Safety Evaluation Report

Related to the Renewal of the Operating License for the Research Reactor at the Missouri University of Science and Technology

Docket No. 50-123

March 2009

Safety Evaluation Report Related to the Renewal of

Facility License No. R-79 for the

Missouri University of Science and Technology Research Reactor,

Rolla, Missouri

March 2009
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 Board of Curators of the University of Missouri (the licensee) for a 20-year renewal of Facility License No. R-79 to continue to operate the Missouri University of Science and Technology (MST) Research Reactor. 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), inspection reports prepared by NRC personnel, and first-hand observations. On the basis of this review, the NRC staff concludes that MST can continue to operate the facility for the term of the renewed license, in accordance with the renewed facility license, without endangering the health and safety of the public, facility personnel, or the environment.

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ABBREVIATIONS

Abbreviation	Definition
ac	alternating current
ADAMS	Agencywide Documents Access and Management System
ALARA	as low as reasonably achievable
ANSI/ANS	American National Standards Institute/American Nuclear Society
C	Celsius
CAM	continuous air monitor
cfm	cubic feet per minute
CFR	*Code of Federal Regulations*
CIC	compensated ionization chamber
cm	centimeter
cm/min	centimeter per minute
cps	counts per second
DNR	Director Nuclear Reactor
DOE	U.S. Department of Energy
EP	emergency plan
F	Fahrenheit
ft	foot or feet
gal	gallon
HEU	high-enriched uranium
HVAC	heating, ventilation, and air conditioning
I&C	instrumentation and controls
in.	inch
in./min	inch per minute
km	kilometer
km/h	kilometer per hour
kW	kilowatt
kW(t)	kilowatt thermal
LCO	limiting condition for operation
LEU	low-enriched uranium
LSSS	limiting safety system setting
µCi/ml	microcurie per milliliter
m	meter
m^3	cubic meter
MHA	maximum hypothetical accident
mi	mile
mi^2	square mile
mph	mile per hour
mR	milliroentgen
mrem/h	millirem per hour
MST	Missouri University of Science and Technology
MSTR	Missouri University of Science and Technology Research Reactor
mSv	millisievert
MTR	materials testing reactor
NRC	U.S. Nuclear Regulatory Commission
NUREG	NRC technical report designation

RAI	request for additional information
RAM	radiation area monitor
RCS	reactor control system
RPS	reactor protection system
RSO	radiation safety officer
s	second
SAR	safety analysis report
SER	safety evaluation report
SNM	special nuclear material
SOP	standard operating procedure
SSC	structure, system, and component
TEDE	total effective dose equivalent
TS	technical specification or technical specifications
UIC	uncompensated ionization chamber
V	volt
W	watt
W/cm^2	watt per square centimeter

1. INTRODUCTION

1.1 Overview

By letter (and supporting documentation) dated August 30, 2004, as supplemented on November 16, November 27, and December 26, 2007, and January 17, March 6, June 26, September 16, and November 7, 2008, the Board of Curators of the University of Missouri (the licensee) submitted to the U.S. Nuclear Regulatory Commission (NRC or the Commission) a timely application for a 20-year renewal of the Class 104c Facility Operating License No. R-79 (NRC Docket No. 50-123).

The NRC staff conducted its review based on information submitted in the renewal application, as supplemented. The renewal application includes the safety analysis report (SAR), environmental report, operator requalification program, emergency plan (EP), statement of financial qualification, proposed technical specifications (TS), and responses to staff requests for additional information (RAIs). The NRC staff also based its review on the licensee's annual reports and NRC inspection reports. The licensee also requested that the NRC staff review and approve a revision of the EP filed with the NRC as part of the application. The licensee has continued to update these documents, both in response to RAIs issued by the NRC staff and as part of its routine document maintenance. Except for the EP that is withheld from public disclosure due to security purposes, the material may be examined, and/or copied for a fee, at the NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, MD. The NRC also 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 may be accessed through the NRC's Public Electronic Reading Room on the Internet at http://www.nrc.gov. Those who do not have access to ADAMS, who have problems accessing the documents located in ADAMS, or who want access to documents published before November 24, 1999, may contact the NRC Public Document Room reference staff at 1-800-397-4209 or 301-415-4737 or by e-mail to pdr@nrc.gov.

The purpose of this safety evaluation report (SER) is to summarize the NRC staff's findings with respect to its safety review of the Missouri University of Science and Technology Research Reactor (MSTR or the facility) and to identify the technical details that the NRC staff considered in evaluating the reactor and the radiological safety aspects of continued operation. This SER and an environmental assessment will serve as the basis for renewing the license for operation of the MSTR at thermal power levels up to and including 200 kilowatts (kW).

In conducting its safety review, the NRC staff evaluated the facility against the requirements of the following regulations:

- Title 10, Part 20, "Standards for Protection against Radiation," of the *Code of Federal Regulations* (10 CFR Part 20)
- 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material"
- 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities"

- 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions"

- 10 CFR Part 55, "Operators' Licenses"

- 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material"

- 10 CFR Part 73, "Physical Protection of Plans and Materials"

In addition to the above-listed regulations, the NRC staff also evaluated the facility against applicable regulatory guides; relevant accepted industry standards, such as the American National Standards Institute/American Nuclear Society (ANSI/ANS) 15 series; and NRC guidance documents, such as NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Research and Test Reactors," issued February 1996. Since no specific accident-related regulations for research reactors exist, the NRC staff compared calculated dose values for accidents to the limits in 10 CFR Part 20. Amendments to 10 CFR Part 20 (10 CFR 20.1001 through 10 CFR 20.2402 and appendices) became effective January 1, 1994. These amendments changed the dose limits for occupationally exposed persons and members of the public, as well as the concentrations of radioactive material that are allowed in effluents released from licensed facilities. The licensee must follow the requirements of 10 CFR Part 20, as amended, for all aspects of operation of the MSTR.

John Nguyen, Project Manager, from the NRC's Office of Nuclear Reactor Regulation, Division of Policy and Rulemaking, Research and Test Reactors Branch B, prepared this SER. Other major contributors to the safety review include Brookhaven National Laboratory, under contract to the NRC, and Daniel Hughes, Paul V. Doyle, Eric Benner, and Larry Pittiglio of the NRC.

1.2 Summary and Conclusions Regarding the Principal Safety Considerations

On the basis of this evaluation, the NRC staff made the following nine findings:

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

(2) The licensee's management organization is adequate to maintain and operate the reactor. Security measures, training programs, and research activities are adequate to ensure safe operation of the facility and protection of its special nuclear material (SNM).

(3) The expected consequences of postulated accidents are not likely to exceed the guidelines specified in 10 CFR Part 20 for doses in restricted as well as unrestricted areas.

(4) Releases of radioactive materials and wastes from the facility are not expected to result in concentrations beyond the limits specified by the Commission's regulations and are consistent with the principle of as low as reasonably achievable (ALARA).

(5) The TS, which state limits for controlling the operation of the facility, provide a high degree of assurance that the facility will be operated in accordance with the assumptions and analyses in the SAR. The licensee's historical data also show no significant degradation of equipment. The TS will continue to ensure that no significant degradation of SSCs will occur.

(6) The financial data submitted with the application show that the licensee has reasonable access to sufficient revenues to cover operating costs and to eventually decommission the reactor facility. Furthermore, the licensee will provide an update including the estimated decommissioning costs of the reactor facility, operating costs, and its ownership under an obligation of 10 CFR 50.9, "Completeness and Accuracy of Information," if there are any changes in financial qualification.

(7) The licensee's program for providing for the physical protection of the facility and its SNM complies with the requirements of 10 CFR 73.67, "Licensee Fixed Site and In-Transit Requirements for the Physical Protection of Special Nuclear Material of Moderate and Low Strategic Significance."

(8) The procedures for training its reactor operators and the plans for operator requalification are adequate and provide reasonable assurance that the reactor will be operated in a competent manner.

(9) The licensee maintains an EP in compliance with 10 CFR 50.54(q) and 10 CFR Part 50, which provides reasonable assurance that the licensee is prepared to assess and respond to emergency events.

On the basis of these findings, the NRC staff concludes that the MSTR can continue to operate in accordance with its application without endangering the health and safety of the public.

1.3 General Description

The licensee states in the SAR and responses to RAIs that the MSTR is housed in an independent building located on the east site of the Missouri University of Science and Technology (MST) campus. The building is a steel-frame structure with insulated metal walls. The MSTR is a pool reactor that utilizes low-enriched uranium (LEU) fuel for the core. Light water is used as the coolant and moderator. Graphite blocks make up the reflector. According to the licensee, the low power level of the core allows for sufficient cooling by natural convection. The core is immersed in highly purified water in an aluminum tank that holds approximately 113,560 liters (30,000 gallons (gal)) of water. Analysis performed by the licensee in the SAR demonstrates that fission product inventories are minimal and will be retained in the fuel with natural convection cooling. Accordingly; the licensee states that active engineered safety features are not necessary.

The reactor's experimental facilities include a pneumatic transfer system, in-core irradiation tube, beam tube, and thermal column. The TS limit the sum of the absolute values of all experiments to a maximum reactivity of 1.2% $\Delta k/k$. The licensee states that the reactivity limit of 1.2% $\Delta k/k$ is well below the maximum excess reactivity limit of 1.5% $\Delta k/k$ established in Section 13.1.2 of the SAR to ensure that the fuel cladding is well below the safety limit established in the TS 2.1 for fuel integrity.

1.4 Shared Facilities and Equipment

The main campus system provides utilities such as electrical supply, sewage, and potable water.

1.5 Comparison with Similar Facilities

The NRC staff notes that the MSTR is based on the design of the bulk shielding reactor at Oak Ridge National Laboratory, which was a materials testing reactor (MTR). Reactors of this type have common features, such as light-water moderation, natural convection cooling, open pools, and plate-type fuel. Many reactors with similar design, construction, and operation characteristics have been safely and reliably operated for more than 40 years. In 1992, the licensee converted the MSTR fuel from high-enriched uranium (HEU) to LEU. The Ohio State University research reactor (which can operate at thermal levels up to 500 kW) is most similar to the MSTR in operating characteristics and facility features. The NRC granted a 20-year extension of the operating license for the Ohio State University research reactor in June 2008.

1.6 Summary of Operations

The SAR states the MSTR provides teaching and research services for students, faculty, and the public in accordance with NRC Facility Operating License No. R-79. From 1999–2007, the reactor has operated for 500–800 hours per year, or about 10–16 hours per week. During that time, the annual thermal energy production of the facility averaged approximately 20 megawatt-hours per year. The licensee states that the reactor expects to maintain or possibly improve its utilization rate during the period of renewed operations.

1.7 Compliance with the Nuclear Waste Policy Act of 1982

To comply with section 302(b)(1)(B) of the Nuclear Waste Policy Act of 1982, the SAR states the following:

> The Nuclear Waste Policy Act of 1982 provides that the NRC may require, as a precondition to issuing or renewing an operating license for a research or test reactor, that the applicant shall have entered into an agreement with the U.S. Department of Energy (DOE) for the disposal of high-level radioactive wastes and spent nuclear fuel. DOE (represented by R.L. Morgan) informed the NRC (represented by H. Denton) by letter dated May 3, 1983, that DOE had determined that universities and other Government agencies operating nonpower reactors had entered into contracts with DOE that provide that the DOE retains title to the fuel and is obligated to take the spent fuel and/or high-level waste for storage or reprocessing.

The NRC staff concludes that, by entering into such a contract with DOE, MST has satisfied the requirements of the Nuclear Waste Policy Act of 1982 as they apply to the MSTR.

1.8 Major Facility Modifications and History

This facility was initially licensed in December 1961, at a maximum thermal power level of 10 kW. In 1967, an amendment to the operating license was granted to increase the reactor maximum thermal power level to 200 kW. In 1985, the NRC granted a 20-year extension of the operating license; in 1992, the Commission issued an order to convert the reactor fuel from HEU to LEU. Other minor changes to the facility or procedures have either enhanced capabilities or improved reactor operations. All of these modifications were subject to an evaluation under 10 CFR 50.59, "Changes, Tests and Experiments," to ensure that there was no impact on the safety of the MSTR. The NRC staff's review of modifications made during the last 20 years reveals no significant or safety-related modifications. The NRC staff's review of the previous license amendments showed that most were administrative in nature. No substantive changes are being advocated for this license renewal.

2. SITE CHARACTERISTICS

2.1 Geography and Demography

The following sections describe the geography of the MSTR site, including the location of the MSTR and the demography of the area around the site.

2.1.1 Geography

MST is located in the City of Rolla in Phelps County, MO. It is about 100 miles (mi) southwest of St. Louis and 180 mi southeast of Kansas City. The MSTR building lies approximately 1 mi east of the main campus.

Rolla has a total area of 11.3 square miles (mi^2), of which 99 percent is land and less than 0.1 percent is water. The surrounding terrain is hilly and rolling. While the land is generally too rocky and sloped to support large-scale agriculture, there is some beef, dairy cattle, hog, and chicken farming in the area.

TS 5.1.1 and 5.1.2 contain features applicable to the MST site, as described below:

> TS 5.1.1. The Nuclear Reactor Building is located on the east side of the Missouri University of Science and Technology campus near 14th Street and Pine Street in Rolla, MO.
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> TS 5.1.2. The reactor is housed in a steel-framed, double-walled building designed to restrict leakage. Air and other gases may be exhausted through vents in the reactor bay ceiling 9.1 m (30 ft) above grade. The Reactor Building's free volume is approximately 1700 m^3.

Design feature TS are defined in 10 CFR 50.36(d)(4) as those features of the facility, such as materials of construction and geometric arrangements, that, if altered or modified, would have a significant effect on safety. TS design features need prior review and approval from the NRC to change. The location, design materials, and geometry defined above are used in calculations that could affect safety, such as public doses from radiation. Because the site is clearly defined, TS 5.1.1 and 5.1.2 are acceptable to the NRC staff.

2.1.2 Demography

According to 2000 U.S. census data, Rolla has a population of 16,367, which represents a 14-percent increase over the preceding 7-year period. Fort Leonard Wood, 25 mi (40 kilometers (km)) southwest of Rolla, is the largest population center, with approximately 13,667 (2000 census data) residents. The current enrollment at MST is about 8,000 persons, including 6,000 students and 2,000 faculty and staff.

2.1.3 Conclusions

The licensee has provided a detailed and accurate description of the geography surrounding the MSTR. The demographic information is sufficient to allow accurate assessments of the potential radiological impact on the public from the continued operation of the MSTR. Based on a review of this information, there is reasonable assurance that no geographic or demographic features render the MSTR unsuitable for continued reactor operation.

2.2 Nearby Industrial, Transportation, and Military Facilities

The licensee states in the SAR and RAI responses that there are no industries, manufacturing facilities, transportation routes, military facilities, or railroads near the MSTR that could pose a significant risk to the continued safe operation of the MSTR.

Table 2.4 in the SAR listed two major manufacturing facilities within 5 mi (8 km) of the MSTR. These facilities produce polyvinyl chloride pipe and pet food. The licensee discussed safety issues with the manufacturers and found that these facilities do not pose any unanalyzed threat to the safe operation of the MSTR.

Section 2.1 of the SAR addressed the effects of transportation facilities on the MSTR during the period of renewed operations. There are three land transportation routes, one railroad, and one airport near the MSTR. Interstate 44 is approximately 0.37 mi (0.60 km) to the northwest, and U.S. 63 is about 0.25 mi (0.4 km) to the northwest. The Burlington Northern Railroad runs 0.25 mi (0.4 km) to the east, with trains usually traveling at slow speed near the reactor community. The wind direction is typically from the southwest to the northeast, which will direct airborne effluents from any interstate or rail accident away from the reactor. No major waterways are located close to the MSTR. The Rolla National Airport at Vichy is located about 13 mi (22 km) north. The airport averages about 15,000 operations yearly and serves as an important transportation hub for large local businesses, government organizations, the university, and the State of Missouri. Airplanes as large as a DC-9 can land at this airport, and as many as 14 large planes can be on the ground at one time. The airport is used mostly by single- and twin-engine planes. The NRC staff independently searched the data maintained by the National Transportation Safety Board and found that only 10 accidents occurred during the 1985–2007 period, and none caused damage to the MSTR.

The SAR identified two military-related facilities located in the vicinity of the MSTR. Fort Leonard Wood, a large military base, is located 25 mi (40 km) southwest of the reactor, and the Army Reserve Center is at least 5 mi (8 km) from the reactor. The licensee states that all field activities of the two bases are conducted outside of the Rolla area. Given their distance from the MSTR and the missions performed at these military installations, the NRC staff concludes that neither of these facilities will pose an unanalyzed threat to the safe operation of the reactor.

2.3 Meteorology

The licensee describes the general climate in the Rolla area as "a continental midwestern type and is not influenced by any local mountains or large bodies of water." According to data provided in the SAR, temperatures in this region ranged from a maximum of 110 degrees Fahrenheit (F) (43.3 degrees Celsius (C)) to a minimum of -27 degrees F (-32.8 degrees C), with a mean annual temperature of 54.5 degrees F (12.5 degrees C) over the past 30 years.
The licensee provided the following metrological data in the SAR and responses to the NRC staff's RAIs. Average annual precipitation from 1961 to 1990 ranged from 28.03 inches (in.) (71 centimeters (cm)) to 41.4 in. (105.1 cm). The highest annual precipitation was 63.06 in. (160.1 cm). Average monthly precipitation ranged from 1.67 in. (4.24 cm) to 5.0 in. (12.70 cm). The average annual snowfall for the region was 18.3 in. (46.5 cm). The maximum monthly snowfall was 25.1 in. (63.8 cm). The highest annual snowfall was 38.1 in. (96.8 cm). Mean annual windspeed in this area is 9.9 miles per hour (mph) (15.9 kilometers per hour (km/h)).

