimum staffing requirements for the NBSR. SAR Section 12.1.3 states that the minimum crew complement during a shift shall be two persons, including at least one licensed senior operator, and that during normal operations, the crew complement for a shift shall be three persons. TS 6.1.3 states that the minimum staffing when the reactor is not secured shall be (1) a reactor operator in the control room, (2) a reactor supervisor present within the reactor exclusion area, and (3) a senior reactor operator present in the facility whenever a reactor startup is performed, fuel is being moved within the reactor vessel, experiments are being placed in the reactor vessel, or a recovery takes place from an unplanned or unscheduled shutdown or a significant reduction in power. This is consistent with ANSI/ANS-15.1 guidance and 10 CFR 50.54(m)(1).

12.2 Review and Audit Activities

Section 12.2.1 of the NBSR SAR describes the SEC, which provides the NCNR with a method for the independent review of the safety aspects of reactor operations. The SEC assists the Director, NCNR, in evaluating reactor operational activities, improving the quality of reactor operational programs, and recommending corrective actions for problem areas.

The SEC comprises at least four senior technical personnel who collectively provide a broad spectrum of expertise in reactor technology (e.g., nuclear, electrical, and mechanical engineering, and radiation protection). The Director, NCNR, appoints the SEC members. At least two members are from the NCNR staff, and one member is from the Health Physics Group. The SEC quorum is three members. The NCNR director may appoint alternates to serve during the absence of regular members. The SEC meets semiannually during reactor operations and as circumstances warrant. Written records of the proceedings, including any recommendations or concurrences, are maintained. The SEC reports directly to the Director, NCNR. TS 6.2.1 specifies the composition of the SEC. TS 6.2.2, “Safety Evaluation Committee Charter and Rules,” contains requirements for the SEC charter.

TS 6.2.3 specifies that the SEC review the following:

- proposed changes to the NBSR facility equipment or procedures when such changes have safety significance, or involve an amendment to the facility license, a change in the TS incorporated in the facility license, or a change pursuant to the applicable criteria in 10 CFR 50.59, “Changes, Tests and Experiments”

- proposed tests or experiments significantly different from any previously reviewed or that involve a change pursuant to the applicable criteria of 10 CFR 50.59

- the circumstances of all reportable occurrences and violations of the TS and the measures taken to prevent a recurrence

- the SEC Charter on a biennial basis and recommend changes to the Director, NCNR

TS 6.2.4, “SEC Audit Function,” specifies the responsibilities of the SEC for conducting audits. The audits include reactor operations, corrective actions, the operator training and requalification program, and emergency planning.

The Safety Assessment Committee (SAC) comprises three senior technical personnel who collectively provide a broad spectrum of expertise in reactor technology. The Director, NCNR, appoints the Committee members. Members of the SAC are not regular employees of NIST. At least two members are required to approve any report or recommendation of the Committee. The SAC meets annually and as required. The Committee audits the NBSR facility operations and the performance of the SEC. The SAC reports in writing to the Director, NCNR.

The NRC staff reviewed the review and audit function against the guidance of ANSI/ANS-15.1 and found the SEC role as described in the SAR and the TS to be consistent with the standard.

12.3 Procedures

The licensee has developed a comprehensive set of written operating procedures for all aspects of facility operation as required by TS 6.3. These procedures address the following:

- startup, operation, and shutdown of the reactor

- fuel loading, unloading, and movement within the reactor

- surveillance checks, calibrations, inspections, and maintenance required by the TS or that may have an effect on reactor safety

- personnel radiation protection, consistent with applicable regulations or guidelines, including commitments and programs to maintain exposures and releases ALARA

- the conduct of irradiations and experiments that could affect reactor safety or core reactivity

- use, receipt, and transfer of byproduct material, if appropriate

Substantive changes to procedures require documented review by the SEC and approval by the Chief Nuclear Engineer. The Chief Nuclear Engineer or his or her deputy may make minor modifications to procedures that do not change their original intent.

12.4 Required Actions

Certain events require specific licensee actions in accordance with the SAR and TS 6.6, "Required Actions."

TS 6.6.1, "Actions to be Taken in the Event the Safety Limit is Exceeded," requires that, if a safety limit is exceeded, the reactor is shut down and operations are not resumed without authorization by the NRC pursuant to 10 CFR 50.36(c)(1). A report to the NRC is then required in accordance with TS 6.7.2, "Special Reports." This report shall contain a complete analysis of the circumstances leading to and resulting from the situation with recommendations to prevent recurrence. Further, the SEC shall review the circumstances of all reportable occurrences and violations of TS and the measures taken to prevent a recurrence and shall recommend action.

TS 6.6.2, "Actions to Be Taken in the Event of an Occurrence of the Type Identified in Section 6.7.2 other than a Safety Limit Violation," requires that if an LSSS is exceeded or an LCO is violated, the reactor is secured and operation is not resumed unless authorized by the Chief, Reactor Operations, or his or her deputy. The occurrence is reported to the Chief, Reactor Operations, or his or her Deputy, and to the NRC, if required by TS 6.7.2. Further, the SEC reviews the occurrence at its next scheduled meeting.

TS 6.7.2 also requires that all reportable occurrences are promptly reported to the Chief, Reactor Operations, or his or her Deputy, and to the NRC and subsequently reviewed by the SEC. Reportable occurrences include the following:

- operation with actual safety system settings less conservative than the LSSSs specified in the TS
- operation in violation of LCO, unless prompt remedial action is taken
- an uncontrolled or unanticipated significant reactivity change
- an uncontrolled or unanticipated significant release of radioactivity from the site
- an engineered safety system component malfunction or other component or system malfunction that could render the affected system incapable of performing its intended safety function
- major degradation of one of the several boundaries designed to contain the radioactive materials resulting from the fission process
- an observed inadequacy in the implementation of major administrative or major procedural controls, such that the inadequacy causes or could have caused the existence or development of an unsafe condition with regard to reactor operation

Based on its review of the information presented in the NBSR SAR and the above-referenced TS, the NRC staff concludes that the required actions specified in the TS are consistent with the guidance of ANSI/ANS-15.1 and 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 protection of public health and safety.

