Safety Evaluation Report Related to the Renewal of Facility Operating License No. R-37 for the Massachusetts Institute of Technology Reactor, Massachusetts Institute of Technology

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Office of Nuclear Reactor Regulation

ABSTRACT

This safety evaluation report summarizes the findings of a safety review conducted by the staff of the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Reactor Regulation. The NRC staff conducted this review in response to a timely application filed by the Massachusetts Institute of Technology (MIT, the licensee) for a 20-year renewal of Facility Operating License No. R-37 to continue to operate the Massachusetts Institute of Technology reactor (MITR-II, the facility). In conjunction with the license renewal, MIT requested an increase in the maximum licensed power level from 5.0 megawatts thermal power (MW(t)) to 6.0 MW(t). The facility is located at the MIT campus in Cambridge, MA. 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 MIT can continue to operate the MITR-II, in accordance with the renewed license, without undue risk to public health and safety, facility personnel, or the environment.

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LIST OF ABBREVIATIONS

Abbreviation	Definition
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
ANSI/ANS	American National Standards Institute/American Nuclear Society
API	American Petroleum Institute
BNCT	boron neutron capture therapy
C	Celsius
CFR	*Code of Federal Regulations*
CHF	critical heat flux
cm	centimeter(s)
cm^3	cubic centimeter(s)
D_2	deuterium
DOE	U.S. Department of Energy
ECCS	emergency core cooling system
EP	emergency plan
ESF	engineered safety feature
F	Fahrenheit
ft	foot (feet)
g	acceleration due to gravity
gal	gallon(s)
gpm	gallon(s) per minute
HEPA	high-efficiency particulate air
HVAC	heating, ventilation, and air conditioning
I&C	instrumentation and control(s)
in.	inch(es)
k_{eff}	k-effective
km	kilometer(s)
kPa	kilopascal(s)
kW	kilowatt(s)
kW(t)	kilowatt(s) thermal
l	liter(s)
LSSS	limiting safety system setting
m	meter(s)
m/s	meter(s) per second
MCNP	Monte Carlo N-Particle

LIST OF ABBREVIATIONS

Abbreviation	Definition
MHA	maximum hypothetical accident
mi	mile(s)
MIT	Massachusetts Institute of Technology
MITR-I	Massachusetts Institute of Technology Reactor (original)
MITR-II	Massachusetts Institute of Technology Reactor
MITRSC	MIT Reactor Safeguards Committee
mph	mile(s) per hour
mrem	millirem
mrem/hr	millirem per hour
mSv	millisievert(s)
MW	megawatt(s)
MWD	megawatt-day(s)
MW(t)	megawatt(s) thermal
NRC	U.S. Nuclear Regulatory Commission
OFI	onset of flow instability
psi	pound(s) per square inch
psig	pound(s) per square inch gauge
RCS	reactor control system
RPS	reactor protection system
RRPO	Reactor Radiation Protection Office
SAR	safety analysis report
SER	safety evaluation report
SSC	structure, system, and component
TEDE	total effective dose equivalent
TS	technical specification(s)
UAl_x	uranium-aluminum
USGS	U.S. Geological Survey
μS/cm	microsiemen(s) per centimeter

1 INTRODUCTION

1.1 Overview

By letter (and supporting documentation) dated July 8, 1999, as supplemented by letters dated February 10 and May 8, 2000; January 29, 2004; July 5 and October 11, 2006; January 26, 2007; February 22, May 29, August 15, August 21, August 26, October 6, October 7, and December 1, 2008; May 26, August 27, October 5, October 9, and November 19, 2009; and March 30, August 6, and August 26, 2010, the Massachusetts Institute of Technology (MIT, the licensee) submitted to the U.S. Nuclear Regulatory Commission (NRC, the Commission) a timely application for a 20-year renewal of the Class 104a and 104c Facility Operating License No. R-37 (NRC Docket No. 50-020). The renewed facility operating license would authorize continued operation of the Massachusetts Institute of Technology reactor (MITR-II) located on the MIT campus in Cambridge, MA. In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 2.109, "Effect of Timely Renewal Application," the current facility operating 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 NRC staff requests for additional information. The NRC staff also based its review on annual reports of facility operation submitted by the licensee and inspection reports prepared by the NRC staff. As part of the review process, the NRC staff conducted site visits to observe facility conditions.

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

This safety evaluation report (SER) summarizes the findings of the NRC staff's safety review of the licensee's application. This SER and the environmental assessment and finding of no significant impact, dated September 27, 2010 (ADAMS Accession No. ML101020281), will serve as the basis for issuance of a renewed license authorizing operation of the MITR-II at power levels up to 6 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, 35, 50, 51, 55, 70, 73, and 140; applicable NRC regulatory guides; and relevant accepted industry standards, such as the American National Standards Institute/American Nuclear Society (ANSI/ANS) 15 series. The NRC staff also referred to the guidance in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," issued February 1996.

William B. Kennedy from the NRC's Office of Nuclear Reactor Regulation, Division of Policy and Rulemaking, Research and Test Reactors Licensing Branch, prepared this SER. Other

contributors to the safety review include Alexander Adams Jr., Stephen Pierce, Marvin M. Mendonca, William C. Schuster, Michael B. Norris, Paul V. Doyle, Ronald B. Uleck, Daniel E. Hughes, and Jo Ann Simpson of the NRC staff. Under contract to the NRC, William Watkins, James Willison, James Wallace and Dennis Gehr 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 MITR-II structures, systems, and components (SSCs) important to safety during normal operation and postulated accident scenarios are acceptable. Safe operation of the facility can reasonably be expected to continue.

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

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

- The NRC staff does not expect exposures from radiation and releases of radioactive effluents and wastes from the facility to result in doses or concentrations in excess of the limits specified by Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage," to 10 CFR Part 20 and finds they are consistent with as-low-as-reasonably-achievable (ALARA) principles.

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

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

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

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

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

On the basis of these findings, the NRC staff concludes that MIT can continue to operate the MITR-II at the increased power level in accordance with the Atomic Energy Act of 1954, as amended (AEA), NRC regulations, and renewed Facility Operating License No. R-37 without undue risk to public health and safety.

1.3 General Facility Description

The MITR-II is owned and operated by MIT, which is a nonprofit educational institution. The reactor is located on the MIT campus in Cambridge, MA, and is a component of the MIT Nuclear Reactor Laboratory. A dedicated containment building houses the reactor and much of the reactor systems. The containment building is constructed primarily of reinforced concrete and steel and is generally fireproof in nature. Other buildings house some auxiliary reactor systems, the main electrical supply equipment, radiation protection equipment, and offices for reactor personnel. Section 2.1 of the SAR describes the reactor site in detail, and TS 5.1, "Site and Facility Description," specifies requirements related to the site and facility. The site consists of an area surrounded by a chain link fence and the adjacent one-story building. The containment building, cooling tower, ventilation stack, and liquid waste storage tanks are within the fenced-in area.

The MITR-II has been analyzed for steady-state operation at power levels not to exceed 6 MW(t) and is not designed for pulsed operation. The reactor uses plate-type fuel elements that are rhombic in shape. The core consists of fuel elements arranged in a compact hexagonal array. Six control blades provide coarse reactivity control for reactor startup, shutdown, and large changes in reactor power. The control blades are coupled to electromagnets that allow the rods to drop into the core by gravity in the event of a power loss or scram signal. A cadmium regulating rod provides fine reactivity control for steady-state operation.

The MITR-II is cooled and moderated by light water in the primary system. The primary system consists of the core tank, piping, heat exchangers, and associated valves and pumps. The light-water primary system transfers heat from the core to the secondary cooling system. The secondary system dissipates the heat to the atmosphere by means of cooling towers located adjacent to the containment building. The MITR-II uses heavy water contained in a tank concentric with the core tank as a neutron reflector. Graphite surrounding the heavy-water reflector provides additional neutron reflection.

The MITR-II is licensed under both AEA Section 104c as a research reactor and AEA Section 104a for medical therapy for humans. A treatment room is located directly beneath the reactor core for neutron exposures. The facility also has a shielded treatment room adjacent to the reactor structure. The fission converter, a subcritical assembly of fuel located near the reactor core, provides a dedicated supply of neutrons to the treatment room adjacent to the reactor structure. The MITR-II is equipped with a number of other experimental facilities. These include beam ports, an automatic sample transfer system, and experiment irradiation locations in the graphite reflector and in the reactor core.

The MITR-II generates high-level waste (spent fuel) and low-level waste. The spent fuel is returned to the U.S. Department of Energy (DOE), as discussed below in Section 1.7 of this report. The low-level waste may be solid, liquid, or gaseous. Radiation monitors examine the liquid and gaseous effluents for abnormal radiation levels. Interlocks with the ventilation system

and liquid discharge piping allow the radiation monitors to halt the release of effluents upon detection of any abnormal radioactivity. The MIT Reactor Radiation Protection Office (RRPO) implements a radiation protection program at the facility. The office is independent of the reactor operations management structure.

1.4 Shared Facilities and Equipment

The MITR-II is located within a dedicated containment building. As a result, it shares only a few systems or equipment with other facilities—electricity supply, water supply, heating, compressed air, and sanitary sewer. In addition to the shared items, other dedicated service systems originate outside the containment building. These include the helium and carbon dioxide cover gas supplies, purified water for use as makeup, and the capability to transfer irradiated samples through pneumatic tubes to laboratories located outside the containment building. Manual or solenoid-operated valves, or both, provide isolation of these systems to maintain containment integrity. The licensee has shown that a malfunction or a loss of function of these shared facilities would not affect the safe operation of the MITR-II. Additionally, the licensee has shown that shared facilities and equipment do not have the potential to damage the MITR-II or preclude the safe shutdown of the reactor.

1.5 Comparison with Similar Facilities

The MITR-II is one of many NRC-licensed, operating research reactors located on university campuses. The MITR-II uses materials testing reactor-type fuel, which is also in use at the National Institute of Standards and Technology reactor and has a long history of safe use in research and test reactors. The MITR-II has a design basis and safety analysis comparable with other facilities of similar fuel type, thermal power level, and site considerations. The history of the MITR-II and similar facilities demonstrates consistent safe operation that has been found acceptable to the NRC staff. The MITR-II does not have any features that would preclude applying general knowledge and experience gained in the operation of these other reactors to operation of the MITR-II.

1.6 Summary of Operation

The licensee has operated the MITR-II in accordance with Facility Operating License No. R-37 and established procedures for approximately 35 years to facilitate experiments, research, and medical studies. The MITR-II has longstanding programs for student laboratory exercises, student operator training, education, and outreach. The reactor typically operates 24 hours a day, 7 days a week, with periodic routine shutdowns for refueling and maintenance. According to the licensee's annual reports for 2005 to 2009, the reactor was operated for approximately 1,124 megawatt-days (MWDs) per year at a normal power level of 5 MW(t). Given the power increase to 6 MW(t), annual MWDs of reactor operation could increase approximately 20 percent during the period of the renewed license.

1.7 Compliance with the Nuclear Waste Policy Act of 1982

Section 203(b)(1)(B) of the Nuclear Waste Policy Act of 1982 states that the NRC may, as a precondition to issuing or renewing an operating license for a research reactor, require the applicant to have reached an agreement with DOE for the disposal of high-level wastes and spent nuclear fuel. In accordance with a letter from DOE (R.L. Morgan) to the NRC (H. Denton) dated May 3, 1983, it has been determined that all universities operating nonpower reactors have entered into a contract with DOE that provides that DOE retain title to the fuel and that

DOE is obligated to take the spent fuel, high-level waste, or both, for storage or reprocessing. Because MIT has entered into such a contract with DOE, it has satisfied the applicable requirements of the Nuclear Waste Policy Act of 1982.

1.8 Facility Modifications and History

On May 7, 1956, the U.S. Atomic Energy Commission (AEC) issued Construction Permit No. CPRR-5 to MIT and construction began on the original reactor, the MITR-I. The MITR-I was heavy-water cooled and moderated. In 1958, the AEC issued Facility Operating License No. R-37, which authorized reactor operation at power levels up to 1 MW(t). The reactor achieved initial criticality on July 21, 1958. In 1961, the AEC approved an increase in the allowed operating power to 2 MW(t). The AEC authorized a further increase to 5 MW(t) in 1965.

The AEC issued Construction Permit No. CPRR-118 in 1973. The permit authorized the modifications to create the currently operating MITR-II. On May 24, 1974, the original MITR-I permanently shut down. Reactor operations at the facility were precluded until completion of the construction specified in the permit. The permit authorized modification of the reactor to use light water as a coolant and moderator and heavy water as a reflector. The new design, named the MITR-II, offered higher flux levels at the same reactor power and significantly reduced tritium production.

On July 23, 1975, the NRC issued Amendment No. 10 to Facility Operating License No. R-37. The amendment authorized the operation of the modified reactor at power levels up to 5 MW(t). The MITR-II achieved initial criticality on August 14, 1975. Significant regulatory actions since 1976 include approval to conduct digital control experiments on the MITR-II, authorization to use the medical facility for neutron capture therapy for humans, approval of the fission converter facility, and extension of the reactor license to August 1999 to recapture time spent in construction.

2 SITE CHARACTERISTICS

2.1 Geography and Demography

2.1.1 Geography

The reactor site is located in the northwest corner of the MIT campus in Cambridge, MA, and is a part of the MIT Nuclear Reactor Laboratory. The reactor site comprises the reactor building and a small area immediately surrounding it, bounded by a chain-link fence, and a portion of an attached multipurpose academic building. Adjacent to the site are an industrial building to the north, a parking lot and warehouse building to the east, a warehouse building to the south, and academic and dormitory buildings to the west. The area surrounding the reactor site is urban with buildings of several stories in most directions. The nearest prominent natural feature is the Charles River Basin at the southern border of the MIT campus. The topography of the land of Cambridge is that of a river basin formed by low-lying hills. However, the urban surroundings dominate the natural topography.

The area adjacent to the south side of the reactor site is the closest publicly accessible area to the reactor building. According to the licensee, the nearest point of normal public occupancy is on Albany Street, approximately 21 meters (m) (68 feet (ft)) northwest of the reactor building. The nearest dormitories are located approximately 100 m (330 ft) west of the reactor building. The nearest non-MIT residence is approximately 250 m (820 ft) from the reactor building.

The NRC staff reviewed the sections of the MITR-II SAR related to geography and finds that the SAR contains sufficient information to appropriately characterize the geography of the area surrounding the facility. Based on the NRC staff's review of the information presented by the licensee and maps of the area, there is reasonable assurance that no geographic features exist that make the site unsuitable for continued operation of the MITR-II.

2.1.2 Demography

The MITR-II is located in the city of Cambridge, MA. The U.S. Census Bureau estimated the population of Cambridge to be 101,355 in 2000. Cambridge is part of the Boston-Cambridge-Quincy Metropolitan Statistical Area, which had an estimated population of 4,483,000 in 2007. According to Chapter 2 of the SAR, the MIT campus, factories, commercial facilities, and a residential section lie within 1 kilometer (km) (0.6 miles (mi)) of the MITR-II. There are approximately 4,150 students living on the MIT campus from September to May, with seasonal variations. Approximately one-third of this number occupies the campus during the summer. The total population within 1 km (0.6 mi) is estimated at 10,300.

A ring 2 km (1.2 mi) from the reactor site includes one-third of Cambridge and some residential areas of Boston and Somerville. The population within this ring is estimated at 73,000. The estimated population within rings of 4 km (2.5 mi), 6 km (3.7 mi), and 8 km (5.0 mi) is 264,000, 570,000, and 850,000, respectively. All population values are expected to increase by 5 percent by 2020. The population distribution in the vicinity of the reactor site is well established and not expected to change significantly over the period of the renewed license.

The NRC staff verified population data by reviewing U.S. Census data. The licensee presented sufficient demographic information to accurately characterize the region surrounding the reactor site and assess the potential radiological impact on the public from operation of the facility.

Based on the NRC staff's review of the demography of the area surrounding the facility, there is reasonable assurance that there are no current or projected demographic features that render the site unsuitable for continued facility operation.

2.2 Nearby Industrial, Transportation, and Military Facilities

Railroad tracks run along the south side of the reactor site approximately 5 m (16 ft) from the site boundary. According to the licensee, the current rail traffic consists of passenger train transfers and freight trains carrying cargo that is not hazardous to the MITR-II. Trains are required to stop at the nearby crossing of Massachusetts Avenue, minimizing the potential for derailments in the vicinity of the MITR-II. The site is also adjacent to Massachusetts Avenue, which is a major route for all types of vehicles. Approximately 5 km (3.1 mi) east of the MITR-II is the Boston inner harbor. The harbor handles all types of ocean-going vessels. The nearest major airport is Logan International, approximately 6 km (3.7 mi) to the east. The airport handles approximately 510,000 aircraft movements per year from 10 major runways. Nine of the runways are in the north-south direction, with only one approached from the eastern direction. The approach for this runway is over Boston, not Cambridge. The NRC staff confirmed the location and distances to nearby transportation facilities through review of local maps and from observation during the facility tour. There are no military facilities in the vicinity of the MITR-II. According to the licensee, facility personnel meet regularly with the City of Cambridge Director of Emergency Planning to keep abreast of any potential hazards to the reactor, and City agencies conduct periodic exercises to test emergency response capabilities.

Based on the character of the local industry in the vicinity and the distances to the facility, the NRC staff concludes that local industry and transportation facilities will not pose a significant risk to the continued safe operation of the MITR-II.

2.3 Meteorology

The MITR-II is located near the Atlantic Ocean in New England. The prevailing wind direction is west to east, but the area is subject to winds from all directions at all times of the year. The mean wind speed is 5.6 meters per second (m/s) (12.5 miles per hour (mph)), being slightly higher in the winter and lower in the summer. Extremes are normally mitigated by the proximity to the ocean. However, hurricanes are possible in August and September. Tornados are extremely rare.

Precipitation is year round, with showers and thunderstorms in the summer and snow in the winter. Precipitation is evenly distributed throughout the year, with monthly averages between 7.6 centimeters (cm) (3 inches (in.)) and 10.2 cm (4 in.). The largest monthly rainfall was 43.4 cm (17.1 in.) in August 1955. Snowfall occurs typically from November to April, peaking in February. The largest snowfall in 24 hours was 59.9 cm (23.6 in.) in 1978. The 100-year snow pack has a water equivalent of 11.7 cm (4.6 in.).

According to the MITR-II SAR, monthly average temperatures range from a low of -5.7 degrees Celsius (C) (21.6 degrees Fahrenheit (F)) in January to a high of 27.7 degrees C (81.8 degrees F) in July. Humidity is mostly constant between 60 percent and 70 percent throughout the year.

According to the MITR-II SAR, the maximum wind speed recorded in the Boston area was a 1-minute sustained value of 24 m/s (54 mph) with a gust of 36 m/s (81 mph) in 1993. According to the SAR, the calculated maximum stress on the containment shell was 862 kilopascals (kPa)

(125 pounds per square inch (psi)). There was no damage to the reactor building or surrounding structures from this storm.

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

2.4 Hydrology

According to the MITR-II SAR, the facility is in the drainage basin of the Charles River, which drains to Boston Harbor and the Atlantic Ocean. The Charles River is dammed at its mouth and, thus, there are no tidal effects at Cambridge. The Charles River is not impounded upstream from Cambridge in any way that could result in dam-failure-related flooding at the reactor site. A seismic event that damaged the Charles River Dam would enhance drainage, as the impounded water could flow out to the ocean. Some flooding of low areas near the river can occur during intense rainstorms. The MITR-II is not located within these areas, nor are the roads used by emergency vehicles that respond to MIT located within these areas.

As discussed in Chapters 4, 5, and 11 of this SER, the MITR-II design features and radiation protection practices minimize the potential for contamination of soil and ground water. The primary coolant system is a closed system that contains multiple means for monitoring coolant inventory. Primary system leak detectors and sampling of the secondary coolant for radioactive contamination provide additional means of detecting a failure of the primary coolant boundary. Liquid radioactive waste generated at the facility is stored in aboveground tanks or in small containers located in designated storage areas. A dedicated structure equipped with leak detection and containment features houses the aboveground tanks. All liquid wastes are appropriately sampled before discharge from the facility.

The NRC staff verified information presented by the licensee by reviewing local maps and making first-hand observations during facility tours. Based on the lack of credible flooding risks, the NRC staff concludes that the local hydrology does not pose a significant risk to the continued safe operation of the MITR-II. The NRC staff evaluated the potential for ground water contamination and concludes that facility design features and radiation protection practices minimize the potential for ground water contamination.

2.5 Geology, Seismology, and Geotechnical Engineering

The MITR-II is located in the vicinity of the Appalachian Mountain system. This geologic formation is old and heavily eroded and was glaciated most recently between 10,000 and 15,000 years ago. The reactor site was originally marsh, but has been filled to a depth of 3 m (10 ft) to 3.7 m (12 ft). Underlying rock in the vicinity of the reactor site is shale-type rock, overlaid by 22.9 m (75 ft) to 30.5 m (100 ft) of clay, topped by sand and gravel, organic material, and fill.

The geology and seismic record of the reactor site provide a good base on which to assess the adequacy of the MITR-II to geologic hazards. The region is not considered seismically active. Earthquakes have been recorded throughout New England, though not in the cities of Boston or Cambridge. Recorded history of earthquakes in the region goes back to 1568. The maximum seismic event recorded in the New England area was the Cape Ann earthquake in 1755. It was estimated to have an intensity of VIII on the Modified Mercalli Index at its epicenter. The risk of a major seismic event in the Boston area is considered low. No surface faulting has been

discovered in the area. The seismic history supports the conclusion that a catastrophic earthquake at or near the site is unlikely during the life of the facility.

The U.S. Geological Survey (USGS) updated its seismic hazard maps for the United States based on new seismological, geophysical, and geological information. 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 reactor site. In conducting its review of seismic hazards, the NRC staff used the 2008 National Seismic Hazard Map produced by USGS. The map shows only a 2-percent probability that in 50 years peak lateral ground acceleration will exceed 0.12 times the acceleration due to gravity (g).

According to the MITR-II SAR, the maximum safe-shutdown earthquake acceleration of 0.15 g was chosen based on seismic studies of the area. This same value is used for the Pilgrim Nuclear Power Station located 97 km (60 mi) southeast of the MITR-II. The potential for soil liquefaction was evaluated during original construction, and the organic and fill material was removed during construction. Based on the sand stratum remaining, a factor of 1.5 is used for determining lateral forces, resulting in a maximum safe-shutdown earthquake acceleration of 0.225 g. As discussed in Chapter 13 of this SER, this value is a fraction of the acceleration required to potentially damage the reactor core tank.

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

2.6 Conclusions

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

3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.1 Design Criteria

The design criteria require the SSCs related to safe operation and shutdown of the reactor to be able to perform their intended functions as described in the MITR-II SAR. The principal safety-related SSCs are the fuel, core support structure, reactor protection system (RPS), reactor coolant systems, and containment 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 RPS 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 containment building must isolate under abnormal conditions and prevent the uncontrolled release of radioactive materials to the environment.
- The reactor containment building must protect the reactor from external environmental conditions.

The Idaho National Laboratory developed and maintains the specifications for the fuel used by the MITR-II. The fuel specifications also incorporate applicable portions of other design standards. The fuel specification includes requirements for plate loading, void volume, fuel homogeneity, cladding and fuel core thickness, evaluation methods, and surface finish. The fuel specification also includes materials for construction, test and inspection requirements, packaging and shipping processes, and acceptance inspections. Chapter 4 of this SER contains additional discussion of the fuel design.

The primary coolant system and core tank designs preclude siphoning of the coolant and provide natural convection cooling if forced cooling is lost. According to the licensee, components of the primary system were designed to standards appropriate for their function. The minimum test pressure for tanks, valves, and piping is 276 kPa (40 pounds per square inch gauge (psig)), which is substantially higher than operating pressures. The heavy-water reflector tank is made from aluminum-6061 and was hydrostatically tested to 138 kPa (20 psig). The licensee tested other portions of the heavy-water system to higher pressures. The fission converter tank is constructed of aluminum-5083 and was hydrostatically tested to 103 kPa (15 psig). The licensee tested all water systems to at least 150 percent of operating pressures. Chapters 4 and 5 of this SER contain additional discussion of these SSCs.

The containment building is designed to withstand 13.8 kPa (2 psig) of internal pressure or 0.69 kPa (0.1 psig) of external pressure. The shell is constructed of American Society for Testing and Materials 283 Grade C steel and is designed to a specification for welded oil storage tanks. The allowable building leak rate is 1 percent of the contained volume per day per

psig of overpressure. The design includes over and under-pressure protection systems. The containment isolation system is designed to close in the event of high airborne radioactivity. Similarly, liquid waste discharge is isolated upon detection of high activity in the discharge line. All effluent paths are monitored for radiation. These paths are designed to either isolate the release or alert operators so as to preclude releases in excess of the applicable limits specified in 10 CFR Part 20. Chapter 6 of this SER contains additional discussion of the containment building and ventilation system designs, and Chapter 11 discusses liquid effluents.

The reactor control system (RCS) is designed so that only a single control device can be withdrawn at a time. As discussed in Chapter 13 of this SER, the maximum ramp or step reactivity insertions will not result in fuel damage or the uncontrolled release of radioactive material. Multiple control devices are used at the MITR-II, and reactor operators have a diverse set of options to control and shut down the reactor. The RPS contains diverse and redundant instrumentation and scram functions and is designed to fail safe in the event of a loss of power. Chapter 7 of this SER contains additional discussion of the RCS and RPS.