The maximum windspeed was 60 mph (97 km/h), with wind gusts of 85 mph (137 km/h). From January 1993 to March 2007, seven tornadoes were observed in Phelps County. Three of these were rated as F0 (Gale Tornado) (windspeed between 40–72 mph (64–116 km/h)) on the Fujita Scale, two were an F1 (Moderate Tornado) (windspeed 73–112 mph (117–180 km/h)), one was an F2 (Significant Tornado) (windspeed 113–157 mph (182–253 km/h)), and one was an F3 (Severe Tornado) (158–206 mph (254–331 km/h)). The licensee states that the reactor building walls and ceiling consist of an I-beam structure with additional angular bracing bars. The building can withstand the live wind loads up to 49.21 pounds per square foot. According to the licensee, the building and reactor pool are constructed of a reinforced steel frame and a poured concrete floor. Therefore, a direct hit by a tornado is not expected to result in damage to the pool integrity or core structure. The SAR also states that no damages resulting from high winds or tornadoes have occurred on the MSTR site since its construction.

The NRC staff concludes that the licensee has provided sufficient historical meteorological data in the SAR and in its response to RAIs to characterize the reactor site. These metrological data will enable the licensee to predict the meteorological impacts on reactor safety and operation. They also provide the licensee sufficient information to analyze the conservative dispersion estimate for postulated airborne release in the unlikely event of a radiological accident. Based on the above information and its independent review of meteorological data maintained by the National Oceanic and Atmospheric Administration, the NRC staff concludes that no weather-related events of credible frequency or consequences will render the MSTR unsuitable for use during the license renewed period of operation. Chapters 3 and 13 of this SER discuss the destruction of the pool integrity, causing a massive loss of pool water, and provide an evaluation demonstrating that an instantaneous loss of pool water would not result in core damage.

2.4 <u>Hydrology</u>

Section 2.3 of the SAR provides detailed hydrology information for the reactor site. The drinking water for the City of Rolla comes mainly from wells that are cased for varying depths from the surface. These wells, in turn, are fed by aquifers that generally run within submerged geologic formations, with occasional outcropping in local streams. Surface waters at the reactor site are drained into streams that flow toward the east, eventually emptying into the Meramec River. There are no known uses of this river's water for drinking until it feeds underground deep-driven wells in the suburbs of St. Louis approximately 150 km from Rolla. The Meramec River finally joins the Mississippi River about 19 km south of St. Louis.

The licensee states that all liquid radioactive releases are analyzed to ensure compliance with regulatory requirements before release to the sanitary sewer system. No direct discharge occurs to the surrounding waterways. In the event of an inadvertent release or leakage of primary coolant, existing procedures require significant dilution before any of the affected water would be used for potential human consumption. The release of liquid radioactive waste is discussed in Section 11.2 of this SER.

Based on the above considerations, the NRC staff concludes that the reactor site hydrology neither poses a significant risk to the continued safe operation of the MSTR nor provides a credible pathway for contamination of the local water supply.

2.5 Geology and Seismology

2.5.1 Geology

Section 2.5.1 of the SAR describes the geology for the MSTR site. The licensee states that Rolla is located toward the northern edge of the Ozark uplift. The sedimentary rock section in this area averages about 1700 ft (518 m) in total thickness. The geographic center of the Ozark uplift lies to the southwest of Rolla. The regional dip in the rocks is toward the northwest, with a very gentle gradient (less than 1 degree). Local sink structures developed in the formations, causing high local dips and even faulting. Soils developed on the surface are mostly of the silt loam type. In floodplains and channels of larger streams, deposits of pure quartz sands and gravels are locally developed.

2.5.2 Seismology

Section 2.5.2 of the SAR describes detailed seismology for the MSTR site. The State of Missouri is divided into six seismic districts (New Madrid, St. Mary's, St. Louis, Hannibal, Springfield, and Northwestern). The City of Rolla is not located in any of these districts, but is located approximately at the center of a square formed by connecting the Springfield, St. Mary's, St. Louis, and Northwestern districts. The MSTR site is located within the Central Stable Region with low probability of seismicity. There is no recorded instance of earthquakes in Rolla reported in the past 140 years. The nearest one, about 50 mi northwest of Rolla (near Camdenton and Lake of the Ozarks) occurred in 1992 with a magnitude of 3.1 (Richter scale). The maximum historic earthquakes in the western section of the region were Modified Mercalli intensity XII in 1811, near the community of New Madrid. Approximately 60 percent of the seismic activity in Missouri originates in the New Madrid district (formed by portions of Missouri, Arkansas, Illinois, Kentucky, and Tennessee), which is 153 miles from the MSTR.

2.5.3 Conclusions

The NRC staff concludes that the licensee has provided sufficient information on the geological features of the region surrounding the MSTR and the potential for seismic activity. Information provided by the licensee, and corroborated by independent reports, indicates that damaging seismic activity for this region during the period of this license is unlikely. Furthermore, as stated in Sections 3.4 and 13.1.8 of the SAR, the worst consequence of an earthquake would be the destruction of the pool integrity, causing a massive loss of pool water. As analyzed in Chapter 13 of this SER, an instantaneous loss of pool water would not result in core damage. Based on this, the NRC staff concludes that there is no significant likelihood that the public would be subject to undue radiological risk following seismic activity.

2.6 Conclusions

The NRC staff concludes that the reactor site has experienced no significant geographical, meteorological, or geological change; therefore, the site remains suitable for continued operation of the MSTR. The infrequency of tornadoes and earthquakes and the robustness of the facility continue to make the site suitable for operation of the MSTR. Hazards related to industrial, transportation, and military facilities will not pose a significant risk to the continued safe operation of the MSTR. Furthermore, the demographics of the area surrounding the reactor have not changed in any way that significantly increases the risk to public health and safety from continued operation of the MSTR.

3. DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.1 Design Criteria

The engineering design criteria for structures, systems and components (SSCs) ensure that they will perform their intended functions to protect the reactor facility, personnel, environment, and the public during normal and abnormal operations. The NRC staff evaluated the specific criteria for safety-related SSCs, including the core support structure, fuel and its cladding, reactor safety system, reactor pool, and reactor building, to ensure the following:

- The reactor building will withstand internal damage from radiation, temperature, and vibration and external damage from meteorology, hydrology, and seismology to protect the reactor from likely detrimental conditions.

- The core support structure will maintain its geometry, orientation, and structure integrity. Chapter 4 of this SER discusses the core support structure design.

- The fuel design and cladding will withstand all credible environmental and radiation conditions during their life cycle. Chapter 4 of this SER discusses fuel design constraints.

- The reactor control system (RCS) will provide safe reactor shutdown and continued safe conditions. Chapters 4 and 7 of this SER discuss RCS design constraints.

- The reactor pool will provide adequate shielding of radiation emitted from the reactor core to control personnel exposure to radiation and provide physical protection of the reactor core from external events.

3.2 Structure Design

According to the SAR, the reactor building was designed and built to meet or exceed building code requirements. The SAR states that in 1999, a structural assessment of the reactor building was conducted by the senior students, who concluded that the reactor building "is far over designed in terms of structural strength and has more than adequate excess capacity to handle a standard 5-ton crane."

Section 3.1 of the SAR describes the structural design of the reactor building. According to the SAR, the reactor building is a rectangular structure 15 m by 10 m by 10 m high. An office/reception/entrance area was added to the building in 1980. The main floor contains a reactor room, control room, counting room, and office space. The reactor is housed in the reactor building, which has a double wall that is constructed of insulated steel frame and is designed to prevent leakage. The doors and windows are weather-stripped and caulked. The vents of the ventilation system automatically close when it is shut down, such as during an abnormal situation, providing confinement of the building air. While the ventilation system is operating normally, a negative pressure is maintained within the reactor building.

The NRC staff notes that the reactor core is located near the bottom of a water-filled pool formed by a reinforced concrete shielding structure. The core and control systems are suspended from a bridge that rides on rails above the reactor pool.

TS 5, 5.1.1, and 5.1.2 contain features applicable to the MST site, as described below:

> TS 5. Only those design features of the facility describing materials of construction and geometric arrangements, that if altered or modified would significantly affect safety and that are not included in Sections 2, 3, or 4 of the TS, are included in this section.
>
> TS 5.1.1. The Nuclear Reactor Building is located on the east side of the Missouri University of Science and Technology campus near 14th Street and Pine Street in Rolla, MO.
>
> 5.1.2. The reactor is housed in a steel-framed, double-walled building designed to restrict leakage. Air and other gases may be exhausted through vents in the reactor bay ceiling 9.1 m (30 ft) above grade. The Reactor Building's free volume is approximately 1700 m^3.

The design criteria are based on applicable standards, codes, and criteria and provide reasonable assurance that the facility SSCs have been built and will function as designed and required by the analyses in the SAR. The NRC staff concludes that the licensee provides sufficient information regarding structure design and has a TS-required process to control design aspects of the facility. Hence, the NRC staff concludes that the MSTR design criteria provide reasonable assurance that the public will be protected from radiological risks resulting from operation of the reactor facility.

3.3 Meteorological Damage

Section 3.3 of the SAR states that very few extreme wind conditions, such as tornadoes or hurricanes, are seen in Rolla, MO. The historical data for the MSTR provided by the licensee in response to the NRC staff's RAI show that tornadoes and hurricanes have had no impact on the reactor structure during the past 40 years. According to the SAR, the reactor building is constructed of a reinforced steel frame and poured concrete floor. Furthermore, the thick concrete walls of the reactor pool (see Section 4.3 of the SAR) and the water within the pool provide additional protection to the reactor core from direct wind damage or debris impact. The licensee states that the university's emergency sirens or a weather alert radio also provide advanced warning of severe weather to ensure that the reactor staff takes appropriate emergency actions.

The NRC staff concludes that even though meteorological damage to the reactor pool walls is very unlikely, it could cause a loss-of-coolant accident. However, Chapter 13 of this SER shows that a loss-of-coolant accident in the MSTR will not lead to fuel failure. On the basis of these considerations, the NRC staff concludes that the risk of metrological damage to the reactor facility is not significant.

3.4 Water Damage

Section 3.3 of the SAR states that the MSTR facility is on sloping terrain, well above any floodplain. Surface drainage from the site is toward the east to Frisco Lake. Therefore, the NRC staff concludes that no significant damage to the building is expected due to flooding in the area. Section 2.4 of this SER presents the detailed analysis of hydrology for the reactor site.

3.5 Seismic Damage

As described in Section 2.5.2 of the SAR, the MSTR is located in a region of minimal seismic activity. The data provided in the SAR show that no severe earthquakes have occurred in this region that would have damaged the reactor building. Furthermore, a recent assessment indicated that the building would withstand an earthquake of the size recommended in the local building code. According to the SAR, if an earthquake were to cause catastrophic damage to the reactor pool walls, a loss-of-coolant accident would occur. However, it concludes in Chapter 13 of the SAR that a loss-of-coolant accident in the MSTR would not lead to core damage, and mechanical damage to fuel cladding would release only a small fraction of the fission product inventory. On the basis of these considerations, the NRC staff concludes that the risk of radiological hazard resulting from seismic damage to the reactor facility is not significant.

3.6 Systems and Components

The MSTR uses a number of diverse systems to reduce and control the potential exposure to radioactivity as a result of reactor operation. These systems include the fuel and its cladding, the control rod scram system, and the confinement and ventilation systems.

Section 4.2.1 of this SER discusses the fuel design, including the fuel cladding requirements. Chapter 13 of this SER examines accident scenarios, and Section 16.1 considers aging issues associated with the fuel cladding. TS 2.1 and 2.2 specify the temperature limits for the fuel cladding to ensure that the integrity of fuel cladding is maintained. TS 3.7.1 specifies the limits of the reactivity in each experiment to ensure that the limiting safety system settings (LSSSs) will not be exceeded. These discussions affirm that the fuel-cladding design basis and related TS are adequate to ensure that fuel-cladding integrity is maintained to guard against an uncontrolled release of fission products under all credible circumstances.

The reactor safety system consists of the control rods, control rod electromagnets, and safety-related instrumentation and controls (I&C). Section 4.2.2 of this SER discusses the control rod design. Section 7.2 examines the design requirements of the safety-related I&C, and Section 16.1 considers aging issues associated with the reactor safety system. TS 3.2.3 requires that the standard rods fully insert to the reactor core from the full-out position in less than 1 second to ensure that the safety limit will not be exceeded in a worst-case delayed critical transient. TS 4.2 specifies the surveillance requirements for the reactor control and safety system. TS 5.3.3 specifies the dimensions, materials, and cladding thickness for the control rods. TS 5.3.4 specifies the speed, a maximum vertical travel, and indicators for the control rods. Again, these discussions affirm that the reactor safety system design bases and related TS provide reasonable assurance that the reactor safety system will function as designed to ensure safe operation and safe shutdown of the reactor.

The confinement and ventilation systems are designed to control the level of airborne radiation in the reactor bay and to discharge facility air at the top of the reactor building. Sections 6.1 and 9.1 of this SER discuss the reactor building ventilation system to ensure that the discharge of air during operation maintains a slight negative pressure in the reactor bay and controls argon-41 concentrations within the bay and at the site boundary to within all applicable limits. TS 3.4, 3.5, 4.4, 4.5, and 5.1 specify that the confinement and ventilation systems must maintain in-leakage to the reactor bay to control the release of radioactivity to the environment. These discussions indicate that the reactor confinement and ventilation systems design bases and related TS provide reasonable assurance that the confinement and ventilation systems prevent any significant radiological risk to the health and safety of the public.

3.7 Conclusions

On the basis of the above considerations, the NRC staff concludes that the design and construction of the MSTR is adequate to withstand and/or ensure safe shutdown as a result of all credible and likely wind, water, and seismic events associated with the site. The design and performance of safety-related systems and components have been verified through safe operation during the period of the current license and routine NRC inspections. Accordingly, the NRC staff concludes that the reactor systems and components are adequate to provide reasonable assurance that continued operation will not cause significant radiological risk to the health and safety of the public, personnel, and the environment.

4. REACTOR DESCRIPTION

4.1 Summary Description

The MSTR is a heterogeneous, swimming-pool type research reactor (nonpower reactor) licensed to operate at power levels of 200 kilowatts thermal (kW(t)) or less. The fuel, control rods, core configuration, and control instrumentation are very similar to other research reactors operating throughout the world.

The reactor core is located near the bottom of a water-filled pool. It is cooled by natural convection, moderated by water, and reflected by water and graphite. The core and control systems are suspended from a bridge that rides horizontally on rails above the reactor pool that allows for positioning in different locations in the pool.

4.2 Reactor Core

Section 4.2 of the SAR describes the MSTR reactor core. The reactor core consists of various components, including fuel elements, a 54-hole grid plate, and control rod elements that connect to control rods. The fuel elements and control rod elements can be arranged in the grid plate to obtain diverse radiation fields. Each unique arrangement is known as a core configuration. The most common one is the core-101 configuration that consists of 14 fuel elements and 4 control rod elements. The licensee has maintained the core-101 configuration since the conversion from HEU to LEU in 1992.

The limitation for a core configuration is given in TS 3.1, “Reactor Core Parameters,” as follows:

1) The maximum excess reactivity for reference core conditions with secured experiments and experimental facilities in place shall be no more than 1.5% Δk/k

2) The minimum shutdown margin under reference core conditions with secured experiments and experimental facilities in place, and with the highest worth control rod and the regulating rod fully withdrawn, shall be no less than 1.0% Δk/k.

3) The excess reactivity limit (Section 3.1(1)) and shutdown margin limit (Section 3.1(2)) may be temporarily exceeded following a core configuration change under the following conditions:

   a) reactor power is limited to 2 kW,

   b) reactor operations are limited to the measurement of excess reactivity, control rod worths, and shutdown margin, and

   c) the reactor is immediately shut down upon discovery of excess reactivity or shutdown margin being in violation of the limits specified in Section 3.1(1) or Section 3.1(2). In such an instance, a core configuration change shall be implemented with the intent of meeting the limits specified in Section 3.1(1) and Section 3.1(2).

4) The reactor shall be operated only when all lattice positions internal to the active fuel boundary are occupied by a fuel element, a control rod fuel element, or an experimental facility.

TS 3.1(1) ensures that a sufficient excess reactivity is needed to provide for temperature effect override, xenon override, and operational and experimental flexibility. Chapter 13 of the SER discusses an insertion of excess reactivity.

TS 3.1(2) ensures that the minimum shutdown margin provides assurance that the reactor can be shut down from any operating condition and remain shut down after cooldown and xenon decay, even if one control rod should become stuck in the fully withdrawn position.

TS 3.1(3) provides for operational flexibility during measurements of excess reactivity and shutdown margin.

TS 3.1(4) precludes the possibility of having an internal vacancy into which a fuel element could be inadvertently inserted.

Based on above discussion, the NRC staff concludes that those related TSs ensure that the MSTR core can operate safely without undue risk to the health and safety of the public during the period of the renewed license. Therefore, those TSs are acceptable to the NRC staff.

4.2.1 Reactor Fuel

The MSTR uses four types of fuel elements in the reactor core: standard fuel elements, half-fuel elements, control rod fuel elements, and irradiation fuel elements.

4.2.1.1 Standard Fuel Elements

Section 4.2.1.1 of the SAR describes a standard fuel element. A standard fuel element contains 18 fuel plates fastened together with aluminum side plates. Each is 34.25 in. (87 cm) tall and has a square 3-in. by 3-in. (7.62-cm by 7.62-cm) cross-sectional area. The nose piece on all of the fuel elements has a circular cross-sectional area that allows for insertion into the grid plate. A fuel element weighs about 11 pounds (5.0 kilograms) in air or 7 pounds (3.2 kilograms) in water. Each fuel plate consists of U_3Si_2-Al fuel "meat" sandwiched in Type 6061 aluminum cladding. The "meat" is approximately 0.02 in. (0.05 cm) thick, 2.4 in. (6.10 cm) wide, and 24 in. (61 cm) tall. The overall plate thickness is 0.05 in. (0.13 cm), in which the cladding is 0.015 in. (0.038 cm). The fuel used at the MSTR is enriched to 19.75 percent uranium-235, and each plate contains 12.5 grams of uranium-235. The active length of the fuel is 24 in. (0.61 m). In addition, the design of the grid plate and the standard fuel elements ensures that the fuel-bearing plates are spaced uniformly to provide a 0.124-in (0.315-cm) water gap between the fuel elements in the core. Both ends of the fuel elements are open to allow for the natural circulation of water to cool the fuel.