12.5 Reports

TS 6.7, "Reports," specifies the reports that the facility is required to make to the NRC. These include an annual operating report and special reports. Annual operating reports include an operational summary, including the number of unscheduled shutdowns and reasons, information regarding safety-significant maintenance activities, effluent releases, environmental surveys, safety evaluations pursuant to 10 CFR 50.59, and any significant exposures to personnel or visitors. Annual reports are required by TS 6.7.1, "Annual Operating Reports."

The NRC staff evaluated these reporting requirements and found that they are consistent with ANSI/ANS-15.1 and provide reasonable assurance that the facility will report appropriate information regarding routine operation, nonroutine occurrences, and changes to the facility and personnel to the NRC in a timely manner.

12.6 Records

Section 12.6 of the NBSR SAR states that, in addition to those records required by applicable regulations, the licensee commits to retaining the following records for period of at least 1 year:

- records of all safety or safety-related equipment maintenance activities
- violations of the TS
- reportable occurrences
- those technical and safety considerations supporting the recommendations of the SEC, including action taken in response to such recommendations
- records and logs of reactor operations
- records of principal maintenance activities
- records of surveillance activities performed in accordance with Section 4 of the TS

TS 6.8 specifies records that the NBSR 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 kept for an appropriate time. The NRC staff evaluated the requirements of TS 6.8 and found that they are consistent with ANSI/ANS-15.1 and applicable regulations and give reasonable assurance that the facility will maintain appropriate records to facilitate NRC inspection and provide adequate history of the facility.

12.7 Emergency Planning

The licensee requested that a revised emergency plan (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; and NRC Information Notice 97-34, "Deficiencies in Licensee Submittals Regarding Terminology for Radiological Emergency Action Levels in Accordance with the New Part 20," issued June 1997. The NRC staff concluded that the NBSR 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 NBSR EP provides reasonable assurance that the licensee can respond appropriately to a variety of emergency situations and that the NBSR EP will be adequately maintained during the period of the renewed license.

12.8 Security Planning

While Facility Operating License No. TR-5 authorizes possession of an overall quantity of special nuclear material of strategic significance, the license stipulates that less than 5.0 kilograms of this amount be unirradiated. According to the licensee and based on the reported operations of the facility, the remaining 40.0 kilograms of ^{235}U meet the exemption from the requirements in 10 CFR 73.60(a)–(e). Therefore, the licensee must maintain security measures that satisfy the requirements of 10 CFR 73.67, "Licensee Fixed Site and In-Transit Requirements for the Physical Protection of Special Nuclear Material of Moderate and Low Strategic Significance," and 10 CFR 73.60(f), as appropriate. The licensee meets these requirements, and the NRC inspects the licensee's measures for physical security and protection of special nuclear material on a routine basis. A recent inspection and site visit verified that the licensee's security measures satisfy applicable regulations and are acceptable.

12.9 Quality Assurance

The NBSR SAR states that the licensee maintains an established quality assurance program that is consistent with the guidance found in ANSI/ANS-15.8, "Quality Assurance Program Requirements for Research Reactors," issued 1995. The licensee states that the quality assurance program is maintained as part of the administrative rules and procedures at the NBSR.

12.10 Operator Training and Requalification Program

Responsibility for the administration of the requalification program rests with the Chief, Reactor Operations. The licensee's requalification program and training program provide reasonable assurance that the licensee will have technically qualified reactor operators. The program is required by TS 6.1.4, "Selection and Training of Personnel." The licensee submitted a revised Operator Requalification Program with 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 guidance contained in ANSI/ANS-15.4, "Selection and Training of Personnel for Research Reactors," issued 1988.

12.11 Conclusions

Based on the above discussions, the NRC staff concludes that NIST has the appropriate organization, experience levels, and adequate controls through the TS to provide reasonable assurance that the NBSR 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

To help establish safety limits, LSSSs, and LCOs for the NBSR, the licensee analyzed potential reactor transients and other hypothetical accidents for the effects of such events on the reactor fuel and the health and safety of facility personnel, the public, and the environment. None of the credible accidents postulated would lead to the failure of the fuel cladding or the uncontrolled release of fission products. However, the licensee postulated an enveloping event involving the complete blockage of flow to one element, leading to complete melting of the fuel plates. This event would lead to the maximum potential radiation hazard to facility personnel and members of the public. The licensee made no assumptions as to the cause of the failure but evaluated only the potential consequences of this event and not the likelihood or mechanisms of its occurrence. This worst-case accident scenario has been designated as the MHA. The licensee has evaluated other possible accident sequences, and none poses a significant risk of cladding failure or release of fission products.

Those accidents considered for evaluation and analysis are as follows:

- MHA (fuel channel blockage and fuel plate melting)
- insertion of excess reactivity (including experiment malfunction)
- loss of primary coolant
- loss of primary coolant flow (including loss of normal power)
- misloading of fuel

In all of the accident scenarios, the reactor is assumed to be operating with all critical parameters at the most limiting extreme value of their normal range. This ensures that the analysis for each accident scenario uses the worst-case initial conditions that might be anticipated, within the normal limits of operation. The following table shows these conditions.