The design criteria are based on applicable standards, guides, codes, and criteria and provide reasonable assurance that the facility SSCs will function as designed and required by the analyses in the SAR. The safety-related SSCs have been maintained or changed using license amendments or licensee review processes, including 10 CFR 50.59, “Changes, Tests, and Experiments”; maintenance; and special procedures, as appropriate, in accordance with the Commission’s rules and regulations and Facility Operating License No. R-37, as amended. The NRC staff previously evaluated all amendments to the facility license, and the NRC inspection program verified that the licensee conducted the proper reviews. Chapter 16 of this SER discusses age-related issues. Based on the above discussions and those in the referenced chapters of this SER, the NRC staff concludes that the design and construction of safety-related SSCs provide reasonable assurance that the SSCs will continue to meet the design criteria for the period of the renewed license.

3.2 Meteorological Damage

Section 2.3 of this SER presents the meteorology of the reactor site. While severe storms or tornados are possible at the site, the reactor and associated safety systems are housed in a steel containment structure that provides considerable protection from meteorological phenomena. According to the licensee, the yield stress of the steel containment shell is 103 megapascals (15,000 psi). Further, the exhaust stack is designed for a 45 m/s (100 mph) wind. With this design level, both structures are designed to withstand the highest recorded wind speeds in the vicinity. The 100-year return snowpack has a water equivalent of 11.7 cm (4.6 in.). However, according to the licensee, this quantity of snow would not accumulate on the containment building because of its rounded shape. The NRC staff reviewed the design criteria of the reactor building and exhaust stack against potential meteorological hazards presented by the licensee and discussed above in this SER and finds that the design criteria exceed the potential wind and water loading associated with the reactor site. Based on this finding, the NRC staff concludes that the designs of the structures are adequate to withstand the potential meteorological conditions that are likely to occur during the period of the renewed license.

3.3 Water Damage

Section 2.4 of this SER presents the hydrology of the reactor site and shows that there is no significant risk of flooding. According to the licensee, water accumulations around the containment structure would not penetrate the building, as it is impermeable. Based on the lack

of flooding risk and the design of the reactor building, the NRC staff concludes that water damage poses no significant risk to the safe operation or shutdown of the reactor.

3.4 Seismic Damage

As discussed in Section 2.5 of this SER, the reactor site is located in a zone of low seismic activity. Chapter 13 of the MITR-II SAR presents an analysis of a postulated seismic event. The MITR-II shim blades are designed to fail safe in that a loss of power causes the blades to drop into the core, shutting down the reactor. As discussed in Chapter 8 of this SER, electricity is not required to accomplish or maintain safe shutdown of the reactor. Disruption of electrical service caused by seismic activity poses no significant risk to the facility. Damage to the core will not occur as long as the core tank is intact. According to the licensee's analysis, rupture of the tank would require a horizontal acceleration of at least 5.1 g or a vertical acceleration of at least 3.5 g. These values are far greater than the maximum safe-shutdown earthquake acceleration of 0.225 g and are in excess of those associated with the Cape Ann earthquake, which was the largest recorded seismic event in New England. Based on these discussions, the NRC staff concludes that the design of the facility provides reasonable assurance that potential seismic events at the facility site will not pose a significant radiological risk to public health and safety.

3.5 Systems and Components

The licensee has a preventive maintenance and surveillance program to provide reasonable assurance that the mechanical systems and components and the electrical and instrumentation systems and components important to safety meet the performance requirements of the TS. Section 4.2.1 of this SER discusses the fuel design requirements. Chapter 13 evaluates accident scenarios, and Chapter 16 considers aging issues. These discussions show that the fuel cladding design basis and related TS provide reasonable assurance that fuel cladding integrity will be maintained under all credible conditions. Chapter 7 of this SER discusses the design of the instrumentation and control (I&C) systems, including the RCS and RPS. Section 4.2.2 discusses the design of the control rods. These discussions show that the reactor safety system design bases and related TS provide reasonable assurance that the reactor safety system will function as designed to ensure the safe operation and shutdown of the reactor.

3.6 Conclusions

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

4 REACTOR DESCRIPTION

4.1 Summary Description

The licensee has analyzed the MITR-II for operation at or below a steady-state thermal power level of 6.0 MW(t). Key parameters of operation are a primary coolant flow rate of 126 liters per second (l/s) (2,000 gallons per minute (gpm)), a coolant outlet temperature of 55 degrees C (131 degrees F), and a coolant level at overflow of 3.14 m (10.3 ft) above the top of the fuel plates. The TS permit operation with forced or natural convection cooling, with the former being the normal mode of operation. The coolant and moderator are both light water, and two reflectors surround the core. The inner reflector consists of heavy water contained in a reflector tank that surrounds the light-water core tank. The outer reflector is constructed of graphite blocks. The MITR-II is equipped with a variety of experimental facilities, including beam ports, irradiation tubes, and medical irradiation rooms.

4.2 Reactor Core

4.2.1 Reactor Fuel

The MITR-II uses flat, aluminum-clad fuel plates that consist of highly enriched uranium-235 in a uranium-aluminum (UAl_x) cermet matrix. The fuel plates have a finned surface to increase the heat transfer area. The plates are arranged in rhomboid-shaped fuel elements to allow configuration into a hexagonal-shaped core. The fuel elements are symmetric in the radial and axial directions, allowing them to be oriented in any configuration in the core. This flexibility allows for more even burnup in the fuel management scheme. Section 4.2.1 of the SAR provides details of the fuel element dimensions, enrichment, and fuel loading.

The maximum fission density for UAl_x fuel used in the MITR-II is limited to 1.8×10^{21} fissions per cubic centimeter (fissions/cm^3), as stated in TS 3.1.6, “Fuel Parameters.” Fuel integrity is monitored by visual inspections required by TS 3.1.6; the frequency of inspections is quarterly, as specified in TS 4.1, “Reactor Core Parameters.” The design of the reactor fuel and core configuration is addressed by TS 5.3, “Reactor Core and Fuel,” which specifies the number of fuel elements, fuel material, and fin and cladding specifications. Prevention of fuel overheating is ensured by TS 2.2, “Limiting Safety System Settings (LSSS),” and TS 3.2.3, “Reactor Protection System.” These TS provide a set of operating restrictions on reactor power, flow, temperature, and coolant level that will prevent exceeding critical heat flux (CHF) values and the fuel temperature safety limit of 450 degrees C (840 degrees F) specified in TS 2.1, “Safety Limits.” In addition, accident analyses performed in Chapter 13 of the MITR-II SAR demonstrate that operation within the limits of the TS ensures no loss of fuel integrity for any credible accident scenarios.

The NRC staff reviewed the MITR-II safety analysis and, based on the information above, determined that the MITR-II fuel design is adequately supported by the materials testing reactor fuel development program. The NRC staff also concludes that the MITR-II limits on fuel temperature and fuel burnup are supported by research and testing performed on similar UAl_x dispersion fuels. 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 public health and safety or the environment.

4.2.2 Control Blades

Section 4.2.2 of the MITR-II SAR describes the RCS, including the materials, components, fabrication specifications, and design bases. The system includes six individual shim blades and a single regulating rod. The shim blades are natural boron-impregnated stainless steel and the regulating rod is a cadmium-wrapped aluminum rod, clad in aluminum. Each shim blade and regulating rod is operated by an independent drive mechanism. The shim blades are coupled to the drive mechanisms by electromagnets, and the nuclear safety channel scram amplifiers provide current to the magnets. Section 7.3 of this SER further discusses the shim blade and regulating rod control systems.

The MITR-II shim blade control system is designed to ensure the capability to provide safe reactor operation and shutdown under all conditions, including that of a single failure or malfunction in the control system. Multiple independent shim blades provide redundancy, and insertion of any five blades results in a shutdown under the most reactive core conditions. This redundancy in ensured by the operability requirements specified in TS 3.1.4, "Core Configuration," and TS 3.2.1, "Operable Control Devices." TS 3.1.2, "Shutdown Margin," ensures a minimum 1 %Δk/k shutdown margin for the reference core condition with the most reactive blade and regulating rod fully withdrawn and all movable experiments in their most reactive state. This TS 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", issued 2007, and the NRC staff finds it acceptable. An independent reflector dump system required by TS 3.2.5, "Backup Shutdown Mechanisms," provides diversity in shutdown capability and is consistent with the defense-in-depth approach to reactor safety.

The MITR-II RCS is a fail-safe design that requires no power source to initiate a shutdown, automatically shuts down on loss of electrical power, relies on a passive motive force (gravity) for control blade insertion, and needs no mechanical action (such as the release of a latch) to drop a blade. This design is implemented through scram signals that deenergize electromagnetic couplings that release the shim blades into the core. According to the licensee, the design of the shim blade electromagnets allows the blades to release within approximately 100 milliseconds. Rapid scram times are required by TS 3.2.1, which specifies a minimum insertion time from scram signal initiation to 80 percent insertion for each blade of less than 1 second. The scram time limit is comparable to that for similar research reactors. Accident analyses performed in Chapter 13 of the SAR demonstrate that this scram time limit is adequate to protect the fuel cladding integrity, and the NRC staff finds it acceptable.

The SAR provides shim blade and regulating rod reactivity worth and insertion and withdrawal speeds. TS 3.2.2, "Reactivity Insertion Rates and Automatic Control," limits the maximum positive reactivity addition rate to $5x10^{-4}$ Δk/k/s and limits shim blade withdrawal to one shim blade at a time. The NRC staff performed check calculations to verify that the maximum differential rod worth and insertion/withdrawal speeds presented in the SAR are consistent with the TS limit and accident analyses. An analysis of a ramp reactivity addition accident in Chapter 13 of the SAR uses a conservative reactivity addition rate of $6.5x10^{-4}$ Δk/k/s. The analysis demonstrates that there will be no loss of fuel integrity, and the NRC staff finds this conservative approach acceptable to justify the maximum reactivity addition rate specified in the TS.

TS 4.2, "Reactor Control and Safety Systems," requires periodic measurement of the reactivity worth of control devices, control device withdrawal and insertion times, scram times, and periodic inspection of shim blade absorbers, electromagnets, and rod drives. The required

surveillance intervals are consistent with the guidance in ANSI/ANS-15.1, and the NRC staff finds them acceptable to ensure the systems meet the operability requirements and limits specified in the TS.

The NRC staff reviewed the MITR-II safety analysis and determined that the control rod section adequately describes the reactivity control and shutdown systems of the MITR-II. The NRC staff finds that the analyses presented in the SAR (including accident analyses in Chapter 13) demonstrate sufficient reactivity worth for control of excess reactivity, adequate shutdown margin, and acceptable control rod dynamic characteristics for both normal and accident conditions. Based on the discussion and findings presented above, the NRC staff concludes that the reactivity control systems and related TS provide reasonable assurance that the reactivity control systems will allow safe and reliable operation and shutdown of the MITR-II.

4.2.3 Neutron Moderator and Reflector

The MITR-II is a light-water-moderated-and-cooled reactor that uses both heavy-water and graphite reflectors. Chapter 5 of the SAR discusses the primary coolant (moderator) properties, including coolant chemistry control, material compatibility, radiation effects, and provisions for removing explosive gasses generated by radiolysis. TS 3.3.6, “Primary Coolant Quality Requirements,” specifies limits on primary coolant pH and conductivity to minimize corrosion of primary system components and fuel. TS 5.2, “Primary Coolant System,” specifies the design requirement that all materials in contact with the primary coolant shall be aluminum alloys or stainless steel except for small noncorrosive components. This design requirement ensures material compatibility between the primary coolant and components in contact with the primary coolant to minimize corrosion of the primary system components and quantities of impurities in the primary coolant that could become activated. TS 4.3, “Coolant Systems,” specifies surveillance requirements for coolant chemistry that are consistent with the recommendations in ANSI/ANS-15.1, and the NRC staff finds them acceptable. Chapter 11 of this SER discusses radioactivity in the primary coolant. Chapter 5 of this SER addresses hydrogen gas generated by radiolysis of the primary coolant.

The heavy-water reflector is contained in the reflector tank, which surrounds the core tank. Components in contact with the heavy water are either aluminum or stainless steel, and free surfaces are blanketed with helium cover gas. The reflector system contains a purification system, heat removal system, and radiolytic decomposition gas recombiner system. TS 3.3.3, “D_2 Concentration Limit,” contains requirements related to the decomposition of the heavy water by radiolysis. The TS specifies a maximum deuterium concentration of 6 volume percent in the helium blanket, which is below the lower explosive limit of 6.8 volume percent estimated from data reported in the literature at the maximum credible helium system temperature. TS 3.3.3 also provides requirements for operation of the recombiner system. These requirements ensure that the recombiner will maintain the deuterium concentration below the TS limit. TS 3.2.3 specifies the minimum reflector tank level and reflector flow rate required for reactor operation. According to the licensee, the requirement for the minimum flow rate ensures adequate system cooling during full power operation. As mentioned in Section 4.2.2 of this SER, dumping of the heavy-water reflector to a dedicated dump tank serves as a backup reactor shutdown mechanism. The requirement for the minimum reflector tank level ensures the reflector dump will add negative reactivity.

Chapter 4 of the SAR discusses the reactivity effects of potential leakage between the moderator and the heavy-water reflector systems. According to the licensee, the system pressures are maintained such that the moderator would leak into the reflector tank, and this

leakage always has a negative reactivity effect. Analysis provided by the licensee shows that leakage of heavy water into the primary system could have a positive reactivity effect if the heavy water were confined to the annular space surrounding the core. However, both forced and natural circulation of the primary coolant through the annular space would carry the heavy water into the core proper, causing a strong negative reactivity effect. Chapter 5 of the SAR and Chapter 13 of this SER discuss leakage of heavy water out of the system.

The graphite reflector is composed of reactor-grade graphite stringers stacked around the sides of the reflector tank. The graphite reflector extends from the heavy-water reflector tank to the radial thermal shield. The graphite reflector is filled with carbon dioxide or other inert gas that is circulated to remove moisture and minimize argon-41 generation. Heat is removed by conduction to the reflector tank and thermal shield. According to the licensee, there is no significant buildup of Wigner energy in the graphite because the neutron flux is low (approximately 10^{13} neutrons per square centimeter per second) and mostly in the thermal energy range. Also, according to the licensee, the peak operating temperature of the graphite is sufficient to cause some annealing. The licensee verified these assertions by testing reflector specimens.

The NRC staff reviewed the neutron moderator and reflector systems presented in the SAR against the guidance in NUREG-1537 and ANSI/ANS-15.1. The NRC staff finds that the systems demonstrate material compatibility with respect to chemical, thermal, and radiation environmental performance. The NRC staff finds that the system designs contain appropriate features to control the buildup of explosive gasses. In addition, the NRC staff finds that the TS governing operating limits and surveillance requirements for these systems are consistent with the guidance in ANSI/ANS-15.1. Based on these findings, the NRC staff concludes that continued operation within the requirements of the TS provides reasonable assurance that the moderator and reflector systems will perform as necessary and will not adversely affect safe reactor operation, prevent safe reactor shutdown, or cause uncontrolled release of radioactive material to the unrestricted environment.

4.2.4 Neutron Startup Source

The power history of the MITR-II is sufficient to maintain a strong photoneutron source, thereby precluding the need for a neutron startup source in all but the initial startups in 1958 and 1975. In the event that the photoneutron source becomes insufficient for reactor startup, a licensee-maintained procedure exists for using a plutonium-beryllium neutron startup source. According to the licensee, the source would be inserted in a coolant-filled aluminum tube near the center of the core. Instruments would then be checked for operability, including registering the minimum number of counts necessary for proper instrument use. After the reactor is critical, the source would be removed before increasing reactor power above 500 watts. TS 3.1.4 specifies this power level restriction to preclude melting of the source encapsulation.

The NRC staff evaluated the use of the neutron startup source against the guidance in NUREG-1537 and requirements for the use of similar neutron sources at other licensed nonpower reactors. The NRC staff finds that the use of the neutron source is consistent with the guidance and, although not normally used, would be similar to that at other licensed nonpower reactors. Based on these findings, the NRC staff concludes that continued use of the plutonium-beryllium source, as needed after extended shutdowns and under the restrictions of the applicable TS and procedures, provides reasonable assurance that the source and holder design can operate safely and reliably.

4.2.5 Core Support Structure

As described in the SAR, the core support structures are designed to ensure that all fuel elements, dummy elements (solid aluminum elements), and in-core experimental facilities are properly secured against all anticipated loads. These loads include the buoyant force of the coolant and the hydraulic forces associated with primary flow. Core components are positioned and secured by means of a lower and an upper grid. The upper grid can be unlatched and rotated to permit refueling operations. The upper grid is mechanically and electrically interlocked to ensure that all six shim blades are fully inserted, a scram signal is present, and primary coolant pumps cannot be operated in the unlatched condition. TS 3.1.4 specifies that the reactor shall not be made critical unless all fuel elements and other core components are secured in position and the upper grid is latched. This requirement ensures that the reactor will not be operated unless the core support structure is capable of maintaining the physical orientation and arrangement of the core components. The core support housing is constructed of aluminum-6061. Coolant corrosion of the core support structures is addressed by proper chemistry control as required by TS 3.3.6, discussed in Section 4.2.3 of this SER. Chapter 16 of the SAR addresses radiation damage effects.

The NRC staff evaluated the core support structures and finds them to be constructed of compatible materials and to be of acceptable design to ensure a stable and reproducible core configuration for all anticipated conditions throughout the reactor life cycle.

4.3 Reactor Tank

The reactor tank design for the MITR-II includes two concentric tanks: the light-water core tank and the heavy-water reflector tank. The design pressure for both tanks is 60 psig, well above the normal operating pressure of the tanks. Both tanks are constructed of aluminum. TS 5.2 specifies the design features of the reactor tank. Corrosion is minimized by proper chemistry control as described in Chapter 5 of the SAR. These requirements are implemented by TS 3.3.6, previously discussed in Section 4.2.3 of this SER. The licensee has evaluated the lifetime corrosion and radiation effects as acceptable for continued safe operation, as described in Chapter 16 of the SAR.

All penetrations through the light-water core tank are above the core, and siphon breakers prevent draining the core tank. Therefore, because of the concentric tank design, no single pipe break or tank failure could result in uncovering the core. Section 13.3 of this SER provides further discussion of loss-of-coolant accidents.

The primary coolant provides shielding above the core along with the top shield lid. The licensee evaluated radiation levels above the core without the top shield lid, and TS 3.1.4 restricts operation in this condition to 100 kilowatts (kW) or below. This power level restriction limits direct radiation from the core and the concentrations of activation products in the primary coolant to control dose rates near the reactor top. The heavy water in the reflector tank provides shielding to the reactor floor along with the biological and thermal shields. A low level in the reflector tank caused by a reflector dump or leak would result in reactor shutdown, as described in Chapter 5 of the SAR.

The NRC staff evaluated the reactor tank design and finds it to be constructed of compatible materials and to be of acceptable design to minimize the potential for leaks or loss of coolant. The design features, related TS requirements, and acceptable past performance provide reasonable assurance of continued reliable operation and integrity.

4.4 Biological Shield

The MITR-II shielding consists of a thermal shield constructed of two 5-cm (2-in.) concentric steel cylinders with a 3.8-cm (1.5-in.) lead-filled space between the cylinders and surrounded by a biological shield constructed of high-density concrete. The design basis discussed in the SAR was a dose rate of 0.002 milliSievert per hour (mSv/hr) (0.2 millirem per hour (mrem/hr)) on the shield surface at the initial operating power of 1 megawatts (MW), corresponding to 0.012 mSv/hr (1.2 mrem/hr) at 6 MW power operation. This level would permit continuous occupational exposure without exceeding the occupational exposure limit of 50 mSv (5000 mrem) specified in 10 CFR 20.1201, "Occupational Dose Limits for Adults." Personnel do not normally work in close proximity to the biological shield for extended periods, so actual radiation dose to personnel from radiation escaping through the biological shield should be a small fraction of the projected maximum. Chapter 11 of this SER discusses the MITR-II radiation protection and ALARA programs.

Equipment penetrations through the shields are stepped or slanted to preclude radiation streaming. Experiment ports are provided with additional shielding such as water-filled shutters, beam stops, or temporary shielding. A 0.64-cm (0.25-in.) boral lining of the inside surfaces of the thermal shield minimizes neutron activation of the shields. The shield coolant system consists of redundant sets of cooling coils within the lead portion of the shield. TS 3.2.3 specifies limits for the minimum shield coolant system flow rate, depending on the mode of reactor operation. According to the licensee, this flow rate is sufficient to remove radiation heat deposited in the shield. The licensee's protocol for increasing reactor power to 6 MW(t) includes checks of reactor system performance to ensure that all cooling systems are adequate for operation at the increased power level.

The NRC staff evaluated the MITR-II biological shield design against the guidance in NUREG-1537 and reviewed personnel dose records and projected radiation dose rates during operation at 6 MW(t). The NRC staff finds that the shield design is consistent with the guidance and personnel dose records demonstrate acceptable past performance. The NRC staff finds that the licensee's projections indicate that the shield will continue to provide adequate shielding for personnel at the increased power level. Based on these findings, the NRC staff concludes that the shield design will continue to support safe operation of the reactor and limit personnel exposures to acceptable levels.

4.5 Nuclear Design

4.5.1 Normal Operating Conditions

Normal operating conditions for the MITR-II include 24-hour-per-day and 7-day-per-week operation, with periodic shutdowns for refueling. The licensee analyzes each reactor core configuration to ensure that the thermal-hydraulic limits, shutdown margin requirement, and fission density limit are met. Burnup calculations and reactivity of other core components and experiment facilities are incorporated into each core configuration analysis. The core is required to contain a fuel element, dummy element, or in-core sample assembly in each of the 27 core positions in order to ensure the proper flow characteristics assumed in the analyses. TS 3.1.4 requires this configuration control. TS 3.1.2 specifies the shutdown margin requirements, and TS 3.1.6 specifies the fission density limit of 1.8×10^{21} fissions/cm^3. The fission density limit is consistent with fuel burnup at similar reactors using comparable fuel, and the licensee has not observed any adverse effects on the MITR-II fuel as a result of using the fuel to this limit.

The licensee provided typical reactivity conditions for normal operations in Section 4.5.3.1 of the SAR. According to the licensee, withdrawal to a typical critical position of 20.3 cm (8.0 in.) on the shim bank and 12.7 cm (5.0 in.) on the regulating rod with the coolant/reflector at 20 degrees C (68 degrees F) inserts a total reactivity of 5.96 Beta. The total worth of the shim bank and regulating rod is 12.79 Beta; therefore, the remaining excess reactivity is 6.83 Beta. Attaining normal operating temperature and equilibrium xenon will insert -0.26 Beta and -4.2 Beta, respectively, leaving 2.37 Beta excess reactivity. A typical experiment in an in-core sample assembly may insert another -0.6 Beta, leaving 1.77 Beta excess reactivity to compensate for fuel burnup. With an estimated change in reactivity of -0.25 milliBeta per megawatt-hour of fuel burnup and neglecting peak xenon startup requirements, this example would allow about 49 days of 6-MW operation. According to the licensee, refueling is performed when a condition of xenon-precluded startup is attained. This example demonstrates compliance with the subcritical interlock (critical position greater than 12.7 cm (5.0 in.)), excess reactivity limit of TS 3.1.1, "Excess Reactivity," and the shutdown margin requirement of TS 3.1.2, while accounting for the reactivity effects of temperature, burnup, and experiment facilities.

Additional TS related to normal operating conditions are provided to ensure adequate reactivity control (TS 3.2.1, 3.2.2, 4.2) and limit the reactivity associated with experiment facilities (TS 3.1.3, "Maximum Safe Step Reactivity Addition"). TS 4.1 provides surveillance requirements for the reactor core parameters such as excess reactivity, shutdown margin, core configuration, and fission density. The NRC staff reviewed the information in the SAR and the TS requirements related to normal operation of the reactor against the guidance in NUREG-1537 and ANSI/ANS-15.1. The NRC staff finds that the licensee considered an appropriate variety of core configurations, including a limiting core configuration, and that these core configurations contained the components required for an operable reactor core. The NRC staff finds that the licensee used input parameters justified by analyses presented in the SAR. The NRC staff finds that the normal operating parameters presented in the SAR are within the limits of the TS, and that the TS requirements are consistent with the guidance in ANSI/ANS-15.1. Based on these findings and the above discussion, the staff concludes that the licensee has adequately analyzed expected normal reactor operation during the period of the renewed license. The staff further concludes that the TS provide reasonable assurance that normal operation of the MITR-II will not pose a significant risk to public health and safety or the environment.

4.5.2 Reactor Core Physics Parameters

According to the licensee, the reactor physics parameters for the MITR-II are determined by both calculation and measurement methods. Neutron lifetime and effective delayed neutron fraction are given as 100 microseconds and 0.00786, respectively, as determined by calculation and confirmed by rod drop and noise analysis experiments performed by the licensee in the 1975 startup testing. These values are not directly comparable to other similar research reactors since the light-water moderator and heavy-water reflector design of the MITR-II is unique. According to the licensee, the MITR-II delayed neutron fraction is larger than most other research reactors because the designs of the core tank and heavy water reflector create a significant photoneutron flux from the reflector to the core.