4.2.1.2 Half-Fuel Elements

Section 4.2.1.2 of the SAR describes half-fuel elements. Half-fuel elements are "identical to the standard fuel elements." According to the licensee, "9 of the 18 plates contain fuel, and the other nine are dummy and contain only aluminum. Depending on the element, either the front or rear nine plates are fueled."

4.2.1.3 Control Rod Fuel Elements

Section 4.2.1.3 of the SAR describes control rod fuel elements. Four fuel elements for the control rods are identical to the standard elements, with the exception that the central eight plates have been removed to accommodate a guide tube for control rods. The licensee states in the SAR that "The guide tube prevents the control rod from coming in contact with fuel plates."

4.2.1.4 Irradiation Fuel Element

Section 4.2.1.4 of the SAR describes the irradiation fuel element. The irradiation fuel element is used for irradiations within the core. It is identical to the standard fuel element, except that eight fuel plates are removed and replaced by two dummy fuel plates and a 1.17-in. (2.97-cm) by 2.36-in. (5.99-cm) space for accommodating irradiation samples.

4.2.1.5 Fuel Design Considerations

The design features of the fuel elements are given in TS 5.3.2 as follows:

> TS 5.3.2 Fuel Elements
>
> 1) Plate fuel elements of the MTR type are used. The overall dimensions of each element are approximately 7.6 × 7.6 × 91.4 centimeters (cm) (3 × 3 × 36 inches (in.)). The active length of fuel is approximately 24 in. and the fuel is clad in aluminum alloy. The fuel elements have 18 fuel plates joined to two side plates. The whole assembly is joined at the bottom to a cylindrical nose piece that fits into the core grid plate. The fuel meat is U_3Si_2 dispersed in an aluminum matrix and is enriched to approximately 20 percent U-235.
>
> 2) Control rod fuel elements are similar to the elements described in (1) with the exception that the center eight plates have been removed and have been replaced with guide plates so that the control rod cannot come in contact with the fuel plates.
>
> 3) Half-fueled elements have nine plates fueled with low-enriched uranium (LEU) (either the front ones or the rear ones, as appropriately marked) and nine dummy (or unfueled) plates.
>
> 1. An irradiation fuel element has six fuel plate positions left unoccupied (plate positions 11 through 16), plates 10 and 17 are unfueled, and all the others (1 through 9 and 18) are fueled.

TS 5.3.2 controls the important aspects of the design of fuel used in the MSTR SAR; therefore, TS 5.3.2 is acceptable to the NRC staff.

TS 2.1, 2.2, 3.2.2, 3.3(1), and 3.3(2) help to protect the fuel integrity and are given as follows:

> TS 2.1. The safety limit shall be on the temperature of fuel element cladding, which shall be 510°C (950°F)
>
> TS 2.2. The LSSS shall be on reactor thermal power, P, which shall be no

greater than 300 kW, or 150% of full power.

TS 3.2.2. The reactor shall not be operated unless the safety system channels presented in Table 3.2 are operable. Values listed in the table are the limiting setpoints. For operational convenience, the actual setpoints may be on more restrictive values.

TS 3.3(1). The reactor shall not be operated unless the water level is at least 4.88 meters (m) (16 feet (ft)) above the core.

TS 3.3(2). The resistivity of the pool water shall be greater than 0.2 megohm-cm as long as there are fuel elements in the pool. This requirement may be waived for a period of up to 3 weeks once every 3 years

TS 2.1 specifies that the temperature of fuel element cladding must not exceed 510 degrees C (950 degrees F) to maintain the integrity of fuel cladding, which is conservatively set 17 degrees C lower than the blister limit of 527 degrees C to provide additional margin, and as such is acceptable to the NRC staff.

TS 2.2 ensures that the maximum cladding temperature of fuel elements is well below 105 degrees C (221 degrees F), or equivalent to the reactor power limit setting of 300 kW. NUREG-1313, "Evaluations of Low-Enriched Uranium Silicide-Aluminum Dispersion Fuel for Use in Non-Power Reactors," issued July 1988, finds that the initial releases of fission products will not occur at cladding temperatures less than the blister temperature of 527 degrees C (981 degrees F). The LSSS specified in the MSTR is much lower than the blister cladding limit in NUREG-1313. The large safety margin between the blister cladding temperature and the LSSSs gives reasonable assurance that the fuel cladding is well protected and that the reactor will continue to operate safely during the period of this license. Therefore, TS 2.2 is acceptable to the NRC staff.

TS 3.2.2 specifies the high-power safety system channels to automatically scram the reactor if its power level were to exceed 300 kW; therefore, TS 3.2.2 is acceptable to the NRC staff.

TS 3.3 will ensure that adequate cooling is provided for the reactor core at all times and that corrosion of the fuel element cladding will be minimized to ensure adequate heat transfer; as such, it is acceptable to the NRC staff.

The NRC staff also reviewed the MTR fuel design in NUREG-1313 and concludes that this type of fuel element has provided safe operation in nonpower reactors. The Ohio State University, the Rhode Island Nuclear Science Center, and the University of Massachusetts at Lowell have used this type of fuel element and acknowledged that this type of fuel element has demonstrated many years of safe operation.

4.2.1.6 Conclusions

The NRC staff finds that the licensee has described the various types of fuel elements used in the MSTR, including appropriate design limits, and the technological and safety-related bases. The NRC staff concludes that the licensee has sufficiently provided information on the constituents, materials, components, and fabrication specifications of the fuel design. The NRC staff notes that this type of fuel element has accumulated safe operating experience at the MSTR and similar research reactor facilities. Furthermore, the NRC staff reviews this type of fuel in NUREG-1313 and concludes it is safe for operations in nonpower reactors. The NRC staff reviews the related TSs and finds that the licensee has included appropriate design limits, safety limits, LSSS, and limiting conditions for operation (LCOs) for the fuel elements. Based on above considerations, the NRC staff concludes that there is reasonable assurance that the fuel will continue to operate safely during the period of the renewed license.

4.2.2 Control Rods

Section 4.2.2 of the SAR provides the detailed description of control rods. The power level and the reactivity in the MSTR are controlled by three shim/safety rods and one regulating rod. Each control rod is sized to fit into the guide tube in the control rod fuel elements, as discussed in Section 4.2.1 of this SER. Each control rod is equipped with position indicators that are accurate to ±0.10 in. (0.25 cm). Lights are provided on the operator's console to indicate the upper and lower position limit for each rod.

According to the SAR, "[E]ach shim/safety rod consists of a grooved, boron stainless steel rod." The nominal dimensions are 0.875 in. (2.23 cm) thick, 2.25 in. (5.7 cm) wide, and 33 in. (83.8 cm) long, with a 24-in. (61-cm) effective poison length. The boron is used for neutron absorption with approximately 1.5 percent natural boron. The reactivity worth of each shim/safety rod varies with the core loading and configuration. For core-101, mode-W configuration (Section 4.2.3 below discusses the reflector modes), the reactivity worth typically ranges from 2.7% $\Delta k/k$ to 3.3% $\Delta k/k$, with the total worth of the three shim/safety rods about 8.7% $\Delta k/k$. The control rods at the MSTR are positioned in the core by an electromechanical linear actuator. The actuator is a ball-bearing type that is screw driven through a gear reduction motor by a low inertia servo-motor. A variable loading ratchet type mechanism connects the screw to the gear reduction unit. During normal operation, the shim/safety rods are driven in or out at a rate of 6 inches per minute (in./min) (15.24 centimeters per minute (cm/min)). When a scram signal is received, the magnets are de-energized and the shim/safety rods drop freely into the core within 1 second. This scram is provided either automatically or manually to maintain the reactor in a safe operating range and for safe shutdown.

The regulating rod, which is used for fine control, is a flattened, stainless steel tube with a wall thickness of 0.065 in. (0.165 cm). The tube's cross-sectional area is 0.875 in. (2.23 cm) wide by 2.25 in. (5.72 cm) long with an oval end. The effective poison length of the regulating rod is also 24 in. (61 cm). The top end of the regulating rod contains a 0.375-in. (0.953-cm) diameter hole to permit free circulation of water through the tube to eliminate any air trappings, which could result in a variable void condition.

A reactivity worth of the regulating rod is less than 0.7% $\Delta k/k$, which varies somewhat with core loading. The regulating rod is permanently fixed to its drive mechanism and travels in or out of the reactor core at a rate of 24 in./min (60.96 cm/min). The regulating rod can be operated manually or automatically for servo-control of reactor power level.

The licensee states that "accidental lifting of a control element out of the core by movement of a shim/safety rod is impossible without first disassembling the rod drive or deliberately omitting the mechanical components." A special adjustable slip clutch, located between the drive motor and the linear actuator, ensures that any excessive loading on the rod drive will cause the clutch to slip, thus preventing rod movement.

The following are LCOs, surveillance requirements, and design specifications related to control rods and their function.

TS 3.2.2. The reactor shall not be operated unless the safety system channels presented in Table 3.2 are operable. Values listed in the table are the limiting setpoints. For operational convenience, the actual setpoints may be on more restrictive values.

Channel	**Setpoint**	**Function**
Manual Scram Button	Not applicable	Scram
Safety # 1	300 kW	Scram
Safety # 2	300 kW	Scram
Reactor Period	5 s	Scram
Bridge Motion	Not applicable	Scram
Loss of Coolant	4.88 m (16 ft) above core	Scram
Log N & Period Not Operative	Not applicable	Scram

TS 3.2.2 specifies LCOs and helps to ensure that the control rods will promptly shut down the reactor upon receiving scram signal; as such, it is acceptable to the NRC staff.

TS 3.2.3. The reactor shall not be operated unless the free-drop time for each of the three shim/safety rods is less than 1 second.

TS 3.2.3 requires that the standard rods fully insert into the reactor core from the full-out position in less than 1 second to ensure that the safety limit will not be exceeded in a worst-case delayed critical transient; as such, it is acceptable to the NRC staff.

TS 4.2.1. Shim/Safety Rods

1) Shim/safety rod drop times shall be measured as follows:

   a) semiannually

   b) for a particular control rod, whenever the magnet assembly is disassembled or reassembled, or if the control assembly is moved to a new grid position

2) The shim/safety rods shall be visually inspected annually for pitting and cracking and whenever rod drop times exceed the LCOs (Section 3.2.3 of these specifications).

TS 4.2.1 specifies the surveillance requirement for measuring the shim/safety rod drop times, as well as for any particular rod, whenever the magnet assembly is disassembled, reassembled, or moved to a different grid position. The surveillance intervals are sufficient to help ensure operability and they are recommended in ANSI/ANS-15.1, "The Development of Technical Specifications for Research Reactors," issued 1990; as such, they are acceptable to the NRC staff.

TS 5.3.3. Control Rods

1) Poison sections of the three shim/safety rods are stainless steel and initially contained approximately 1.5 percent natural boron. The rods' dimensions are 5.7 × 2.2 cm (2¼ × 7/8 in.) and are approximately 83.8 cm (33 in.) long.

2) The poison section of the regulating rod is a stainless steel oval-shaped tube, 25 in. long, has a wall thickness of 0.065 in., and is mechanically coupled to the rod drive.

TS 5.3.4. Control Rod Drive Mechanisms

1) The shim/safety rod drives have a maximum vertical travel of 24 in. and a withdrawal rate of approximately 6 in. per minute. The shim/safety rods are magnetically coupled to the drive mechanisms and drop into the core, by gravity, upon a scram signal.

2) The regulating rod drive has a maximum vertical travel of 24 in. and a withdrawal rate of approximately 24 in. per minute. The regulating rod is mechanically coupled to its rod drive and does not respond to a scram signal.

3) Lights are provided on the operator's console to indicate the upper limit, lower limit, and shim range for each shim/safety rod.

These TS control the important aspects of the design of control rods used in the MSTR SAR; as such, they are acceptable to the NRC staff.

The NRC staff finds that the licensee has described the control rods and drive system used at the MSTR and included sufficient information on the materials and individual components. Based on a review of this information, the NRC staff concludes that there is reasonable assurance that the control rods and associated drives conform to all applicable design bases and can continue to control and shut down the reactor safely from any operating condition. Based on a review of the scram design for the control rods, the NRC staff concludes that there is reasonable assurance that these features will perform as required to ensure fuel integrity and protect public health and safety during the period of this license. After review of the related TS the NRC staff concludes that the licensee has included appropriate design limits, LCOs, and surveillance requirements for the control rods and control rod drives at the MSTR.

4.2.3 Neutron Moderator and Reflector

According the SAR, the reactor pool water serves as both the moderator and reflector and the MSTR can be operated in two reflector modes: the W-mode and the T-mode. In the W-mode, the reactor is rolled away from the thermal column, allowing the reactor to be water-reflected on all sides. In the T-mode, the reactor is positioned such that the rear face of the core is essentially touching the thermal column. In this location, the rear face of the core is reflected by the graphite thermal column. The licensee states in SAR Section 4.2.3 that the core excess reactivity in the T-mode is about 0.4% $\Delta k/k$ higher compared to the W-mode, depending on the particular core configuration.

During operation, the MSTR bridge, from which the reactor core is suspended, is secured in place by two manually operated bridge clamps. Additionally, a bridge motion detector switch is installed and will result in a reactor scram if the bridge is inadvertently moved (TS 3.2.2). TS 3.3(1) through 3.3(3) ensure that the reactor will not be operated if one of the following conditions exists:

- Water level is less than 16 ft (4.88 m) above the core. This condition ensures a sufficient depth of water for radiation shielding and natural convection flow.
- The resistivity of the pool water is less than 0.2 megohm-cm while fuel is present. This condition ensures that water quality control is maintained to prevent the corrosion rate.
- The pool water temperature exceeds 60 degrees F (15.5 degrees C). This condition guarantees that the excess reactivity, mixed-bed demineralizer, and shutdown margin are within the limit.

In addition, TS 4.3, "Coolant System," ensures that the licensee performs water quality surveillance to prevent the deterioration of water over extended periods of time (including when the reactor is not operating). The licensee, through performance of TS surveillance, ensures that the pool water is maintained so that it does not deteriorate and negatively impact the other elements of the reactor core.

Based on the above considerations, the NRC staff concludes that the design limits, LCOs, and surveillance requirements are sufficient for the safe operation of the MSTR during the period of the renewed license.

4.2.4 Neutron Startup Source

One of the primary functions of a neutron source is to provide sufficient counts such that instrumentation will function properly during startup. The licensee states that the MSTR uses a plutonium-beryllium startup source. TS 3.2.1 specifies that a minimum count rate of at least 2 counts per second (cps) on the startup channel to ensure that sufficient neutrons are available for proper operation of the startup channel and for a controlled approach to criticality.

The neutron source used at the MSTR is similar to that used in other nonpower reactors. Based on a review of the information provided by the licensee in the SAR, the NRC staff concludes that the source has sufficient strength to allow controlled reactor startup.

4.2.5 Core Support Structure

The MSTR core consists of the fuel elements, control rods, and, if needed, in-core experimental facilities. Each core component is positioned in the grid plate, which is supported by an inverted aluminum tower suspended from the bridge that spans the reactor pool.

The structural steel bridge (about 11 ft (3.33 m) long and 4.5 ft (1.4 m) wide) is wheel mounted on rails located parallel to the long axis of the reactor pool atop the pool walls. This allows for the bridge to move a distance of about 6 ft (1.8 m) from its normal operating position, thus providing water shielding between the experimental facilities and the reactor core when required. Mechanical stops on the bridge rails limit bridge travel to within the pool area, and any inadvertent reactor bridge movement will result in a reactor scram as indicated in TS 3.2.2. The 5-in.- (12.7-cm)-thick aluminum grid plate has 54 element holes arranged in a 6 by 9 array. Each hole, about 2.42 in. (6.91 cm) in diameter, passes through the grid plate to facilitate coolant circulation through the core. Holes that do not contain an element are not plugged. Smaller auxiliary coolant holes (0.875 in. (2.22 cm) in diameter) are provided between the larger holes to allow coolant flow between the outside plates of the fuel elements. TS 3.1(4) and 4.1.1 specify that all the lattice positions internal to the active fuel boundary must be occupied by a fuel element, a control rod fuel element, or an experimental facility before the reactor can be operated. This specification prevents the possibility of having an internal vacancy into which a fuel element could be inadvertently inserted.

On the basis of its review of the SAR, the NRC staff concludes that the core support assembly accurately positions and aligns the fuel elements for all anticipated operating conditions. Coolant holes in the grid plate allow for sufficient natural circulation coolant flow to prevent loss of fuel integrity and overheating. The grid plate and tower support structure are constructed of aluminum, which is resistant to radiation damage and corrosion. The MSTR core support structure is designed to ensure a stable and reproducible core configuration for all anticipated conditions throughout the reactor life cycle. The licensee's TS justify appropriate LCOs and surveillance requirements for the core support structure. The core support structure has operated satisfactorily during the current plant life, and reasonable assurance exists that it will continue to operate safely during the period of the renewal.

4.3 <u>Reactor Tank or Pool</u>

The rectangular reactor pool at the MSTR is about 19 ft (5.79 m) long by 9 ft (2.74 m) wide by 27 ft (8.23 m) deep and holds about 113,560 liters (30,000 gal)) of water. The pool houses the reactor core, a beam port, and a thermal column. The pool walls are made of reinforced concrete and have a thickness of 12 in. (30.5 cm) at the top of the pool and taper up to a thickness of 22 in. (55.9 cm). The walls are thicker at a beam hole and thermal column end (78 in. (1.98 m) thick). The internal pool walls and floor have several coats of a protective vinyl paint applied to prevent excessive mineral leaching and the leakage of water to the environment. The reactor pool is set in bedrock and has no drains. Pool water suction to the demineralizer system is through piping located 16 ft (4.88 m) above the core. The demineralizer system contains a siphon break that precludes the possibility of pumping or siphoning pool water below this level. If the pool needs to be drained, the fuel elements are transferred to the storage racks located in the fuel storage pit. This pit is formed by a reinforced concrete bulkhead extending 16 ft (4.88 m) above and 3.5 ft (1.07 m) below the pool floor. This bulkhead design ensures that at least 16 ft (4.88 m) of water will cover the stored fuel elements at all times. The fuel storage pit has no drains or drainage pipes that could lead to inadvertent drain down.