Parameter	**Limit**	**Value**
Reactor Power	102% of Nominal Rating	20.4 MW(t)
Reactor D_2O Level	Low	3.81 m (150 in.)
Core Inlet Temperature	High	43.3 ºC (110 ºF)
Main Primary Coolant Flow	Low	549 L/s (8,700 gpm)

13.1 Maximum Hypothetical Accident

The licensee has postulated the MHA as a complete blockage of flow to one fuel element by unspecified means. The flow blockage is assumed to result in complete melting of the fuel plates and the release of all fission products to the primary coolant. The reactor is assumed to be shut down by one of the following mechanisms:

- Fuel plates melt and drop out of the core region, which leads to loss of reactivity and shutdown.
- A manual reactor scram occurs as a result of the fission product monitor alarm.
- An automatic major scram (TS 3.2.2, Table 3.2.2) is actuated by the stack monitor alarm.

Once the major scram is actuated by the high stack activity, normal ventilation is secured, confinement is isolated, and emergency ventilation is automatically established. This condition is assumed for the duration of the accident. Since the MHA does not involve a release of primary coolant, only volatile fission products are postulated for release. The licensee uses the ORIGEN2 computer code to determine the inventory of noble gas and iodine fission products in the most heavily irradiated element. TS 3.4, "Confinement System," requires confinement operability for this scenario, and TS 3.5 requires emergency ventilation operability.

All of the noble gas fission products are assumed to be released into the primary coolant and quickly collect in the helium space at the top of the reactor vessel. For the temperatures and coolant chemistry expected in the NBSR MHA, most of the iodine is assumed to be in the form of cesium iodide (CsI), and would remain in solution in the primary coolant water. Accounting for radiolysis of CsI to iodine (I_2) gas, it is assumed that approximately 3 percent of the total iodine release is present as I_2. Based on a review of the literature on iodine releases from fuel failures, the NRC staff concludes that this is a reasonable assumption for the conditions that would be expected for the NBSR MHA.[11,12]

The gaseous fission products in the helium space at the top of the reactor vessel are released to the confinement building, along with helium at a rate based on historical leak rate measurements characteristic of the primary system under emergency ventilation conditions (no normal building exhaust). The emergency ventilation system then determines the exhaust rates to the stack from these spaces. The NRC staff concludes that the removal rates assumed in the analysis by the licensee are reasonable relative to the maximum allowable leak rate in the annual integrated leak rate test required by TS 4.4.

The release of the fission product gases provides the source term for estimating doses to the general public at the 400-m (1,300-ft) exclusion radius established in TS 5.1, "Site Description." The licensee used standard techniques to calculate doses to the public. The codes HOTSPOT (first day) and CAP88 (>1 day) were used for doses from the passage of radioactive clouds. The direct doses were calculated following the methods used in NBSR-9, allowing for the

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11 C.F. Weber, E.C. Beahm, and T.S. Kress, "Models of Iodine Behavior in Reactor Containment," ORNL/TM-12202, Oak Ridge National Laboratory, Oak Ridge, TN, October 1992.

12 T.P. McLaughlin, et al., "A Review of Criticality Accidents," LA-13638, Los Alamos National Laboratory, May 2000.

depletion of the fission products in the building as the gases are released. The licensee calculated the scattering from the air above the confinement building with SKYDOSE. As a direct result of the aqueous iodine chemistry, the mitigating effect of the filters, and the closed primary system, the licensee found that the iodine dose to the public is entirely negligible. All dose components are below the regulatory requirements of 10 CFR Part 20 and the test reactor criteria in 10 CFR Part 100. The predominant portion of the dose to the public is from the noble gas release; independent dispersion calculations performed by the NRC staff verified these results.

Postulated Doses to the Public (Unrestricted Area)

Location	Type of Dose	Dose in mSv (mrem)
Exclusion Boundary (400 m)	Whole Body (TEDE)	0.07 (7.0)
	Thyroid (CDE)	0.001 (0.1)

The licensee estimated the dose to the NBSR staff on the basis of the fission product gas concentrations in rooms C-100 (the experimental floor) and C-200 (the operations level, where the control room is located) and the amount of time spent in the area. By procedure, the operators would evacuate the building of all nonessential personnel immediately upon seeing the high readings on the stack monitor and fission product monitor. They would then proceed to place the reactor in a safe condition and leave the building. For purposes of dose estimation, it is assumed that this takes 10 minutes. Independent calculations by the NRC staff verified that the results for the thyroid dose to the worker are reasonable.

Postulated Doses to the Workers (Restricted Area)

Location	Type of Dose	10-Minute Dose in Sv (rem)
Experimental Floor (C-100)	Whole Body (TEDE)	0.0107 (1.07)
	Thyroid (CDE)	0.0001 (0.01)
Operations Level (C-200)	Whole Body (TEDE)	0.0406 (4.06)
	Thyroid (CDE)	0.0002 (0.02)

The NRC staff concludes that these calculated doses are based on conservative assumptions and show that the reactor can be put into a safe condition and all personnel evacuated within the dose limits allowed for an emergency. On the basis of these calculations, the NRC staff finds that the projected doses from the MHA are acceptable for both the general public and the NBSR staff. Independent calculations performed by the NRC staff, using the same assumptions as the licensee, corroborate these conclusions.

13.2 Insertion of Excess Reactivity Accidents

To determine credible accident scenarios, the licensee considered accidents involving the insertion of excess reactivity. A step reactivity insertion involving mishandling of a fuel element was determined not to be credible because of administrative controls, ESFs, passive safety features, and the TS. TS 3.1.3 requires all core grid positions to be filled, and TS 3.9.2.1, “Fuel Handling within the Reactor Vessel,” requires that all fuel be latched, thereby precluding any movement of fuel elements during reactor operation. Two scenarios for a ramp insertion of excess reactivity were analyzed: startup accident and rapid removal of experiments. The consequences of these accidents bound any other credible scenario.