Table 4-5 and Section 4.5.1.6 of the SAR present void coefficients and temperature coefficients, respectively. These values were measured during the 1975 startup testing or other special test programs and vary approximately ±20 percent with radial power distribution. These values are negative over the normal range of reactor operation, demonstrate compliance with TS 3.1.5,

"Reactivity Coefficients," and are consistent with the guidance in NUREG-1537. TS 7.2.2, "Reportable Occurrence Reports," requires the licensee to report any significant variation of these parameters to the NRC, including any variations observed as a result of the increase in reactor power to 6 MW(t).

The computer codes CITATION, Monte Carlo N-Particle (MCNP), or both are used to perform power density distribution calculations. CITATION is a diffusion theory code available from the Radiation Safety Information Computational Center at Oak Ridge National Laboratory, and MCNP is a Monte-Carlo nuclear physics code developed at Los Alamos National Laboratory. The licensee measured the axial and radial neutron flux densities as part of the startup testing in 1975. Axial neutron flux densities were measured again about a year later using a different method to confirm the effect of shim bank height on flux profile. According to the licensee, both radial and axial flux profiles are independent of power and dependent on the position of control devices. The licensee used both sets of experimental data to normalize flux profiles calculated with the computer codes, and used the corrected power distributions for thermal-hydraulic limits analyses. The licensee determined the radial and axial peaking factors to be 2.0 and 2.1, respectively.

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

4.5.3 Operating Limits

The subcritical interlock required by TS 3.2.4, "Control System Interlocks," limits the excess reactivity of the MITR-II. The subcritical interlock restricts operation to a critical position height of no less than 12.7 cm (5.0 in.). According to the licensee, this effectively limits the maximum possible excess reactivity to about 9.4 Beta. Surveillance requirements in TS 4.1 specify at least annual verification of the operability of the subcritical interlock. The licensee analyzed the operational requirements for excess reactivity for a typical core, including factors for temperature, burnup, neutron poisons, and experiments. This analysis demonstrates a readily operable reactor with a modest excess reactivity that can be fully controlled under normal operating conditions.

The shutdown margin requirement for the MITR-II is that it be possible to shut the reactor down by at least 1 %Δk/k using shim blades from the cold (10 degrees C (50 degrees F)), xenon-free condition with the most reactive blade and the regulating rod fully withdrawn and all movable experiments in their most reactive states. This requirement is specified in TS 3.1.2 and satisfies the "stuck rod" criterion found in the guidance in NUREG-1537 and ANSI/ANS-15.1. The licensee provided an example shutdown margin calculation in Section 4.5.3.3 of the SAR, using a total shim blade reactivity worth of 12.63 Beta. Assuming the most reactive shim blade (2.26 Beta) and regulating rod are full out, and accounting for a temperature drop to 10 degrees C (50 degrees F), xenon decay, and experiment reactivity, the shutdown margin was 2.53 Beta, which is within that required by TS 3.1.2 (the required minimum of 1 %Δk/k is approximately 1.27 Beta). The NRC staff reviewed the calculations provided in the SAR and

determined that they demonstrate the adequacy of the shutdown margin for all core configurations. Surveillance requirements in TS 4.1 require verification of the shutdown margin at least annually and after significant changes in core configuration. The licensee determines the shutdown margin before every refueling and estimates the accuracy of the calculations at ±10 percent.

Section 13.7 of this SER discusses a loss of power accident; however, the fail-safe design of the MITR-II control system means that loss of electrical power has no effect on the shutdown margin. A control rod withdrawal accident is not explicitly analyzed, but it would be bounded by the analysis of a ramp reactivity insertion of 6.5×10^{-4} $\Delta k/k/s$ presented in Chapter 13 of the SAR. This reactivity insertion rate is greater than the maximum controlled reactivity addition rate specified in TS 3.2.2. The analysis also bounds the credible withdrawal of the most reactive control rod (differential worth of 0.860 Beta per inch) at the maximum rod withdrawal rate of 10.8 cm per minute (4.25 in. per minute), resulting in approximately 4.8×10^{-4} $\Delta k/k/s$. The transient analysis demonstrates that the limit on the reactivity insertion rate precludes fuel damage from a ramp reactivity insertion.

The thermal-hydraulic analysis of the MITR-II core with a peak thermal power density is based on the assumption of five nonfueled positions in the core. Since the power peaking factors are a function of the number of nonfueled positions and the shim bank height, the licensee analyzed this core at a series of bank heights to determine the worst case values. The licensee determined radial and axial peaking factors of 2.0 and 2.1, respectively. Also, TS 3.1.4 requires nearly uniform alignment of the shim bank to preclude a skewed flux profile that might invalidate the chosen peaking factors. These worst case peaking factors were used in the determination of safety limits specified by TS 2.1 and LSSSs specified by TS 2.2, and in the transient accident analyses discussed in Chapter 13 of this SER.

The NRC staff reviewed the operating limits for the MITR-II against the guidance in NUREG-1537 and ANSI/ANS-15.1. The NRC staff finds that the licensee adequately justified the limits on excess reactivity and shutdown margin through conservative analyses of reactivity transients and shutdown requirements. The NRC staff finds that the related TS requirements are consistent with ANSI/ANS-15.1 and contain appropriate reactivity requirements to define an envelope of safe operating conditions. Based on these findings and the above discussion, the NRC staff concludes that the operating limits provide reasonable assurance that the reactor can be shut down safely under all credible reactivity transients.

4.5.4 Conclusions

The NRC staff reviewed the MITR-II 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 reactivity coefficients provide inherent safety characteristics during normal operation and transients. The NRC staff reviewed the shutdown margin requirements and finds them comparable to requirements for similar reactors and sufficient to allow the reactor to be shut down safely and remain subcritical under all credible conditions without risk of fuel damage. The NRC staff considers the worst case peaking factors to be sufficiently conservative for use in determining the thermal-hydraulic limits. Based on the above, the NRC staff concludes that the nuclear design and the requirements of the TS provide reasonable assurance that the reactor can be operated and shut down safely.

4.6 Thermal-Hydraulic Design

The objective of the thermal-hydraulic design for the MITR-II is to guarantee the structural integrity of the fuel. The aluminum fuel cladding begins to soften at about 450 degrees C (842 degrees F); therefore, sufficient cooling must be ensured to prevent the fuel cladding from reaching this temperature. For the MITR-II core design, the phenomena that could result in insufficient cooling are CHF as a result of channel dryout or the onset of flow instability (OFI). Empirical correlations are used to set the safety limits on reactor power and coolant temperature, flow, and height required to ensure that these phenomena do not occur.

As required by 10 CFR 50.36(c)(1)(i)(A), safety limits are limits on process variables to reasonably protect barriers that guard against uncontrolled release of radioactivity. Safety limits are determined for both forced and natural convection modes of operation and for a fast reactivity transient. The safety limits for the MITR-II specified in TS 2.1.1, and TS 2.1.2 restrict the reactor power, coolant flow, coolant height above the core, and coolant outlet temperature to ensure that the thermal-hydraulic phenomena of CHF or OFI do not occur. Additionally, TS 2.1.3 limits fuel temperature to below the clad softening temperature. TS 5.3 and 3.1.4 specify the core configuration and fuel design upon which the thermal-hydraulic analysis and power distributions are based. These specifications provide the bases for which the safety limits and LSSSs are determined.

For forced convection mode, the licensee compared the CHF and OFI limits to demonstrate that the OFI limit is the most conservative for the MITR-II. The licensee calculated the safety limits specified in TS 2.1.1 using the Whittle-Forgan correlation to produce a curve, at a constant coolant height, of power/flow versus reactor outlet temperature below which the OFI phenomena are not expected to occur (Whittle, 1967). The licensee's calculations appropriately accounted for uncertainties in the power and flow distributions and other engineering variables through the use of hot channel factors.

For natural convection mode, the licensee calculated the safety limits specified in TS 2.1.2 using a Sudo-Kaminaga CHF correlation for a low-flow condition (Sudo, 1993). The calculations included uncertainties in the power and flow distributions and other engineering variables through the use of hot channel factors and an additional uncertainty factor for the CHF correlation. The result is a maximum power limit of 250 kW for natural convection operation with a minimum coolant height of 6 ft above the top of the fuel plates.

For fast reactivity transients, the licensee specified in TS 2.1.3 a safety limit of 450 degrees C (842 degrees F) for the fuel cladding temperature. This is the softening temperature of the aluminum cladding specified in the literature and is consistent with the fuel cladding temperature limit specified at other similar nonpower reactors. This safety limit is necessary to satisfy the requirement of 10 CFR 50.36(c)(1) that reactor conditions do not exceed the safety limits during transients. According to the licensee's calculations, core conditions may exceed the safety limits in TS 2.1.1 and TS 2.1.2 for a very short period of time during a fast reactivity transient. However, the licensee calculated a maximum fuel temperature of approximately 86 degrees C (187 degrees F) during the transient, which is well below the temperature required for potential fuel damage. For this reason, the licensee included the limit on fuel cladding temperature and specified that TS 2.1.1 and TS 2.1.2 do not apply during fast reactivity transients. This satisfies the requirement of 10 CFR 50.36(c)(1) for fast reactivity transients and the NRC staff finds it acceptable. Section 13.2 of this SER discusses the transient calculations in more detail.

As required by 10 CFR 50.36(c)(1)(ii)(A), the LSSSs are settings for the RPS selected to provide automatic protective action to prevent exceeding the safety limits during both normal and abnormal operations. For the MITR-II, the onset of nucleate boiling is chosen as the criterion for the LSSS on reactor power, coolant flow, coolant height above the core, and coolant outlet temperature. The onset of nucleate boiling precedes both CHF and OFI. This guarantees that the reactor will automatically shut down before CHF or OFI occur, and the safety limits will not be approached. The LSSSs also ensure that a fast reactivity transient will not result in exceeding the safety limit on the fuel cladding temperature.

For forced convection, the licensee calculated the LSSSs specified in TS 2.2 using the Bergles-Rohsenow correlation. The licensee generated curves, at constant coolant height and coolant flow rate, of power versus coolant outlet temperature below which the onset of nucleate boiling is precluded. Curves were determined for both one- and two-primary-coolant-pump operation using worst case power distribution and uncertainties in other engineering variables through the use of hot channel factors.

For natural convection, the licensee used the MULCH-II computer code to determine the maximum clad temperatures at 100-kW power operation with a reactor inlet temperature of 60 degrees C. The calculations incorporate the worst case power distribution and engineering uncertainty hot channel factors. The result is a maximum clad temperature of 104 degrees C (219 degrees F) for a minimum coolant height of 3 m (10 ft) above the top of the fuel plates. Since the saturation temperature for this coolant height is about 107 degrees C (225 degrees F), no boiling will occur in the core. Therefore, the LSSSs for natural convection are a maximum reactor power of 100-kW and a coolant outlet temperature of 60 degrees C (140 degrees F) with a minimum coolant height of 3 m (10 ft) above the top of the fuel plates.

The NRC staff reviewed the MITR-II safety analysis and determined, by verification of the calculations provided in the SAR and by independent calculations of CHF and OFI limits using alternative correlations, that the thermal-hydraulic design limits are adequate to ensure fuel integrity under all reactor conditions including accidents. The use of hot channel factors in the analyses conservatively includes uncertainties in the analyses resulting from design variations and analytical methods. The safety limits and LSSSs were calculated using methods and assumptions typical of those for other similar research reactors and are justified for use on the MITR-II. The resulting restrictions on reactor power, coolant flow, coolant outlet temperature, coolant height above the core, and fuel cladding temperature are sufficiently conservative to ensure that steady-state operation within these limits will not cause a loss of fuel integrity. As discussed in Chapter 13 of this SER, the NRC staff finds that these limits are sufficiently conservative to preclude any credible accident from resulting in the loss of fuel integrity. Based on these findings, the NRC staff concludes that the thermal-hydraulic design, as limited by the TS, is adequate for continued safe operation of the MITR-II at the increased power level.

4.7 Conclusions

The NRC staff concludes that the licensee has adequately described the bases and functions of the reactor design to demonstrate that it can be safely operated and shut down from any operating condition or accident assumed in the safety analysis. The systems provide adequate control of reactivity, containment of coolant, and barriers to the release of radioactive material as well as sufficient radiation shielding for the protection of facility personnel. Nuclear and thermal-hydraulic design and operating limits as established by the TS adequately provide for the protection of fuel integrity. The NRC staff concludes that continued operation of the MITR-II

within the limits of the TS and facility license will not result in undue risk to public health and safety, facility personnel, or the environment.

5 REACTOR COOLANT SYSTEMS

5.1 Summary Description

The MITR-II coolant systems include the primary, secondary, reflector, shield, experiment, and fission converter coolant systems. Each system operates independently with its own separate coolant and pumps, piping, and related equipment. The primary system can operate in forced or natural convection modes, with the natural convection mode capable of removing decay heat following a reactor shutdown. The secondary system is designed to transfer the heat load from the primary, reflector, shield, experiment, and fission converter coolant systems to the atmosphere through the cooling towers. Chapter 4 of this SER discusses requirements for the reflector and shield coolant systems. Chapter 10 of this SER discusses experiments and the fission converter.

5.2 Primary Coolant System

The primary coolant system can transfer at least 6 MW of heat from the primary to the secondary system with a minimum forced flow of 114 l/s (1,800 gpm) and maintain the core free of boiling under steady-state operation. Natural convection operation, facilitated by the natural convection and antisiphon valves, is capable of removing at least 100 kW of heat from the core. This heat removal capacity is adequate to cool the core during low-power operation and reactor shutdown. The natural convection valves are a passive safety feature, and in the event of an abnormal condition, such as pump failure or loss of offsite electricity, the transition from forced to natural convection occurs automatically. TS 3.3.1, “Natural Convection and Anti-Siphon Valves,” requires the valves to be operable before reactor operation. Regardless of the cause, if a loss of forced convective flow occurs during extended full-power operation, the reactor will scram and decay heat will be removed effectively. This scenario is analyzed in Chapter 13 of the SAR and discussed in Section 13.4 of this SER.

The primary coolant is contained within the reactor tank and the primary coolant piping system, which is designed to minimize the potential for leaks. The reactor tank penetrations are all above the core, and siphon breakers prevent draining the core tank in the case of a pipe break. The primary coolant system is equipped with water-sensitive tapes or probes at probable leak areas that provide alarm indication to the control room. In addition, the reactor tank is concentrically contained within the reflector tank such that draining of the core tank because of a rupture would require a simultaneous rupture of the reflector tank. These design features provide reasonable assurance that the reactor is adequately protected from a loss of coolant that could drain the core tank.

The chemical environment of the primary coolant is maintained at a pH between 5.5 and 7.5 with an electrical conductivity of less than 10 microsiemens per centimeter (μS/cm) by TS 3.3.6. TS 4.3 requires weekly surveillance of the coolant conductivity whenever the reactor is operating. The reactor may be operated outside the limits in TS 3.3.6, provided the chloride ion concentration is kept to below 6 parts per million. This environment ensures that corrosion of the fuel, core components, and the primary coolant loop structure is maintained within an acceptable limit.

The radiolytic decomposition of the primary coolant may generate hydrogen gas. During normal operation, a continuous purge of the air space above the core prevents the accumulation of hydrogen. In the event of isolation of the core purge, TS 3.3.2, “H_2 Concentration Limit,”

requires reactor power to be reduced to below 100 kilowatts thermal power (kW(t)) within 15 minutes. The licensee's analysis shows that the requirements in the TS ensure that hydrogen concentrations will not exceed the lower explosive limit of 4.1 percent in air during the 15-minute period and during reactor operation below 100 kW(t). This analysis is consistent with and provides justification for TS 3.3.2, which specifies a maximum hydrogen concentration of 3.5 volume percent in the air space above the core.

The reactor coolant above the core provides a significant part of the radiation shielding. The 3 m (10 ft) of coolant is sufficient to allow operation of the reactor up to a power level of 100 kW(t) with the top shield lid removed. Above 100 kW(t), the top shield lid must be installed, as required by TS 3.1.4. These restrictions, along with the design basis of the shielding discussed in Chapter 4 of the SAR, provide reasonable assurance that personnel exposures from radiation above the core tank will be maintained below the limits of 10 CFR Part 20 and ALARA.

TS 3.2.7, "Control Systems and Instrumentation Requirements for Operation," requires core tank level, primary coolant flow, and coolant outlet temperature instrumentation to be operable before reactor startup and during reactor operation. This ensures that the necessary system information is available to the reactor operator and that the safety system channels related to the coolant parameters are operable. Chapter 7 of this SER further addresses the primary coolant system instrumentation.

TS 5.2 specifies design requirements that ensure that the design of the primary coolant system is consistent with the analyses in the SAR. TS 5.2 also specifies that the primary system components in contact with the primary coolant must be chemically compatible with the coolant. TS 4.3 requires annual inservice inspection of primary coolant system components.

The NRC staff reviewed the primary coolant system design described in the SAR and the related requirements of the TS. The NRC staff finds that the TS require adequate coolant system equipment and instrumentation for safe reactor operation. The TS also require the use of appropriate construction materials for primary system components and contain appropriate limits on coolant chemistry to ensure that corrosion will be maintained within the acceptable limits analyzed in the SAR. The NRC staff finds that the surveillance requirements for the primary coolant system are adequate to ensure that system integrity can be maintained and that any degradation will be detected in a timely manner. Based on these findings, the NRC staff concludes that continued operation in accordance with the TS provides reasonable assurance that the primary system can perform all intended functions, as described in the MITR-II SAR.

5.3 Secondary Coolant System

The SAR states that the MITR-II secondary coolant system is designed to transfer a total heat load of about 7 MW from the primary, reflector, shield, experiment, and fission converter coolant systems to the atmosphere through the cooling towers. This capacity is adequate to handle the increase in the licensed power level to 6 MW(t) and simultaneous operation of the fission converter at 250 kW(t). According to the licensee, the cooling towers have a heat load capacity of 10 MW(t) under the most adverse conditions of humidity and temperature. For shutdown cooling, secondary flow can be reduced to the heat exchangers or the cooling towers can be bypassed to prevent overcooling other systems.

Flow, temperature, and pressure instrumentation provides indication in the control room of secondary system process parameters. Low temperature alarms set at 10 degrees C

(50 degrees F) notify the reactor operator to prevent freezing of the heavy water in the reflector system heat exchangers. A nonchemical treatment system provides corrosion control in the secondary coolant system. Polymer and algaecide systems may also be used. A filtration system and feed-and-bleed purge removes accumulated dissolved solids.

Redundant online gamma scintillation detectors monitor the secondary coolant for radioactivity. Leaks from the primary or reflector systems would be detected through the activity of nuclides such as nitrogen-16, fluorine-18, and sodium-24. TS 3.3.5, "Coolant Radioactivity Limits," requires that monitoring for gamma, beta, and tritium-specific beta radiation is performed daily during reactor operation or any day that secondary flow is supplied to a reflector system heat exchanger. TS 3.7.1, "Monitoring Systems," requires daily tritium content sampling or continuous monitoring using a tritium-sensitive radiation monitor installed in the secondary system. TS 3.7.1 also requires monitoring of the levels in the primary storage tank, reflector dump tank, and the fission converter tank and a secondary water monitor that indicates and alarms in the control room any time secondary coolant is flowing through the heavy-water heat exchangers to the cooling tower. In the event that activity is detected, cooling tower spray and discharge would be secured and the leaking equipment isolated, as required by TS 3.7.2, "Effluents."

The NRC staff reviewed the MITR-II secondary coolant system design, as described in the SAR, and the related TS requirements. The NRC staff finds that the secondary system has adequate heat transfer capacity to cool the various reactor and experiment facility cooling systems. The NRC staff also finds that the TS requirements provide diverse means of detecting the leakage of radioactivity into the secondary coolant and appropriate actions to minimize the release of radioactive material to the environment in the case a leak does occur. Based on these findings, the NRC staff concludes that continued operation in accordance with the descriptions in the SAR and the requirements of the TS provides reasonable assurance that the secondary coolant system can support safe reactor operation during the period of the renewed license.

5.4 Primary Coolant Cleanup System

The design basis of the primary coolant cleanup system is to ensure that corrosion of the fuel, core components, and the primary coolant loop is maintained within acceptable limits. The primary coolant cleanup system uses particulate filters and demineralizers to maintain the pH between 5.5 and 7.5 with conductivity below 10 μS/cm in order to minimize corrosion. Conductance probes monitor demineralizer inlet and outlet conductivity and provide a control room alarm if values exceed 2.0 μS/cm. TS 3.3.6 requires chemistry control, and TS 4.3 specifies minimum surveillance requirements.

The primary coolant system demineralizers are located in a shielded area within a high-radiation area of the primary coolant system equipment room. Controlled access prevents inadvertent radiation exposure. The demineralizers are replaced on an as-needed basis as determined by the inlet and outlet conductivity readings. The exhausted column is temporarily stored to allow for radioactive decay; then, the spent resin is discharged, dewatered, and packaged for shipment offsite.

Leak detection on system components is provided by water-sensitive tapes or probes located at probable leak areas that provide alarm indication to the control room. In addition, level alarms in the primary storage tank and reactor tank overflow would notify operators of a leak in the system.

In addition to maintaining primary coolant chemistry, the system provides shutdown cooling through its heat exchanger, which is cooled by secondary cooling or city water, if offsite electricity is unavailable. The system also provides one of the redundant water supplies to the emergency core cooling system (ECCS).

The NRC staff evaluated the MITR-II primary coolant cleanup system and finds that it is of similar design and has coolant chemistry limits that are similar to other licensed nonpower research reactors. Based on this evaluation, the NRC staff concludes that continued operation in accordance with the TS provides reasonable assurance that the corrosion of the fuel, core components, and the primary coolant loop will be maintained within acceptable limits.

5.5 Primary Coolant Makeup Water System

The primary coolant makeup water system supplies deionized water to the light-water process systems and fuel storage pool. City water is processed through particulate filters, activated charcoal, and ion exchange columns before storage in the makeup water storage tank. A recirculation system maintains purity sufficient to meet primary water chemistry requirements. Makeup water can be provided through a permanent connection to the primary coolant storage tank and by hose connections to the shield water and fuel pool systems. Under normal operation, surge capacity of primary coolant is provided by the primary storage tank as part of the normal flow through reactor overflow and the primary coolant cleanup system. According to the licensee, the makeup water and primary coolant storage tanks normally contain at least 50 percent of their 5,450-liter (l) (1,440-gallon (gal)) and 7,570-l (2,000-gal) capacities, respectively. The primary coolant system contains a check valve to prevent backflow of contaminated water to the makeup water system. The NRC staff reviewed the design and operation of the primary coolant makeup water system against the guidance in NUREG-1537. The NRC staff finds that the system design is consistent with the guidance because the system prevents backflow, contains purification equipment to control water chemistry, and has sufficient capacity to compensate for normal losses of primary coolant. Based on these findings, the NRC staff concludes that the makeup water system is adequate to support continued reactor operation.

5.6 Nitrogen-16 Control System

Radiation exposure from nitrogen-16 is controlled within the requirements of 10 CFR Part 20 and the ALARA program by both design and administrative features. The primary coolant system, except for the core purge, is a closed system. The piping and equipment is enclosed in a limited access high-radiation area. The core purge system draws gases from above the core tank out through a storage tank and radiation monitor before discharging to the main ventilation exhaust plenum. TS 3.7.1 requires the operability of this radiation monitoring equipment, and TS 4.7.1, “Radiation Monitoring Systems,” provides surveillance requirements. Coolant sample lines are designed to allow sufficient decay time before the coolant reaches the sample station. For areas where nitrogen-16 may present a hazard, such as portions of the reactor top and the equipment room, administrative controls are used to limit exposure. Additionally, TS 3.1.4 requires the reactor top shield lid to be in place prior to operation above 100 kW(t). This limits nitrogen-16 production in the coolant and consequently the dose rate at the reactor top. The NRC staff concludes that these design and administrative features provide reasonable assurance that personnel exposure will be minimized and remain within the regulatory guidelines.

5.7 Conclusions

Based on the above discussions, the NRC staff concludes that the MITR-II cooling systems, as described in the SAR, are adequate for the removal of heat generated during continuous full-power reactor operation and for the removal of decay heat after 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 fission product release to the environment. The NRC staff concludes that the coolant systems and related TS requirements are sufficient for continued safe reactor operation during the period of the renewed facility operating license.

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 MITR-II ESFs include natural convection and antisiphon valves, an ECCS, and the containment building. The ESFs can be actuated automatically by the protection instrumentation that monitors various parameters during reactor operation, manually by the reactor operator, or passively. The ESFs provide protection against overheating of the core, uncovering the core, and uncontrolled release of radioactive material to the surrounding environment.

6.2 Detailed Descriptions

6.2.1 Natural Convection and Antisiphon Valves

The natural convection valves consist of four ball-type check valves located at the top flange of the core support structure. The valves are held closed by the hydraulic pressure of the coolant during forced convection operation, and they are held open by gravity when forced flow ceases. With the valve open, heated coolant flows up through the core and into the bulk volume of the core tank while cooler coolant flows down through the valves into the space below the core. According to the licensee, three of the four valves are adequate to remove the decay heat from extended full-power operation at 6 MW(t). A heat sink is provided by the heat capacity in the large volume of coolant in the tank, which is sufficient to prevent the loss of fuel integrity as demonstrated by analysis in Chapter 13 of the SAR. Also, the antisiphon valves discussed below provide an additional natural circulation pathway and additional redundancy to the natural convection valves. The natural convection valves facilitate normal low-power operation and shutdown cooling and ensure a passive transition from forced convection to natural convection cooling in the case of a loss-of-coolant-flow accident. The NRC staff reviewed the design and operation of the natural convection valves and finds that the valves are appropriately placed in the primary system and that there is sufficient redundancy to ensure that natural convection cooling will be available when required.