At the opposite end of the pool from the fuel element storage pit is the thermal column that protrudes through the pool wall and sits behind the reactor core. The thermal column assembly, as described in SAR Section 10.2.1, consists of a door that opens into the basement experimental level, a graphite assembly, and a shield. The reactor end of the thermal column is covered with a 4-in. (10.2-cm) lead shield to reduce gamma flux. The graphite assembly measures 3.5 ft by 3.5 ft by 5.75 ft (1.1 m by 1.1 m by 1.75 m) and extends from the pool wall by 3.3 ft (1 m). Five horizontal irradiation ports are filled with graphite stringers when not in use. A beam port provides neutron beams for reactor experiments. This port is constructed of aluminum and is closed at the reactor end. Section 10.2.2 of the MSTR SAR describes the detailed beam port.

The licensee states in SAR Section 13.1.3 that the potential for a sudden loss of coolant from the reactor pool is considered to be an extremely remote possibility. Nevertheless, the licensee analyzed the instantaneous draining of pool water. In this case, the reactor would shut down immediately as the result of loss of moderator, and heat transfer would occur by natural convection of the ambient air. The licensee states that the clad temperature would remain well below the melting temperature for the aluminum cladding in this case. Since no forced cooling is used at the MSTR, no loss of coolant flow scenarios are postulated. The MSTR standard operating procedures (SOPs) require that the core be inspected visually before operation in accordance with SAR Section 13.1.4. Any obstruction to the coolant channels would be detected at that time.

A review of 20 years of personnel exposure monitoring (1984–2003) for workers in the vicinity of the pool during normal reactor operations, as discussed in SAR Section 11.1.2, shows that all exposures are well within the limits specified in 10 CFR Part 20.

Radiation monitors, as discussed in TS 3.6.1 will detect radiation levels in the vicinity of the pool. In the case of a significant drop in pool water level, appropriate actions are initiated. In accordance with TS 4.3, the coolant water quality will be checked to ensure that no degradation occurs over extended periods of time, including shutdown times.

On the basis of its review of the SAR, the NRC staff concludes that the reactor pool has been designed to withstand all anticipated mechanical loads and stresses to prevent loss of integrity that could result in a loss of coolant or other malfunction that could interfere with, or prevent, safe reactor operation. The thermal shield and the beam port are designed to ensure safe reactor operation. Control of the water chemistry and the coatings applied to the interior pool wall surfaces will prevent chemical interactions. Based on a review of personnel exposure, the NRC staff concludes that the licensee demonstrates sufficient radiation shielding that allows individuals to work in the vicinity of the pool. The NRC staff also finds that the licensee has justified appropriate LCOs and surveillance requirements for the reactor pool. The NRC staff concludes that the reactor pool will continue to function as designed and the integrity of the structure will not be degraded for the period of the renewal. The design of the reactor pool provides reasonable assurance that there will be no undue risk to public health and safety.

4.4 Biological Shield

The MSTR biological shield consists of the pool water and the reactor pool walls. The coolant in the reactor pool provides the majority of shielding above the reactor core. TS 3.3(1) and 3.2.2 specify that the reactor water level will be maintained with a minimum of 16 ft of water above the core. The reactor will automatically scram when a water pool level drops below 16 ft... The thickness of the concrete walls and experimental facility shielding provide the majority of shielding in the horizontal direction.

The licensee states that a radiation survey in 1999 during a 200-kW(t) period of operation demonstrated the adequacy of the biological shield. Measured dose rates on the midlevel basement shield wall were less than or equal to 0.13 millirem per hour (mrem/h). Dose rates at the lower level basement shield wall were less than or equal to 0.5 mrem/h. Slightly higher dose rates were measured at the thermal column and beam port facilities. The dose rate at the closed beam port face was 5 mrem/h. Some minor streaming was found around the periphery of the thermal column with dose rates of 8 mrem/h. Measurements directly over the surface of the reactor pool revealed dose rates of less than or equal to 5.0 mrem/h. The licensee states that annual radiation surveys confirm that the dose rates are likely the same as the 1999 survey. Chapter 11 of this SER discusses and evaluates radiation protection, the biological shield, and the ALARA program in terms of reducing direct radiation exposure from the reactor core.

Based on the above considerations, the NRC staff concludes that the MSTR biological shielding is adequate to ensure safe operation and provide reasonable assurance that facility personnel will not exceed to the radiation limits specified in 10 CFR Part 20.

4.5 Nuclear Design

The MSTR is operated at a maximum steady-state power level of 200 kW, with a relatively low burnup of less than 10 megawatts thermal per hour per year. The MSTR had previously been converted from HEU to LEU fuels and is presently operating with LEU. The system is designed to have negative temperature and void coefficients of reactivity. The total loss of coolant will remove the principal neutron moderator and shut down the reactor chain reaction. The reactor can be and has been operated with different core configurations. Chapter 13 of this SER discusses the detailed evaluation of step insertions of reactivity.

4.5.1 Normal Operating Characteristics

The core is supported with a 54-hole grid plate that can be inserted for the various items, including fuel elements, experimental facilities, and control rods. The most commonly used core configuration, core-101, has 14 full-fuel elements, 3 control rod fuel elements, and 1 regulating rod fuel element. This configuration has been the standard configuration since 1992, and it represents a minimum core configuration for criticality. The licensee states that there is not enough excess reactivity in this core to become critical if one of the full-fuel elements is replaced with a half-fuel element. TS 3.1 specifies the LCOs, shutdown margin, regulating rod worth limit, and excess reactivity to ensure that the reactor is operated safely and maintained in a shutdown condition.

The fuel elements are removed from the core in two situations: (1) performance of the annual visual inspection of the control rods, and (2) arrangement of a core configuration other than core-101. In both cases, the licensee follows a well-defined procedure for reconfiguring the core. The SOPs address loading fuel to the core. Once criticality is obtained, excess reactivity and shutdown margin are determined. Excess reactivity is increased by adding half-fuel element increments. After reloading the core, the licensee measures the reactivity worth of the control and regulating rods using the positive period and rod drop methods.

The NRC staff reviewed the information provided and found that the licensee considered the limiting core configurations that contained components required for an operable reactor core. The licensee used appropriate input parameters to analyze the reactivity effects of the individual components. The TS include limits on excess reactivity, minimum shutdown margin, fuel lattice positions, and maximum reactivity insertion rate. In addition, the TS specify surveillance requirements for core reactivity parameters, fuel lattice positions, and reactivity worth of the control rods. Based on these considerations, the NRC staff concludes that the licensee has adequately analyzed expected normal operation during the period of the renewed license.

4.5.2 Reactor Core Physics Parameters

The licensee recalculated many of the reactor core physics parameters in 1988[1] in support of the conversion process to change the MSTR from HEU to LEU fuel. The moderator and fuel temperature coefficients of reactivity are both negative at -1.3×10^{-4} Δk/k/°C and -1.1×10^{-5} Δk/k/°C, respectively. Measurements made to confirm the moderator temperature coefficient of reactivity indicated good agreement. The licensee stated that the void coefficient of reactivity is negative, -9.0×10^{-7} Δk/k/cm^3. The xenon poisoning had been previously measured with the HEU fuel after 8 hours of operation and had a reactivity of -2.0×10^{-3} Δk/k. The effective delayed neutron fraction was calculated to be 0.0079, and the prompt neutron lifetime was calculated to be 50 microseconds. Chapter 13 of the SAR states that the reactor period for the maximum credible step insertion of 1.5% Δk/k would be about 6 milliseconds, and the maximum fuel temperature is about 435 degrees C (815 degrees F), which is still distinctly below the blistering temperature of the cladding of 527 degrees C (981 degrees F). The reactor core parameters are similar to those of other research reactors and have shown safe operations for many years. The staff concludes that both the temperature and void coefficients of reactivity are negative, so no unexpected small temperature spike or voiding of the reactor will result in a power excursion.

TS 3.1 limits the maximum excess reactivity for normal operations in the MSTR core to 1.5% Δk/k and states that the reactor may not be operated if any internal lattice position is not filled. The latter ensures that a situation will not occur in which excess reactivity is inadvertently inserted by dropping a fuel element into the center of the core. The value of the core excess reactivity is based on the need to be able to overcome negative reactivity resulting from temperature changes, xenon buildup, and experiments inserted into the reactor. The sum of these three effects adds up to 1.2% Δk/k. An additional 0.3% Δk/k is included for adequate reactor period and operational flexibility. A full-fuel element located at the center of the core has a reactivity worth between 2.5% Δk/k and 5.6% Δk/k; a full-fuel element located on the periphery has a reactivity worth between 0.5% Δk/k and 1.5% Δk/k. Each of the three control rods is worth approximately 3.0% Δk/k, and the regulating rod is worth 0.5% Δk/k.

TS 3.1(2) specifies that minimum shutdown margin of 1.0% Δk/k to ensure that reactor can be shut down from any conditions.

Based on the above considerations, the NRC staff concludes that the nuclear design is appropriate for safe operation of the MSTR.

4.6 Thermal-Hydraulic Design

The MSTR is cooled by natural convection. The operating power is 200 kW, and there is no forced flow through the core. An auxiliary cooling system containing a heat exchanger near the output of the demineralizer tank is available to reduce pool water temperature, if needed. The pool contains approximately 113,560 liters (30,000 gal) of water. This large amount of thermal mass is heated by the reactor, and the heat is transferred to the ambient air. The heat exchange with the ambient air and evaporation is the only mechanism for cooling the reactor pool water.

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1 Covington, L., "A Neutronic and Thermal-Hydraulic Study of the Conversion of the University of Missouri-Rolla Reactor to Low Enriched Uranium Fuel," M.S. Thesis, University of Missouri-Rolla, December 1988.

The coolant velocity through the core is approximately 0.1 meter per second (0.33 feet per second), which is too low to cause damage to the fuel elements.

The licensee performed calculations, considering numerous core configurations, which have shown that the maximum power peaking factor is 2.4, resulting in the maximum heat flux of 2.2 watts per square centimeter (W/cm^2). The calculated peak cladding temperature associated with full power (200 kW) is 90 degrees C (194 degrees F), which is well below the safety limit specified in TS 2.1 of 510 degrees C (950 degrees F). TS 2.2 ensures that the maximum cladding temperature of fuel element is well below 105 degrees C (221 degrees F), or equivalent to the reactor power limit setting of 300 kW.

The NRC staff concludes that the thermal-hydraulic analysis for the MSTR adequately demonstrates that the reactor can operate at its licensed power level with sufficient safety margins in regard to thermal-hydraulic conditions. The analyses are done with qualified calculation methods and acceptable assumptions.

4.7 Conclusions

The NRC staff concludes that the licensee has presented adequate information and analyses demonstrating the ability to operate the MSTR core without undue risk to the health and safety of the public and to the environment. The MSTR TS requirements related to reactor design, reactor core components, reactivity limits, and related surveillance requirements provide reasonable assurance that the reactor will be operated safely during the period of the renewed license.

5. REACTOR COOLANT SYSTEM

5.1 Summary Description

The reactor coolant system of the MSTR consists of the primary cooling system, the secondary cooling system, the primary coolant cleanup system, the primary coolant makeup water system, and the nitrogen-16 control system. The MSTR cooling system serves five major functions as follows:

(1) remove and dissipate heat generated in the reactor
(2) control primary water conductivity and radioactivity
(3) provide radiation shielding of the reactor core
(4) maintain optical clarity of the primary water
(5) provide neutron moderation and reflection in the core

The open, nonpressurized MSTR pool contains approximately 30,000 gal of high-purity, light, demineralized water. The low power level of the reactor core allows for sufficient cooling by natural convection. Heat from the water pool is dissipated primarily by evaporation into the reactor bay and discharged into the environment by the ventilation system. The auxiliary cooling system with a heat exchanger is also available to reduce the water temperature if needed. Release of thermal effluents from the MSTR will not have a significant effect on the environment. The primary coolant cleanup and primary coolant makeup water systems maintain water purity, optical clarity, and inventory of the primary coolant.

5.2 Primary Coolant System

The primary cooling system consists of the reactor pool, the primary pump, and associated piping, valves, and fittings. The primary pump draws water from the reactor pool through the suction line, passes the water through the filter and demineralizer system, and then returns the water to the reactor pool through the return line. In addition, a siphon break located in the demineralizer inlet piping, approximately 4.88 m (16 ft) above the core, prevents the possibility of pumping below that level.

Natural convective cooling is a primary design feature of the reactor as given in TS 5.2 as follows:

> The reactor is cooled by natural convection of light water. The core is submerged in the reactor pool assuring a pathway for natural convection flow. The pool also serves as a heat sink, neutron moderator and reflectors, and radiation shield.

The thermal-hydraulic analysis discussed in Chapter 4 of this SER shows that the heat produced by the reactor can be safely dissipated to the primary coolant by natural convective water flow. The purpose of TS 5.2 is to require prior NRC review and approval before a change is made to the basic arrangement and design features of the reactor coolant system. Because TS 5.2 controls a basic design feature of the reactor coolant system, it is acceptable to the NRC staff.

In addition to its cooling function, the reactor pool normally provides 20 ft (6.1 m) of radiation shielding directly above the reactor core. To ensure that adequate cooling is provided for the reactor core at all times and that there is sufficient biological shielding available, TS 3.3(1) specifies the following:

> The reactor shall not be operated unless the water level is at least 16 feet (4.88 meters) above the core.

Since TS 3.3(1) provides assurance that there is adequate biological shielding and sufficient water available for natural convective cooling of the reactor and as such is acceptable to the NRC staff.

Isolation of the pool lines through the primary valves mitigates the loss of primary coolant as a result of a pipe break or maintenance operation. The reactor operator will immediately recognize a major loss of coolant either by the illuminating indicators on the control panel or by an automatic scram resulting from the control rod drop. TS 3.2.2, Table 3.2, "Safety System Channels," specifies that the "Loss of Coolant" reactor safety system channel will automatically scram the reactor when the water pool level is below 16 ft of water above the core. A daily surveillance channel test is specified by TS 4.2.2(1) for each of the reactor safety system channels. Increased radiation levels on the radiation area monitors (RAMS) will also indicate loss of coolant, as described in Section 11.1.4 of this SER. Since the "Loss of Coolant" reactor safety system channel scram function specified in TS 3.2.2, Table 3.2, and its associated surveillance channel test in TS 4.2.2(1) provide assurance that the loss of coolant from the reactor pool will automatically scram the reactor, these TS are acceptable to the NRC staff.

Natural convection in the reactor pool provides reactor cooling. Therefore, the only parameter that can be used to limit the fuel cladding temperature is the reactor power. The licensee's analysis in Section 4.6 of the SAR shows that at a reactor power of 300 kW, the maximum cladding temperature is well below 221 degrees F (105 degrees C). The LSSS for reactor power is given in TS 2.2:

> The limiting safety system setting shall be on reactor thermal power, P, which shall be no greater than 300 kW, or 150% of full power.

TS 2.2 applies to the scram setpoints, described in Chapter 7 of this SER, for the reactor safety system channels monitoring reactor thermal power, P. TS 2.2 limits the reactor thermal power to 150 percent of full power during an accident so that the convective cooling provided by the water in the reactor pool is sufficient to maintain the maximum cladding temperature well below 221 degrees F (105 degrees C), which ensures that the safety limit is not exceeded. Therefore, TS 2.2 is acceptable to the NRC staff.

The reactor core has a negative moderator reactivity effect that provides an increase in excess reactivity when the reactor pool is at lower temperatures and lower reactivity at higher pool temperatures. Consequently, coolant system TS 3.3(3) specifies the minimum allowable temperature for the reactor pool water during operation:

> The minimum temperature of the reactor pool should be no less than 15.5°C (60°F) when the reactor is operational.

Because the licensee maintains a minimum reactor pool water temperature of 15.5 degrees C (60 degrees F) or greater to ensure that the excess reactivity will not significantly increase, nor will the shutdown margin decrease, TS 3.3(3) is acceptable to the NRC staff.

To minimize corrosion of the fuel element cladding and to minimize neutron activation of dissolved materials, TS 3.3(2) and surveillance tests TS 4.3(1) and TS 4.3(2) set limits on the reactor pool water quality:

TS 3.3(2): The resistivity of the coolant water shall be greater than 0.2 megohm-cm as long as there are fuel elements in the pool. This requirement may be waived for a period of up to three weeks once every three years.

TS 4.3(1): The resistivity of the coolant water shall be measured at least once every two weeks when the reactor is operated.

TS 4.3(2): If the reactor is not operated, conductivity shall be measured monthly.

These TS provide a chemical environment that limits corrosion of reactor pool, fuel cladding, and control and safety rod surfaces and minimizes the radioactivity of the pool water. Therefore, TS 3.3(2) and the associated surveillance requirements TS 4.3(1) and TS 4.3(2) are acceptable to the NRC staff.

5.3 Secondary Coolant System

The licensee states that the natural convection cooling system in the MSTR is adequate to remove the heat produced during extended periods of full-power operation or decay heat following reactor shutdown without a heat exchanger. In addition, a heat exchanger near the output of the demineralizer tank is available to reduce the pool water temperature if needed. Because the capacity (30,000 gal) of reactor pool water is adequate to remove the heat generated from the reactor core by natural convection, the NRC staff concludes that the secondary system is sufficient under licensed operating conditions.