13.2.1 Startup Accident

The licensee evaluated a postulated startup accident assuming that the reactor operator, in violation of training and procedures, continuously withdraws the shim safety arms from the reactor initially critical at a power level of 10^{-4} MW(t). The reactivity insertion rate is conservatively assumed to be 5×10^{-4} $\Delta\rho$/s (a rate in excess of the maximum measured differential shim arm worth and limited to this maximum by TS 3.2.1). The transient is terminated by a high-power-level trip assumed to occur at 130 percent of full power (the TS 2.2 LSSS). An insertion time of 241 msec is assumed to insert the shim safety arms 5º, consistent with the limit of 240 msec specified in TS 3.2.1. Conservatively, no temperature or other reactivity feedback mechanism is credited. This scenario was analyzed for both the startup core and the EOC core. Using these assumptions, RELAP5 was used to study the transient behavior, with the results indicating a minimum critical heat flux ratio (MCHFR) greater than 1.7 for the EOC core and 1.55 for the SU core. Both MCHFRs provide ample margin to ensure that no fuel damage will result.

13.2.2 Rapid Removal of Experiments

TS 3.8.1 limits the maximum absolute reactivity of any single experiment in the NBSR to 0.5% $\Delta\rho$. Thus, the maximum credible excess reactivity insertion that removal of a single experiment could cause would be 0.5% $\Delta\rho$. The licensee evaluated a postulated accident scenario in which an experiment containing the maximum allowed reactivity (0.5% $\Delta\rho$) is removed in 0.5 seconds (a 1.0% $\Delta\rho$/s ramp). This postulated accident has been analyzed with the following assumptions:

- Initial power is 20.4 MW(t).
- Reactor power scram occurs at the LSSS of 26 MW(t) (130 percent) (TS 2.2).
- Negative feedback from increasing fuel and coolant temperatures is neglected.
- The shim safety arm is inserted 5º in 241 ms (consistent with the TS 3.2.1 maximum of 240 ms).
- The prompt neutron lifetime is 650 μs.

The scenario was analyzed for both BOC and EOC cores, with the limiting case occurring at BOC. The lowest value of the critical heat flux ratio (CHFR) is 1.74, which provides an adequate margin against fuel damage. The licensee’s analysis is conservative in that it does not take credit for the scram setpoint of 125 percent of full power (25 MW(t)) required by TS 3.1.1, “Reactor Power.”

The NRC staff concludes that the inputs and assumptions for the startup accident and rapid removal of experiments analyses are reasonable and consistent with the TS.

13.3 <u>Loss of Primary Coolant</u>

The licensee evaluated a postulated loss of primary coolant scenario in which a major pipe break in the process room allows all of the primary coolant to drain from the reactor vessel into the process room located under the reactor while the reactor is operating at 20 MW(t). Approximately 11,400 L (3,000 gal) of primary coolant is trapped in the process room by a dam built for this purpose. The reactor scrams immediately on a loss-of-flow signal (TS 3.2.2). Primary coolant is passively added to the core from the inner reserve tank which drains to a distribution pan that directs the coolant to the individual elements for over 20 minutes. Operation of a single valve adds the 11,400-L (3,000-gal) capacity of the D_2O emergency cooling tank by which the core is fully protected for about 2.5 hours. During this time, a system can be started by which lost primary water can be pumped from the dammed area in the process room up to the D_2O emergency cooling tank, providing virtually unlimited cooling time. Alternatively, through the addition of a single spool piece, H_2O can be piped into the system to provide cooling. TS 3.3.2 requires the emergency core cooling system to be available during reactor operation. With the cooling provided by this system, the temperature of the clad will remain well below the blistering temperature. Thus, no fission products will be released during this accident.

However, ^3H contained in the primary water would be released as a result of the loss of coolant. The confinement system, required during reactor operation by TS 3.4, would mitigate the release to the environment. The licensee made the following conservative assumptions in estimating the release:

- The ^3H concentration in the primary coolant is at the maximum level permitted by TS 3.7.1 (5 Ci/L).
- After the break, emergency ventilation is immediately established. TS 3.5 ensures that emergency ventilation is available during reactor operation.
- The process room is not isolated from the emergency ventilation system (ACV-10 is left open).
- The emergency ventilation system pulls the maximum design flow of 7 L/s (15 ft^3 per minute) from this area.
- Equilibrium is established immediately between the spilled D_2O at an assumed temperature of 42 degrees C (108 degrees F) and the air in the process room.

Using these assumptions, the licensee determined that the rate of ^3H released to the stack is 1.8×10^{-3} Ci/s. Using this release rate and a variety of weather conditions and codes from the U.S. Environmental Protection Agency, the licensee calculated that the effluent concentration at or beyond the 400-m (1,300-ft) boundary established by TS 5.1 is less than 1,000 nanocuries/m^3. The licensee stated that this value was determined for extremely stable conditions and low windspeeds that could not persist over any significant length of time. The licensee further stated that any release would be terminated within 24 hours, as remedial measures (pumping water into tanks, closing ACV-10, covering spilled water with plastic) would be taken immediately. Therefore, the licensee concluded that the maximum total dose to an individual member of the public at the 400-m (1,300-ft) boundary would be 0.002 mSv (0.2 mrem), well below the limits in 10 CFR Part 20 and 10 CFR Part 100. The NRC staff corroborated these results by independent calculations of the source term and dispersion analysis.

The NBSR SAR states that worker exposures for a loss-of-primary-coolant scenario would be determined by access to the process room, which is always strictly controlled. The ^3H levels would result in a concentration approaching 1.25×10^4 DAC. The NRC staff performed this calculation and independently verified the result. In the NBSR SAR, the licensee stated that if prolonged access were required, special provisions would be implemented to control exposure to acceptable levels.