Two antisiphon ball-type check valves are located at the top of the inlet plenum. These passive safety features function similarly to the natural convection valves, but they contain a perforated section of tubing. Upon loss of coolant to the level of the inlet plenum, the perforations allow air to enter the primary piping to break a siphon caused by a rupture in the primary piping system. In the case of siphoning, the antisiphon valves ensure that the core tank level would remain approximately 2 m (6 ft) above the top of the core. The NRC staff reviewed the design and operation of the antisiphon valves and finds that the valves are appropriately placed in the primary system and that there is sufficient redundancy to ensure a siphon break if required.

TS 3.3.1 requires verification of the operability of the valves before reactor operation and checks that the valves are closed before operation above 100 kW(t). TS 4.3 requires annual inservice inspections of the valves. The NRC staff reviewed these TS and finds that they provide reasonable assurance that the valves will be able to perform their intended function when required.

6.2.2 Containment

The MITR-II is located within a containment structure consisting of a domed, steel cylindrical shell, within which is a 0.6-m-thick (2-ft-thick) concrete wall providing additional radiation shielding to the unrestricted area. According to the licensee, the shell is constructed to the specifications in American Petroleum Institute (API) Standard 12C, issued 1958, and API-620, issued 1958. The containment system is designed to withstand an internal pressure of 13.8 kPa (2 psig) above and 0.7 kPa (0.1 psig) below atmospheric pressure, with a maximum permissible leakage rate of 1 percent of the building volume per day per pound per square inch of building overpressure. TS 3.4, "Reactor Containment Integrity and Pressure Relief System," specifies the maximum leakage rate, and TS 4.4, "Containment Surveillance," requires surveillance tests to verify leak rate and ensure building integrity.

During normal operation of the ventilation system, the intake and exhaust air ducts remain open. Upon detection of abnormal radiation levels in the exhaust air, the building automatically isolates by closing hydraulically actuated butterfly dampers in these ducts, as required by TS 3.7.1. Containment isolation is a fail-safe design actuated on demand or loss of signal. Isolation is actuated if any one of the four plenum exhaust monitors (gas or particulate) exceeds its setpoint, or if sample flow or electrical power is lost to any plenum monitor. Section 7.5 of this SER provides further discussion of the instrument and control system of the containment ESF. All other penetrations maintain a gas-tight seal, with the exception of small air lines for the personnel locks, instruments, and compressed air system. The compressed air lines contain check valves, and the integrated leak rate test conservatively accounts for the potential failure of small instrument lines.

The containment was successfully tested at design pressure after construction, and an integral leakage rate test is performed every 2 years as required by TS 4.4. Pressure protection is provided by a relief system that includes two high-efficiency particulate air (HEPA) filters and an activated charcoal filter for the removal of elemental iodine. The minimum efficiency of the charcoal absorber is specified in TS 3.4 as 95 percent for the removal of iodine. This system is manually operated to relieve excess pressure to the ventilation exhaust stack. Two sets of vacuum breakers provide vacuum protection. TS 3.4 requires the vacuum breaker system to operate if the building pressure is 0.7 kPa (0.1 psig) below atmospheric pressure. This is consistent with the design pressure of the containment building. TS 4.4 requires surveillance testing for the relief system, vacuum breakers, and isolation valves.

The MHA analyzed in Chapter 13 of the SAR assumes the complete melting of four fuel plates with subsequent release of fission products to the containment. Fuel failure would be detected by the radiation monitor in the core purge gas effluent and containment isolation initiated by the stack effluent monitors, as required by TS 3.7.1. Release estimates are conservatively based on the maximum permissible leakage rate with the containment at design pressure of 13.8 kPa (2 psig) for the duration of the accident. Dose assessment for this accident demonstrates that the design and functional features of the containment system ensure that exposures will be below the regulatory limits of 10 CFR Part 20.

Based on the above discussion, the NRC staff concludes that the design, testing, surveillance provisions and intervals, and related TS provide reasonable assurance that, if required, the containment ESF will be operable and capable of mitigating the design-basis MHA to doses below the regulatory limits of 10 CFR Part 20.

6.2.3 Emergency Core Cooling System

The ECCS consists of two independent subsystems, each with its own spray nozzle to deliver coolant to each fuel element in the core. The licensee determined the flow from either subsystem to be sufficient to protect the core from a loss of fuel integrity. The water supply for the system is available from the primary storage tank, primary makeup water, or city water. This ensures that normal electrical power is not required for the system to function and that there are diverse coolant sources. TS 3.3.4, "Emergency Cooling Requirements," requires the availability of the ECCS for operation above 100 kW(t) and specifies the minimum system flow rate. TS 4.3.1 requires annual testing of the system including operability of the manual water supply valves and core spray nozzles. This requirement is consistent with the guidance in ANSI/ANS-15.1 and the NRC finds it acceptable.

According to the licensee, a loss-of-coolant accident scenario that could uncover the core is not considered credible for the MITR-II because the core is contained in two concentric tanks (primary and reflector) and siphoning is prevented by passive and redundant antisiphon valves. Nevertheless, the licensee maintains an ECCS to mitigate this contingency. The licensee postulated an experimental malfunction that resulted in draining the core tank. The licensee provided an analysis that shows that a loss of coolant from this hypothetical accident would not result in the loss of fuel integrity. As discussed in Chapter 13 of this SER, the NRC staff reviewed the licensee's analysis and finds it to be reasonable, conservative, and consistent with the requirements of the TS.

Based on the above discussion, the NRC staff concludes that the ECCS design and related TS requirements provide reasonable assurance that the ECCS will adequately cool the reactor core and maintain fuel integrity in the case of a loss-of-coolant accident.

6.3 Conclusions

Based on the above discussions and evaluations performed in the referenced sections of this SER, the NRC staff concludes that the designs of the ESFs, as described in the SAR, provide adequate protection to prevent the loss of fuel integrity from a loss-of-coolant or loss-of-flow accident. Further, the NRC staff concludes that the ESFs provide reasonable assurance that the potential release of radioactive material during the MHA will not pose an undue risk to public health and safety or the environment.

7 INSTRUMENTATION AND CONTROL SYSTEMS

7.1 Summary Description

The MITR-II I&C systems consist of five major subsystems: the RCS, the RPS, the engineered safety features actuation system, the main control panel, and the radiation monitoring system. The overall function of the I&C systems is to monitor and control the reactor and to bring the reactor to a shutdown condition. The RCS has both manual and automatic control capability.

The original I&C equipment for the reactor was analog in design. Subsequent instrument upgrades have been mostly digital or a hybrid of digital and analog technology. The nuclear safety system and the radiation monitoring system's effluent monitors are entirely analog.

7.2 Design of Instrumentation and Control Systems

The MITR-II I&C systems design criteria consist of the following elements:

- The RPS and the ESF actuation system automatically initiate operation to mitigate the consequences of abnormal conditions.
- Elements of the I&C systems that are important to safety include both redundancy and diversity, and their signal cables are routed separately to prevent common-mode failures.
- The I&C systems are designed to be fail safe, through the use of deenergized interlocks or the loss of motive air supply in conjunction with components subsequently transitioning to desired safety positions.
- A single failure will not prevent a safe shutdown because of the redundancy and diversity of components that are important to safety.

The sections below discuss specific design elements for the separate I&C systems.

7.3 Reactor Control System

The RCS provides for the insertion and withdrawal of the MITR-II's six shim blades and its fine-control regulating rod. TS 3.2.1 designates the minimal operable components for operation. According to the licensee, the RCS consists of the following individual absorber insert/withdraw circuits, their associated interlocks, manual and automatic insertion circuitry, and an automatic control circuit:

- The withdraw permit circuit is a startup interlock required by TS 3.2.4 that consists of a string of relays and contacts in series.

- The subcritical interlock (1) establishes a level, below the critical position, to which the shim blades may be individually withdrawn in one step, (2) provides a convenient reference point at which the operator can pause to make a complete instrument check before bringing the reactor to criticality, and (3) ensures that the shim blade bank is at a uniform height before the final approach to criticality. TS 3.2.4 requires the subcritical interlock to be operable before the reactor is made critical.

- Synchrotransmitter units provide fine and coarse position indication for the shim blades' upper and lower limits.

- Blade withdrawal circuits include the following features: (1) as required by TS 3.2.2.2, only one shim blade can be withdrawn at a time, (2) the shim blade absorbers may be dropped from their drives at any position of travel, (3) each shim blade may be run in at its normal speed without interrupting its magnet current, (4) all six shim blades may be run in simultaneously at their normal speed without interrupting their magnet currents, and (5) the fine-control regulating rod operates independently of the shim blades.

- The MITR-II has two modes of steady-state operation: manual control by the operator and automatic control of the regulating rod by a servo mechanism. The automatic control mode cannot be initiated until certain requirements imposed by the "automatic control permit" circuit are met as follows: (1) all shim blades must be above the subcritical interlock position, (2) the deviation between the power set and the actual power must not exceed 1.5 percent, (3) the regulating rod control switch must be in the neutral position, and (4) the regulating rod must be withdrawn beyond its near-in position (~4 cm (1.6 in.)). TS 3.2.2.3, TS 3.2.2.4, and TS 3.2.2.5 contain requirements related to automatic control.

- The automatic rundown circuit is part of the automatic control system. A buzzer and visual alarm activate when the reactor is in automatic control and the regulating rod reaches the "near in" position. If the reactor operator does not assume manual control within 30 seconds, one shim safety blade is driven into the reactor core at its normal drive speed to automatically reduce reactor power.

- The all-rods-in circuit allows the reactor operator to shut the reactor down completely by simultaneously lowering all six shim blades and the regulating rod at their normal drive speeds. This activates the "withdraw permit circuit open" alarm, but does not interrupt current to the shim blade electromagnets.

- The main control panel provides the operating controls and positioning information.

The nuclear instrumentation system for the MITR-II consists of nine neutron or gamma flux monitoring channels. Each channel consists of a detector, high-voltage and signal cabling, an output display device, and associated alarm, scram, or control circuitry. Channels 1 and 2 are used as startup channels and, with Channel 3, have associated scram trips at a period of 10 to 11 seconds. Channels 4, 5, and 6 are used as power-range channels and have high-flux scram trips corresponding to a reactor power level of 6.6 MW(t), as determined by correlating the previous equilibrium value of each detector's output with the thermal power. These six instruments comprise the reactor's nuclear safety system. Channels 7 and 8 are part of the control/console display instrumentation system. Channel 7 provides a linear indication of the flux level, and Channel 8 provides a flux indication if electrical power is lost. Channel 9 is part of the RCS and provides a signal to the automatic control permit circuit.

Section 7.4.2 of the SAR describes the process instrumentation system for the MITR-II. Cooling system instrumentation displays and system controls are provided in the control room. Some system instrumentation also provides local displays. Instrumentation is provided for coolant system process variables, including coolant temperatures, pressures, flow rates, and valve positions. Controls are provided for system components such as pumps, fans, and valves. The NRC staff evaluated the cooling system I&C and finds them to be adequate to allow the reactor

operator to assess current system conditions and make appropriate adjustments to system operation. The NRC staff also finds that the instrumentation provides process variable information for the locations of greatest interest in the system.

TS 4.2 specifies periodic surveillances of required elements of the RCS. For most elements, these are at least an annual test of operability. These elements include measurement of the reactivity worth of control devices, rod speed, scram times, calibration and trip point verification for scram signal instruments, and thermal power. A monthly heat balance is required. All control devices are inspected at least annually. The NRC staff reviewed the requirements of TS 4.2 including specified intervals and finds that the requirements are consistent with the guidance in ANSI/ANS-15.1 and NUREG-1537 and are therefore acceptable.

7.4 Reactor Protection System

The RPS consists of both nonnuclear and nuclear components. Table 3.2.3-1 of TS 3.2.3 specifies the required safety channels for reactor operation, including channel setpoints, protective actions, and minimum number required. The table includes RPS requirements for three modes of reactor operation: two-pump operation, one-pump operation, and natural convection operation. For parameters that have LSSSs, TS 3.2.3 requires the RPS setpoints to be more conservative than the LSSSs. This ensures that the RPS will prevent the safety limits from being exceeded. The RPS setpoints required by TS 3.2.3 are consistent with the assumptions in the SAR. The analyses in the SAR demonstrate that all RPS setpoints are appropriate to ensure the reactor will be safely shut down under all credible accident conditions.

The nuclear instrumentation system for the MITR-II consists of nine neutron or gamma flux monitoring channels. This system provides interlock signals to the RPS to initiate reactor shutdown actions. A trip of any of the scram relays will cause a loss of power to all of the shim safety blade magnets, thereby decoupling the blades from their respective drive motors and dropping the blades into the core by gravity. The system provides scrams for short reactor period and high neutron flux level (reactor power). The TS require redundant safety channels for these parameters. These scrams ensure that reactor power remains below the safety limit and that rapid changes in reactor power are terminated. The system provides a scram if less than two period channel signals are on-scale. This ensures the nuclear instrumentation has the required redundancy. The NRC staff finds these requirements to be consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1 and acceptable.

The non-nuclear portion of the RPS monitors a variety of process variables, as well as mechanical systems in the nonnuclear safety system. The RPS provides redundant scrams on both low primary coolant flow and high primary coolant outlet temperature. The system also provides scrams on low core tank and reflector tank levels and low reflector and shield coolant flow rates. Manual scrams include pushbuttons for a major scram (scram with a reflector dump and containment closure) and minor scram. Minor scrams, which may be initiated at the reactor control room or the medical therapy room consoles, will cause all six blades to drop into the core, the blade drives to run in, and the regulating rod to run in. A reflector dump switch provides an additional manual scram. Operation of the reflector dump valve will also decrease reactivity by rapidly decreasing the level of the reflector. Mechanical scrams include building overpressure and loss of containment integrity due to deflation of both sets of either airlock gaskets. Unlatching the hold-down grid plate that secures the fuel elements in the core also causes a scram. Scrams associated with experiments may be required as part of the experiment review and approval process. TS 3.2.3 and 3.2.4 allow limited scram and interlock bypasses for the purpose of reactivity measurements at low power. The NRC staff reviewed the

RPS system requirements against the guidance in NUREG-1537 and ANSI/ANS-15.1. The NRC staff finds that the TS requirements are consistent with the guidance and cover the full range of safety parameters assumed in the licensee's accident analyses.

TS 4.2 requires surveillance of RPS instruments, setpoints, and protective actions. Surveillance requirements include channel checks, channel test, and channel calibrations, as defined in TS 1.3, "Definitions." Surveillance requirements also include verification of instrument setpoints and tests of interlocks. TS 4.2.3 requires measurement of scram times to ensure the RPS can shut down the reactor within the time assumed in the SAR. The NRC staff reviewed the surveillance requirements against the guidance in ANSI/ANS-15.1. The NRC staff finds that the types and frequency of the surveillance requirements are consistent with the guidance.

Based on the above discussion and findings, the NRC staff concludes that the RPS design and TS requirements provide reasonable assurance that all safety-related parameters are appropriately monitored and that the RPS will safely shut down the reactor, if required.

7.5 Engineered Safety Features Actuation

Chapter 6 of this SER discusses the MITR-II ESFs. The ESFs include the natural circulation and antisiphon valves located in the primary coolant system. These valves actuate passively in the event of a loss of coolant flow or siphoning of the core tank. The ECCS is a third ESF provided to protect against overheating during a loss of primary coolant. It consists of two independent subsystems that provide water to two nozzles located above the core. The reactor operator manually initiates operation of the system. The containment structure and isolation interlocks comprise an ESF system. TS 3.4 details containment integrity requirements. The ducts for the containment building ventilation system are automatically sealed upon detection of an abnormal radiation level in the exhaust air.

Containment structure protection against vacuum is provided by two sets of vacuum breakers installed to protect the integrity of the containment in the event of excessive underpressure within the building. According to Section 6.5.4.1 of the SAR, the inner set of vacuum breakers are set to automatically open when internal pressure is between 100 and 250 pascals below atmospheric (-0.015 and -0.036 psig), while the exterior breakers are set to open between 250 and 430 pascals below atmospheric (-0.036 and -0.062 psig). TS 4.4 requires an annual test of the proper functioning of the vacuum breakers. Containment structure protection against excessive pressure is provided if the building pressure should approach its pressure rating (2.0 psig). Relief is manually actuated and achieved by use of the pressure relief system. Using HEPA and activated carbon filters, this system filters the exhaust air and discharges it to the ventilation exhaust stack.

The NRC staff reviewed the ESF actuation mechanisms and methods against the guidance in NUREG-1537 and finds them to be consistent with the guidance. Based on the above discussion and this finding, the NRC staff concludes that the ESF actuation mechanisms and methods allow for reliable and timely ESF actuation in response to abnormal conditions.

7.6 Control and Console Display Instruments

The MITR-II console displays or otherwise provides the information needed by licensed personnel to operate the reactor. TS 3.2.7 details the required display information. The console consists of three panels. One panel provides information on area and effluent radiation levels. The second panel provides the position of the control devices, the reactor power level

(overlapping indicators from startup to full power), and the reactor period. The third panel shows primary flow and temperature as well as additional displays for reactor power. In addition, the console contains a centralized alarm annunciator panel that provides individual alarms from the process equipment. These are color-coded to indicate severity. Red implies a reactor scram, white is for information, and green indicates that the alarm also registers at a remote panel that is exterior to the containment building. Each alarm is labeled with the underlying cause (e.g., low-level core tank, high conductivity, or low-pressure compressed air). In addition, each alarm shows the corresponding procedure number so that the operator can quickly locate the appropriate response in the procedure manual.

A second function of the display system is to provide essential information at locations outside of the reactor building during emergencies that result in the reactor building becoming inaccessible. Each of the eight weekend alarm conditions has an indicating light that reads out in the Reactor Operations Office in Building NW12. These lights are functional regardless of whether or not the weekend alarm system is activated. In addition, certain instruments may be read out remotely. These include indications of building pressure, core tank level, wind speed and direction, radiation levels on the reactor floor, and airborne radiation levels both inside and outside the building.

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

7.7 Radiation Monitoring Systems

The radiation monitoring system comprises 12 effluent monitors and 10 area monitors. The effluent monitors are Geiger-Mueller detectors on the nine air monitors, and scintillation detectors on the three liquid monitors. The air effluent monitors are located at various locations in the exhaust system, including the base of the stack and in the gas and particulate plenums. The liquid effluent monitors are on the secondary water system and the outlet to the sewer. Area radiation monitors containing energy-compensated Geiger-Mueller tubes are located throughout the reactor facility to provide warning to personnel of increased dose rates. The output of each effluent and area monitor is displayed in the control room. Effluent monitors provide indication of the radioactivity of the air and water that leaves 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).

Redundant plenum gaseous and particulate monitors, located in the equipment room, continually sample the effluent air at the upstream end of the exhaust holdup plenum. A high-level alarm on any one of these four plenum monitors will cause the building ventilation system intake and exhaust fans to stop and the isolation dampers to close before the sampled gas can pass by the exhaust damper. The alarm is displayed on the annunciator panel in the control room.

The NRC staff reviewed the design and operation of the radiation monitoring systems against the guidance in NUREG-1537 and ANSI/ANS-15.1 and systems at other nonpower reactors. The NRC staff finds that the systems are consistent with the guidance and comparable to other systems. Based on these findings, the NRC staff concludes that the radiation monitoring

systems have sufficient instruments and locations to ensure that effluents can be monitored, recorded, and controlled. The NRC staff further concludes the systems are adequate to provide information about the magnitude of the radiation fields of greatest interest in the reactor building and to alert personnel to the existence of any abnormally elevated radiation fields.

7.8 Conclusions

The NRC staff reviewed the design of the I&C systems, as described in the 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 operating conditions. The RCS and the nuclear and process instrumentation are sufficient to provide for the safe control of reactor power and the monitoring of reactor safety parameters. The RPS is adequate for protecting the safety limits, and the ESF actuation is 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 control console, and sufficient radiation monitoring is provided to detect abnormal radiation levels and prevent excessive radiation exposure to personnel or release to the environment.

8 ELECTRICAL POWER SYSTEMS

8.1 Normal Electrical Power Systems

Section 8.1 of the SAR describes the design and construction of the normal electric power system. The functional requirements of the MITR-II normal electrical power distribution system are to supply electrical power for all motors, pumps, and instrumentation associated with the operation of the reactor and fission converter, and to facilitate a safe reactor shutdown if the offsite power supply is interrupted. The normal electrical power system supplies all electrical needs of the facility. Normal power is supplied by two 13.8-kilovolt power lines that feed into separate circuit breakers. Under normal conditions, one of the two feeder line circuit breakers is closed. Should offsite power be lost from the selected feeder line, the circuit breaker can be closed to feed loads from the alternate feeder line. The two supply lines feed into a single transformer, which steps the voltage down to 480 volts. Normal electrical power is not necessary to achieve and maintain safe-shutdown conditions. Upon loss of normal electric power, the following events occur to ensure safe shutdown:

- The six reactor shim blades, which are held up by electromagnets, will drop into the core by gravity. This shuts down the reactor.
- The solenoid holding the heavy-water dump valve closed deenergizes, dumping the heavy water from the reflector. This is sufficient to shut down the reactor even if the shim blades do not insert.
- The isolation dampers in the containment building ventilation system close, thereby isolating the containment and precluding any potential release of radioactive material.
- The natural convection valves and antisiphon valves open to ensure adequate cooling of the core and prevent uncovering the core.

The NRC staff reviewed the normal electrical power system and finds that it is adequate to support normal MITR-II operations. The reactor will safely shut down with loss of normal power.

8.2 Emergency Electrical Power System

The MITR-II emergency electrical power system is designed to provide power for at least 1 hour following a loss of offsite power. Power from the emergency system is allocated for lighting, communications, reactor monitoring, and removal of decay heat. TS 3.6, “Emergency Power,” specifies the equipment to be supplied by the emergency electric power system. The following is the minimum specified equipment:

- one neutron flux level channel
- core tank coolant level indicator
- primary coolant outlet temperature
- radiation monitors specified by TS 3.7, “Radiation Monitoring Systems, Effluents, Hot Cells, and Byproduct Material”
- containment intercom system
- primary coolant auxiliary pump
- lighting required for personnel safety

Emergency electrical power is supplied from a motor-generator set. Energy for the motor of the motor-generator is provided from a bank of 60 lead-calcium storage cells that are rated for 577 ampere-hours at an 8-hour discharge rate. At a nominal battery load of 72 amperes, the batteries can meet emergency loads for approximately 8 hours after the loss of both external electrical power feeders.

A fused disconnect switch is used to connect the 130-volt direct current battery power supply to the emergency electrical power system. The system is normally configured so power will be available to start the motor-generator set. Power to operate the direct current motor-drives that operate each of the two main 13.8-kilovolt circuit breakers comes from a separate line from the batteries. The batteries also supply direct current lights in the utility room.

Upon interruption of normal power, the following automatic actions occur:

- Emergency lighting is shifted to the batteries.
- The motor-generator starts (12-second delay).
- Emergency loads are transferred to the motor-generator output.

Surveillance testing requirements for the emergency electrical power system are specified in TS 4.6, “Emergency Electrical Power Systems,” and consist of quarterly, annual, and biannual measurements and tests. The design of the emergency electrical power system has sufficient redundancy and capacity to easily meet the required electrical needs in the event of a loss of normal power.

8.3 Conclusions

The NRC staff reviewed the design of the electrical power systems, as described in the SAR, and concludes that the systems and related TS requirements are adequate to support normal reactor operation and to achieve and maintain safe reactor shutdown under all abnormal operating conditions. The normal electrical power system is sufficient to provide power to all equipment loads required for reactor operation and instrumentation needed for the safe control of reactor power and the monitoring of reactor safety parameters. The emergency electrical power system is adequate to maintain the reactor in a safe shutdown condition and provides reasonable assurance that a loss of normal electrical power at the MITR-II will not adversely affect public health and safety, facility personnel, or the environment.

9 AUXILIARY SYSTEMS

9.1 Heating, Ventilation, and Air-Conditioning System

Section 9.1 of the SAR describes the design and construction of the containment building heating, ventilation, and air-conditioning (HVAC) system. The MITR-II ventilation system is a single-pass system that supplies filtered, conditioned air to the containment building, which is then collected and exhausted through a stack. The system consists of fans, blowers, filters, dampers, heating and cooling equipment, ductwork, and the exhaust stack. Additionally, the system contains instruments, controls, interlocks, and radiation detectors.