5.4 Primary Coolant Cleanup System

The primary coolant cleanup system consists of a pump; two conductivity cells; a flow meter; a water particulate filter; a mixed-bed demineralizer; and associated valves, piping, and fittings. The pump draws the primary water through the coolant cleanup system at a flow rate of about 30 gal/min. The primary water passes through the filter, conductivity cells, and the mixed-bed demineralizer. The filter removes large particles from the primary water. Pressure gauges measuring the pressure drop across the filter determine when the filter needs to be changed. The conductivity cell located in the outlet piping of the demineralizer measures the conductivity of the primary water to determine whether the water quality deteriorates beyond the setpoint. The conductivity readout associated with an alarm is provided in the reactor control room. Another conductivity cell located between the filter and demineralizer measures the inlet water quality to determine the condition of the mixed-bed demineralizer. The demineralizer removes small particulates and ions from the primary coolant system. Pool water activity is monitored monthly to ensure that no gross pool contamination or fuel cladding rupture has occurred. Liquid effluents, used resins, filters, and various activation products in the coolant system are analyzed for radioactive contamination and approved by the MSTR radiation safety officer (RSO) before discharge.

The temperature of the demineralizer resins should remain below the suggested limit of 140 degrees F (60 degrees C) to prevent degradation of the resin. TS 3.2.1, Table 3.1 specifies that the core inlet pool water temperature RCS channel will automatically provide a rod withdrawal prohibit to the RCS at the core inlet pool water temperature limit of 135 degrees F (57 degrees C) to maintain the integrity of demineralizer resins. This ensures that the demineralizer resin temperature will be kept below its maximum suggested temperature limit, thereby maintaining the quality of the reactor pool water. The core inlet pool water temperature

RCS channel specified in the TS is acceptable to the NRC staff.

The NRC staff has reviewed the design of the primary coolant cleanup system and associated TS and concludes that the system will acceptably control quality of the primary coolant to limit corrosion of the reactor fuel and other systems that contact primary coolant.

5.5 Pool Water Makeup System

The SAR states that the pool water makeup system replaces water that has been lost by evaporation from the reactor pool surface. There are no pool overflow drains. According to the SAR, makeup water supplied by the campus water system is added to the reactor pool after it passes through a particulate filter and a mixed resin bed ion exchange demineralizer. A siphon break is installed between the reactor pool and the makeup system to prevent siphoning potentially contaminated water from the reactor pool into the potable water supply.

The NRC staff has reviewed the design of the primary coolant makeup water system and concludes that the system is sufficient to replace normal primary water loss and will protect against the entry of primary coolant into the city water system.

5.6 Nitrogen-16 Control System

The NRC staff notes that a nitrogen-16 control system reduces the gamma dose rate resulting from nitrogen-16 at the reactor pool surface. Nitrogen-16 is a short-lived, high-energy gamma source (with a half-life of 7.13 seconds) produced by the high-energy neutron irradiation of oxygen in the pool water ($^{16}O\ (n,p)\ ^{16}N$).

As described in Section 5.3 of the SAR, the nitrogen-16 control system consists of two water pumps installed to direct surface water downward above the reactor core. The water pumps draw and discharge water in a tangential direction inside the reactor pool. This imparts a swirling motion to the water, which breaks the large gas bubbles into smaller ones, thereby decreasing the buoyancy vector and increasing the travel time to escape the reactor pool. All piping and pumps for this system are located within the confines of the pool perimeter, so any leakage would be into the pool. When the reactor is at full power with one pump operating, it measures about 3 mrem/h at 1 ft above the pool surface.

TS 3.6.1 and the associated surveillance requirements in TS 4.6.1 discussed in Section 11.1.4 of this SER, provide reasonable assurance that the reactor operator can identify and respond to any radiation hazard resulting from the nitrogen-16 produced during reactor operation.

The NRC staff concludes that the design of the nitrogen-16 control system, along with the licensee's radiation protection program and ALARA program, provides sufficient reduction of radiation fields at the top of the reactor tank from nitrogen-16 to maintain personnel exposures below the limits in 10 CFR Part 20.

5.7 Conclusions

On the basis of the above evaluation, the NRC staff concludes that the design and construction of MSTR coolant systems are adequate to remove heat from the fuel and prevent loss of fuel integrity under normal full-power operating conditions. The reactor pool water and air cooling by natural convection can provide adequate decay heat removal from the reactor core without damaging the fuel. In addition, coolant temperature limits prevent operation above the limits analyzed.

The pool water cleanup system is designed to provide reasonable assurance that the required water quality is maintained and designed to minimize corrosion of fuel cladding, systems, and structure and to keep contamination levels very low such that no malfunction or leakage will lead to an uncontrolled release of reactor coolant or radioactive materials. The nitrogen-16 control system minimizes personnel dose such that the total dose does not exceed the requirements of 10 CFR Part 20.

The related TS provides reasonable assurance that the cooling system will operate as designed and be adequate for normal operations as described in the SAR.

6. ENGINEERED SAFETY FEATURES

6.1 Introduction

The licensee has demonstrated in Section 13.1.1 of the SAR that the consequences from a maximum hypothetical accident (MHA) involving the failure of a fueled experiment do not exceed the regulatory limits to individuals in unrestricted areas and to members of the public without using any engineered safety features. According to the licensee, engineered safety features are not needed since the reactor operates at low power (200 kW) and the design of the reactor building is conservative (discussed in Section 3.2 of the SER). The licensee performed dose calculations for members of the public that demonstrated that doses from a fission product release are within regulatory limits. The licensee's analysis of the loss-of-coolant accident assumed no emergency core cooling system, just natural convection cooling from the reactor room ambient air.

6.2 Engineered Safety Features

The most common engineered safety features found in research reactors are confinement or containment, including an associated ventilation system, and an emergency core cooling system. These engineered safety features and their applicability to the MSTR are discussed below.

6.2.1 Confinement and Ventilation Systems

The MSTR reactor bay ventilation system and reactor building, which act as a confinement, are discussed in Section 9.1 of this SER and are described in Chapter 9 of the SAR. ANSI/ANS-15.1 defines a confinement as an enclosure of the overall facility that is designed to limit the release of effluents between the enclosure and its external environment through controlled or defined pathways. The reactor bay is kept at a negative pressure in relation to the outside ambient air. If an abnormal situation arises, the ventilation system can be automatically shut down, which effectively isolates the reactor bay free air volume from other areas of the reactor building and the environment. The confinement and ventilation systems help to control releases from the reactor building and serve to reduce doses to members of the public during normal operation and potential accident conditions. The NRC staff concludes that the confinement and ventilation systems as used at the MSTR are not engineered safety features.

6.2.2 Containment

Most research reactors can be designed, sited, and operated such that a containment is not required for normal operation or accident mitigation. ANSI/ANS-15.1 defines a containment as an enclosure of the facility designed to be at a negative internal pressure to ensure in-leakage, control the release of effluents in the environment, and mitigate the consequences of certain analyzed accidents or events. Containments are much more robust than confinements. If a facility does not need a confinement engineered safety feature, it also does not need a containment. Staff review of the accidents analyzed in Chapter 13 of the SAR demonstrated that a containment is not necessary for the MSTR.

6.2.3 Emergency Core Cooling System

Section 13.3 of this SER discusses a potential loss-of-coolant accident. Evaluations performed by the licensee show that air cooling after a loss-of-coolant accident is sufficient to remove decay heat from the fuel and prevent loss of fuel element integrity. Therefore, the NRC staff concludes that an active emergency core cooling system is not necessary.

6.3 Conclusions

The NRC staff has reviewed information presented in the licensee's SAR and concluded that the reactor confinement and building ventilation systems, reactor containment, and procedures are adequate to control release of radiological effluents during normal and abnormal operations. The NRC staff concludes there is no need for any engineered safety features to mitigate the consequences of the potential accidents analyzed in Chapter 13 of the SAR.

7. INSTRUMENTATION AND CONTROL SYSTEMS

7.1 Summary Description

The MSTR SAR states that the I&C systems of the MSTR consist of the reactor control system (RCS), reactor protection system (RPS), control console, and radiation monitoring system. The RCS consists of the instrumentation channels, control rod drive circuit and interlocks, and an automatic flux controller. The primary function of the RCS is to provide the operator with the information and capability to safely control the reactor. The RPS consists of scram instrumentations that will shut down the MSTR when certain conditions reach their predetermined limits. The primary function of the RPS is to rapidly place the reactor in a subcritical condition by automatically inserting the shim/safety control rods to prevent fuel damage when specific limits are exceeded. The control console consists of control panels, indicators, and alarms. The function of the control console is to provide the reactor operator with the information status and control capability necessary to safely operate the reactor. The radiation monitoring systems consist of radiation area monitors (RAMs) and continuous air monitors (CAMs). The function of RAMs and CAMs is to provide reliable indication of the presence of radiation or a release of radioactive materials in the reactor building to ensure the safe operation and shutdown of the reactor and protection of personnel. The I&C systems at the MSTR are functionally similar to those at similar research reactors in the United States.

7.2 Design of Instrumentation and Control Systems

According to the SAR, the I&C systems used at the MSTR provide the operator with information needed to properly manipulate the nuclear controls and initiate automatic protective (scram and rundown trip) functions. Tables 7.1 and 7.2 and Figure 7.1 of the SAR describe the nuclear instrumentation channels and the reactor instrumentation protective actions. The NRC staff summarizes the I&C systems in Table 7.1 (see Section 7.2.3 of this SER).

7.2.1 Design Criteria

The I&C systems provide functions as follows:

- information on the status of the reactor
- the means for insertion and withdrawal of control rods
- automatic control of reactor power level
- the means for detecting overpower, excessive rate of change of power, high pool water temperature, loss of operability of the power-measuring channels, and loss of detector voltage and for automatically shutting down the reactor to terminate operation
- radiation monitoring

7.2.2 Design-Basis Requirements

The licensee states that the current MSTR I&C designs are not being changed as part of license renewal. The primary function of the I&C systems is to control the reactor power in a manner by which the temperature of fuel element cladding does not exceed its safety limit. The designs contain appropriate interlocks, redundancy, reliability, and common-mode failure protection. Following are the significant attributes of the system:

- The RCS at the MSTR helps to prevent the operator from unintentionally inserting large amounts of positive reactivity through various interlock systems. TS 3.2.1 specifies these requirements, including the inability to withdraw the control rods with fast reactor period, with high pool water temperature, and without the presence of a minimum signal from the neutron detectors. TS 3.2.1, Table 3.1, specifies the channels and setpoints for the rundown. The rundown is defined as a situation in which the control rod drives automatically insert the control rods into the reactor core to reduce the power level. The rundown can be reset only after the condition causing it has been removed and the reset relay has been manually energized by pushing the reset button.

- The primary functions of the RPS are to automatically insert all of the control rods into the reactor core when certain parameters reach their setpoints, whether limiting safety systems or redundant systems (reactor scram) as specified in TS 2.2 and TS 3.2.2. TS 4.2.1 also specifies the surveillance requirements for the control rods, including visual inspection for pitting and cracking, and the control rods' drop time measurement.

- The radiation monitoring system, RAMs and CAMs, provides the reactor operator with information on the actual radiation environment inside the reactor building, including alarms to warn personnel of dangerous conditions. TS 3.2.1 and TS 3.6.1 specify the radiation instruments to be operable and their setpoints to provide protection against excessive radiation levels for personnel. TS 4.6.1 specifies the surveillance requirements.

- The physical layout of the control console and display instruments places them within sight and easy reach of the reactor operators. All push buttons required for reactor control are located on the console.

7.2.3 System Description

The I&C systems used at the MSTR are composed of process and control instrumentation channels that provide the means to safely control the reactor and to avoid or mitigate accidents. The systems provide the operator with audible and/or visual indications of key operating parameters and the means to manually scram the reactor at the operator's discretion. Actuation of the scram logic will occur automatically if setpoints established in the TS are exceeded. In addition, the I&C systems include various interlocks that prevent a particular action from occurring unless all the prerequisites for that action are satisfied. The required reactor measuring channels consist of startup channel, log and linear channel, linear channel, safety 1 channel, and safety 2 channels. TS 3.2.1, Table 3.1, and TS 3.2.2, Table 3.2, provide the setpoints and their associated functions.

The following describes the instruments that monitor the status of the reactor:

- Startup Channel. Sections 4.2.4 and 7.2.2.1 of the MSTR SAR describe the startup channel. The startup channel indicates reactor power from startup to low power (1×10^{-4} watt (W) to 1 W). The startup channel consists of a neutron detector, high-voltage power supply, signal preamplifier, linear amplifier, log count rate meter, and recorder. A startup neutron source provides an initial population of neutrons in the reactor core. During the startup, the fission chamber is fully inserted near the core, and the channel is able to measure the count rates. TS 3.2.1 specifies that the reactor is automatically shut down, or remains shut down, if the startup channel count rate is less than or equal to 2 cps, or if the startup channel recorder is off. The startup channel described in the MSTR SAR is commonly used in similar research reactors for startup channels. The NRC staff evaluated the startup channel and finds that it is capable of detecting low neutron count rates, and as such the NRC staff concludes that the startup channel is adequate for safe and reliable startup for the MSTR.

- Linear Power Channel. Section 7.2.2.2 of the MSTR SAR describes the linear power channel. This channel consists of a gamma compensated ionization chamber (CIC) detector, a linear pico-ampmeter, and an analog strip-chart recorder. This channel indicates reactor power level in the range of less than 0.1 W to 300 kW and also provides the signal for automatic servo-control of reactor power. TS 3.2.1 specifies this channel to be operable and activate a reactor rundown if the reactor power exceeds 120 percent of the demand power, or if the high voltage supplied to the gamma CIC detector drops to less than 80 percent of its nominal value, or if the recorder is off. The NRC staff evaluated this channel and finds that it has sufficient range and sensitivity to detect linear power, provides adequate signal for automatic servo-control of the reactor, and activates a reactor rundown to protect the reactor power from exceeding its safety limit. Therefore, the linear power channel is acceptable to the NRC staff.

- Log and Linear Power Channels. Section 7.2.2.3 of the MSTR SAR describes the logarithmic power monitoring (log) channel and linear power monitoring (linear) channel. Each channel consists of a detector, high-voltage power supply, signal amplifier, meters, and a chart recorder. These channels indicate logarithmic power (10^{-6} to 140 percent of full-power level), power range (0 to 125 percent of full-power level), and reactor period. The remote meters and the chart recorder are located in the control room. The log and linear channels are independent and redundant. The combination of both channels is commonly used for research reactors. The NRC staff evaluates these channels and finds that they have sufficient range and sensitivity to detect reactor power and reactor period over all regimes of operation analyzed in the MSTR SAR; therefore, it is acceptable to the NRC staff.

- Safety Channels. Section 7.2.2.4 of the MSTR SAR describes the safety channels. The safety channels consist of two redundant channels to provide the reactor scram when reactor power exceeds 300 kW. Each safety channel consists of a gamma uncompensated ionization chamber (UIC), high-voltage power supply, signal amplifier, meter, and a scram circuit. TS 3.2.2 specifies these channels to be operable and the setpoints to include a reactor scram if the reactor power exceeds 300 kW. The licensee states that a scram check of the safety channels is made on each day that the reactor is operated. In addition, the reactor is shut down immediately if the safety channel indicators show a margin drift that compares to other channels, including the linear power channel and the log and linear power channel. The NRC staff evaluates these channels and finds that they have sufficient sensitivity to detect high power level and activate the reactor scram to protect the fuel cladding; therefore, it is acceptable to the NRC staff.

- Servo Amplifier System. Section 7.2.2.6 of the MSTR SAR describes the servo amplifier system. This system permits reactor power to be automatically controlled during steady-state operations. In automatic mode, reactor power is controlled to within ±2 percent of an adjustable setpoint on the linear power channel recorder. The servo system is interlocked so that the power level must be within about ±2 percent of the setpoint before the system may be engaged. If the power level deviates outside of the ±2 percent limit, control of the reactor will revert to manual control, and visual and audible alarms will be actuated. The NRC staff evaluated the servo amplifier system and finds that it has sufficient sensitivity to automatically control the reactor power and provide adequate signal for engaging servo-control of reactor; therefore, it is acceptable to the NRC staff.

Table 7.1 Safety and Control Instrumentation

Situation	Detector	Unit Initiating Action	Resulting Action	Annunciation	Limiting Values[1]
Manual scram	Operator	Scram button	Scram	Yes	Operator
Period ≤5 s	CIC	Log N and period amplifier	Scram	Yes	5 s
High reactor power	UIC	Safety amplifier	Scram	Yes	300 kW
Bridge motion	Motion switch	Motion switch	Scram	Yes	1.3 cm horizontal travel
Log N and period amplifier not operative	Log N period amplifier	Relay	Scram	Yes	
Power demand	CIC	Linear recorder	Rundown	Yes	120% of selected scale
Period ≤15 s	CIC	Period recorder	Rundown	Yes	15 s
Regulating rod insert limit on automatic	Microswitch	Microswitch	Rundown	Yes	

Situation	Detector	Unit Initiating Action	Resulting Action	Annun-ciation	Limiting Values[1]
Low CIC voltage	dc relay	dc relay	Rundown	Yes	400 V
≥120% full power	CIC	Log N recorder	Rundown	Yes	240 kW
High radiation[2,3] at RAM points	GM tubes	Remote area monitoring system	Rundown	Yes	20 mR/h
Any recorder off[4]	Relay	Relay	Rod prohibit	Yes	
Period ≤30 s	CIC	Period recorder	Rod prohibit	Yes	30 s
Log count rate ≤2 cps[3]	Fission chamber	Log count rate system	Rod prohibit	Yes	2 cps
Safety rods below range or regulating rod above insert limit[3]	Microswitch	Relay	Rod prohibit	No	
Loss of coolant ≤16 ft above core	Relay	Relay	Scram	Yes	16 ft
Core inlet water temperature ≥ 57 °C	Thermocouple	Relay	Rod prohibit	Yes	57 °C
Interlock bypassed	Key switch	Key switch		Yes	
Pool water resistivity ≤2 megohm-cm	Resistivity bridge	Relay		Yes	2 megohm-cm
High neutron flux in beam room	BF-3 neutron detector	Relay		Yes	
Evacuation alarm	GM tubes	RAM system	Initiate evacuation sequence both automatic and manual actuation	Yes	50 mR/h

Situation	Detector	Unit Initiating Action	Resulting Action	Annun-ciation	Limiting Values[1]
Airborne particulate radioactivity[5]	GM	Building CAM		Yes	

Notes:

[1] These are limiting values; operational setpoints may be more conservative.
[2] Radiation detector on the reactor bridge causes building alarm.
[3] Indicates that the situation may be key bypassed around safety circuitry.
[4] The drive motor on startup channel recorder may be off.
[5] These are 50 percent of the limits in 10 CFR Part 20, Appendix B, Table 2.