13.4 Loss of Primary Coolant Flow

The licensee evaluated five different scenarios for loss of primary coolant flow:

(1) loss of offsite power
(2) seizure of one primary coolant pump
(3) throttling of coolant flow to the outer plenum
(4) throttling of coolant flow to the inner plenum
(5) loss of both shutdown coolant pumps

When the licensee analyzed each of the scenarios with the RELAP5 thermal-hydraulics analysis code, none of the scenarios resulted in fuel damage. The licensee found that the minimum value of the CHFR during the transients analyzed was 2.17 for the case of loss of offsite power. Based on the following discussions and the ample margin between the minimum CHFR and the CHFR required to exceed the safety limit, the NRC staff concludes that a loss of primary coolant flow will not result in fuel damage and that the consequences of the accident are bounded by the MHA.

13.4.1 Loss of Offsite Power

This accident scenario analyzed by the licensee assumes that all three primary pumps trip upon loss of offsite power. The three primary coolant pumps will coast down, and primary coolant flow will drop to a value where one or more of the primary coolant flow monitors will generate a delayed scram signal. TS 3.2.2 requires flow monitoring scram signal channels. The coolant flow coastdown in the RELAP5 model is validated by comparison with measured data. It is assumed that the scram occurs 400 msec after flow has reached the trip value. This allows for instrumentation sensing and scram actuation delays (TS 3.2.1 sets the maximum scram initiation time at 240 msec). The licensee’s transient analysis shows that the earliest flow scram is from the outer plenum flow at approximately 1 second after the primary pumps trip. After the 400-msec delay, a reactor scram is initiated that terminates the transient. The minimum CHFR at the hot spot is 2.17, and the maximum fuel centerline temperature is 136.9 degrees C (278.3 degrees F), which is below the TS 2.1 safety limit value of 450 degrees C (842 degrees F) and thus provides an adequate margin against fuel damage. The TS cited in this section apply to all of the loss-of-flow accident scenarios below.

On February 4, 2009, the Advisory Committee on Reactor Safeguards (ACRS) held a subcommittee meeting to review the licensee’s application for license renewal and the NRC staff’s draft SER. The subcommittee requested an analysis and review of an additional loss-of-flow accident due to closure of primary coolant system valve DWV-19 (see Section 13.4.4 of this SER). As a result of performing the new analysis, the licensee identified an error in the flow coastdown data set used to benchmark the RELAP5 model used in the analysis of the loss-of-

offsite-power accident. The licensee promptly notified the NRC staff of the issue by telephone on March 30, 2009, and agreed to take new measurements of the flow coastdown, and re-analyze the loss-of-offsite-power accident. The licensee submitted the results of the new analysis and the new flow coastdown data set. The NRC staff reviewed this information and found that the new analysis remained conservative because the licensee did not change any of the conservative assumptions used in the original analysis. The staff also found that the new analysis confirms that there will be an adequate safety margin in the maximum fuel temperature and the minimum CHFR. In addition, the staff found that the new flow coastdown data set is nearly identical to the original data set during the period immediately following the loss of offsite power, which is the period of greatest concern in terms of safety margins. Based on these findings, the NRC staff concludes that there is reasonable assurance that a loss of offsite power will not result in fuel damage and that the consequences of the accident are bounded by the MHA.

13.4.2 Seizure of One Primary Coolant Pump

In this scenario, the licensee assumed that through some failure, such as a faulty bearing, the rotor of one pump suddenly becomes locked, imposing a rapid flow decrease. The RELAP5 code models the pump seizure as a one-third flow reduction over a time span of 3.5 seconds to ensure that the flow reduction transient is over and the discharge valve on the seized pump is closed (3-second closure time). At 3.5 seconds, it is assumed that flow-sensing instrumentation initiates a scram. In response to the reduced flow, the minimum CHFR (at the hot spot in the outer plenum) decreases from an initial value of 2.67 to a new minimum value of 2.23, which still provides an adequate margin against fuel damage.

13.4.3 Throttling of Coolant Flow to the Inner or Outer Plenums

Because of the two-plenum configuration of the NBSR primary system, the possibility of inadvertent blockage of flow to either the inner or outer plenum exists. The licensee evaluated both scenarios. For the outer plenum scenario, the flow control valve DWV-1 is assumed to close, reducing the flow through the outer plenum. A scram is initiated when the flow reaches the LSSS low-flow trip point of 297 L/s (4,700 gpm) (TS 2.2), and the scram is completed after 400 msec. The complete closure of the flow control valve, after its 30-second stroke time, isolates the lower plenum of the reactor and cuts off the supply of forced coolant flow. The licensee's RELAP5 calculation shows that closed loop recirculation flow paths are established between the inlet and outlet plenums, and this recirculation flow removes heat from the core region by natural convection. When the buoyancy head is insufficient to induce closed-loop recirculation, boiling by itself is sufficient to remove decay heat from the core region provided that the power is below the flooding condition. The licensee calculated the flooding-limited power for the NBSR coolant channels to be 3.58 kW. The hottest coolant channel in the outer core region is estimated to be 1.5 times the power of an average channel. With the hottest channel at the flooding-limited power, the corresponding core power is 1.2 MW(t). The core power would fall below 1 MW(t) in less than 30 seconds after a reactor scram. The licensee concluded that forced circulation (until full closure of DWV-1) and closed-loop recirculation flow provide adequate cooling time for the reactor power to decrease below 1.2 MW(t), and therefore the throttling of a flow control valve would not lead to fuel damage.