According to the licensee, the ventilation system has four primary functions. First, the system provides temperature and humidity control and fresh air to the containment building. The system includes dedicated climate control equipment for the control room instruments. Second, the ventilation system provides airflow into clean areas and exhausts air out of areas with the potential for airborne radioactivity or contamination through filtered exhaust ducts. The system also provides fresh air directly from the containment building ventilation inlet to the control room at a higher pressure than the rest of the containment building. This minimizes the potential for airborne radioactivity to enter the control room. Third, the system maintains the containment building at a negative pressure with respect to the atmosphere. This ensures that any leakage is into the building and that all air exhausts through a filtered and monitored pathway. Fourth, as required by TS 3.7.1.2, the ventilation system provides rapid isolation of the containment building in the case of abnormal airborne radiation levels. The system contains primary and auxiliary dampers in the ventilation inlet and exhaust. The sets of dampers have automatic and manual closure mechanisms that are redundant and diverse. The NRC staff reviewed the ventilation system design and finds that it is consistent with the guidance in NUREG-1537 and the licensee's analyses in Chapters 11 and 13 of the SAR.

TS 3.5, "Ventilation System," specifies requirements for operation of the ventilation system. TS 3.5.1 requires a minimum exhaust flow rate of 3.5 cubic meters per second (7,500 cubic feet per minute) when the reactor is operating at power levels greater than 250 kW and TS 3.5.2 requires the reactor operator to reduce power to less than 250 kW if the ventilation system stops operating. As discussed in Chapter 11 of this SER, these requirements limit the concentration of argon-41 in the containment building to an acceptable level. Also, the required minimum ventilation exhaust flow rate is consistent with the licensee's assumption for dilution of airborne radioactive effluents. TS 3.5.3 requires the containment building pressure to be below atmospheric pressure prior to reactor startup. A negative pressure differential ensures that leakage is into the containment building, thus ensuring that airborne radioactive effluents are exhausted through a monitored pathway. According to the licensee, internal procedures require the reactor operator to reestablish a negative pressure differential within 5 minutes if ventilation stops operating or the pressure differential becomes positive during reactor operation. Given that the reactor is located within a containment building, brief periods of small positive pressure differential should not cause significant leakage. TS 3.5.4 specifies the equipment required for ventilation system operability. The NRC staff reviewed the ventilation system requirements against the guidance in ANSI/ANS-15.1 and NUREG 1537 and finds that the requirements are consistent with the guidance.

As discussed in Section 10.2.8 of this SER, the reactor building contains several hot cells for conducting experiments involving radioactive materials. The hot cells have ventilation systems that interface with the containment building ventilation system. TS 3.7.3, "Reactor Floor Hot

Cells," specifies requirements for the hot cell ventilation systems, including an interlock that prevents operation of the hot cell ventilation unless the containment building ventilation system is operating. The requirements ensure that the hot cell ventilation systems will not exacerbate upset conditions in the hot cells involving airborne radioactivity or fire.

TS 4.5, "Ventilation Systems," specifies surveillance requirements for the ventilation system. The requirements include measuring the system flow rate annually, testing interlocks quarterly and before reactor startup, calibrating building differential pressure monitoring equipment annually, and monitoring filter performance. The requirements also include testing the interlock with the hot cell ventilation systems. The types and periodicity of the surveillance requirements are consistent with the guidance in ANSI/ANS-15.1 and the NRC staff finds them acceptable.

Based on the above discussion and findings, 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 features 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

The licensee's fuel handling operations include refueling and reconfiguring the core, transferring fuel between the core and the fuel storage ring, transferring fuel in and out of the reactor tank, transferring fuel into the storage pool, loading fuel into shipping containers, and receipt of new fuel elements. As part of refueling the reactor or reconfiguring the core, the licensee can flip fuel elements axially and cycle elements in and out of the storage ring to better control fuel burnup. Tools for handling of the MITR-II fuel assemblies are designed to preclude damage to the fuel during handling and have positive latching mechanisms to minimize the potential for fuel drops. The licensee uses a variety of shielded containers for fuel movements outside the reactor tank. According to the licensee, the designs of the shielded containers preclude loading enough fuel to achieve criticality under any conditions. TS 3.4 requires containment integrity during all fuel movements. This provides reasonable assurance that the containment building will be available to mitigate the consequences of any potential fuel handling accident. TS 5.4.4 permits movement of only one fuel element outside the core at a time and limits the power history of the element prior to removal from the reactor tank. The restriction on power history ensures that the element will not overheat during movement. TS 3.2.3 requires operating neutron flux detectors during fuel movement in the core to ensure detection of unanticipated reactivity changes.

Aside from the reactor core, irradiated fuel assemblies can be stored in analyzed and approved locations. TS 5.4, "Fissile Material Storage," specifies storage locations and conditions for fresh and irradiated fuel elements and fuel plates. TS 5.4.4 requires all storage locations, other than the reactor core, to have a k_{eff} of less than 0.90 to preclude inadvertent criticality. According to the licensee, fuel storage location designs include neutron poisons and/or geometry considerations to ensure criticality safety. TS 3.3.6 specifies limits on pH, conductivity, and chloride ion concentration for the water in the reactor tank and the fuel storage pool. The limits minimize corrosion of the fuel cladding. According to the licensee, operating experience has shown that the coolant chemistry limits are adequate to prevent significant corrosion.
TS 3.1.6.2 and TS 4.1.5 require annual fuel inspections using a variety of techniques. TS 4.3 requires quarterly surveillance of the reactor tank and storage pool water chemistry. The NRC

staff reviewed the TS requirements against the guidance in ANSI/ANS-15.1 and NUREG-1537 and finds them to be consistent with the guidance and acceptable.

Up to 11 irradiated fuel assemblies may be stored in the fission converter tank. This facility has its own cooling system and different safety limits and TS requirements than those covering the MITR-II reactor. TS 6.6, "Design and Operation of the Fission Converter Facility," specifies requirements related to the use of fuel elements in the fission converter. Amendment No. 31 to Facility Operating License No. R-37, dated December 21, 1999, authorized the licensee to operate the fission converter. The NRC's safety evaluation of the amendment request considered storage and use of irradiated fuel elements in the fission converter tank. The NRC staff concluded that there is reasonable assurance that storage and use of fuel elements in the fission converter would not pose an undue risk to public health and safety. The licensee requested no changes to the design or operation of the fission converter that would invalidate the NRC's prior safety conclusions regarding storage and use of fuel elements in the fission converter.

Based on the above, the NRC staff concludes that the fuel storage facility designs and TS requirements and fuel handling requirements provide reasonable assurance that adequate measures exist 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 MITR-II containment 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. The building contains automatic fire detection systems supplemented by manual pull boxes that the NRC staff observed throughout the facility. Fire extinguishers are located throughout the facility and are inspected on a regular basis. All fire detection and alarm systems tie into the reactor control room. This system annunciates in the control room and includes visual and audible alarms throughout the building. For fires beyond the incipient phase, local fire departments can provide assistance, if needed. According to the licensee, these departments tour the facility on a regular basis to ensure familiarity with the site. The licensee's EP, discussed further in Section 12.7 of this SER, includes measures to deal with fires.

According to the licensee, the reactor safety system would fail safe in the case of fire damage. This would de-energize the shim blade electromagnets and drop the shim blades into the core, shutting down the reactor. As discussed in Section 9.1 of this SER, the reactor hot cells have ventilation systems connected to the containment building ventilation system. TS 3.7.3 requires the hot cell ventilation systems to automatically isolate upon detection of a fire in the hot cells. This requirement minimizes the potential for a fire to spread outside the hot cell or for fire-generated airborne radioactivity to escape from the hot cells to the uncontrolled environment.

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

9.4 Communication Systems

Primary communication throughout the facility is by telephone. Telephones are located in the control room, throughout the reactor building, and at the medical therapy areas. All of these

telephones are supplied with emergency power, are part of the MIT telephone system, and can make and receive calls from within and outside of MIT. An intercom system provides communication between the control room and other locations throughout the reactor building and adjacent support buildings. The intercom system is supplied with emergency power. Using this system, the control room may communicate with any of the intercom units throughout the facility. Any of these remote units may communicate with the control room, but they may not communicate with other remote intercom units. Based on the above discussion, the NRC staff concludes that adequate communication systems are in place at the MITR-II 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 at the MITR-II are allowed primarily for materials research and radiation protection purposes. The renewed license authorizes the licensee to receive, possess, and use not more than 100,000 curies of byproduct materials with atomic numbers 3 through 83 in solid form. The NRC staff reviewed this authorization against the requirements in Section 274 of the AEA, “Cooperation with States,” and finds it acceptable because the byproduct materials are possessed and used only within the restricted area specified in renewed Facility Operating License No. R-37. Each sample is limited to 1,000 curies and an unshielded radiation flux of 1 Gray (100 rads) per hour at 1 m (3.3 ft). This material may be irradiated in the reactor in accordance with the provisions of TS 3.7.4, “Byproduct Material.” TS 3.7.4 also allows possession of byproduct material for calibration, characterization, and detection for radiation protection purposes. TS 7.4.3, “Scope of Procedures,” requires written procedures for the use, receipt, and transfer of byproduct material. The license authorizes possession and use of two plutonium-beryllium sources for reactor startup purposes, as well as the uranium contained in fission chambers. MIT has separate licenses for byproduct and special nuclear material not held on the reactor license.

The NRC inspection program verifies that the licensee properly uses and maintains procedures related to special nuclear material. The NRC staff most recently reviewed selected procedures related to special nuclear material during an inspection in April 2010 (NRC Inspection Report No. 50-020/2010-201, ADAMS Accession No. ML100920003). The inspection concluded that the licensee satisfied the procedural requirements. The NRC staff reviewed the possession limits in the renewed license and finds them consistent with the inventory requirements for reactor operations and the experiment program. As discussed in Section 12.8 of this SER, the licensee maintains an NRC-approved physical security plan.

9.6 Cover Gas Control Systems

The MITR-II has a helium cover gas system over the heavy-water reactor reflector. The system is designed to minimize contamination of the heavy water and lessen the corrosion that could be caused by air leaking into the system. The helium also provides an inert medium to circulate disassociation products to the recombiner for conversion of deuterium and oxygen to heavy water. The helium is supplied from a high-pressure manifold outside the containment building through reducers and isolation valves. The system is instrumented to alarm for both high-pressure and low-pressure conditions.

Carbon dioxide is used as a cover gas for the graphite reflector. The cover gas is supplied by the carbon dioxide system manifold from outside the containment building through an isolation valve. This system is instrumented to alarm for both high-pressure and low-pressure conditions.

The purpose of the cover gas system is keep air from leaking into the graphite reflector to minimize production of argon-41.

Based on this information, the NRC staff concludes that the cover gas control and processing systems are adequate to prevent the intrusion of ambient air into the reflector coolant systems and minimize radiation doses to facility personnel from the generation of argon-41.

9.7 Compressed Air System

The MITR-II has a compressed air system to supply air for various functions throughout the facility. The system has two compressors located outside the containment building. One compressor is used for containment building loads. The other compressor is used for laboratory loads and serves as a backup for the containment building compressor. The air system supplies the airlock gaskets and various isolation and control valves throughout the facility. The system includes various pressure gauges and moisture indicators to allow operations personnel to identify potential problems in the system. Additionally, an alarm in the control room alerts operators if the system pressure decreases significantly. According to the licensee, the loss of compressed air pressure would not prevent the safe shutdown of the reactor. A loss of air pressure causes the air-operated reflector dump valve to open, dumping the reflector and shutting down the reactor. According to the licensee, a high pressure air cylinder provides an emergency supply of compressed air for the airlock gaskets.

9.8 Conclusions

Based on the above discussions, the NRC staff concludes that the auxiliary systems at the MITR-II support the safe operation of the facility and aid in the safe shutdown of the reactor. Further, the NRC staff concludes the TS provide reasonable assurance that fuel elements will be appropriately handled and that there will be no undue risk to public health and safety, facility personnel, or the environment from the storage and movement of fuel.

10 EXPERIMENTAL FACILITIES AND PROGRAMS

10.1 Summary Description

The MITR-II supports educational and research activities at MIT. In addition to beam irradiation ports, the reactor has unique facilities for medical research involving neutron irradiation. Types of experiments include materials testing, neutron activation analysis, reactor control studies, boron neutron capture therapy, and neutron and reactor physics experiments. The TS provide limitations on experiments that ensure that experiments and experiment facilities will not interfere with safe reactor operation or shutdown. The TS also require administrative controls for the review and approval of experiments to ensure that the experiment program continues to meet the limitations in the TS.

10.2 Experimental Facilities

10.2.1 Fission Converter Medical Irradiation Room

The fission converter is an experiment facility located adjacent to the reactor that uses MITR-II fuel elements to produce a specialized neutron beam. Thermal neutrons from the MITR-II graphite reflector enter the fuel assemblies in the fission converter. The resulting fission neutrons are filtered to generate a beam of neutrons in the epithermal energy range that is directed to a shielded medical therapy room. The neutron beam is used for boron neutron capture therapy research, and the licensee is authorized to conduct trials on patients in accordance with TS 6.5, “Generation of Medical Therapy Facility Beams for Human Therapy.” MIT submitted a separate SAR for the fission converter as part of the application for authorization to use the fission converter, and the MITR-II SAR does not include a detailed discussion of this activity. TS 6.6, “Design and Operation of the Fission Converter Facility,” includes a complete set of TS covering design and operation of the fission converter facility. The NRC previously reviewed and approved the SAR and TS related to the fission converter in Amendment No. 31 to Facility Operating License No. R-37, dated December 21, 1999. Since that amendment, the NRC has issued one amendment authorizing several minor changes to the requirements in TS 6.6.

As part of the license renewal review, the NRC staff did not perform an indepth review of the fission converter design or operation. The NRC staff reviewed the potential effect of the license renewal and increase in reactor power to 6 MW(t) on fission converter design and operation. The NRC staff finds that license renewal should not significantly affect the fission converter because the licensee did not request any significant changes to the design or operation of the fission converter as part of the license renewal. The NRC staff finds that the increase in reactor power level will not have a significant effect on the fission converter design or operation because the NRC safety evaluation for Amendment No. 31 considered reactor power levels up to 10 MW(t). As part of license renewal, TS 6.5 and TS 6.6 were slightly modified for clarity and typographical errors. The NRC staff reviewed the modifications and finds that they do not alter the technical bases or technical content of the TS. Based on these findings, the NRC staff concludes that prior approval of the fission converter design, operation, and TS requirements remains valid. Section 16.2 of this SER contains additional discussion of the requirements of TS 6.5.

10.2.2 Basement Medical Irradiation Room

Section 10.2.2 of the SAR describes the design and construction of the basement medical irradiation room. The basement medical irradiation room is located directly beneath the reactor. The room is shielded by 1.4 m (4.5 ft) of concrete, except for a shielded and shuttered room observation window. Four different shutters are used to control the passage of neutrons from the reactor core to the therapy room. These shutters are the heavy-water blister tank, the water shutter tank, the boral shutter, and the lead shutter. Shielding in the room allows personnel to be in adjacent rooms during irradiations with average radiation levels of less than 0.005 mSv/hr (0.5 mrem/hr).

The control panel for the shutters is located outside of the therapy room and requires a key for operation. TS 6.5.3 requires a minor scram pushbutton on the control panel. TS 6.5.5 requires interlocks that prevent the beam shutters from being opened unless the medical irradiation room shield door is closed and cause closure of the beam shutters if the door is opened. These interlocks ensure that the proper personnel shielding is in place during irradiations and prevent inadvertent access to the room when the shutters are open. The shield door for the irradiation room is motor driven, but it can be manually opened in the event of power failure. Indication of door position is provided in the control room. Radiation detectors in the area are provided to alert personnel to elevated radiation levels.

10.2.3 Beam Ports

Section 10.2.3 of the SAR describes the design and construction of the beam ports. A total of 23 beam ports are available for experiments and sample irradiation. The main radial ports extend through the biological shield to the edge of the heavy-water reflector tank. Five of the radial ports are equipped with shutters. An additional six radial ports exist, but they are not equipped with shutters. Each port is plugged when not in use. Two semiradial ports are present and have a similar configuration to the radial ports, but they are offset from the center of the tank. One of these ports houses the end of the high-flux pneumatic tube. Six through ports are present, as well as four horizontal instrument ports. Beam catchers and alarming barriers are used to protect personnel from beam radiation. Purge gases are used in beam ports where needed to reduce the generation of argon-41.

10.2.4 High-Flux Pneumatic Tube

Section 10.2.4.1 of the SAR describes the design and construction of the high-flux pneumatic tube. A pneumatic transfer system is available to insert samples into the reflector tank reentrant thimble for irradiation. The sample containers are commonly known as “rabbits.” Samples irradiated in this system are exposed to a higher flux than in the other pneumatic tube system. The send/receive station for this system is in a shielded cell in the secondary chemistry area. The sample exit point is equipped with a radiation detector with a remote readout, so that radiation levels can be evaluated before handling a sample. The cell is equipped with remote manipulators to allow personnel to remotely handle samples.

10.2.5 Intermediate-Flux Pneumatic Tubes

Section 10.2.4.2 of the MITR-II SAR describes the design and construction of the intermediate-flux pneumatic tubes. These four additional pneumatic transfer tubes are available for sample irradiation. There are two chemistry areas in the basement area of the reactor. The secondary chemistry area is equipped with two manipulator arms and a leaded-glass window to protect

workers from sample radiation. Each chemistry area is equipped with two pneumatic tubes. All four of these tubes can send samples inside the graphite reflector region of the reactor for irradiation. Samples are injected or ejected from the irradiation location through the use of compressed air. Samples are received in a shielded ventilated cell in the chemistry area. An extension of the pneumatic system allows samples to be sent to a shielded area of the nuclear chemistry laboratory in the adjacent building. Samples being sent to the nuclear chemistry laboratory must first be measured for radiation before transfer interlocks are satisfied and the sample can continue to the laboratory. Carbon dioxide is used as fill in all of the pneumatic tubes when not in use to minimize the production of argon-41.

10.2.6 Graphite Reflector Irradiation Facilities

Section 10.2.6 of the SAR describes the design and construction of the graphite reflector irradiation facilities. Six vertical thimbles are located in the graphite reflector for sample irradiation. Should cooling be required, cooling water is available from the shield coolant system. Carbon dioxide is used to blanket these thimbles during operation to minimize the production of argon-41.

10.2.7 In-Core Sample Assemblies

Section 10.2.7 of the SAR describes the design and construction of the in-core sample assemblies. These are experiment locations that may occupy one or more fuel element positions in the reactor core. Up to four in-core sample assemblies may be present in the core and still meet core thermal-hydraulic and shutdown requirements. Each use of these assemblies requires an individual safety evaluation. TS 4.3.2 requires ex-core mock-up testing of new assemblies if the assembly has the potential to obstruct the emergency core cooling system. This ensures that the assembly will not impede proper operation of the cooling system. Each assembly consists of an aluminum thimble positioned inside an aluminum jacket with the same outside geometry as a standard fuel element.

TS 6.7, “Experiments Involving In-Core Irradiation of Fissile Materials,” allows the licensee to irradiate fissile materials in the reactor core. The NRC approved TS 6.7 by amendment dated April 16, 2003. As part of the license renewal review, the NRC staff did not perform an indepth review of TS 6.7. The NRC staff reviewed the potential effect of the license renewal and increase in reactor power to 6 MW(t) on the requirements of TS 6.7. The NRC staff finds that license renewal should not affect the radiological safety bases for fissile material experiments because the bases assume an experiment power level of 100 kW(t) required by TS 6.7.5 and a maximum fissile material loading required by TS 6.7.4. These requirements determine the potential radiological effects of experiment malfunction and are independent of reactor power. The NRC staff finds that TS 6.7 specifies other requirements, including design criteria and administrative controls, that are also independent of reactor power level. Based on these findings, the NRC staff concludes that prior approval of TS 6.7 remains valid.

10.2.8 Reactor Floor Hot Cell

Section 10.2.8 of the SAR describes the design and construction of the hot cell on the reactor floor. This hot cell is used for the storage and examination of highly radioactive samples. The cell has two subcells, each approximately 1.5 m (5 ft) wide by 1.2 m (4 ft) deep. The walls of the cell are a minimum of 46 cm (18 in.) of high-density concrete with a shielding factor of 1,250 for 2.0 megaelectron volt gamma rays. The hot cell is directly over the control room, but additional shielding in the floor provides a shielding factor of 5,000 to that location. Stepped,

leaded-glass windows with a substantially higher shielding factor are used for observation. The cells are kept at a negative pressure relative to the rest of the containment building. Previous sections of this SER discuss the ventilation systems and requirements of the hot cells. Access to the hot cells is through concrete plugs in the roof of the cells that require the use of the reactor crane to remove. The hot cells have radiation monitors both inside and outside to alert personnel to radiation levels.

10.2.9 Gamma Irradiation Facility

Section 10.2.9 of the SAR describes the design and construction of the gamma irradiation facility. The gamma irradiation facility is a steel-lined concrete tank used for storage of spent fuel and spent control blades. Gamma radiation from the spent elements can be used for sample irradiation. Water in the tank provides personnel shielding.

10.3 Experiment Review and Approval

TS 6.1, "General Experiment Criteria," lists the general criteria for experiment design. The requirements include limits on reactivity effects, thermal-hydraulic effects, chemical effects, radiolytic decomposition, and radioactive releases. TS 6.1.1 specifies limits on reactivity associated with moveable (0.5% $\Delta k/k$), nonsecured (1.0% $\Delta k/k$), and secured experiments (1.8% $\Delta k/k$). As discussed in Chapter 13 of this SER, the licensee analyzed a reactivity insertion accident that assumed failure of a secured experiment and determined that the failure would not cause fuel damage. TS 6.1.2 requires experiments to be able to withstand conditions in the reactor corresponding to the limiting safety system settings, not to cause boiling in the reactor coolant, and not to cause a significant redistribution of the reactor coolant flow. These requirements ensure that experiments are properly designed for the environment in the reactor and will not cause thermal-hydraulic effects that could invalidate the related assumptions in the analyses in the SAR. TS 6.1.3 specifies design requirements for encapsulation of metastable, explosive, and corrosive materials. The requirements include prototype testing and monitoring of capsules during irradiation. These requirements ensure that chemical effects will not cause damage to the reactor core components or reactor systems. TS 6.1.4 specifies encapsulation and venting requirements for experiments containing materials subject to radiolytic decomposition. The requirements ensure that capsules are designed to withstand internal pressure buildup and that vented material will not result in exceeding the limits in 10 CFR Part 20. Similarly, TS 6.1.7 requires all experiment designs to preclude exceeding the limits in 10 CFR Part 20 for doses to members of the general public and airborne effluent concentrations. This ensures that experiments and potential experiment malfunctions will not pose a significant radiological risk to public health and safety. TS 6.1.5 specifies provisions for adding scram functions to experiments to protect the experiment and reactor from potential experiment malfunctions. TS 6.1.6 requires prototype testing and conservative operation for experiments containing materials whose properties are uncertain. This requirement minimizes the potential for experiment malfunction due to unanticipated irradiation effects. The NRC staff reviewed the MITR-II general experiment criteria against TS requirements at other nonpower reactors and the guidance in ANSI/ANS-15.1 and NUREG-1537. The NRC staff finds that the requirements in TS 6.1 are comparable to those common for research reactors, are consistent with the guidance, are adequate to minimize the potential for experiment malfunctions, and are therefore acceptable.

TS 7.5.1, "Review Process," specifies the requirements and process for experiment review. Any use of experiment facilities at the MITR-II for irradiation or nonroutine use or operation of the reactor requires a written description, safety evaluation, and procedures. The experiment

proposals must describe the experiment in detail, the types of samples to be studied, and the potential risks. TS 7.5.1 requires a safety review of the written description, safety evaluation, and procedures in accordance with TS 7.4.1, "Review Process." Review requirements include consideration of whether the new experiment requires an amendment to the reactor license or could impact the ALARA program. TS 7.4.3 requires written procedures for all experiments that could affect reactor safety or core reactivity. Additional review criteria are employed for patient trials involving the medical therapy facilities. These review criteria, given in TS 6.5, include those of the affected hospital's NRC or State license, the MIT Committee on the Use of Humans as Experimental Subjects, the MIT Committee on Radiation Exposure to Human Subjects, hospital institutional review boards, and the U.S. Food and Drug Administration.

TS 7.5.2, "Approval Process," specifies the experiment approval process. All new experiments and substantive changes to previously-reviewed experiments must be approved in writing by two licensed senior reactor operators, the Director of Reactor Operations, and the MIT Reactor Safeguards Committee (MITRSC). The MITRSC is discussed in detail in Chapter 12 of this SER. As specified in TS 7.5.2.2, minor changes that do not significantly alter an experiment may be approved by two senior reactor operators and the Director of Reactor Operations. TS 7.5.2.3 allows a senior reactor operator to authorize performance of a previously approved experiment. According to the licensee, no experiments may be performed without the permission of the reactor operator. Section 10.3.4 of the SAR provides a general outline of the areas typically covered as part of experiment review and approval. The NRC staff reviewed the experiment review and approval process required by the TS and described in the SAR against the guidance in NUREG-1537, ANSI/ANS-15.1, and Regulatory Guide 2.2, "Development of Technical Specification for Experiments in Research Reactors," issued 1973. The NRC staff finds the licensee's experiment review and approval program to be consistent with the guidance and the TS limits on experiments and the related review criteria, along with the experiment review and approval process, provide reasonable assurance that any experiment performed at the MITR-II can be carried out safely without undue risk to public health and safety.