7.3 Reactor Control System

Section 7.2.2.6 of the MSTR SAR describes the RCS. Controls for the reactor and reactor systems, indication of key reactor operating parameters, and a means for selecting the reactor operating mode (manual or automatic) are provided at the reactor console. Control of the nuclear fission process is achieved using four motor-driven control rods (three shim/safety rods and one regulating rod). Shutdown (scram) capability is provided by insertion of the three shim/safety rods. The regulating rod is used for fine control of reactor power and does not have scram capabilities. All four control rods are equipped with console-mounted electronic position indicators that measure the heights of withdrawal of each respective rod in inches. The position indicators are accurate to within about 0.10 in. (±0.25 cm).

According to the SAR, the control rods are driven by an electromechanical linear actuator located at the bridge. An adjustable slip clutch arrangement is incorporated between the drive motor and the linear actuator to ensure that excessive loading on the rod drive will cause the clutch to slip, thereby preventing movement of the shim/safety rods. The clutch is designed so that the force available to insert the rod is always greater than that available for withdrawal, regardless of the clutch adjustment setting. The regulating rod drive assembly is identical to that of a shim/safety rod drive assembly.

The SAR states that shutdown of the reactor (scram) may be accomplished manually, at the discretion of operators, or automatically, in response to a loss of electrical power, movement of the bridge or in the event that predetermined limits of power and/or reactor period are exceeded as specified in TS 3.2.2. The safety amplifier provides the electric current to the electromagnets holding the control rods in position. Upon receipt of a scram signal, the safety amplifier interrupts current to the electromagnets, thereby allowing the rods to drop, by gravity, into the core.

The NRC staff concludes that the design of the RCS ensures that it can maintain the reactor in a shutdown condition, change reactor power, maintain operation at a fixed power level, and select the reactor operating mode as derived from the SAR analysis and in accordance with the TS.

7.4 Reactor Protection System

The primary purpose of the RPS is to initiate, automatically or manually, a reactor scram to protect the reactor. As specified in TS 3.2.2, Table 3.2, the minimum reactor safety channels required to be operational include the following:

- manual scram button
- reactor safety channels (reactor power)
- reactor period
- bridge motion
- loss of coolant
- log N and linear not operative

The manual scram allows the operator to shut down the reactor at any time for any reason that the operator feels is justified. The power-level scrams provide protection to ensure that the power level is limited to protect against abnormally high fuel temperatures. The period scram is provided to ensure that the power level does not increase on a period of less than 5 s. The bridge motion scram shuts the reactor down in the event that the bridge is moved. The pool water level ensures that a loss of biological shielding results in a reactor shutdown. The log N and linear not operative channel ensures that the log N channel and linear channel are operable.

TS 4.2.2 gives the following surveillance requirements for the RPS:

(1) A channel test of each of the reactor safety system channels shall be performed before each day's operation or before each operation expected to extend more than 1 day, except for the bridge motion monitor, which shall be done weekly.

(2) A channel calibration of the reactor power range safety channel and period channel shall be performed annually.

(3) The thermal power shall be experimentally verified annually.

The NRC staff finds that these intervals are consistent with the guidance found in ANSI/ANS-15.1 and the intervals used at similar research reactors. The NRC staff concludes that the specified intervals provide reasonable assurance that I&C component failure and degradation will be detected in a timely manner and that specified calibration frequencies are adequate to prevent significant drift in instrument setpoints and detection ranges.

Based on the above considerations and years of safe operation with the current systems in research reactors and at the MSTR, the NRC staff concludes that the protection channels and protective responses are sufficient to ensure that no safety limit or LSSSs specified in the TS will be exceeded, and that the full range of reactor operation poses no undue radiological risk to the health and safety of the public, the facility staff, or the environment.

7.5 Control Console and Display Instruments

Section 7.3 of the SAR describes the MSTR control console and display instruments. The physical layout of the control console and display instruments is within easy sight and reach of the reactor operator. An annunciator console provides audible and visual warning of off-normal conditions, including the alarm status of various instrumentation. Irradiation facility status, power supplies, safety channels, and pool light controls are available on a rack located at the right of the console. Table 7.1 of this SER summarizes indications, controls, and alarms at the reactor console.

The NRC staff concludes that the annunciator and alarm indications on the control console give assurance that the status of systems important to adequate and safe operation will be presented to the reactor operator. The NRC staff compared the general arrangement and types of controls and displays provided by the control console to those at similar research reactors and found that the designs are similar. The NRC staff observed the control console during a site visit and found that the control console provides the reactor operator with the types of information and controls necessary to facilitate reliable and safe operation of the reactor.

7.6 Radiation Monitoring Systems

Section 7.4 of the MSTR SAR describes the radiation monitoring systems. According to the SAR, three systems (the RAMs, the basement neutron monitor, and the CAMs) provide protection against excessive radiation levels for personnel in the reactor building. Three RAMs are placed at appropriate locations, including the reactor pool area (main level), the demineralizer area (intermediate level), and the experimental area (basement level) to measure the radiation levels. Each consists of a Geiger-Muller detector, a remote readout unit with an audible and visual alarm, and a local readout in the control room. The RAM located near the demineralizer area also monitors the activity of the reactor coolant that is passing through the demineralizer tank. In addition to audible and visual alarms, each monitor will initiate a reactor rundown if radiation levels exceed the 20 mrem/h limit specified in TS 3.6.1. In addition, the RAM located above the reactor pool also initiates a building evacuation alarm if radiation levels exceed the 50 mrem/h limit (TS 3.6.1).

According to the SAR, the basement neutron monitor consists of a BF-3 detector mounted on the wall in the basement experimental area, adjacent to the beam port and thermal column. Its output is displayed on an analog meter located in the control room. The meter output ranges from 0.1 mrem/h to 10,000 mrem/h. If neutron radiation exceeds the predetermined setpoint, audio and visual alarms are actuated on the control console.

According to the SAR, the CAM system consists of a monitor, recorder, and associated alarm and warning circuitry. The function of the CAM is to measure the radioactivity of airborne particulates in reactor bay air. The CAM is equipped with an alarm system to give audio and visual warning if the reading exceeds the alarm setpoint. The CAM system is a standalone unit and is not interfaced with the control console. Chapter 11 of this SER discusses the radioactive airborne effluent, specifically argon-41, and concludes that no changes in reactor operation that would affect offsite or onsite radiation levels are expected as a result of license renewal.

TS 4.6.1 and 4.6.2 give surveillance requirements and interval testing for the radiation monitoring systems. These TS provide reasonable assurance that the radiation monitoring systems will be capable of performing their intended functions

Based on its review of the SAR and observations made during an onsite visit, the NRC staff concludes that the radiation monitoring systems described in the SAR provide reasonable assurance that all anticipated sources of radiation will be identified and accurately evaluated.

7.7 Conclusions

Based on the above discussion, the NRC staff concludes that the nuclear and nonnuclear I&C systems are adequately designed and implemented to provide for safe and reliable startup, operation, and shutdown of the reactor during normal facility operation. The NRC staff concludes that the RPS is adequate to protect fuel cladding and maintain the reactor in a state analyzed in the MSTR SAR. The NRC staff also concludes that the I&C systems used for radiation monitoring are positioned appropriately within the facility, use appropriate detectors and displays, and provide reasonable assurance that facility personnel will be aware of area radiation levels.

8. ELECTRIC POWER

8.1 Normal Electrical Power System

Chapter 8 of the MSTR SAR states that the MSTR receives standard (110/204-volt (V) alternating current (ac), three-phase, 60-hertz) electrical power from the campus substation. This power to the reactor facility is supplied to various low-voltage loads such as lighting, receptacles, and the reactor console. An ac voltage conditioner prevents transient power surges from overloading circuits in the reactor console. Although this circuit has a battery backup supply that could be connected, it is normally not in use.

8.2 Emergency Electrical Power System

As noted in Chapter 7 of this SER, a loss of the normal ac power source will initiate an automatic scram of the reactor by causing the control rods to drop, by gravity, into the core. A loss of electric power places the reactor in a safe-shutdown condition. The licensee states in Section 13.1.2 of the SAR that electrical power is not needed to maintain the reactor in a safe-shutdown condition because the decay heat generated will not cause fuel heating above acceptable levels. Therefore, an emergency source of ac electrical power is not required.

According to the SAR, battery-operated emergency lighting is provided to permit evacuation of the reactor building and the performance of emergency activities within the building in the event of a loss of normal power. Additionally, the security and fire alarm systems have individual battery-operated backup power supplies allowing them to function during a power outage to the MSTR. The licensee states that the RAMs are also powered from an uninterruptible power supply that will supply power for approximately 20 minutes. Finally, as many as six hand-held radiation meters (battery powered) are available for use in monitoring radiation during loss of electrical power to the facility. These meters are calibrated yearly.

8.3 Conclusions

Based on the above discussion, the NRC staff concludes that the facility's normal electrical power system provides reasonable assurance of adequate operation. In addition, the NRC staff concludes that no emergency backup power is needed for the RCS, since any electrical power interruption results in a reactor scram. Additionally, there is no need for forced convective cooling of the core. As discussed in Chapter 13 of the SAR, natural convection of the pool water through the reactor core is sufficient to prevent damage to the fuel cladding.

9. AUXILIARY SYSTEMS

9.1 Heating, Ventilation, and Air Conditioning System

According to Section 9.1 of the SAR, a commercial-grade system provides recirculating conditioned air to the MSTR, and the campus provides steam heat for the reactor building. The licensee states that when the reactor is operational, the radioactivity in the reactor building atmosphere is monitored. If a radiation alarm occurs in the reactor building, the heating, ventilation, and air conditioning (HVAC) exhaust fans are shut down manually by the reactor operator, whereupon the dampers close automatically by springs on loss of air flow. Dilution with the reactor building free air volume (6.1×10^4 cubic ft) is used to ensure that worker and offsite dose rates are within regulatory limits during both normal operations and under accident conditions. The NRC staff has reviewed the history, current practices, and future expectation of operations related to the radioactive effluent and has concluded that the potential dose rates to members of the general public and the reactor staff are well below the regulatory limits specified in 10 CFR Part 20. Chapters 11 and 13 of this SER discuss releases from the HVAC system under normal operation and accident conditions.

The associated TS for the reactor building confinement and ventilation systems are given as follows:

> TS 3.4: Unless the reactor is secured the truck door is to be closed and the ventilation intake and exhaust duct louvers operable or secured in a closed position.
>
> TS 3.5: A ventilation fan with a rated capacity of at least 4,500 cubic feet per minute (cfm) (127.4 m^3/min) shall be turned on within ten minutes after the reactor reaches full power.

The NRC staff finds that these specifications provide assurance that the ventilation and exhaust systems are operable and that the reactor building can be quickly isolated in case of an unexpected release of airborne radioactivity. Therefore, TS 3.4 and TS 3.5 are acceptable to the NRC staff.

TS 4.4 and 4.5 provide the surveillance requirements for the reactor building confinement and ventilation systems as follows:

> TS 4.4: A test shall be performed quarterly to assure that the following equipment is operable or can remain permanently closed: bay door, ventilation inlet and exhaust duct louvers, and the personnel security door.
>
> TS 4.5: Ventilation fans and intake/exhaust louvers shall be visually checked quarterly for proper operation.

The NRC staff finds that these quarterly surveillance tests will verify that the confinement of the reactor bay can be maintained, if confinement is needed, and will ensure that the ventilation fans and closure devices will perform their functions satisfactorily. Therefore, the surveillance tests in

TS 4.4 and TS 4.5 are acceptable to the NRC staff.

The NRC staff has reviewed the design, operation, and TS requirements of the HVAC system. The NRC staff concludes that the HVAC system is adequate to provide controlled release of airborne radioactive effluents during normal operations and in the event of abnormal or accident conditions. The reactor staff, researchers, and the public will be adequately protected from airborne radioactive hazards related to reactor operations. Based on the NRC staff's review of the operational experience of the facility and TS requirements for operability and testing of the system, the NRC staff concludes that degradation of components will be detected; therefore, there is reasonable assurance that the HVAC system discussed in this section of the SER can continue to operate safely, as limited by the TS, for the proposed license renewal period.

9.2 Handling and Storage of Reactor Fuel

According to Section 9.2 of the MSTR SAR, two fuel storage racks, submerged a minimum of 16 ft beneath the pool water surface, permit temporary storage of up to 30 fuel elements in the fuel storage pit. The fuel storage pit is a reinforced concrete bulkhead (16 ft above and 3.5 ft below the pool floor) located at the end of the west site pool. The bulkhead and the main water pool are separated by a 16-ft concrete wall. Since the wall is 16 ft above the pool floor, natural circulation occurs between the reactor pool and the storage pit. The SAR states that the fuel storage racks ensure that stored fuel elements will not become critical and will not reach an unsafe temperature. The NRC staff has reviewed the criticality measurements by the licensee from the previous HEU fuel elements and found that the k-effective was less than 0.6 for the loaded fuel pit. Because the reactivity of the LEU fuel elements is very similar to HEU fuel elements, the k-effective of 0.6 for the load fuel pit is acceptable to the NRC staff. The fuel is cooled by natural circulation of the reactor pool water. The licensee states that use of the fuel storage racks allows the licensee to remove entire irradiated fuel elements from the core without using fuel handling casks. In addition, the main pool water can be drained for maintenance, while the storage racks can hold all fuel elements in a safe condition.

The licensee states that the fuel handling tool allows the reactor staff to individually move an irradiated fuel element within the core and place it into and out of the storage pit. The purpose of the fuel handling tool is to grasp, move, and position fuel elements under the water to minimize radiation exposure to the reactor staff during fuel handling. It also helps to prevent mechanical damage to the fuel elements. When not in use, the fuel handling tool is kept secured.

TSs related to the fuel handling and storage of the MSTR are given as follows:

- TS 3.3(1) specifies that the reactor shall not be operated unless there are at least 16 ft of water above the core. This condition ensures a sufficient depth of water for radiation shielding and for a natural convection flow, and as such is acceptable to the staff.
- TS 3.3(2) requires the resistivity of the pool water to be greater than 0.2 megohm-cm while fuel is present. This condition ensures that water quality control is maintained to minimize the corrosion rate; therefore, TS 3.3(1) is acceptable to the NRC staff.

- TS 5.4, "Fissionable Material Storage," requires the neutron multiplication factor (k-effective) of the fully loaded storage pit to be less than 0.9 under any conditions. Because TS 5.4 ensures that the reactivity of the fully loaded storage pit is subcritical at all times, it is acceptable to the NRC staff.

- TS 4.3 specifies the surveillance requirements for the reactor pool water. This TS ensures that water quality does not deteriorate over extended periods of time even if the reactor is not operated. As such, TS 4.3 is acceptable to the NRC staff.

- TS 6.1.3(3) requires that any fuel movement be performed in the presence of a senior reactor operator in accordance with 10 CFR 55.13, "General Exemptions." The presence of a senior reactor operator during fuel movement meets the requirement of 10 CFR 55.13 and as such is acceptable to the NRC staff.

Based on its review of the licensee's fuel handling and storage and related TS above, the NRC staff concludes that there is adequate assurance that fuel elements will be stored and handled safely.

9.3 Fire Protection System

The licensee describes the fire protection system in Section 9.3 of the MSTR SAR. Fixed thermal fire detection and manual pull stations are located throughout the MSTR building. The audible and visual alarms are installed in various locations in the reactor facility, including the control room, the front office, the lower level, and the reactor bay. The licensee states that the Rolla Fire Department, located less than 1 mi from the reactor facility, provides fire response. According to the licensee, the Rolla Fire Department personnel receive annual MSTR-specific training in radiological hazards and specific familiarization training, as required under the Emergency Response Plan. Fire extinguishers are located throughout the MSTR. The MSTR operating staff maintains the detection system and tests it for operability twice per year. Hydrants are located near the site boundary of the reactor building. The NRC staff has toured the facility and noted that housekeeping is good and combustible loading is controlled.
The NRC staff has reviewed the fire protection systems at the MSTR, which are typical for a research reactor and consist of fire detection, alarm, and trained response. The NRC staff concludes that the fire protection systems are capable of detecting, alarming, and responding to fires. In addition, as discussed in Chapter 7 of this SER, the NRC staff concludes that failures of the I&C systems because of the potential consequences of a fire will not prevent a reactor scram. Based on its review, the NRC staff concludes that fire protection at the MSTR is acceptable.

9.4 Communication Systems

The MSTR building has public address and intercom capability, which allows communication within the reactor facility. Outside telephone capability also exists. Because these communication systems allow the MSTR staff to meet the communication requirements of the emergency and security plans, the NRC staff concludes that the communication systems are adequate.

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

The license for the MSTR authorizes the receipt, possession, and use of special nuclear and byproduct materials. SNM consists of such material as the uranium-235 in the reactor fuel and fission chambers and SNM produced by operation of the reactor. Byproduct material consists of such material as activation products produced by operation of the reactor in the fuel, experiments and reactor structure, and plutonium-beryllium neutron sources. The facility has designated storage areas for the possession and use of special nuclear and byproduct material, as well as source material under NRC license. Existing operational and health physics procedures control the handling of these materials. These procedures are written to comply with 10 CFR Part 20 and the MSTR ALARA program described in Chapter 11 of this SER. These procedures and processes provide reasonable assurance that no uncontrolled release of radioactive material to unrestricted areas will occur.

TS 5.1.1 defines the nuclear reactor building, and all activities performed within this area fall under the jurisdiction of the reactor license. All of the laboratories are located within the reactor building and are under the jurisdiction of the reactor license. The NRC inspection program has shown that the licensee has procedures and equipment to safely handle licensed material within the restricted area.