For the inner plenum scenario, the flow control valve DWV-2 is conservatively assumed to close in 15 seconds, decreasing the flow through the inner plenum, with a scram completed 400 msec after the flow reaches the low-flow trip point of 76 L/s (1,200 gpm). In the most limiting BOC equilibrium core, the minimum CHFR at the hot spot (in the inner plenum) decreases from an initial value of 4.01 to a new low of 2.80. The transient is over once the shim arms begin to move in at 9.99 seconds. The discussions of heat removal by natural convection and boiling apply to the inner plenum flow transient, as well. Again, the licensee concluded that no fuel overheating occurs as the result of the throttling of coolant flow to the inner plenum.

13.4.4 Throttling of Coolant Flow to Both Plenums

In response to the ACRS subcommittee's request for an additional loss-of-flow accident analysis, the licensee analyzed the closure of primary coolant system valve DWV-19. This valve is a motor operated valve that is normally open and has a stroke time of 21 seconds. In the case that a spurious signal caused the valve to close during full power operation, the reactor would scram due to low primary coolant flow approximately 17 seconds after the valve began to close. Upon full closure of DWV-19, primary coolant flow through the primary system would cease, natural circulation cooling within the reactor vessel would be established, and the bulk coolant temperature would begin to rise. In the accident analysis, the licensee made conservative assumptions about the boundary conditions and initial reactor conditions. These assumptions include treating the reactor vessel as an adiabatic system, assuming the accident occurred at the end of an operating cycle, and assuming reactor power, temperature, and coolant flow were at their limiting values at the time the accident occurred. The NRC staff agrees that these are conservative and appropriate assumptions for the accident analyzed.

The licensee modeled the accident using RELAP5 and several different appropriate correlations to determine the minimum CHFR and the OFI ratio for the worst-case position in the core. In all cases these ratios were greater than 2.18 and the NRC staff finds this acceptable. Given the assumption that the reactor vessel is an adiabatic system, the primary coolant in the vessel serves as the ultimate heat sink for decay heat produced in the reactor fuel. The licensee compared the energy required to heat the primary coolant one degree kelvin to the energy released by the decay of fission products in the fuel and determined that the bulk coolant temperature would remain below the saturation temperature for at least 2 hours. The licensee found that this provides ample time for the licensee to establish coolant flow through the primary system. The licensee observed that, in the case that DWV-19 could not be reopened, there are multiple pressure relief pathways that ensure that the reactor vessel would not fail due to over-pressurization. The NRC staff reviewed the licensee's analysis and found it to be conservative and adequate to demonstrate that the consequences of the closure of DWV-19 are bounded by the consequences of the MHA. The licensee's analysis and the NRC staff's finding were presented to the ACRS during the ACRS full committee meeting held on April 2, 2009. The ACRS did not have any further questions regarding this accident scenario.

13.4.5 Loss of Both Shutdown Coolant Pumps

TS 3.6 requires a diesel generator and a battery power system to be available when the reactor is operating. However, in this scenario, the licensee analyzed a loss of offsite power followed by a complete failure of all backup power sources. Similar to the scenario evaluated in Section 13.4.1 of this SER, a reactor scram occurs on low primary flow with a 400-msec delay. In addition, both the shutdown primary coolant pumps and all of the secondary coolant pumps coast down because of the failure of all backup power sources. The licensee performed a RELAP5 simulation of this scenario to the point when the fuel reaches a relatively stable temperature and is being cooled by the natural convective flow of water up through the fuel elements and down around the outside of the core. According to the licensee, the coolant would gradually warm up; however, it is expected to take several hours for the bulk water temperature to reach the boiling point, which allows ample time for shutdown cooling to be restored. The NRC staff reviewed the analysis presented in the NBSR SAR and found that it is bounded by the consequences of the accident evaluated in Section 13.4.1 of this SER. Accordingly, the NRC staff concludes that there is reasonable assurance that a loss of both shutdown coolant pumps will not result in fuel damage and that the consequences of the accident are bounded by the MHA.

13.5 Misloading of Fuel

The fuel misloading accident scenario analyzed by the licensee assumes that an error occurs in the rotation of fuel elements and an irradiated fuel element is left in the M-4 location, which is normally filled by a fresh fuel element. Instead, it is assumed that the fresh fuel element is placed in the remaining empty position in the core. The licensee determined power distributions for a set of 26 different postulated cases in which the positions of two elements are exchanged. The results indicate that the radial power distributions for the misloading of fuel accident have peak values of 1.68 and 1.51 in the inner and outer core regions, respectively. The minimum CHFRs for the worst cases of the misloading of fuel accident can be inferred from the peak wall heat fluxes and the critical heat fluxes calculated previously. The licensee-calculated minimum CHFRs are 2.50 and 2.01 for the misloaded fuel (worst case) in the inner and outer core, respectively. Thus, the licensee asserted that the minimum CHFRs for the misloaded fuel provide adequate safety margins, and therefore no fuel damage is anticipated for the misloading of fuel accident. The NRC staff reviewed the licensee’s analysis and found that the calculated minimum CHFRs do provide adequate safety margins. On the basis of its review, the NRC staff concludes that there is reasonable assurance that a misloading of fuel will not result in fuel damage and that the consequences of the accident are bounded by the MHA.

13.6 Conclusions

The NRC staff has reviewed the accident analyses presented in the NBSR SAR and concludes that the licensee has considered a sufficient range of accident categories and analyzed limiting scenarios for each category to bound all credible accidents for the NBSR. In addition, the licensee analyzed a hypothetical accident scenario (the MHA) that bounds all credible accident scenarios and is shown to result in radiological consequences below the regulatory limits. Therefore, the NRC staff concludes that continued operation within the limitation of the TS and facility license provides reasonable assurance that any credible accidents would not result in a violation of facility safety limits or loss of fuel integrity.