10.4 Conclusions

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

11 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

11.1 Radiation Protection

11.1.1 Radiation Sources

Section 11.1.1 of the SAR describes the radiation sources at the MITR-II. The fuel assemblies in the reactor are the primary source of radiation at the MITR-II. The reactor core is surrounded by heavy water, graphite, steel, and concrete, which reduce direct radiation doses to surrounding areas. Beam ports provide radiation from the reactor core to experiments and are controlled through shutters and/or local shielding and alarmed barriers. The reactor also has a pneumatic transfer system for in-core irradiation of experiments. Radioactive materials are created using this system by neutron activation. The MITR-II possesses a number of sources used for calibration and checks of radiation detection instruments. The sources range from microcurie-level mixed and single nuclide calibration standards to a cesium-137 source for high-range instrument calibration. As discussed in Section 9.5 of this SER, the licensee possesses two plutonium-beryllium startup sources and one antimony-beryllium source. The latter requires radioactive antimony and is normally not active. These sources are controlled by written procedures as required by TS 7.4, “Procedures,” and are periodically leak tested.

The airborne radioactive materials generated during reactor operation of principal concern are argon-41 and tritium. Argon is a natural component of the atmosphere and becomes activated to argon-41 upon neutron bombardment. The production of argon-41 is minimized by surrounding the core structure with a helium or carbon dioxide cover gas and conducting activities such as maintenance in ways that lessen air intrusion into volumes subjected to neutron flux. Tritium is generated by the neutron bombardment of deuterium in the heavy-water reflector and builds up during reactor operation. The reflector system is closed to the atmosphere and according to the licensee all flanges and pump seals are continuously monitored for leaks. The presence of significant quantities of airborne tritium would be an abnormal occurrence. Fission product gases could be generated from off-gassing of the fuel, although this is also an abnormal occurrence These gases would be collected under the reactor top and routed to the ventilation system for filtering before release through the stack. The NRC staff reviewed the information contained in the SAR against the guidance in NUREG-1537 and finds that the SAR contains sufficient information to provide a reasonable understanding of the airborne radiation sources at the MITR-II.

Licensee calculations of the airborne concentrations of radioactive materials are significantly less than 1 derived air concentration, and calculated occupational dose rates from argon-41 are significantly less than 0.001 mSv/hr (0.1 mrem/hr). The NRC staff reviewed the licensee’s calculations and finds them to be reasonable. Measured annual doses at rooftop monitors on nearby buildings are typically less than 0.005 mSv (0.5 mrem). This demonstrates compliance with the annual dose constraint on air emissions of radioactive material of 0.1 mSv (10 mrem) specified by 10 CFR 20.1101(d).

Liquid radiation sources at the MITR-II consist primarily of activation products in the primary coolant, principally nitrogen-16, sodium-24, and aluminum-28. Nitrogen-16 has a 7-second half-life and is only a radiation hazard during reactor operations or immediately after reactor shutdown. Aluminum-28 has a 2.25-minute half-life and also decays rapidly after reactor shutdown. Sodium-24 has a 14.96-hour half-life. The primary system is sealed and has a leak-detection system in use. Sodium-24 concentrations in the primary coolant range from 0.2 to

0.5 microcuries per milliliter as measured by the licensee. The primary system piping and equipment is located in heavily shielded areas with appropriate access controls. Tritium is generated in the heavy-water reflector. According to the license the equilibrium concentration of tritium for 6-MW operation is expected to be 5 curies per liter. The NRC staff reviewed the information contained in the SAR against the guidance in NUREG-1537 and finds that the SAR contains sufficient information to provide a reasonable understanding of the liquid radiation sources at the MITR-II.

MITR-II 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. Chapter 9 of this SER discusses spent fuel movement and storage. Other solid radioactive sources include ion exchange resins and filters, shielding plugs, and activated reactor components. Shims blades are also activated after time in the reactor. These are stored for decay after discharge from the core tank. Dummy elements and plugs for the experimental beam ports are also activated by MITR-II operations and are stored at various locations for decay after use. Other solid radioactive sources include material activated as part of an experiment. The licensee monitors the material for radiation upon removal from the reactor. The receiver station for the pneumatic transfer system has local lead shielding. Lead-lined holders are available for higher activity samples. Solid radioactive waste is disposed of in accordance with appropriate NRC regulations and is transferred to organizations authorized to receive the material. The NRC staff reviewed the information contained in the SAR against the guidance in NUREG-1537 and finds that the SAR contains sufficient information to provide a reasonable understanding of the solid radiation sources at the MITR-II.

Based on the above discussion and findings, the NRC staff concludes that the description and characterization of the radiation sources at the MITR-II are reasonable for a research reactor of this type and size and that this information provides sufficient detail to evaluate the radiation protection program and controls described in the remainder of Section 11.1 of this SER.

11.1.2 Radiation Protection Program

Section 11.1.2 of the SAR summarizes the radiation protection program required by 10 CFR 20.1101, “Radiation Protection Programs.” As stated in TS 7.3, “Radiation Safety,” the radiation program is designed to achieve the requirements of 10 CFR Part 20. This program includes the stated policy to employ the ALARA concept in all operations at the MITR-II. RRPO is responsible for administering the radiation protection program at the MITR-II. This group is part of the MIT Environmental Medical Service and is in a separate reporting chain from reactor operations through the Vice President level. The Reactor Radiation Protection Officer oversees RRPO activities and is responsible for implementing the radiation protection program at the MITR-II. RRPO responsibilities include calibration of survey instruments, effluent monitoring, radiation and contamination surveys, training, sample analysis, and personnel monitoring. TS 3.7.3 authorizes the Reactor Radiation Protection Officer to interdict or terminate activities that may compromise radiation safety.

According to the licensee, both operations and health physics personnel review procedures for implementing the radiation protection program before adoption. All individuals with access to the reactor restricted areas receive radiation safety training commensurate with their duties. Training is only required for those individuals likely to receive an occupational exposure in excess of 1 mSv (100 mrem) in a year. Escorted individuals, emergency responders, experimenters, and reactor operators receive different levels of training. Escorted individuals receive awareness training. Fire and police personnel are trained on facility layout, hazards,

and radiation safety and are given a facility tour. Further, police are trained on the use of portable detectors and participate in emergency drills and exercises. All experimenters and reactor operators receive training as specified by 10 CFR Part 19, "Notices, Instructions and Reports to Workers: Inspection and Investigations." Reactor operators and senior experimenters receive additional training, including topics such as ALARA, the use of instruments and protective clothing, and contamination control.

Records relating to the radiation exposure of facility personnel and others who enter radiation controlled areas, as well as effluent records, are retained for the life of the facility. This satisfies the records retention requirements specified by 10 CFR 20.2106, "Records of Individual Monitoring Results," and 10 CFR 20.2107, "Records of Dose to Individual Members of the Public." TS 7.8.1, "Five-Year Record Retention," requires retention of records of radiation and contamination surveys for a period of 5 years. This exceeds the 3-year retention period required by 10 CFR 20.2103, "Records of Surveys."

The NRC staff reviewed the structure and strategy of the MITR-II radiation protection program against the guidance in NUREG-1537 and ANSI/ANS-15.11, "Radiation Protection at Research Reactor Facilities," issued 2009. The NRC staff finds that the licensee's radiation protection program is consistent with the guidance. Based on this finding, the NRC staff concludes that the radiation protection program for the MITR-II as implemented is adequate to provide reasonable assurance that personnel are protected from radiation hazards.

11.1.3 ALARA Program

As described in Section 11.1.3 of the SAR, the licensee maintains an ALARA program as required by 10 CFR 20.1101(b) and TS 7.3.2. The Reactor Operations Office and RRPO administer it jointly. All procedure or equipment changes that have safety significance must be reviewed for ALARA considerations in accordance with TS 7.4. Maintenance outages must be preplanned and follow a written schedule. Both the Reactor Operations Office and RRPO review the schedule to identify activities that can be arranged so as to minimize radiation exposure to personnel. Opportunities to use temporary shielding are also identified. According to the licensee, planning meetings include dose minimization as a discussion item. The ALARA program is a fixed item of discussion for the meetings of the MITRSC. The TS contain requirements for proper consideration of the ALARA principle to ensure that ALARA is a principal concern in all MITR-II operations. The licensee's methods of reducing exposures are typical for ALARA programs at research reactors. Based on the above discussion the NRC staff concludes that the ALARA program provides reasonable assurance that personnel exposures will be minimized.

11.1.4 Radiation Monitoring and Surveying

RRPO maintains numerous fixed and portable radiation detection instruments throughout the MITR-II. Chapter 11 and Section 7.7 of the SAR discuss the instruments. Ten fixed gamma area radiation monitors are located throughout the containment building and at the stack to alert staff and operators to changing radiation conditions. Some of the area monitors are equipped with remote readout units that provide information to control room personnel. Other fixed radiation monitors are used to detect personnel contamination and include hand and foot monitors and portal monitors. These contamination monitors are located at the entrance to the containment building and at other locations as needed. Additional monitoring is performed on an as-needed basis to support nonroutine activities.

Portable instrumentation is available to survey areas in the MITR-II facility for all types of radiation and radioactive contamination that may be present from facility operations. This includes alpha, beta, gamma, and neutron survey instruments. Survey frequency is at least weekly during reactor operations and is increased as needed to monitor work activities. Air monitoring equipment including fixed and portable continuous air monitors are used in the facility. This includes particulate and tritium monitoring. Instruments are calibrated in accordance with the manufacturer's recommendations.

TS 3.7.1 specifies the required radiation monitoring equipment, including instrument setpoints and alarms. TS 3.7.1.1 requires an operable continuous air monitor with an alarm and data recording capability whenever the containment building is occupied. TS 3.7.1.2 requires a containment ventilation plenum monitor interlocked with the ventilation dampers, a stack effluent monitor, and measurement of the stack effluent tritium concentration whenever the containment is not isolated and containment integrity is required. TS 3.7.1.3 requires a core purge gas monitor capable of detecting fission products. TS 3.7.1.4 requires at least one operable area gamma monitor on the first floor of the reactor building any time the floor is occupied. TS 3.7.1.8 specifies the required setpoints for the radiation monitors. The setpoints are appropriately conservative to ensure that personnel will be alerted to abnormal radiation conditions in a timely manner. The NRC staff reviewed the radiation monitor requirements against the guidance in NUREG-1537 and ANSI/ANS-15.1 and finds them to be consistent with the guidance and appropriate to monitor expected radiation conditions at the MITR-II.

TS 4.5.2 requires testing of the plenum/damper interlock quarterly and following maintenance. TS 4.7.1.1 requires channel checks of the area and effluent monitors on any day the reactor is operating above 250 kW for more than 12 hours. According to the licensee, the power level and time limit are necessary to provide meaningful instrument indication to perform the channel checks. TS 4.7.1.2 requires monthly channel tests of the area radiation monitors using a radiation source and quarterly channel tests of the plenum, stack, and core purge monitors using an electrical pulse. TS 4.7.1.3 requires channel checks of the plenum, stack, and core purge monitors using a radiation source. TS 4.7.1.4 and TS 4.7.1.5 require initial and annual calibration and setpoint verification of the radiation monitors. The NRC staff reviewed the radiation monitor surveillance requirements against the guidance in NUREG-1537 and ANSI/ANS-15.1 and finds them to be consistent with the guidance and acceptable.

Based on its review, the NRC staff concludes that the installed and available radiation detection equipment is of the proper type, range, and sensitivity to detect and quantify the types of radiation at the MITR-II. Furthermore, the programs to use and maintain the equipment, as well as the frequency of surveys, satisfy the requirements of 10 CFR 20.1501(a) and (b) and provide reasonable assurance that doses to personnel will be kept below the limits specified in 10 CFR 20.1201

11.1.5 Radiation Exposure Control and Dosimetry

Only a small fraction of individuals working at the MITR-II meet the monitoring requirements of 10 CFR 20.1502, "Conditions Requiring Individual Monitoring of External and Internal Occupational Dose." However, most individuals entering the restricted area receive a monitoring device. Regular personnel are assigned an individual monitoring device for appropriate characterization of dose received from gamma, beta, and neutron radiation. A National Voluntary Laboratory Accreditation Program-certified vendor supply and process these monitors as required by 10 CFR 20.1501(c). Badges are processed on a quarterly basis and provide the dose of record. Extremity dosimeters are available if needed. This monitoring

program meets the requirements of 10 CFR 20.1502, and is consistent with the guidance of ANSI/ANS-15.11.

Radiation monitoring is supplemented with pocket ionization chambers to allow the estimation of personnel dose between badge readouts. Pocket ionization chambers are calibrated periodically and the results recorded at preset levels. Internal monitoring is not normally required at the MITR-II. Bioassay may include in vivo or in vitro measurements, or both, and is performed as needed for ongoing operations. Normal frequency of bioassay is baseline, annual, and termination.

Radiation exposure is controlled through the use of training, postings, and physical barriers to higher levels of radiation. Control of high- and very high-radiation areas is accomplished with local and remote audible alarms, controlled key access to locked high-radiation areas, and control devices to prevent unauthorized access to very high-radiation areas. The NRC staff reviewed the access controls and finds that they satisfy the requirements of 10 CFR 20.1601, "Control of Access to High Radiation Areas," and 10 CFR 20.1602, "Control of Access to Very High Radiation Areas."

The NRC staff evaluated the last five annual reports submitted to the NRC to gauge the historical effectiveness of the exposure control and dosimetry programs related to personnel and experimenter exposures. During the years reviewed, the average dose ranged from 0.318 mSv (31.8 mrem) to 0.549 mSv (54.9 mrem), and the maximum dose ranged from 7.5 mSv (750 mrem) to 22.5 mSv (2,250 mrem). These doses are below the occupational dose limit of 50 mSv (5,000 mrem) specified in 10 CFR 20.1201. The licensee does not expect the increase in reactor power level to cause a proportional increase in personnel doses. However, even with a 20-percent increase, personnel doses will remain well below the regulatory limit and the licensee's radiation protection program should continue to keep personnel doses ALARA.

Based on the above discussion, the NRC staff concludes that the exposure control and dosimetry program at the MITR-II is adequate to monitor and control exposures to personnel below the limits in 10 CFR Part 20, Subpart C, "Occupational Dose Limits."

11.1.6 Contamination Control

Contamination control at the MITR-II includes a combination of personnel monitoring and area surveys for contamination. Personnel contamination monitors, including hand-and-foot monitors and/or friskers are located at the containment building exits, restricted area exits, and the control room entrance. As noted in Section 11.1.4 of this SER, area surveys for contamination are performed at least weekly. According to the licensee, the frequency of contamination surveys depends on the day's scheduled work, and may be performed daily or more often. Any contamination found is isolated and decontaminated as soon as practicable. Confirmatory surveys after decontamination are then performed. For work involving potential contamination, preplanning and mockups, including engineering controls, are used. Personnel exiting contamination areas are required to monitor both themselves and any items removed from the area for contamination as soon as practicable. The licensee labels and bags or covers contaminated equipment and components. According to the licensee, the personnel and experimenter training program includes discussion of contamination risks and decontamination methods. TS 7.8.1 requires the retention of contamination survey records for 5 years. This exceeds the 3-year retention period required by 10 CFR 20.2103. The NRC staff reviewed the licensee's contamination control program described in the SAR and finds that the contamination control measures are common for research reactors similar to the MITR-II and appropriate to

detect and control contamination. Based on this finding, the NRC staff concludes that the licensee's contamination control program provides reasonable assurance that contamination at the MITR-II will not pose a significant risk to public health and safety, facility personnel, or the environment.

11.1.7 Environmental Monitoring

As described in Section 11.1.7 of the SAR, environmental monitoring at the MITR-II is accomplished through a combination of effluent monitoring and environmental thermo-luminescent dosimeters located outside the containment building. Direct real-time monitoring of the effluent is supplemented by the use of thin-walled Geiger-Mueller detectors at a quarter-mile radius from the reactor building. Each of these monitors is connected via dedicated telephone line to RRPO. Environmental thermo-luminescent dosimeters are placed and processed on a quarterly basis to confirm offsite radiation levels. The licensee reported that measured offsite doses are an average of 0.002 mSv (0.2 mrem) annually since 2003. The methodology for the collection and analysis of environmental samples is appropriate for determining compliance with the requirements in 10 CFR 20.1301, "Dose Limits for Individual Members of the Public." TS 3.7.1.7 requires at least one environmental monitor at the site and one within approximately 0.4 km (0.25 mi) of the site. TS 7.7.1, "Annual Report," requires the licensee to report the results of the environmental monitoring to the NRC annually. TS 7.8.3, "Life of Facility," requires the licensee to retain records of the environmental monitoring program for the life of the facility. The NRC staff reviewed the environmental monitoring program and TS requirements against the guidance in NUREG-1537, ANSI/ANS-15.1, ANSI/ANS-15.11, and the requirements of 10 CFR 20.1302, "Compliance with Dose Limits for Individual Members of the Public," and 10 CFR 20.2103. The NRC staff finds that the monitoring program and TS requirements are consistent with the guidance and satisfy the applicable regulatory requirements. Based on this finding, the NRC staff concludes that the licensee's environmental monitoring program provides reasonable assurance that radiation and radioactive effluents from the facility will be detected, recorded, and reported in a manner that protects public health and safety and the environment.

11.2 Radioactive Waste Management

11.2.1 Radioactive Waste Management Program

According to the licensee, the principal objective of the radioactive waste management program is to minimize the creation of radioactive waste. Materials that can be decontaminated or reused are not classified as waste while they still have potential use. The Reactor Operations Office is primarily responsible for the MITR-II radioactive waste management program, with guidance and assistance from RRPO. The ALARA principle applies to all aspects of the program, with emphasis on the minimization of waste generation. According to the licensee, all personnel that work in or frequent the restricted area receive training on waste minimization practices. Waste generated by experimental groups is the responsibility of the principal investigator for that experimental group. TS 7.8.3 requires the retention of records of effluent releases and radioactive material shipments for the life of the facility. This meets the requirement of 10 CFR 20.2108, "Records of Waste Disposal," for records retention. The radioactive waste management program includes periodic audits of the program to assess its status. A quarterly audit by the Reactor Operations Office includes all reactor activities, including radioactive waste management. The NRC staff reviewed the radioactive waste management program presented in the SAR and finds it to be comparable in scope to programs successfully implemented at similar research reactors. The NRC staff finds the program to be consistent with the guidance in ANSI/ANS-15.1 and NUREG-1537. Based on these findings

and compliance with the applicable regulatory requirements, the NRC staff concludes that the program contains appropriate provisions for training, review, and oversight that are commensurate with the types and quantities of radioactive wastes expected as a result of facility operations. Further, the NRC staff concludes that the program provides reasonable assurance that the licensee's management of radioactive waste will not pose an undue risk to public health and safety, facility personnel, or the environment.

11.2.2 Radioactive Waste Controls

The licensee controls gaseous radioactive wastes (gaseous effluents) by minimizing the production of the waste products and discharging the effluents to the unrestricted area. The generation of argon-41 is minimized through the use of cover gases and sealed primary water systems. Gaseous radioactive effluent is filtered through HEPA filters and monitored before release through the containment building stack. The licensee calculated a routine air effluent dilution factor of at least 50,000 that accounted for the stack height and nearby buildings. The NRC staff reviewed the parameters used in the licensee's calculation and finds them reasonable and consistent with atmospheric dispersion methodology. The licensee uses the dilution factor for determining compliance with the air effluent concentration limits specified in Appendix B to 10 CFR Part 20. TS 3.7.2 specifies the dilution factor and a dose scaling factor of 1,200 for particulates and iodine with half lives greater than 8 days. According to the licensee, the dose scaling factor is based on a comparison of iodine exposure pathways to submersion dose from noble gasses. TS 3.7.1 requires monitoring of the stack effluent. The NRC staff reviewed the requirements related to gaseous waste controls and finds them to be consistent with the guidance in ANSI/ANS-15.1 and NUREG-1537. As previously stated in this SER, gaseous releases from the MITR-II result in potential offsite doses that are a small fraction of public dose limits specified in 10 CFR 20.1301.

Liquid wastes at the MITR-II are collected in two waste storage tanks or in dedicated containers near the point of waste generation. The storage tanks are located above ground and outside of the reactor building. Each is located on a concrete pad with curbing that can contain the entire tank contents. The tanks are also equipped with a leak-detection capability. Tank contents are circulated and sampled to ensure concentrations meet the requirements of 10 CFR 20.2003, "Disposal by Release into Sanitary Sewerage," prior to release to the sanitary sewer system. The final discharge valve is normally locked to prevent inadvertent discharge. TS 3.7.1.8 requires radiation monitoring of the effluent during discharge and an interlock that terminates the discharge upon detection of abnormal radiation levels. TS 4.7.1 requires quarterly channel tests and annual calibration of the monitor. The NRC staff reviewed the liquid effluent discharge requirements and finds them consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1 and acceptable. As described in Section 4.4.4 of the SAR, the MITR-II is designed to preclude the possibility of soil or ground water contamination. The NRC staff reviewed the design features and finds them to be adequate to minimize the potential for soil or ground water contamination.

TS 3.3.5 requires the periodic sampling of primary coolant to detect any degradation of fuel integrity that could lead to increased levels of waste and effluents. This sampling, along with required actions for elevated radioactivity in primary cooling, provides reasonable assurance that the degradation of fuel integrity will be detected promptly and allow for timely corrective and mitigative actions. TS 3.7.1.5 and TS 3.7.1.6 specify requirements related to monitoring the secondary coolant for radioactivity. In the case of a leak in a reflector system heat exchanger, the secondary coolant could become contaminated with tritium. TS 3.7.1.5 requires daily sampling or continuous monitoring of the secondary coolant for tritium whenever secondary

coolant is flowing through a reflector system heat exchanger. This requirement ensures the licensee can detect tritium in a timely manner and take corrective actions required by TS 3.7.2 to minimize the release of contaminated water to the environment. TS 3.7.1.6 requires a secondary coolant water monitor with an alarm in the control room whenever the reactor is operating and secondary coolant is circulating to the cooling tower. This ensures the reactor operator will be alerted to abnormal radioactivity in the secondary coolant that could result from a leak between the secondary system and the primary or reflector systems. TS 4.7.1 requires monthly (electrical pulse) and quarterly (source test) channel tests and annual calibrations of the secondary coolant monitoring equipment. The NRC staff reviewed the primary and secondary coolant monitoring systems and requirements against the guidance in ANSI/ANS-15.1, ANSI/ANS-15.11, and NUREG-1537. The NRC staff finds that the systems and requirements are consistent with the guidance and appropriate to detect abnormal radioactivity in the systems in a timely manner and minimize the potential for inadvertent releases of liquid radioactive wastes.

The licensee classifies solid low level radioactive wastes generated at the MITR-II as either wet or dry waste. Wet waste includes filters and ion exchange resins. Dry waste includes ventilation filters and contaminated materials such as paper, cloth, metals, and other items used for routine facility operations. Solid waste may also include reactor components and experiment materials. Solid waste management is divided into four processes: collection, pretreatment, solidification, and packing. According to the licensee, volume reduction methodologies are applied to all processes and solid wastes are stored onsite for decay. After solid waste is processed, it is sent to a designated waste facility in accordance with all applicable regulations.

Based on the above discussion and findings, the NRC staff concludes that the licensee's radioactive waste controls are appropriate to classify, store, and prevent unmonitored releases of radioactive wastes generated at the MITR-II. Further, the NRC staff concludes that there is reasonable assurance that the licensee's radioactive waste controls provide adequate protection of public health and safety and the environment.

11.2.3 Release of Radioactive Waste

The licensee releases gaseous and liquid wastes as effluents as permitted by 10 CFR 20.2001, "General Requirements." Gaseous releases are released through the containment building stack. These effluents first pass through a roughing filter and then a HEPA filter. As previously discussed, liquid wastes are discharged to the sanitary sewer after sufficient sampling and monitoring to ensure all releases meet the applicable regulatory requirements. The licensee monitors and records effluent measurements of all releases. TS 7.8.3 requires retention of records of effluents for the life of the facility. This satisfies the requirement of 10 CFR 20.2108. TS 3.7.2 requires effluent concentrations to meet the requirements of Appendix B to 10 CFR Part 20. TS 4.7.2, "Effluents," requires continuous monitoring of all effluents and collection of effluent information sufficient to fulfill the annual reporting requirements. 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 applicable NRC regulations and the applicable licenses of the recipient.

Based on above information and the licensee's compliance with the applicable regulatory requirements, the NRC staff concludes that administrative controls, radioactive release methods, and TS requirements provide reasonable assurance that the release of radioactive waste from the MITR-II will not pose an undue risk to public health and safety or the environment.

11.3 Conclusions

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

12 CONDUCT OF OPERATIONS

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

12.1 Organization

Section 12.1 of the SAR describes the licensee's organizational structure. Figure 7.1-1 of TS 7.1, "Organization," specifies the management structure including reporting and communication lines. The management structure consists of three separate reporting lines that all ultimately report to the President of MIT. The three reporting lines are the radiation protection organization, the operations organization, and the MITRSC. The radiation protection and operations organizations have independent reporting lines to minimize the potential for a conflict of interest. Lines of communication exist between the three reporting lines, including communication lines between the Reactor Radiation Protection Officer and the Director of Reactor Operations and Superintendent of Reactor Operations and Maintenance. According to the licensee, the RRPO is located at the reactor site and works closely with the operations organization. TS 7.3.3 authorizes the Reactor Radiation Protection Officer to interdict or terminate activities that may compromise radiation safety. Communication lines also exist between the MITRSC and the Director of Reactor Operations and Superintendent of Reactor Operations and Maintenance. TS 7.1.2, "Responsibility," specifies responsibilities of the Director of Reactor Operations and the Reactor Radiation Protection Officer and permits delegation of authority contingent upon appropriate qualifications. The NRC staff reviewed the TS requirements against the guidance in ANSI/ANS-15.1 and finds them to be consistent and acceptable.