The licensee states that all current material possession limits will be carried over into the renewed license unchanged. As is current practice, the NRC staff is clarifying the byproduct possession license condition to allow separation of byproduct material produced in experiments. The license condition on separation of SNM and byproduct material produced in the reactor fuel remains in the license unchanged. The NRC staff has reviewed the possession limits of the license and concludes that they are acceptable for continued operation of the reactor.
Based on the NRC staff's review as discussed above and the acceptable results of the NRC inspection program, the NRC staff concludes that the licensee has procedures and equipment in place to safely receive, possess, and use the materials authorized by the reactor license.

9.6 Conclusions

Based on the above discussions, the NRC staff concludes that the auxiliary systems in place at the MSTR generally enhance safe and reliable operation of the reactor and are acceptable. Additionally, the NRC staff concludes that the related TS and procedures provide reasonable assurance that possession and use of byproduct material and SNM at the MSTR will not pose a significant risk to the health and safety of the public, MSTR personnel, or the environment.

10. EXPERIMENTAL FACILITIES

10.1 Summary Description

The NRC staff notes that the MSTR experimental program provides a wide range of experiments utilizing a variety of experimental facilities. Experiments include, but are not limited to, isotope production, neutron activation analysis, materials study, and medical experiments. In accordance with Subsection 104c of the Atomic Energy Act, as amended (the Act), the facility is licensed as a production or utilization facility, which is useful in the conduct of research and development activities of the types specified in Section 31 of the Act. The licensee may conduct a variety of research experiments that are preapproved by this license or may develop, review, and approve new experiments and experiment facilities using 10 CFR 50.59. However, the experimental program is subject to the limiting conditions for experiments that are in the TS, such as reactivity limits and materials to be activated.

10.2 Experimental Facilities

According to the SAR, the design and location of the different facilities provide unique flexibility for experimenters to obtain various neutron energies and neutron fluxes. Experimenters can extract radiation beams from the reactor core or perform irradiations in the active region of the core. The experimental facilities are comparable in design, construction, utilization, and purpose to experimental facilities at other similar research reactors. The licensee states that the MSTR experimental facilities have been successfully and safely used during the past 20 years of the operation license.

Chapter 13 of this SER discusses accidents such as loss of coolant and reactivity insertion that could occur at experimental facilities. The design, construction, and utilization of the experimental facilities are such that these accidents are extremely unlikely. Chapter 11 of this SER discusses radiation hazards and radiation protection at the MSTR. The licensee states that access to experimental facilities is controlled by the use of operating and radiation protection procedures. The NRC staff has reviewed the use of appropriate radiation detection equipment, radiation protection practices (including the ALARA program), and established experiment review procedures. Based on this, the staff concludes that the licensee has provided reasonable assurance that doses from experimental facilities will meet the requirements of 10 CFR Part 20 for personnel and members of the general public.

The following sections describe the experimental facilities, which include a thermal column, beam port, pneumatic transfer system, sample rotor assembly, core access and isotope production elements, and a void tube.

10.2.1 Thermal Column

Section 10.2.1 of the MSTR SAR describes the detailed information for the thermal column. The thermal column is a large, boral-lined, graphite-filled aluminum container to provide thermal neutrons for experimental purposes. It consists of two sections. The outer section is embedded in the concrete biological shield wall, and the inner portion is placed directly behind the reactor core. The outer section of the thermal column assembly consists of a door, a graphite assembly, and a shield. The thermal column door at the outer section provides access to the graphite assembly. This door is filled with concrete and has a front plate made of boral (B_4C). The RAMs are located in the experimental basement area to alert personnel working in the experimental area of any change in radiation levels, since the open beam port provides a pathway for radiation leakage. These monitors have both local and remote audio and visual alarms, as well as readouts that are displayed in the control room.

The NRC staff has reviewed the description, materials, and historical performance of the MSTR thermal column and concludes that its design is sufficient to limit the expected radiation dose to experimenters, reactor operators, and other personnel to levels below those required by 10 CFR Part 20. The NRC staff concludes that the consequences from any loss-of-coolant accident created from the leakage of the thermal column are acceptable and are bounded by the analysis performed in Section 13.1.3 of the SAR and discussed in Chapter 13 of this SER.

10.2.2 Beam Port

Section 10.2.2 of the MSTR SAR describes the beam port. A permanent beam port penetrates the pool shield wall in the basement experimental area. The beam port is constructed of aluminum and lined with Type 304 stainless steel. There is an additional lining of boral, which reduces the activation of the stainless steel liner and the surrounding concrete. The beam port is closed on the reactor end since it is surrounded with reactor pool water. According to the SAR, experiments performed before the HEU to LEU core conversion demonstrated that flooding of the beam tube with pool water (if a leak developed in the tube wall) will have no significant effect on the reactivity of the reactor core. Beam port plugs are stainless steel discs filled with concrete. This design prevents neutron streaming when the shield plugs are in place. There is also a lead plug for the experimental end of the tube to reduce the gamma radiation to personnel in the experimental area when the reactor is shut down. The licensee states that since an open beam port provides a pathway for radiation leakage, gamma and neutron RAMs are located in the experimental basement area to alert personnel working in the experimental area of any change in radiation levels. Section 11.1.2 of this SER discusses the radiation protection program at the MSTR.

According to the SAR, a two-part shutter assembly inside the tube is used to produce a collimated beam of neutrons. It has a shutter that provides an extension to the beam guide when open and provides additional radiation shielding when closed. Shutter position is controlled remotely from the control room, and shutter position indication is also displayed in the control room.

The NRC staff has reviewed the description, materials, and historical performance of the MSTR beam port and concludes that its design is sufficient to limit the expected radiation dose to experimenters, reactor operators, and other personnel to levels below those required by 10 CFR Part 20. The NRC staff concludes that the consequences from flooding of the beam tube with pool water will have no signal effect on the reactivity of the reactor core.

10.2.3 Pneumatic Transfer System

Section 10.2.3 of the MSTR SAR describes the pneumatic transfer system. The pneumatic transfer system facilitates the rapid transport of small sealed samples to and from the core region. It is used for the production of short-lived radioisotopes, primarily to support neutron activation analysis. Two stainless steel rabbit tubes make up the system that penetrates the core grid plate with the same orientation as the fuel elements on the periphery of the core. Each rabbit tube is made up of two concentric stainless steel tubes; one contains the sample, and the other supplies the gas pressure differential that provides the transport mechanism for samples. According to the SAR, nitrogen gas is used as the transport medium and to displace the air in the tubes to reduce argon-40 activation. This tube gas is vented through a filter to minimize release of particulate activity from the system. The system is controlled either remotely by a computer system located in the control room or manually by bypassing the computer controller.

The NRC staff reviewed the description, materials, and historical performance of the MSTR pneumatic transfer system and concludes that its design, including the system controls and the use of nitrogen gas, is sufficient to limit the expected radiation dose to experimenters, reactor operators, and other personnel to levels below those required by 10 CFR Part 20.

10.2.4 Sample Rotor Assembly

Section 10.2.4 of the MSTR SAR describes the sample rotor assembly. According to the SAR, the purpose of this device is to rotate samples next to the core to produce uniform irradiation of up to eight samples simultaneously. The assembly is placed into the grid plate with the same orientation as a fuel element in an external core position. Positioning of the rotor assembly is by a motor/gear arrangement that is mounted from the reactor bridge. All samples placed in the rotor must conform to the material and reactivity requirements specified in the TS, since they are classified as secured experiments.

Since the experiments irradiated in the sample rotor assembly must comply with TS 3.7.1 and 3.7.2, the NRC staff concludes that the sample rotor assembly is sufficient to limit the expected radiation dose to experimenters, reactor operators, and other personnel to levels below those required by 10 CFR Part 20.

10.2.5 Core Access and Isotope Production Elements

Section 10.2.5 of the MSTR SAR describes the core access and isotope production elements. These elements contain no fuel and are similar in shape to the fuel elements. They are used to provide experimental access to various grid locations within the core. The core access element is used to house only dry samples for irradiation, while the isotope production element can handle both wet and dry irradiation samples.

Because all samples placed in either element must conform to the material and reactivity requirements specified in the TS and both of these elements are classified as secured experiments, the NRC staff concludes that the sample rotor assembly is sufficient to limit the expected radiation dose to experimenters, reactor operators, and other personnel to levels below those required by 10 CFR Part 20.

10.2.6 Void Tube

Section 10.2.6 of the MSTR SAR describes the void tube. The void tube is a hollow cylindrical aluminum tube similar in dimension to a fuel element, which can be filled and sealed with either air or water, or a mixture of both, and can be located at various grid plate locations. It is considered a movable experiment, and it is primarily used to demonstrate concepts associated with criticality in a reactor core.

The NRC staff reviewed the description, materials, and historical performance of the MSTR void tube and concludes that its design is sufficient to limit the expected radiation dose to experimenters, reactor operators, and other personnel to levels below those required by 10 CFR Part 20.

10.3 Restrictions on Experiments

The TS place requirements on the conduct of experiments to help ensure that they are carried out safely. These limitations are placed on reactivity, experimental materials, and failures and malfunctions. Section 13.6 of this SER discusses experiment malfunction.

TS 1.2, "Definitions," defines the terms "experiment," "movable experiment," and "secured experiment" as follows:

> *experiment*—any apparatus, device, or material installed in or near the core or which could conceivably have a reactivity effect on the core and which itself is not a core component or experimental facility.
>
> *movable experiment*—an experiment which is intended to be moved in or near the core or into and out of the reactor while the reactor is operating.
>
> *secured experiment*—any experiment, experimental facility, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining forces must be substantially greater than those to which the experiment may normally be subjected.

These are standard definitions used in research reactor TS; they meet the guidance of ANS/ANSI-15.1 and are therefore acceptable to the NRC staff. The definitions of "secured" and "movable" experiments are used primarily when specifying experimental reactivity limitations.

TS 3.7.1 limits the effect that all experiments can have on reactivity as described below:

> The reactor shall not be operated unless the following conditions exist:
>
> 1) Experiments worth more than 0.4% $\Delta k/k$ shall be:
>
>   a) a secured experiment,
>
>   b) inserted and removed with the reactor shut down, and
>
>   c) inserted and removed from the reactor with a procedure approved by the Radiation Safety Committee.

2) The sum of the absolute values of all experiments shall be no greater than 1.2% Δk/k.

3) Experiments having moving parts shall not have a continuous insertion rate greater than +0.05% Δk/k per second. This requirement does not apply to the experiment's insertion to or removal from the core.

TS 3.7.1 establishes the experiment reactivity worth limit of 0.4% Δk/k, above which experiments must be classified as "secured" and are thereby required to be restrained whenever the reactor is operating. Additionally, all experiments with reactivity worth greater than 0.4% Δk/k must be inserted and removed only when the reactor is shut down using a procedure approved by the Radiation Safety Committee. Since the experiment is held stationary in the reactor, the likelihood that it would fall away from the core to produce an undesirable step increase in reactivity is minimized.

"Movable" experiments that are inserted and removed from the reactor during operation are limited by TS 3.7.1 to an absolute reactivity worth of 0.4% Δk/k. Chapter 13 of the SAR analyzes failure of this type of experiment.

TS 3.7.1 limits the reactivity worth of all experiments in the reactor at any given time to 1.2% Δk/k. This places the upper limit on the reactivity worth of all experiments below the value of assumed reactivity insertion of 1.5% Δk/k used in the accident analyzed in Chapter 13 of the SAR.

Experiments with moving parts that, because of their movement, can create reactivity changes are limited to a reactivity insertion rate below +0.05% Δk/k per second, which is below the insertion rate of 0.074% Δk/k per second for the accident analyzed in Chapter 13 of the SAR that did not lead to significant consequences. Review of the licensee's operating procedures showed that they specify this requirement and that opening of the beam port or thermal column requires either the senior reactor operator or a health physics officer to be present.

Materials used in experiments are controlled in TS 3.7.2, as described below:

The reactor shall not be operated unless the following conditions governing experiments exist:

1) All materials to be irradiated in the reactor shall be either corrosion resistant in reactor pool water or encapsulated within corrosion resistant containers.

2) Explosive material shall not be allowed in or near the reactor unless specifically approved by the Radiation Safety Committee. Experiments reviewed by the Radiation Safety Committee in which the material is potentially explosive, either while contained or if it leaked from the container, shall be designed to prevent damage to the reactor core or to the control rods or instrumentation, and to prevent any changes in reactivity. Known explosives in the amount greater than 25 milligrams

shall not be irradiated in or near the reactor core. In addition, the pressure shall be calculated or experimentally determined such that it will not cause the sample container to fail.

3) Fueled experiments shall not be allowed in or near the reactor unless specifically approved by the Radiation Safety Committee. Fueled experiments in the amount which would generate a power greater than 25 W shall not be irradiated at the MSTR facility. Fueled experiments which generate more than 1 W power shall be irradiated in the reactor pool at least 4.88 m (16 ft) deep under the pool water surface. Fueled experiments which generate less than 1 W power may be irradiated anywhere in the facility. Fueled experiments shall be encapsulated to contain all fission products during irradiation. The encapsulation device shall be designed to prevent degrading of the device due to pressure and temperature of the fueled experiment

4) Cooling shall be provided to prevent the surface temperature of an experiment being irradiated from exceeding the boiling point of the reactor pool water.

TS 3.7.2(1) requires experiments either to involve corrosion-resistant material or use corrosion-resistant encapsulation to ensure that irradiation samples will not contaminate the reactor pool water. This requirement reduces the potential for damage to reactor components from corrosive material and is therefore acceptable to the NRC staff.

TS 3.7.2(2) requires that special case-by-case precautions be taken before irradiation of experiments containing highly reactive chemicals or explosive materials is allowed. The 25-milligram limit is a longstanding limit discussed in Regulatory Position C.2.d of Regulatory Guide 2.2, "Development of Technical Specifications for Experiments in Research Reactors," issued November 1973. Explosive material up to 25 milligrams may be irradiated provided that the pressure produced upon detonation of the explosive has been calculated and/or experimentally demonstrated to be less than half the design pressure of the irradiation container, as discussed in Regulatory Guide 2.2, Regulatory Position C.1.c. TS 3.7.2(2) requirements for experiments containing highly reactive chemicals or explosive materials are within the guidance of Regulatory Guide 2.2 and therefore are acceptable to the NRC staff.

TS 3.7.2(3) ensures that potential releases of radioactive material from fueled experiments are bounded by the dose limits in 10 CFR Part 20 for MSTR staff and members of the public. This includes failures under normal reactor operations, credible reactor accident conditions, and accident conditions in the experiment. TS 3.7.3(3) is acceptable to the NRC staff because it limits doses from potential experiment failure or malfunction to exceed the 10 CFR Part 20 limits and is also bounded by the maximum hypothetical accident discussed in Chapter 13.1 of this SER.

TS 3.7.2(4) requires experiments to be cooled sufficiently to prevent the surface temperature of an experiment being irradiated from exceeding the boiling point of the reactor pool water. Samples or containers irradiated in the pool are in contact with a large heat sink. However, independent cooling of experiments must also be provided, if the surface temperature of the experiment being irradiated could exceed the boiling point of the reactor pool water in order to ensure that departure from nucleate boiling does not occur. TS 3.7.2(4) is acceptable to the NRC staff because it ensures that the surface temperature of the experiment could not exceed the boiling point of the reactor pool water.

TS 3.7.3, "Failure and Malfunction," specifies the design of experiments whose failure or malfunction could adversely affect the proper operation of the reactor controls as described below:

> Experiments shall be designed such that they will not contribute to the failure of other experiments, core components, or cause other perturbations that may interfere with the safe operation of the reactor. Experiments shall be designed such that no credible reactor transient could cause the experiment to fail in such a way as to contribute to a reactor accident.

TS 3.7.3 ensures that a new experimental design will not interfere with the reactor safety systems, contribute to the failure of other experiments, or result in a higher probability of reactor accident. Therefore, TS 3.7.3 is acceptable to the NRC staff.

The NRC staff has reviewed the licensee's limitations on experiments. The licensee's TS and the associated surveillance requirements cover the TS areas suggested by ANSI/ANS-15.1 and NUREG-1537. The technical content of the MSTR TS is consistent with the guidance and provides an envelope of performance against which proposed experiments can be evaluated. Therefore, the licensee's limitations on experiments are acceptable to the NRC staff.

10.4 Experimental Review

The Radiation Safety Committee must approve any new experiment involving the reactor. The reactor staff must review all experiments to ensure compliance with applicable TS. If safety issues are involved, the reactor staff performs its evaluation in accordance with 10 CFR 50.59. Changes to existing experiments that do not impact reactor operations can be approved by the reactor manager, subject to Radiation Safety Committee approval. The reactor operator controls loading, unloading, and movement of experiments affecting the reactivity of the core. Specific restrictions on the types and quantities of materials, the effects on reactivity, the physical locations and restraints, and the administrative procedures for review and approval of experiments allowed in the reactor and its experimental facilities are described in MSTR TS 3.7.1 and TS 3.7.2.

10.5 Conclusions

The NRC staff concludes that the design of the MSTR experimental facilities, combined with the detailed review and TS applied to all experimental research activities, is adequate to ensure that all experiments (1) are not likely to fail, (2) are unlikely to release significant radioactivity directly to the environment, and (3) are unlikely to cause damage to either the reactor or its fuel. Therefore, the NRC staff considers that reasonable controls are in place to prevent a significant release of radiation to the public resulting from experimental programs at the MSTR. In summary, the NRC staff concludes that the MSTR experimental program will not pose a significant risk to the health and safety of the public, facility personnel, experimenters, or the environment during normal operations or credible accidents.

11. RADIATION PROTECTION AND RADIOACTIVE WASTE MANAGEMENT

11.1 Radiation Protection

Activities involving radiation at the MSTR are controlled under the radiation protection program. This program is designed to meet the requirements of 10 CFR 20.1101, "Radiation Protection Programs," and minimize radiation exposure. The regulations in 10 CFR 20.1101 specify, in part, that each licensee shall develop, document, and implement a radiation protection program and shall use, to the extent practical, procedures and engineering controls based on sound radiation protection principles to achieve occupational doses and doses to members of the public that are ALARA. The regulations also provide that the licensee shall periodically (at least annually) review the radiation protection program content and implementation. The NRC inspection program routinely reviews radiation protection and radioactive waste management at the MSTR. The NRC staff notes that the licensee's performance in this area has been acceptable.