14 TECHNICAL SPECIFICATIONS

The NRC staff evaluated the TS as part of its review of the application for renewal of Facility Operating License No. TR-5. The TS define certain features, characteristics, and conditions governing the operation of the NBSR. 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. The NRC staff concluded that the NBSR TS do 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 NBSR 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 NBSR 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 LSSSs for the RPS to preclude reaching the safety limit.

- 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).

- 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 NBSR 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 occupational exposures. The NRC staff also finds that the TS provide reasonable assurance that the facility will be operated as analyzed in the NBSR SAR and adherence to the TS will limit the likelihood of malfunctions and the potential accident scenarios discussed in Chapter 13 of this SER.

15 FINANCIAL QUALIFICATIONS

15.1 Financial Ability To Operate the Facility

As stated in 10 CFR 50.33(f), "Except for an electric utility applicant for a license to operate a utilization facility of the type described in 10 CFR 50.21(b) or 10 CFR 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."

NIST does not qualify as an "electric utility," as defined in 10 CFR 50.2. Further, 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. Therefore, the NRC staff has determined that NIST 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. NIST 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. It must submit estimates for the total annual operating costs for each of the first 5 years of facility operations from the expected license renewal date and indicate the source(s) of funds to cover those costs.

According to the application, the NBSR is operated within the NCNR. The projected operating costs for the NBSR are estimated to be $10,626,000 per year for FY 2009 through FY 2013. Funds to cover operating costs will come from the appropriated budget of the U.S. Department of Commerce. The applicant expects that this funding source will continue for the above-referenced fiscal years. The NRC staff reviewed the applicant's estimated operating costs and projected source of funds and found them to be reasonable.

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

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

The application referenced a decommissioning cost estimate, developed by Duratek, Inc., of $56.1 million in FY 2003 dollars. The applicant updated the cost estimate to be $87,223,357 in

FY 2009 dollars. The cost estimate shows costs summarized by labor, radwaste shipment and disposal, energy, other (i.e., spent fuel shipment), and a contingency factor of 25 percent. The applicant stated that it will update its decommissioning cost estimate using an algorithm prepared by Duratek, Inc., based on NUREG-1307, “Report on Waste Burial Charges,” where changes in labor, energy, and transportation costs will be estimated from producer price indices (available from the U.S. Bureau of Labor Statistics), adjusted for local conditions as described in the Duratek, Inc., cost estimate. Based on the NRC staff’s review of the information submitted by NIST regarding decommissioning of the NBSR, the NRC staff finds that the decommissioning cost estimate submitted by NIST is reasonable.

The applicant has elected to use a statement of intent (SOI) to provide financial assurance, as allowed by 10 CFR 50.75(e)(1)(iv) for a Federal, State, or local government licensee. The SOI must contain or reference a cost estimate for decommissioning and indicate that funds for decommissioning will be obtained when necessary.

The applicant provided an SOI, dated October 10, 2008, that states the signator will “request as necessary from the U.S. Congress through the Department of Commerce and the Office of Management and Budget, external and NIST direct cost funds for decommissioning activities….” The decommissioning cost estimate is approximately $87,223,357 for the DECON option (“full scale decommissioning, decontamination, and demolition, restoring the site for unrestricted use…,” as stated by NIST).

To support the SOI and the applicant’s qualifications to use an SOI, the application states that NIST is a Federal agency and includes documentation to corroborate this statement. The application also provides information supporting NIST’s representation that the decommissioning funding obligations of NIST are backed by the full faith and credit of the U.S. Government. NIST also provided documentation verifying that Patrick Gallagher, Deputy Director of NIST (the signator of the SOI), is authorized to execute contracts on behalf of NIST.

The NRC staff reviewed the applicant’s information on decommissioning funding assurance as described above and finds that the applicant is a Federal Government licensee under 10 CFR 50.75(e)(1)(iv), the SOI is acceptable, the decommissioning cost estimate for the DECON option is reasonable, and NIST’s means of adjusting the cost estimate and associated funding level periodically over the life of the facility is reasonable.

15.3 Foreign Ownership, Control, or Domination

Section 104d of the AEA prohibits the NRC from issuing a license under Section 104 of the AEA to “any corporation or other entity if the Commission knows or has reason to believe it is owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government.” The NRC regulation at 10 CFR 50.38, “Ineligibility of Certain Applicants,” contains language to implement this prohibition. According to the application, NIST is a Federal agency within the U.S. Department of Commerce and is not owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government. The NRC staff does not know or have reason to believe otherwise.

15.4 Nuclear Indemnity

The NRC staff notes that the applicant currently has an indemnity agreement with the Commission, and said agreement does not have a termination date. Therefore, NIST will continue to be a party to the present indemnity agreement following issuance of the renewed license. Under 10 CFR 140.51,"Scope," NIST, as a Federal Government licensee, is not required to provide nuclear liability insurance. The Commission will indemnify NIST for any claims arising out of a nuclear incident under the Price-Anderson Act (Section 170 of the AEA, as amended) and in accordance with the provisions under its indemnity agreement pursuant to 10 CFR 140.94, "Appendix D—Form of Indemnity Agreement with Federal Agencies," up to $500 Million. Also, NIST is not required to purchase property insurance under 10 CFR 50.54(w).

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

16 OTHER LICENSE CONSIDERATIONS

Previous sections of this SER concluded that normal operation of the reactor causes insignificant risk of radiation exposure to the public and that only an accident event could cause some exposure. Chapter 13 of the SER concluded that the MHA is shown to result in radiological consequences that are below regulatory limits.