TS 7.1.3, "Staffing," specifies minimum staffing requirements. These require at least one senior reactor operator as well as an additional licensed operator when the reactor is not shut down. At least one of the operators must be in the control room. When the reactor is not secured, at least one senior licensed operator as well as an additional person must be on site, with a licensed operator in the control room. In addition, a representative of the Radiation Protection Office must be either on site or on call. TS 7.1.3.3 requires a list of facility personnel by name and telephone number be available in the control room. The NRC staff reviewed the staffing requirements against the guidance in ANSI/ANS-15.1 and finds them to be consistent and acceptable. TS 7.1.4, "Selection of Personnel," specifies the minimum educational and/or experience requirements for the Director of Reactor Operations, the Superintendent of Reactor Operations and Maintenance, shift supervisors, and reactor operators. TS 7.1.5, "Training of Personnel," requires the licensee to maintain a training program for initial certification and requalification. Section 12.10 of this SER discusses the licensee's training and requalification program. The NRC staff reviewed the requirements against the guidance in ANSI/ANS-15.4, "Selection and Training of Personnel for Research Reactors," and finds them to be consistent and acceptable.

Based on the above discussion and findings, the NRC staff concludes that the licensee's management organization, staffing, and selection and training of personnel provide reasonable assurance that the licensee can continue to operate the facility without undue risk to public health and safety, facility personnel, or the environment.

12.2 Review and Audit Activities

TS 7.2, "Review and Audit," specifies the requirements for the MITRSC and its review and audit functions. The MITRSC is responsible for review and audit functions at the MITR-II. TS 7.2 details the requirements for the MITRSC. MITRSC approval is necessary for all new operating plans and policies and all significant modifications thereto that may involve questions of nuclear safety. The MITRSC is also responsible for auditing operation of the reactor as detailed in TS 7.2.3, "MITRSC Audit Function." The Chairman of the MITRSC reports directly to the President of MIT.

The MITRSC is composed of a minimum of nine persons with not more than one-third of the total membership chosen from the reactor staff organization and a minimum of three members from outside MIT. The President of MIT selects all members and the Chairman. TS 7.2.1.1 details specific educational and experience requirements for the committee. The MIT Radiation Protection Officer and the MIT Environment, Health, and Safety Officer are included as ex-officio members of MTRSC. TS 7.2.1.2 specifies the MITRSC charter and rules governing meeting frequency, quorum composition, use and membership of subcommittees, preparation of meeting minutes, and access to reactor records.

TS 7.2.2 specifies the MITRSC's review function. The MITRSC reviews violations, reportable occurrences, audit reports, new procedures and experiments, license amendments, and changes made under 10 CFR 50.59, including changes to safety-significant reactor facility equipment or systems. The review findings are documented in the MITRSC meeting minutes. TS 7.2.3 specifies the MITRSC's audit function. TS 7.2.3.1 allows MITRSC members without line organization responsibility and qualified consultants to perform audits. TS 7.2.3.2 requires annual audits, and TS 7.2.3.3 specifies the scope of the audit program. TS 7.2.3 requires written audit reports and immediate reporting of deficiencies to the Director of the Nuclear Reactor Laboratory.

Nuclear medicine activities have additional oversight outside of reactor operations. In addition to the oversight provided by the MITRSC, MIT committees with responsibility include the Committee on the Use of Humans as Experimental Subjects and the Committee on Radiation Exposure to Human Subjects. The chairmen of both entities report directly to the President of MIT.

The NRC staff reviewed the structure and conduct of review and audit activities described in the SAR and required by the TS and finds they are consistent with the guidance of ANSI/ANS-15.1. Based on this finding, the NRC staff concludes that the review and audit functions are sufficient to ensure that safety-related matters will be appropriately reviewed.

12.3 Procedures

Written, approved procedures govern all aspects of the reactor facility's operation and use. TS 7.4 requires written procedures include, but are not limited to, the following areas:

- Startup, shutdown, and operation of the reactor.
- Refueling operations.
- Maintenance of components that have nuclear safety significance.
- Surveillance and testing as required by these technical specifications.

- Personnel radiation protection consistent with applicable regulations or guidelines. These procedures shall include a management commitment and programs to maintain experiments and releases in accordance with the ALARA concept.
- Administrative controls for operation and maintenance and for the conduct of irradiations and experiments that could affect reactor safety or core reactivity.
- Implementation of required plans such as emergency or security plans.
- Use, receipt, and transfer of byproduct material.

Changes to procedures go through a written safety review process. The process includes a description, safety evaluation, and evaluation for impacts on emergency and security planning, the ALARA program, the reactor license, and the requalification program. The review process includes a determination that the change or new procedure meets the requirements of 10 CFR 50.59. The approval process requires at a minimum the signatures of two senior operators, the Reactor Radiation Protection Officer (if radiation is involved), and the Director of Reactor Operations. New procedures and modifications to existing procedures having safety significance also require MITRSC review and approval. The MITRSC is informed of all other procedures.

The NRC staff reviewed the procedure scope and review and approval process described in the SAR and required by the TS and finds they are consistent with the guidance of ANSI/ANS-15.1. Based on this finding, the NRC staff concludes that the process and methodology described in the SAR and required by the TS provide reasonable assurance that procedures will be properly controlled and reviewed.

12.4 Required Actions

Certain events require specific licensee actions in accordance with the SAR and requirements of TS 7.6, “Required Action.” TS 2.1 specifies safety limits for different operational modes. In the event of a safety limit violation, the MITR-II would be shut down and the senior management of the facility notified. TS 7.6.1, “Action To Be Taken in Case of Safety Limit Violation,” requires notification of the NRC, investigation of the event, and preparation of a safety limit violation report. The reactor would not be restarted without prior NRC approval. TS 1.3.32, “Reportable Occurrence,” defines reportable events. TS 7.6.2, “Action To Be Taken in the Event of a Reportable Occurrence,” lists the required licensee actions. TS 7.7.2, “Reportable Occurrence Reports,” specifies the requirements for reporting to the NRC.

The NRC staff reviewed the requirements of TS 7.6 and finds that they are consistent with ANSI/ANS-15.1 and satisfy the requirements of 10 CFR 50.36, “Technical Specifications.” Based on these findings, the NRC staff concludes that the TS requirements provide reasonable assurance that the facility will respond to unanticipated occurrences in a manner that emphasizes reactor safety and the protection of public health and safety, facility personnel, and the environment.

12.5 Reports

TS 7.7, “Reports,” specifies reports that the licensee is required to make to the NRC. These include an annual operating report and special reports. TS 7.7.1 lists the required contents of the annual operating report, including operational history, major maintenance performed, approved major changes to the facility, and radioactive effluents. TS 7.7.2 discusses how to file

special reports for a safety limit violation, a release of radioactivity from the site above allowed limits, and a reportable occurrence. TS 7.7.3, "Special Reports," requires written special reports for permanent changes in facility management at levels 1 and 2, and for significant changes in the transient or accident analysis as described in the SAR.

The NRC staff evaluated these reporting requirements and finds 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

TS 7.8, "Records Retention," specifies records retention requirements, including records scope and retention periods. According to the licensee, recordkeeping is the responsibility of the Quality Assurance Supervisor, who reports directly to the Director of Reactor Operations. Records may be in the form of logs, data sheets, or other suitable forms, including electronic data storage. When the latter is used, a capability to read the storage medium is also maintained. The licensee specified the types of records that the facility will retain and the period of retention, to ensure that important records will be retained for an appropriate time. The NRC staff evaluated the requirements of TS 7.8 and finds that they are consistent with the guidance in ANSI/ANS-15.1 and applicable regulatory requirements. Based on this finding, the NRC staff concludes that the TS requirements provide reasonable assurance that the licensee will maintain appropriate records to facilitate NRC inspection, including an adequate history of the facility.

12.7 Emergency Planning

The NRC staff reviewed the MITR-II 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, and the requirements of 10 CFR 50.34(b)(6)(v). The NRC staff concluded that the MITR-II 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 MITR-II EP provides reasonable assurance that the licensee can respond appropriately to a variety of emergency situations and that the MITR-II EP will be adequately maintained during the period of the renewed license.

12.8 Security Planning

The NRC staff reviewed the licensee's measures for physical security and protection of special nuclear material against the applicable requirements of 10 CFR Part 73, "Physical Protection of Plants and Materials." Additionally, the staff reviewed the licensee's measures using the guidance contained in Regulatory Guide 5.59, "Standard Format and Content for a Licensee Physical Security Plan for the Protection of Special Nuclear Material of Moderate or Low Strategic Significance," issued February 1983. The NRC staff finds that the licensee's measures for physical security and protection of special nuclear material meet the intent of the guidance and satisfy the applicable regulatory requirements. Additionally, the NRC routinely inspects the licensee's measures for physical security and protection of special nuclear material

to verify that the licensee continues to satisfy the applicable regulatory requirements. In 2009, an NRC inspection verified that the licensee's security measures continued to satisfy applicable regulatory requirements and were acceptable. Based on the NRC staff review and routine inspection program, the NRC staff concludes that the licensee's measures for physical security and protection of special nuclear material provide reasonable assurance that continued possession and use of licensed special nuclear material at the facility will not pose an undue risk to public health and safety and will not be inimical to the common defense and security.

12.9 Quality Assurance

Section 12.9 of the SAR describes the quality assurance program in place at the MITR-II. According to the licensee, the program covers procedures and equipment important to safety. Changes to existing procedures and the creation of new procedures undergo a safety review by operations personnel or the MITRSC, depending on the nature of the procedure. According to the licensee, procurement of new equipment and upgrades to existing equipment are subject to a quality assurance review. The review includes completion of a quality assurance form that contains program requirements specific to the type of equipment modification or procurement. The form also specifies who is qualified to verify that each quality assurance requirement has been met. Two individuals are required to verify that each requirement is satisfied. The quality assurance program includes independent checks, reviews by the MITRSC, quarterly audits by the quality assurance supervisor, and annual spot-check audits by independent auditors. The licensee maintains quality assurance records for at least 5 years.

12.10 Operator Training and Requalification Program

As required by TS 7.1.5 responsibility for the administration of the requalification program rests with the Superintendent of Operations and Maintenance and/or the Director of Reactor Operations. The NRC staff reviewed the program and finds that it satisfies 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. Based on this finding, the NRC staff concludes that the licensee's requalification program and training program provide reasonable assurance that the licensee will have technically qualified reactor operators and senior reactor operators.

12.11 Startup Plan

The licensee requested an increase in the maximum licensed reactor power level from 5 MW(t) to 6 MW(t) as part of the license renewal application. The licensee provided protocol for the initial increase in reactor power to 6 MW(t). The protocol involves verifying all reactor systems, including heat removal, instrumentation, and shielding are operable for reactor operation at 6 MW(t). The licensee intends to establish equilibrium conditions at 5 MW(t) before beginning the power ascension. The licensee intends to raise reactor power to 5.5 MW(t), verify adequate reactor system performance, establish equilibrium conditions, and re-verify adequate reactor system performance. The licensee will repeat this process to raise power to 6.0 MW(t). In the case of any observed inadequacies or unanticipated instrument response, the licensee will reduce reactor power to the previous equilibrium level. The NRC staff reviewed the power ascension protocol against the guidance in NUREG-1537. The NRC staff finds that the protocol is consistent with the guidance because it includes verification of adequacy of shielding and provisions for testing instrumentation and heat removal system performance prior to beginning routine operations at 6 MW(t). Additionally, the protocol requires the reactor to be in a stable, safe condition prior to the power increases. Based on these findings, the NRC staff concludes

that the power ascension protocol provides reasonable assurance that the licensee will increase the reactor power in a conservative manner.

12.12 Conclusions

Based on the above discussion, the NRC staff concludes that the MITR-II has the appropriate organization, experience levels, and adequate controls through the TS to provide reasonable assurance that the MITR-II 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 establish safety limits, LSSSs, LCOs, and design features for the MITR-II, the licensee analyzed potential reactor transients and other hypothetical accidents for the effects of such events on the reactor fuel and public health and safety, facility personnel, 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 blockage of fuel element flow channels, leading to complete melting of several fuel plates. This event would lead to the maximum potential radiation hazard to facility personnel and members of the public. The licensee evaluated only the potential consequences of this event and not the likelihood of its occurrence. This worst-case accident scenario has been designated as the MHA. The licensee evaluated other possible accident scenarios, and none pose a significant risk of cladding failure.

The accidents considered for evaluation and analysis are as follows:

- MHA (fuel channel blockage and fuel plate melting)
- insertion of excess reactivity
- loss of primary coolant
- loss of primary coolant flow
- external events
- experiment malfunction
- equipment malfunction
- loss of normal electrical power
- mishandling of fuel

In all of the accident scenarios, the reactor is assumed to be operating with all critical parameters at the most limiting values authorized by the TS. This ensures that the analysis for each accident scenario uses the worst-case initial conditions and that the TS define an envelope of safe reactor operation.

13.1 Maximum Hypothetical Accident

The MHA for the MITR-II is the complete melting of four fuel plates caused by a coolant flow blockage in a fuel element. The blockage is assumed to be caused by a foreign object dropped into the core tank that becomes lodged against the underside of a fuel element. Fission products are assumed to be released to the coolant and containment based on reasonable release fractions and evaporation calculations. The release to the containment conservatively assumes that the reactor top shield is not in place, although it is required to be in place if the reactor power exceeds 100 kW. The NRC staff finds that this is an extremely conservative assumption because presence of the top shield lid would greatly reduce the rate of release of fission products from the primary coolant to the reactor containment. Upon detection of fission products, radiation monitors in the exhaust plenum would automatically isolate the containment building and activate an alarm in the control room (prompting the reactor operator to shut down the reactor), as required by TS 3.7.1.

The licensee calculated the maximum effective doses to the public, including contributions from containment leakage and direct radiation. The containment leakage dose assumes a containment building leak rate consistent with the maximum leak rate allowed by TS 3.4.3.

Dispersion dose estimates are made using reasonable assumptions based on guidance in Regulatory Guide 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants,” Revision 1, issued November 1982. The direct radiation doses account for direct doses from the building airlocks (which do not have the same concrete shielding as the rest of the containment) and scattered dose from the containment dome. The licensee assumed a 2-hour exposure time for members of the public, which assumes that the EP for the protection of the public will be implemented in less than 2 hours. This is reasonable for moving people away from the site boundary. The licensee calculated a maximum whole body dose to a member of the public in the unrestricted area of 3 mSv (300 mrem), and a maximum committed effective dose equivalent to the thyroid 0.041 mSv (4.1 mrem). These doses result in a total effective dose equivalent (TEDE) of approximately 3.04 mSv (304 mrem). The NRC staff performed independent calculations of the doses to the public from the MHA and obtained results that were similar to those presented by the licensee.

Although the TEDE is not below the annual limit of 1 mSv (100 mrem) usually used for nonpower reactors (as discussed in NUREG-1537), the TEDE is below the annual limit of 5 mSv (500 mrem) specified in 10 CFR 20.1301(d). To satisfy the requirement of 10 CFR 20.1301(d)(1) the licensee demonstrated the need for the higher dose limit by showing that a worst-case, hypothetical situation could potentially cause a dose greater than 1 mSv (100 mrem). As discussed elsewhere in this SER, the TS requirements ensure that the licensee can assess potential doses to the public, and the licensee maintains an approved emergency plan that includes provisions for controlling dose to members of the public. This satisfies the requirement of 10 CFR 20.1301(d)(2). TS 7.4.3.e requires the licensee to maintain procedures for keeping radiation exposure ALARA. This satisfies the requirement of 10 CFR 20.1301(d)(3). Based on the licensee satisfying the requirements in 10 CFR 20.1301(d), the NRC staff finds the higher annual dose limit for members of the public to be acceptable in the case of the MHA. However, based on its review of the TS requirements and administrative controls, the NRC staff finds that fuel damage comparable to the MHA is extremely unlikely and there is reasonable assurance that doses to the public from the MITR-II will not exceed the lower dose limit of 1 mSv (100 mrem) during the period of the renewed license.

The licensee estimated both submersion and inhalation doses to reactor staff from the fission product release to the containment. The licensee conservatively assumed the worst-case fission product inventory, no barriers to release to the containment atmosphere (no top shield in place), and no airborne radioactivity removal mechanisms (no plateout or deposition). These assumptions are very conservative because the top shield lid would minimize the release of radioactivity from the primary coolant system to the containment atmosphere and plateout and deposition would reduce inhalation of airborne radioactive material. Using these very conservative assumptions, the licensee calculated a maximum stay time of approximately 20 minutes without exceeding the 25-rem emergency protective action guideline. This corresponds to a dose of approximately 50 mSv (5,000 mrem), the occupational dose limit specified in 10 CFR 20.1201, in 4 minutes. Doses to facility personnel would be controlled by evacuation of the containment building and use of proper protective equipment and radiation protection procedures. The NRC staff considers that for the more reasonable scenario with the top shield in place, as required by TS 3.1.4, the release of fission products to the containment would be a very small fraction of that assumed by the licensee.

The NRC staff reviewed the MHA for the MITR-II against the guidance in NUREG-1537. The NRC staff finds that the analysis is consistent with the recommendations of the guidance and uses extremely conservative assumptions. The NRC staff finds the scenario, source term, release fractions, dispersion analyses, and dose estimates to be very conservative. The NRC

staff finds that the licensee's dose estimates for members of the public in the unrestricted area are below regulatory limits. The NRC staff finds that the licensee's calculations of submersion and inhalation doses and stay time for facility personnel are extremely conservative. Additionally, the NRC staff finds that the MHA is extremely unlikely to occur. Based on these findings, the NRC staff concludes that the MHA will not pose an undue risk to public health and safety, facility personnel, or the environment. Based on the discussions in the remainder of this chapter of this SER, the NRC staff concludes that the potential consequences of the MHA bound all other accident scenarios.

13.2 Insertion of Excess Reactivity

13.2.1 Step Reactivity Insertion Accident

The licensee analyzed a reactivity insertion of 1.8% Δk/k ($2.30) over a period of 0.5 seconds using the PARET computer code. The reactor conditions for the event were an initial operating power of 6 MW, flow conditions at the LSSS limits of 114 l/s (1,800 gpm), a power scram at 7.4 MW, and a period scram at 7 seconds, each with a 0.1-second delay and control rod insertion time of 1 second to 80-percent insertion. These assumptions are consistent with the requirements in the TS and the design of the RPS. The calculated peak cladding temperature was 83.4 degrees C (182 degrees F), which is well below the safety limit of 450 degrees C (842 degrees F). The licensee also performed parametric analysis by varying inputs for reactor parameters (flow, power, channel width, power peaking, moderator void coefficient, and moderator temperature coefficient) by up to 5 percent in the most conservative direction. According to the licensee, an uncertainty of 5 percent for any of the parameters is conservative based on instrument performance and manufacturing tolerances. The licensee determined that for the worst case uncertainty of 5 percent overpower, the cladding temperature would increase an additional 2.2 degrees C (4 degrees F), which results in a cladding temperature well below the safety limit.

The NRC staff reviewed the licensee's PARET inputs and finds that the parameters are consistent with the description of the analysis in the SAR and the requirements of the TS. Further, the NRC staff finds that the licensee used the PARET code for transient regimes within existing benchmarks for the code. The NRC staff reviewed the licensee's analysis against transient analyses for similar nonpower reactors and finds that the licensee's analysis is consistent. Based on the above discussion and findings, the NRC staff concludes that there is reasonable assurance that the design of the RPS and the TS requirements will protect the cladding temperature safety limit and preclude a loss of fuel integrity due to a rapid insertion of positive reactivity.

13.2.2 Ramp Reactivity Insertion Accident

The licensee evaluated a postulated ramp reactivity insertion accident assuming the limiting reactivity insertion rate of $5x10^{-4}$ Δk/k/s specified by TS 3.2.2. The licensee did not postulate the means of reactivity insertion, but the NRC staff finds that the use of the TS limit is appropriate for the analysis. The transient is assumed to be terminated by either the reactor power or period scram at the LSSSs required by TS 2.2. This is a conservative assumption because TS 3.2.3 requires scram setpoints to be more conservatively than the corresponding LSSSs. The licensee assumed an instrument delay time of 1 second following the initiation of the scram signal, followed by a control blade insertion rate of 1 dollar per second. These control system response assumptions are conservative relative to the minimum 1% Δk/k shutdown margin required by TS 3.1.2 and the maximum 1 second to 80-percent insertion time for each control

blade required by TS 3.2.1. For a worst case initial reactor power of 5.3 MW, both the high-power and short-period scrams occur approximately at the same time, resulting in a calculated peak power of 8.9 MW. This peak power is below the safety limit on reactor power of 9.1 MW for the LSSSs for core flow rate of 1,800 gpm and core outlet temperature of 60 degrees C (140 degrees F). The NRC staff reviewed the analysis against the guidance in NUREG-1537 and finds that the analysis is consistent with the guidance because the licensee used initial conditions and RPS response consistent with the TS requirements. Based on this finding and the results of the licensee's analysis, the NRC staff concludes there is reasonable assurance that operation within the limits of the TS will preclude fuel damage from a ramp reactivity insertion accident.

13.3 Loss-of-Coolant Accident

The licensee described several potential loss-of-coolant accidents, including pipe and tank ruptures and experiment malfunction. According to the licensee, a loss-of-coolant accident resulting in an uncovered core is not credible for the MITR-II because of the primary system design features and TS requirements for experiments. Sections 4.3, 5.2, and 6.2.1 of this SER discuss the primary system design features that protect against a complete loss of primary coolant. Specifically, the core tank is wholly contained in the reflector tank, all penetrations in the core tank are above the level of the core, and the two redundant antisiphon valves in the primary inlet plenum would prevent the core tank from draining due to siphoning. These design features ensure that the core would remain covered during any loss-of-coolant accident, and decay heat would be removed by natural circulation. According to the licensee, the core could only be uncovered by a simultaneous rupture in both the core tank and the reflector tank. As discussed in Section 3.4 of this SER, the licensee demonstrated that the tanks are designed to withstand seismic loading with a large safety margin. As discussed in Section 6.2.3 of this SER, TS 3.3.4 requires an ECCS capable of providing adequate coolant flow to cool the core in the case of a complete loss of coolant from the core tank. The ECCS has redundant and diverse coolant supplies, including recirculation of lost coolant and direct feed of city water. The NRC staff reviewed the licensee's discussion of potential loss-of-coolant accidents against the guidance in NUREG-1537 and finds that the licensee considered an appropriate spectrum of initiating events. The NRC staff finds that the primary system design features and engineered safety features are adequate to minimize the potential for a complete loss of coolant and to cool the core in the extremely unlikely case of a complete loss of coolant. Based on the above discussion and findings, the NRC staff concludes that there is reasonable assurance that a loss-of-coolant accident will not result in a loss of fuel integrity.

13.4 Loss-of-Flow Accident

A loss of flow to the MITR-II may be caused by a loss of offsite power or reactor pump malfunction. The licensee analyzed a complete loss of flow from the most conservative initial conditions corresponding to the LSSS values of 7.4 MW power, an outlet temperature of 60 degrees C (140 degrees F), and a core flow of 114 l/s (1,800 gpm). The licensee assumed an instrument delay time of 1 second following initiation of the scram signal, followed by 80-percent control blade insertion within 1 second. These control system response assumptions are conservative relative to the requirements of TS 3.2.1. The licensee performed the analyses with both the MULCH-II and RELAP5 computer codes. Both codes provided similar results and were benchmarked against flow coastdown and coolant temperature measurements taken from MITR-II startup test data. The licensee's most conservative analysis, using the RELAP5 code, resulted in a peak cladding temperature in the hot channel of approximately 125 degrees C (257 degrees F), which is well below the temperature required for

fuel damage. The NRC staff reviewed the analysis against the guidance in NUREG-1537 and finds that the analysis is consistent with the guidance because the licensee used initial conditions and RPS response consistent with the TS requirements. The NRC staff finds the licensee's analytical methods are appropriate because the licensee used two benchmarked codes that provided consistent results. Based on these findings and the results of the licensee's analysis, the NRC staff concludes there is reasonable assurance that operation within the limits of the TS will preclude fuel damage from a loss-of-flow accident.

13.5 Mishandling or Malfunction of Fuel

Mishandling of the MITR-II fuel could result in physical damage to the fuel, although historically no damage has occurred during handling. Actual or suspected damage would result in inspection and isolation of the fuel element, as required. According to the licensee, a significant release of fission products from physical damage is unlikely because of the construction of the fuel. A scratch on a fuel plate could expose a small area of fuel, but the structure of the fuel would retain fissions products not in the immediate vicinity of the scratch. Compared to the complete release of fission products from four fuel plates assumed in the MHA, physical damage would result in a minimal fission product release.

Fuel malfunctions such as excess outgassing have occurred during the operating history of the MITR-II. The radiation monitors in the core purge gas system required by TS 3.7.1 identified these in the incipient stages. Individual elements were identified by fuel inspections and removed from service, consistent with the requirements of TS 3.1.6, to prevent excessive release of radioactive material. The licensee methods for detection of such fuel failures minimize additional releases of radioactive material and are consistent with methods used at similar nonpower reactors.

Criticality safety is ensured by restricting fuel storage outside of the reactor core to locations where k_{eff} is less than 0.90, as required by TS 5.4. This requirement is consistent with the guidance in NUREG-1537 and ANSI/ANS 15.1. TS 3.2.3.2 requires appropriate operable nuclear safety channels, reflector dump, and major scram capability during fuel movement. Additionally, the upper core grid plate only allows access to one fuel element position at a time. These requirements ensure that refueling operations are adequately monitored and controlled to prevent inadvertent criticality. The licensee uses a transfer cask to minimize radiation levels in the containment building during fuel transfers outside of the core tank. TS 5.4 requires a minimum decay time before fuel movement outside the core tank. According to the licensee's analysis, this requirement reduces decay heat to a level that precludes fuel melting.

The NRC staff reviewed the licensee's discussions of fuel mishandling and malfunctions against the guidance in NUREG-1537 and finds that the discussions are consistent with the guidance because the licensee considered potential mechanisms for fuel damage at all stages of fuel handling. The NRC staff finds that the TS requirements are adequate to protect the fuel from overheating during transfer and protect against inadvertent criticality during storage. Additionally, the NRC staff finds that the radiological consequences of the MHA bound the consequences of fuel mishandling or malfunctions because no credible mishandling or malfunctions would result in a greater release of fission products to the containment atmosphere. Based on the above discussion and these findings, the NRC staff concludes there is reasonable assurance that mishandling or malfunctions of the fuel will not pose an undue risk to public health and safety, facility personnel, or the environment.

13.6 Experiment Malfunction

Chapter 10 of this SER discusses the licensee's experiment facilities and experiment review and approval processes. All experiments are subject to general experiment criteria and a review and approval process specified in TS 6.1 and TS 7.5, "Experiment Review and Approval," respectively. The general criteria include restrictions on reactivity effects, thermal-hydraulic effects, explosive energy, corrosion potential, radiolytic decomposition, internal heating, pressurization, and radioactive release potential. Provisions are also included for experiment scrams and prototype testing. The review and approval process requires a written safety review approved in writing by two licensed senior reactor operators, the Director of Operations, and the MITRSC. These requirements are consistent with the guidance in ANSI/ANS-15.1 and the NRC staff finds them acceptable to minimize the potential for experiment malfunctions. As discussed elsewhere in this SER, the licensee analyzed potential experiment malfunctions including reactivity effects and damage to the primary coolant boundary. The licensee's analyses show that experiment malfunctions will not cause fuel damage. The licensee analyzed the potential effects of malfunctions of experiments that contain radioactive material. Using assumptions consistent with the requirements of the TS, the licensee showed that all radiological consequences are bounded by the consequences of the MHA. This is consistent with the guidance in NUREG-1537 and the NRC staff finds it acceptable. Based on the above discussion and findings, the NRC staff concludes that the performance of experiments within the restrictions of the TS provides reasonable assurance that experiment malfunctions do not pose an undue risk to public health and safety, facility personnel, or the environment.

13.7 Loss of Normal Electric Power

The loss of normal power is an anticipated event for the MITR-II, and passive safety features incorporated into the design ensure safe shutdown of the reactor without external power sources. The loss of offsite power initiates the following safe-shutdown measures:

- The shim blade electromagnets deenergize, causing the control blades to insert under the influence of gravity and shut down the reactor.
- The reflector dump valve opens automatically, providing redundant shutdown capability.
- The containment building ventilation dampers automatically close, minimizing the release of any radioactive material to the environment.
- The reactor passively transitions from forced to natural convection cooling to remove decay heat from the core.

TS 3.6 requires battery backup power and specifies instruments and equipment powered by the batteries. These include select reactor and radiological instrumentation, lighting, and an auxiliary pump that may be used to transfer heat from the primary system to the secondary system. TS 4.6 provides surveillance requirements for backup power. As discussed in Section 8.2 of this SER, the emergency electrical power system is not required to maintain safe shutdown of the reactor. Based on the above discussion, the NRC staff concludes that there is reasonable assurance that a loss of normal electrical power will not result in fuel damage.

13.8 External Events

The SAR contains discussions and analyses of various external events, including the following:

- lightning
- flooding
- meteorological disturbances (e.g., tornadoes, hurricanes)
- seismic events
- mechanical impacts or collisions with the building (including stack collapse)

As discussed in Chapter 3 of this SER, the NRC staff has evaluated the facility design and analysis with respect to external events. This evaluation concluded that the facility design is capable of withstanding each of the described events without compromising the safe shutdown or containment functions of the facility. Therefore, the NRC staff concludes there is reasonable assurance that the above events would not pose an undue risk to public health and safety from damage to the facility.

13.9 Mishandling or Malfunction of Equipment

The licensee also considered and analyzed four additional potential accidents related to the mishandling or malfunction of equipment:

(1) operation with shim blades in a nonuniform bank position
(2) use of massive objects over the core tank
(3) spill of heavy water
(4) mixing of light and heavy water

For operation with a nonuniform shim bank, an analysis by the licensee for an assumed severe misalignment shows an increase in the radial peaking factor of 1.04 for the hot channel. The resulting hot channel power would still be well within the safety limits defined in TS 2.1 for normal operations. In addition, the subcritical interlock specified in TS 3.1.4 and the requirement that shim blades are within 2.0 in. of the average shim bank height specified in TS 3.2.4 minimize the potential for shim bank misalignment.

During the removal of spent fuel from the reactor core tank, certain massive objects and the fuel transfer cask must be moved over the top of the core. The licensee imposes administrative requirements that the reactor top shield lid must be in place before any lead fixtures are moved across the top of the core tank and that the fuel transfer cask must be kept within 15 cm (6 in.) of the surface of the top shield lid. The licensee's analysis has shown that should one of the fixtures or cask drop onto the lid, the lid would not rupture and the core would be protected.

The heavy-water system is equipped with a leak detection system that will alarm to notify the operator of a leak. The licensee analyzed a spill of heavy water assuming the maximum tritium concentration of 5 Ci per liter required by TS 3.3.5. According to the licensee, the 24-hour offsite dose would be several millirem. This is well below the consequences of the MHA and within the annual dose limit of 1 mSv (100 mrem) for members of the public specified in 10 CFR 20.1301. The analysis assumed a conservative liquid temperature of 60 degrees C (140 degrees F), which is above the expected maximum reflector system temperature. The NRC staff performed independent source term and dispersion calculations and confirmed that

the results of the licensee's calculations reflect a conservative estimate of the accident consequences.

Chapter 4 of the SAR analyzes the reactivity consequences of mixing of light and heavy water through leakage between the core and reflector tanks. Leakage of light water from the core tank into the reflector tank results in a negative reactivity effect, tending to shut down the reactor. The effect of leakage of heavy water into the core tank is positive for areas above or below the core, or for the annular space around the core. However, according to the licensee, any reactivity addition would be bounded by the previously analyzed reactivity transient accidents. The presence of the heavy water in the core proper results in a negative effect. Since the core tank is at a higher pressure, the most likely leakage would cause a negative reactivity effect.

The NRC staff reviewed the licensee's analyses of the effects of the mishandling and malfunctions of equipment. The NRC staff finds that these accidents do not pose a significant risk of loss of fuel integrity and any potential release of radiological material is bounded by the MHA. Based on these findings, the NRC staff concludes that mishandling and malfunctions of equipment do not pose an undue risk to public health and safety, facility personnel, or the environment.

13.10 Conclusions

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

14 TECHNICAL SPECIFICATIONS

The NRC staff evaluated the TS as part of its review of the application for renewal of Facility Operating License No. R-37. The TS define certain features, characteristics, and conditions governing the operation of the MITR-II. The renewed license includes the TS as Appendix A. The NRC staff reviewed the format and content of the TS for consistency with the guidance 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 they meet the requirements in 10 CFR 50.36. The NRC staff concludes that the MITR-II TS meet the requirements of the regulations based on the following findings:

- To satisfy the requirements of 10 CFR 50.36(a), the licensee provided 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 MITR-II is a facility of the type described in 10 CFR 50.21(a) and (c); 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 MITR-II SAR.

- To satisfy the requirements of 10 CFR 50.36(c)(1), the licensee provided TS specifying safety limits and LSSSs for the RPS to preclude reaching the safety limits.

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

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

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

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

The NRC staff finds the TS to be acceptable and concludes that normal operation of the MITR-II within the limits of the TS will not result in radiation exposures in excess of the limits specified in 10 CFR Part 20 for members of the general public or for occupational exposures. Based on the above findings and evaluations of individual TS contained elsewhere in this SER, the NRC staff concludes that the TS provide reasonable assurance that the facility will be operated as analyzed in the MITR-II SAR and will limit the likelihood of malfunctions and potential accidents.

15 FINANCIAL QUALIFICATIONS

15.1 Financial Ability To Operate the Facility

As stated in 10 CFR 50.33(f)—

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

MIT does not qualify as an "electric utility," as defined in 10 CFR 50.2, "Definitions." Further, pursuant to 10 CFR 50.33(f)(2), the application to renew or extend the term of any operating license for a nonpower reactor shall include the financial information that is required in an application for an initial license. The NRC staff has determined that MIT must meet the financial qualifications requirement pursuant to 10 CFR 50.33(f) and is subject to a full financial qualifications review. MIT must provide information to demonstrate that it possesses or has reasonable assurance of obtaining the funds necessary to cover estimated operating costs for the period of the license. Therefore, MIT must submit estimates of 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 these costs.

In supplements to the application dated October 7 and December 1, 2008, MIT submitted its projected operating costs for the MITR-II by year for fiscal years 2009 through 2013. The projected operating costs for the MITR-II are estimated to range from $3,038,957 in fiscal year 2009 to $3,420,373 in fiscal year 2013. Funds to cover operating costs will come from a university allocation, research, the MIT-National Scientific User Facility collaboration, and commercial services. The NRC staff reviewed MIT's estimated operating costs and projected sources of funds and found them to be reasonable.

The NRC staff finds that MIT has demonstrated reasonable assurance of obtaining the necessary funds to cover the estimated operating costs for the MITR-II for the period of the license. Accordingly, the NRC staff has determined that MIT has met the financial qualification requirements pursuant to 10 CFR 50.33(f) and is financially qualified to engage in the proposed activities regarding the MITR-II.

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. In 10 CFR 50.33(k), the NRC 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. Under 10 CFR 50.75(d), each nonpower reactor applicant for or holder of an operating license shall submit a decommissioning report that contains a cost estimate for decommissioning the facility, 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 appear in 10 CFR 50.75(e)(1).

The application submitted on July 8, 1999, referenced a conceptual decommissioning plan and cost estimate developed by General Electric Company for the MITR-II of $21.55 million in 1996 dollars. In a supplement to the application dated October 11, 2006, MIT submitted the "Duke Executive Summary" and "Decommissioning Costs," which provided a decommissioning cost estimate of $23.0 million in 2005 dollars based on the "Duke Engineering Study 2001." According to Duke Engineering & Services, the decommissioning cost estimate is a "bottom-up" cost estimate and "provides MIT with a baseline estimate to use for the eventual decommissioning planning and execution." In additional supplements to the application dated May 29, 2008, and May 26, 2009, MIT updated the decommissioning cost estimate to be $29.8 million in 2008 dollars, assuming that the DECON decommissioning method is used. The MITR-II decommissioning cost estimate summarized costs by labor, waste disposal, other activities (e.g., energy, equipment, and supplies), and a 10-percent contingency factor. Contingency is required to address unforeseeable elements of cost within the defined scope. Since MIT has provided a "hands-on cost estimate" developed by Duke Engineering & Services, the cost estimate has identified key elements and greatly reduced the possibility of unforeseeable elements. It is the basis for which the NRC is accepting the lower (10 percent) contingency. In reviewing the decommissioning cost estimate submitted by MIT ($29.8 million), the NRC staff took into consideration experience at other facilities with similar construction and operational history and concludes that the decommissioning cost estimate for the MITR-II is reasonable.

MIT stated that it will update its decommissioning cost estimate using the General Electric Company cost escalation formula that was included in the conceptual decommissioning plan and cost estimate, applying inflator figures based on the most recent revision of NUREG-1307, "Report on Waste Burial Charges: Changes in Decommissioning Waste Disposal Costs at Low-Level Waste Burial Facilities," Revision 13, issued November 2008, page D.1, Example 2 for the Northeast Region. According to MIT, the costs associated with energy, equipment, and supplies were apportioned between labor and burial.

Until July 2010, MIT provided financial assurance for decommissioning through an escrow account, as allowed by 10 CFR 50.75(e)(1)(ii), which states that "[p]repayment may be in the form of a trust, escrow account, or Government fund with payment by, certificate of deposit, deposit of government or other securities or other method acceptable to the NRC." MIT submitted an amendment to its escrow agreement on August 27, 2009, dated as of June 18, 2009, requesting NRC consent to increasing decommissioning funding for the MITR-II to $30 million. MIT did not propose changes to the $1.125 million decommissioning funding provided for the special nuclear material license (SNM-986). NRC consent was required as stipulated by the MIT Amended Escrow Agreement, dated November 30, 2005, for the Decommissioning Trust currently held for eventual decommissioning of the MITR-II. Based on its review of the information submitted by MIT, the NRC staff approved the proposed decommissioning funding increase to $30 million for the MITR-II. Accordingly, an amendment to the MIT Amended Escrow Agreement to reflect this increase is acceptable. However, as described in the following paragraph, the escrow account method of providing decommissioning funding assurance has been superseded by recent NRC staff approval of the licensee's application to change its method of providing decommissioning funding assurance to the self-guarantee method.

On October 27, 2009, as supplemented on December 22, 2009, and May 11, 2010, MIT submitted an application requesting to change its method of providing decommissioning funding assurance from an escrow account to the self-guarantee method allowed by

10 CFR 50.75(e)(1)(iii) for nonprofit entities such as universities. The NRC staff reviewed the documentation submitted by MIT and approved the application by letter dated July 30, 2010 (ADAMS Accession No. ML102090007). The findings of the letter, duplicated below, remain valid for license renewal:

> The NRC staff reviewed the self-guarantee agreement and corroborating documentation to cover the cost of decommissioning the above-mentioned NRC Licenses and found that the self-guarantee agreement meets or exceeds the financial test criteria for a non-profit university that issues bonds, that it is acceptable for providing decommissioning funding assurance, and that it is in accordance with the provisions of Appendix E to 10 CFR Part 30.

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

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." NRC regulation 10 CFR 50.38, "Ineligibility of Certain Applicants," contains language to implement this prohibition. According to the application and its supplements, MIT is a organized as a State of Massachusetts nonprofit corporation, principally doing business within the State of Massachusetts, and it is not owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government. MIT provided the names, addresses, and citizenship of MIT's principal officers, all of whom are U.S. citizens. The NRC staff does not know or have reason to believe otherwise.

15.4 Nuclear Indemnity

The NRC staff notes that MIT currently has an indemnity agreement with the Commission that does not have a termination date. Therefore, MIT will continue to be a party to the present indemnity agreement following issuance of the renewed license. Under 10 CFR 140.71, "Scope," MIT, as a nonprofit corporation licensee, is not required to provide nuclear liability insurance. The Commission will indemnify MIT for any claims arising out of a nuclear incident under the Price-Anderson Act, Section 170 of the AEA, as amended, and in accordance with the provisions under its indemnity agreement pursuant to 10 CFR 140.95, "Appendix E—Form of Indemnity Agreement with Nonprofit Educational Institutions," for up to $500 million and above $250,000. Also, MIT 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 MITR-II and, when necessary, to shut down the facility and carry out decommissioning activities. In addition, the NRC staff concludes there are no problematic

foreign ownership or control issues or insurance issues that would preclude the issuance of a renewed facility operating license.

16 OTHER LICENSE CONSIDERATIONS

16.1 Prior Use of Reactor Components

16.1.1 History of the MITR-II

Initial criticality of the original MIT reactor occurred in 1958. The licensed power level was 1 MW until 1961, when it was increased to 2 MW. The power level was increased to 5 MW in 1965. The reactor was modified in 1974 and 1975. The initial criticality for the modified reactor, the MITR-II, occurred in 1975 and routine operation at 5 MW began in 1976. In 1995, the NRC authorized the licensee to continue operation until 1999 to recapture time spent constructing the modified reactor.

16.1.2 Component Assessment

No MITR-II components were used at other reactor facilities. All components and systems are covered by a maintenance and surveillance program. The containment building is functionally tested on a biennial basis as required by TS 4.4. Further, individual components, including gaskets, are tested as needed, but at least annually. TS 4.4 further requires annual inspections or tests of the ventilation isolation dampers, vacuum relief breakers, and charcoal filters. The principal mechanism for deterioration of the steel shell would be galvanic attack below ground. Electronic equipment is subject to less physical degradation than mechanical components as most equipment is in environmentally controlled areas. Periodic surveillance and testing meeting the requirements of TS 4.2 are used to detect components that need replacement. The current cathodic protection system replaced the original system in 1994. Neutron absorbers are replaced, as needed, for functional purposes. Twenty years of semiannual measurements of absorbers have demonstrated no swelling of components or other dimensional changes. Other in-core structures (light-water core tank, deuterium oxide reflector tank) are protected from deterioration either through water chemistry or by inert cover gases. The reactor core tank and core support structure were replaced during the construction of the modified reactor in 1974. The licensee inspected the tank for corrosion several times since then and noted no significant corrosion of the inner or outer surfaces. The licensee performed a study of expected irradiation damage and compared the expected irradiation conditions to those at similar research reactors. According to the licensee, irradiation of the core tank will not cause significant degradation of the material properties over the life of the tank. The NRC staff reviewed the information in the SAR related to age-related degradation of components and systems against the guidance in NUREG-1537. The NRC staff finds that the information is consistent with the guidance and that the licensee considered an appropriate range of systems and components.

16.1.3 Preventive and Corrective Maintenance Program

Regular preventive and corrective maintenance are performed at the MITR-II. Preventive maintenance is performed on components with importance to safety and subject to possible wear, including includes gaskets and reactivity control drives. The frequency of maintenance activities is based on manufacturer recommendations. Operational experience is used to set appropriate maintenance frequency for MIT-designed components. Maintenance activities are audited quarterly by the Quality Assurance Supervisor or designate.

16.2 Medical Use of Nonpower Reactors

The traditional use of research reactors for medical use is through the production of isotopes for medical use, as licensed under AEA Section 104c. While the MITR-II is used for this function, it is also licensed under Section 104a., related to generation of neutron beams for medical therapy. As described in Chapter 10 of the SAR, the MITR-II has two medical therapy rooms for neutron irradiation. The older room is directly beneath the reactor and has been used for thermal neutron irradiation of brain tumors. As this energy of neutrons will not effectively penetrate the scalp and skull, this therapy room was equipped as an operating room, and patients had surgery to expose the brain tissue for irradiation.

More recent use of the MITR-II for medical therapy has involved boron neutron capture therapy (BNCT). For this procedure, a patient is given a boron-tagged compound that is preferentially taken up by the diseased tissue. The patient is then exposed to a beam of neutrons in the epithermal energy range. These neutrons are more effective at penetrating the skull than thermal neutrons and, thus, no surgery is required. The newer therapy room is located at the side of the reactor adjacent to the fission converter. Neutrons from the reactor are thermalized in the graphite reflector and then enter the fission converter, where they cause fission in the fission converter fuel elements. Radiation from the fission converter fuel is filtered to remove gamma and other high-energy components, and an epithermal neutron beam is collimated and directed to the therapy room. The beam is controlled through a series of shutters that block the beam when not needed. The beam can also be shut off by shutting down the reactor.

16.2.1 Responsibilities for Use of Medical Therapy Facility Beams

The delivery of a medical therapy beam to a patient is a medical treatment and requires the supervision and approval of a licensed physician using approved protocols for patient treatment. The use of the beam must be through a medical organization that is licensed for such treatment. TS 6.5 specifies the requirements for the generation of medical therapy beams for human therapy. This TS specifies MIT's responsibilities associated with medical use of the beam as well as the design requirements for the beam irradiation facilities. MIT is responsible for providing current and accurate beam characteristic parameters to the medical use licensee and for delivery of the appropriate beam in accordance with written directives from the supervising physician. The MITR-II staff is also responsible for all health physics considerations associated with the beam, except with regard to the patient. The MITRSC, the Committee on the Use of Humans as Experimental Subjects, and the Committee on Radiation Exposure to Human Subjects all have some oversight responsibility for the use of the facilities. The requirements of TS 6.5 include the ability to scram the reactor from the medical therapy area (TS 6.5.3), interlocks to prevent beam operation without the shield doors closed (TS 6.5.5), indication of shutter status and radiation levels (TS 6.5.6 and 6.5.7), and training requirements for personnel (TS 6.5.16). Further, TS 6.5.18 requires a quality management program.

16.2.2 Regulatory Commitment

The regulations in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," do not contain explicit requirements governing the use of special nuclear material for medical therapy at research reactors. The regulations in 10 CFR Part 35, "Medical Use of Byproduct Material," contain requirements governing the use of byproduct material for teletherapy. The use of neutron beams for medical therapy has many similarities to teletherapy. Although 10 CFR Part 35 may be used as a guide to regulate BNCT, it is not directly applicable to BNCT and the use of special nuclear material for human therapy at the MITR-II.

On February 16, 1993, the NRC issued Amendment No. 27 to Facility Operating License No. R-37 which authorized MIT to generate neutron beams for human therapy. As explained in the NRC's safety evaluation for the amendment, the NRC used 10 CFR Part 35 (as published in the *Federal Register* on October, 16, 1986, at 51 FR 36951 and amended through 1992) as a guide for developing a regulatory commitment for generation of neutron beams for medical therapy at the MITR-II. To address the regulatory commitment, MIT developed TS 6.5 which contained requirements for generation of neutron beams for medical therapy. The NRC safety evaluation of the amendment concluded that, "there is assurance that the use of the MIT medical therapy facility and beam for human therapy will be in accordance with the treatment plan established by the physician authorized user, that non-therapeutic radiation exposure will be ALARA, and that the medical therapy facility will function as designed." The requirements of TS 6.5 were amended by Amendment No. 30 dated April 3, 1997, and Amendment No. 32 dated April 2, 2001. These amendments included changes to the TS requirements prompted by Massachusetts becoming an Agreement State in 1997, and licensing of the fission converter facility as a second human therapy facility in 1999. The license amendments did not significantly change the 1993 regulatory commitment.

The NRC published a final rule in the *Federal Register* on April, 24, 2002, at 67 FR 20370 that revised the regulations in 10 CFR Part 35. This revision affected several of the regulations used as guidance for developing the regulatory commitment for generation of neutron beams for medical therapy. As part of the license renewal review, the NRC staff reviewed the changes to the regulations and compared the regulatory commitment and TS 6.5 to the current regulations in 10 CFR Part 35 (2002 version as amended through 2009). Based on the review of the changes to 10 CFR Part 35, the NRC staff finds that the regulatory commitment continues to parallel the related requirements for teletherapy. Additionally, the NRC staff finds that the current 10 CFR Part 35 does not contain any new regulations that would necessitate a new regulatory commitment for generation of neutron beams for medical therapy. Based on the comparison of TS 6.5 against the current regulations in 10 CFR Part 35, the NRC staff finds that TS 6.5 is consistent with, or more conservative than the regulations. Based on these findings, the NRC staff concludes that the requirements of TS 6.5 continue to provide assurance that the use of the MIT medical therapy facilities and beams for human therapy will be in accordance with the treatment plan established by the BNCT physician authorized user, that non-therapeutic radiation exposure will be ALARA, and that the medical therapy facilities will function as designed.

Section 17.3 of the SAR discusses MIT's regulatory commitment for generation of neutron beams for medical therapy and provides a comparison between the requirements of TS 6.5 and the regulations in 10 CFR Part 35 prior to the 2002 revision. The revision to 10 CFR Part 35 resulted in renumbering, reorganizing, and revising or deleting regulations referenced in the SAR and the bases for TS 6.5. Given that TS 6.5 was developed based on the version of 10 CFR Part 35 in place in 1993, and the above finding that TS 6.5 remains technically consistent with, or more conservative than the current regulations in 10 CFR Part 35, the NRC did not request that MIT update the references in the SAR or the bases for TS 6.5 as part of license renewal.

16.3 Conclusions

The NRC staff reviewed the prior use of reactor components as well as the aging of safety components, as described in the 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. The NRC staff also reviewed the use of the MITR-II for medical uses and concludes that the facilities and programs for use of these facilities are sufficient to provide reasonable assurance that medical operations will not result in exposures to facility personnel over regulatory limits and that neutron beam characteristics for medical therapy will be within design parameters.

17 CONCLUSIONS

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

- The application for license dated July 8, 1999, as supplemented by letters dated February 10 and May 8, 2000; January 29, 2004; July 5 and October 11, 2006; January 26, 2007; February 22, May 29, August 15, August 21, August 26, October 6, October 7, and December 1, 2008; May 26, August 27, October 5, October 9, and November 19, 2009; and March 30, August 6, and August 26, 2010, complies with the standards and requirements of the AEA and the Commission's rules and regulations set forth in 10 CFR Chapter I, "Nuclear Regulatory Commission."

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

- There is reasonable assurance that (1) the activities authorized by the renewed license can be conducted at the designated location without undue risk to public health and safety, 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 facility operating license will not be inimical to the common defense and security or to public health and safety.

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