In this section, the NRC staff reviews the impact of prior operation of the facility on the risk of radiation exposure to the public. The two parameters involved are the likelihood of an accident and the consequences if an accident occurs.

Because the NRC staff concluded that the reactor was initially designed and constructed to meet nuclear safety requirements, the NRC staff must also consider whether operation will cause significant degradation in safety features. Furthermore, because melting of a fuel assembly is the MHA, the NRC staff has considered mechanisms that could increase the likelihood of fuel damage resulting in release of radionuclides. Possible mechanisms include the following:

- radiation degradation of fuel element cladding strength
- high internal pressure caused by high temperature leading to exceeding the elastic limits of the cladding
- corrosion or erosion of the cladding leading to thinning or other weakening
- mechanical damage as a result of handling or experimental use
- degradation of safety components or systems

16.1 <u>Prior Use of Reactor Components</u>

16.1.1 Reactor Vessel and Components

16.1.1.1 Reactor Vessel

Limited opportunities are available to inspect the NBSR vessel. In 2000, when the refueling head was removed, the licensee performed a visual inspection using binoculars and observed no flaws or unusual discolorations. In January 2002, the licensee conducted an inspection using remote imaging equipment while the reactor vessel was fully defueled but still filled with D_2O. This inspection included a detailed inspection of the vessel interior, the vessel exterior, and the interior of the thermal shield. The inspection revealed no flaws that indicated a decrease in the integrity of the welds or defects that would affect the vessel and component design function. Visual inspections are repeated if the refueling head is removed from the vessel. Remote imaging equipment inspections are repeated when the core is offloaded.

16.1.1.2 Reactor Vessel Exterior Inspection

During maintenance activities, NIST personnel were able to insert a camera through a 1-in. gap between the thermal shield and the reactor vessel to examine the reactor vessel wall and bottom and the thermal shield. The examination did not uncover any material condition that could lead to a failure.

16.1.1.3 Reactor Vessel Embrittlement

The NBSR vessel is fabricated from aluminum alloys 5052 and 6061. Oak Ridge National Laboratory studied heavily irradiated samples of the 6061-T6 alloy obtained from a control rod drive follower tube used in the High Flux Beam Reactor at Brookhaven National Laboratory. The most heavily irradiated portions of the NBSR vessel are the tips of the beam tubes. These components will conservatively have accumulated less thermal neutron fluence than the irradiated samples by 2024. At that time, according to the licensee, the ductility, while reduced, will retain approximately 70 percent of the original value; the Charpy energy will have dropped by over a factor of 6; and the component materials will remain ductile, with reduced impact strength and toughness, but with reduced resistance to crack propagation under tensile stress and reduced resistance to sudden pressure applications and impacts. The licensee updated this analysis to extend to 2030, and the updated analysis shows that the material properties of the reactor vessel will remain adequate to ensure vessel integrity beyond 2030.

The licensee claims that the vessel components are never under significant tensile stress; therefore, tensile stress is not an issue. The licensee further asserts that while the surrounding D_2O does exert minimal compressive force, this stress will not create cracks that can propagate quickly. As the vessel is entirely closed, the licensee concludes that there can be no impacts against these components while the core is loaded. The NRC staff reviewed the licensee's analysis of reactor vessel embrittlement and found that the analysis is consistent with methods and data reported in the literature. Based on its review of reactor vessel embrittlement, the NRC staff concludes that the predicted 2030 embrittlement state of the reactor vessel and components creates no hazard to continued operation of the NBSR.

16.1.2 Fuel Element and Control Rods

16.1.2.1 Fuel Element Aging

The NBSR core is completely replaced in eight refueling outages over a nominal period of 1 year. Therefore, the NRC staff concludes that fuel aging creates no hazard to continued operation of the NBSR.

16.1.2.2 Control Rod Aging

The NBSR has three reactivity control systems consisting of four shim safety arms, a single regulating rod, and a moderator dump system. The shim safety arms are the primary means of reactivity control and shutdown capability. TS 4.1.2 requires annual determination of control rod worth. TS 4.2.1 requires semiannual verification of shim arm motion.

As discussed in Section 4.2.2 of this SER, the lifetime of the shim arms is affected by poison burnup, corrosion, and radiation damage. The finding that corrosion does not limit the shim arm lifetime is supported by experience with similar arms that remained in the CP-5 reactor for approximately 8 years until poison burnup necessitated their replacement. Over 30 years of operation of the NBSR reactor have shown this to also be true for the NBSR shim arms. The radiation damage to the shim safety arms is not significant during reactor operation since the shim arms are in the top reflector above the core where the fast neutron flux is relatively low. Shim arm sets have been replaced at the NBSR reactor three times, with no radiation damage apparent in the shims removed. The NRC staff also considered aging of the regulating rod. Since the regulating rod "poison" is aluminum, the NRC staff views regulating rod burnup as insignificant. The staff determined that corrosion damage is similar to that of other aluminum core components and minimal.

The NRC staff concludes that normal replacement of the shim arms based on burnup limits the effect of aging on the NBSR control elements. As a result, the NRC staff concludes that control rod aging should not be a significant concern for the continued safe operation of the NBSR.

16.2 Conclusions

The NRC staff has reviewed the prior use of reactor components as well as the aging of safety components, as described in the NBSR SAR, and concludes that there has been no significant degradation of reactor components to date. Further, the surveillance requirements in the TS provide reasonable assurance that the reactor components 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 April 9, 2004, as supplemented on October 2, 2006, May 30 and August 14, 2007, September 16, October 21, and December 8, 2008, and March 3, March 19, and April 22, 2009, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended, and the Commission's rules and regulations set forth in Chapter I of Title 10 of the *Code of Federal Regulations*.

- The facility will operate in conformity with the application, as well as the provisions of the Act 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.

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