11.1.1 Radiation Sources

The NRC staff has reviewed the descriptions of potential radiation sources, including the inventories of each physical form and their locations. This review of the radiation sources included the identification of potential radiation hazards as presented in the SAR and a verification that the hazards were accurately depicted and comprehensively identified.

11.1.1.1 Airborne Radiation Sources

Section 11.1.1.1 of the MSTR SAR describes airborne radiation sources. The SAR states that the primary airborne sources of radiation are argon-41 and nitrogen-16 throughout a normal operation. According to the SAR, argon-41 results from irradiation of the air in experimental facilities and dissolved air in the reactor pool water. The primary means of argon-41 production is the reactor pool. On the other hand, nitrogen-16 is produced when oxygen in the pool water is irradiated by the reactor core.

The licensee states that the dose rate at the reactor bridge with the reactor operating at 200 kW is less than 5 mrem/h, and nitrogen-16, argon-41, and direct radiation from the reactor core contribute to this dose rate. According to the SAR, nitrogen-16 has a very short half-life (7.13 s), and the reactor has a core diffuser system (see Chapter 5 of the SAR) that creates a water circulation pattern designed to suppress nitrogen-16 transported to the surface of the pool and reduce the reactor pool surface dose rate. Therefore, exposure to the public due to nitrogen-16 is negligible. The NRC staff concludes that the licensee's assumption (exposure to the public due to nitrogen-16 is negligible) is adequate due to the short half-life of nitrogen-16 compared to the transit time.

The licensee measured the dose from normal operations to a person in the unrestricted area from argon-41. The concentration of argon-41 leaving the reactor roof fan exhaust where argon-41 is released to the general public was measured at 4.24×10^{-10} microcuries per milliliter (μCi/ml). The calculations very conservatively assume that the reactor operates continuously for a year and that the member of the public stands at the point of maximum exposure continuously for the year. Using the worst-case conditions results in a dose of 2 mrem. This is below the 10 CFR Part 20, Appendix B, Table 2, limit of 50 mrem.

The licensee also measured occupational exposure to argon-41 in the reactor bay that affects the MSTR personnel's dose. Using the worst-case conditions, the conservative measurement of argon-41 concentration in the reactor building from reactor pool release was 1.80×10^{-7} μCi/ml, more than a factor of 10 below the regulatory limit of 3.0×10^{-6} μCi/ml (10 CFR Part 20, Appendix B).

Because the MSTR dose measurements are below the regulatory limits for individual members of the public and the MSTR personnel, the NRC staff concludes that the production of argon-41 is acceptable.

The regulations in 10 CFR Part 20, Appendix B, Table1, column 3, specify that the limit of the argon-41 concentration in the reactor bay is 3.0×10^{-6} μCi/ml. The NRC staff examined the MSTR argon-41 concentration values for the time period 1999–2007 and concludes that those concentration values were less than the Federal limits. Based on a review of MSTR argon-41 data, the NRC staff concludes that the control and radiation protection program for argon-41 is adequate for ensuring safety.

The NRC staff has reviewed the licensee's calculations of the production and release of routine airborne radioactive effluents and the resultant doses to the public and MSTR staff. The staff concludes that routine airborne effluent releases from the MSTR are well within 10 CFR Part 20 criteria and are therefore acceptable.

11.1.1.2 Liquid Radiation Sources

Section 11.1.1.2 of the MSTR SAR describes the liquid radiation source. The NRC staff acknowledges that impurities in the primary coolant become activated by neutrons as they pass through the reactor core. According to the SAR, the licensee controls the impurity levels by maintaining water quality to reduce corrosion and filtering the coolant with the demineralizer system. The NRC staff reviewed the equilibrium concentrations of predominant radionuclides in the primary coolant and concludes that they are within the 10 CFR Part 20, Appendix B, limits for release to the sewer.

Nitrogen-16 also contributes to the dose rate from the primary piping, since this piping carries pool water that has been circulated through the reactor core. The pool surface exposure rate is discussed under Section 11.1.1.1 above. Measurements taken by the licensee at full-power operation indicate dose rates of 1 to 5 mrem/h on the surface of the primary piping, all of which is within a designated and controlled radiation area as specified in 10 CFR Part 20.

The licensee states that primary water will be sampled on a monthly basis for radioactive content to help detect potential fission product leakage from the reactor fuel. This record will be used to determine the total radioactive release if an uncontrolled release of coolant occurs. There is a RAM located near the demineralizer to provide information to operating personnel about the radiation level of primary water piping. TS 3.6.1 specifies that the reactor starts rundown when the demineralizer RAM exceeds the limit of 20 mrem/h.

The NRC staff notes that radiation exposures from these liquid radiation sources at the MSTR are small, and access to them is controlled. Therefore, the NRC staff concludes that these sources do not present a significant hazard to either the operating personnel or the public.

11.1.1.3 Solid Radiation Sources

The SAR states that the solid radiation sources are coming from the fission products in the reactor fuel and nonfuel sources. However, the fuel design and related TS discussed in Section 4.2.1 of this SER ensure that the fuel cladding integrity keeps the fission products inside the fuel element. The SAR also states that spent fuel elements belong to DOE. If needed, those fuel elements will be transferred to DOE in accordance with agreement discussed in Section 1.7 of this SER. Nonfuel sources include activated reactor components, resins from the primary water demineralizer, filters, and irradiated samples. According to SAR, final radioactivity is estimated before experimental irradiations are performed, so both shielding and storage duration requirements will be known. The radiation protection program monitors and controls these sources. The licensee states that solid radioactive waste handling has not resulted in any significant personnel exposure at the MSTR.

The NRC staff notes that radiation exposures from the solid radiation sources at the MSTR are small, and access to them is controlled. Therefore, the NRC staff concludes that these sources do not present a significant hazard to either the operating personnel or the public.

11.1.2 Radiation Protection Program

The regulation in 10 CFR 20.1101(a) requires that each licensee shall develop, document, and implement a radiation protection program. The NRC staff notes that MST has a structured radiation protection program with a health physics staff that is appropriately independent and equipped with radiation detection capabilities to determine, control, and document occupational radiation exposures at the MSTR. The NRC regularly inspects the radiation protection program and finds that the program as implemented meets the requirements of the regulations. SAR Sections 12.1 and 12.2 describe the management of the radiation protection program. Section 11.1.2.3 of the SAR describes the health physics procedures and document control. These procedures include testing and calibration of the monitors and detection instrumentation; administrative guidelines for receiving, monitoring, handling, transporting, and testing radioactive materials; decontamination; investigation; training; the ALARA program; and personnel access. TS 6.4, "Operating Procedures," specifies that radiation control procedures will be maintained and made available to all operations personnel. Any substantive changes to the radiation control procedures must be approved by the Radiation Safety Committee and the Director Nuclear Reactor.

According to the SAR, all personnel entering the facility are issued the appropriate monitoring devices and any required protective clothing. All personnel permitted unescorted access to the MSTR reactor building shall receive training in radiation protection as required by 10 CFR 19.12, "Instruction to Workers." Experimenters permitted unescorted access to the MSTR experimental facilities shall receive additional training to include access control rules, emergency procedures, dosimetry requirements, key checkout and return, security procedures, reactor top safety, communication systems, security door requirements, general checkout procedures when exiting the reactor bay, and emergency equipment location and use.

During the site visit, the NRC staff noted that radiological conditions are posted on appropriate caution signs for required areas within the facility. The licensee stated that experimental and reactor equipment areas are surveyed regularly. According to the SAR, all liquid and gaseous effluents are monitored before release to comply with 10 CFR Part 20 limits.

Section 12.3 of the MSTR SAR states that health physics/radiation protection program procedures are audited annually by the Radiation Safety Committee. This includes all procedures, personnel radiation doses, radioactive material shipments, radiation surveys, and radioactive effluents released to unrestricted areas. According the SAR, the health physics staff maintains radiation protection program records, including radiological survey data, personnel exposure reports, training records, inventories of radioactive materials, environmental monitoring results, and waste disposal records. Records are kept for the life of the facility.
Based on above considerations, the NRC staff concludes that the MSTR radiation protection program as described in Section 11.1 of the SAR complies with 10 CFR 20.1101(a), is acceptably implemented, and provides reasonable assurance that, for all facility activities, the program will protect the MSTR staff, the environment, and the public from unacceptable radiation exposures. Chapter 12 of this SER reviews the organization and oversight of the program.

11.1.3 As Low As Reasonably Achievable Program

To comply with the regulations in 10 CFR 20.1101, the NRC staff notes that the MSTR has established and implemented a policy that all operations are to be planned and conducted in a manner to keep all exposures ALARA. The program to implement this policy is based on the guidelines of ANSI/ANS-15.11, "Radiation Protection at Research Reactor Facilities," issued 1993. The program is applied through written procedures and guidelines described in SAR Section 11.1.3. The NRC staff also acknowledges that all proposed experiments and operational procedures at the MSTR are reviewed by appropriate levels, including the Radiation Safety Committee, for ways to minimize potential exposure to personnel. According to the SAR, the health physics staff participates in experiment planning to minimize both personnel exposure and the generation of radioactive waste. The NRC regularly inspects the ALARA program and has found that the program as implemented meets the requirements of the regulations. The licensee states that the annual exposure history for MSTR personnel for the last 20 years is well below the limits in 10 CFR Part 20. Based on the exposure history for MSTR personnel and the NRC inspection report results, the NRC staff concludes that the MSTR ALARA program as described in Section 11.1.3 of the SAR complies with 10 CFR 20.1101, is acceptably implemented, and provides reasonable assurance that for all facility activities, radiation exposure will be maintained ALARA.

11.1.4 Radiation Monitoring and Surveying

The regulations in 10 CFR 20.1501(a) requires that each licensee shall make, or cause to be made, surveys that—

(1) May be necessary for the licensee to comply with the regulations in this part; and
(2) Are reasonable under the circumstances to evaluate—

(i) The magnitude and extent of radiation levels; and
(ii) Concentrations or quantities of radioactive material; and
(iii) The potential radiological hazards.

The regulation in 10 CFR 20.1501(b) requires that the licensee shall ensure that instruments and equipment used for quantitative radiation measurements (e.g., dose rate and effluent monitoring) are calibrated periodically for the radiation measured.

Section 11.1.4.4 of the SAR shows that the licensee has a comprehensive set of portable radiation survey instruments with sufficient ranges for the various types of radiation. The licensee also has other specialized radiation monitoring equipments, such as a gamma "frisker," portable neutron survey meter, gamma spectrum analyzer, and low- and high-range portable beta-gamma meters. According to the SAR, those monitors are located at strategic points throughout the facility where radiation levels could exceed normal levels. The monitors include local audible and visual alarms. The alarms are set at levels based on anticipated normal or potentially abnormal radiation levels.

The discussion in Section 11.1.1.1 of this SER shows that routine effluent releases are within regulatory limits, and the discussion in Chapter 13 of this SER shows that the consequences of accidents are acceptable. TS 3.6.1 requires sufficient monitors to evaluate potential radiation hazards and also provides the period of time required for a single safety channel out of service. The following requirements for the radiation monitoring system appear in TS 3.6.1:

> The reactor shall not be operated unless the Constant Air Monitor (CAM) is operable and the Radiation Area Monitors (RAMs) located at the reactor bridge, at the demineralizer, and in the basement experimental area are operable. Table 3.3 as specifies the approximate locations, set points and functions. Values listed are the limiting set points. For operational convenience the actual set points may be on more restrictive values.
>
> The reactor may be operated with one or more of the RAM channels inoperable under the following conditions:
>
> 1) The period of operations with the RAM channel(s) inoperable does not exceed 1 week.
>
> 2) A portable gamma radiation instrument is placed in the same vicinity as the inoperable RAM detector(s), with a local audible alarm setpoint of 20 mrem/h or less.
>
> 3) If the inoperable channel is the bridge RAM, the control room operator must be able to visually monitor the radiation level of the portable unit.

Because TS 3.6.1 provides sufficient radiation monitors to protect the MSTR personnel and to prevent a significant release of radiation to the public, and to provide conditions to replace an inoperable channel, therefore, TS 3.6.1 is acceptable to the NRC staff.

Table 3.3-Radiation Area Monitors.		
Location	**Set Point**	**Function**
CAM	1500 cpm	Alarm
Reactor Bridge	20 mrem/h 50 mrem/h	Rundown Building Evacuation
Demineralizer	20 mrem/h	Rundown
Basement Experimental Area	20 mrem/h	Rundown

The following surveillance requirements for the radiation monitoring systems are given in TS 4.6.1:

a. A channel check shall be performed on each gamma RAM channel daily before reactor startup.

b. Calibration of the RAMs shall be performed annually.

The frequency of channel checks and calibration is based on experience and is consistent with frequencies recommended in ANSI/ANS-15.1. The NRC staff concludes that the surveillance requirements in TS 4.6.1 for the radiation monitoring channels are acceptable.

The licensee states that radiation and contamination surveys are both performed on a regular basis (daily, weekly, and monthly) by the health physics staff. To confirm that safe radiation working conditions exist, the licensee states that routine and nonroutine facility operations and contamination-level surveys of specific areas are also performed. In addition, the licensee conducted a pool water analysis to determine whether there is leaking in fuel elements. The licensee states that no fission products for early detection of a fuel leak are found in the pool water. The NRC staff concludes that surveys and observations/corrections from the MSTR are sufficient to ensure that the operational and radiation protection programs help to keep doses ALARA; therefore, the MSTR radiation surveying program is acceptable to the NRC staff. Based on above consideration, the NRC staff concludes that the equipment used by the licensee is appropriate for detecting the types and intensities of radiation likely to be encountered within the facility at appropriate frequencies to ensure compliance with 10 CFR 20.1501(a) and (b).

11.1.5 Radiation Exposure Control and Dosimetry

The licensee states that the radiation surveys performed in 1999 during 200-kW(t) operation show that the dose rates were ≤8 mrem/h in the reactor facility. The licensee also states that radiation surveys are conducted annually, and those results are consistent with the measurements in the 1999 survey. Section 4.4 of this SER provides the detailed results for each location.

The ventilation system as described in Chapter 9 of the SAR and in Section 9.1 of this SER keeps the reactor bay at negative pressure with respect to outside areas and maintains argon-41 and nitrogen-16 levels below the limits prescribed in 10 CFR 20.1201, "Occupational Dose Limits for Adults."

Section 11.1.5 of the SAR states that personnel exposure is monitored by film badges and pocket ion dosimeters, which are assigned to individuals who may be exposed to radiation. These personnel monitoring devices measure all types of radiation found at the MSTR. Section 11.1.6 of the SAR shows the licensee has sufficient equipment for controlling contamination of personnel, the facility, and equipment. Procedures governing the use of this equipment also exist.

The NRC staff reviewed the average and highest annual dose equivalent incurred by MSTR staff and concludes that the licensee demonstrates compliance with the facility's ALARA program as well as the efficacy of the radiation protection and control program. The regulations in 10 CFR 20.1201 limit the annual occupational total effective dose equivalent (TEDE) to less than 5 rem and the shallow dose to the skin of the whole body or any extremity to 50 rem. The NRC staff examined the average individual annual exposures for the time period 1999–2007. The average annual dose over that period was less than 100 mrem, and all MSTR staff received less than the Federal limits. Based on a review of MSTR staff exposures, which are significantly below regulatory limits, the NRC staff concludes that the radiation protection and control program is adequate for ensuring safety.

11.1.6 Contamination Control

Section 11.1.6 of the SAR describes MSTR contamination control. According to the SAR, contamination surveys are performed daily, weekly, or monthly, depending on the frequency with which the radioactive material is used or handled. According the SAR, written procedures control the handling of any radioactive material within the MSTR. The licensee states that the facility surveys have routinely shown no detectable contamination in nonradiological areas of the facility. The NRC inspection program has confirmed that the licensee has effective contamination control. The NRC staff concludes that adequate controls exist to prevent the spread of radiological contamination within the facility.

11.2 Radioactive Waste Management

The purpose of the radioactive waste management program is to minimize radioactive waste and ensure that it is properly handled, stored, and disposed. Section 11.2 of the SAR describes the radioactive waste management program, which addresses solid, liquid, and gaseous radioactive wastes. The NRC staff notes that all radioactive waste handling operations are controlled by procedure and overseen by the MSTR heath physics staff.

TS 6.7.1, "Operating Reports," specifies that, annually, the facility reports the following to the

NRC:

5) A summary of the nature and amount of radioactive effluents released or discharged to environs beyond the site boundary. The summary shall include to the extent practicable an estimate of individual radionuclides present in the effluent. If the estimated average release after dilution or diffusion is less than 25 percent of the concentration allowed, a statement to this effect is sufficient.

6) A summarized result of environmental surveys performed outside the facility.

7) A summary of exposures received by facility personnel and visitors where such exposures are greater than 25 percent of that allowed.

In accordance with 10 CFR 50.36a(2), the licensee shall submit a report to the NRC annually that specifies the quantities of each of the principal radionuclides released to unrestricted areas. Since TS 6.7.1 provides the information specified in 10 CFR 50.36a(2), TS 6.7.1 is acceptable to the NRC staff.

11.2.1 Solid Waste

Section 11.1.1.3 of the SAR indicates that uncompacted solid low-level radioactive waste at the MSTR consists of gloves, pads, used resins, filters, and various activation products from experiments conducted using the MSTR. According to the SAR, these solid wastes typically contain a few millicuries of radionuclide per year. When filled, the low-level waste containers are sealed and transferred to the MST Radiation Safety Office until they are shipped off campus by a licensed carrier to a licensed facility for disposal. Activated equipment and activated irradiation samples are stored in the reactor bay area for reuse or to decay to low-level activity limits. The NRC staff reviewed the MSTR radiation procedures and concludes that procedures at the MSTR are adequate to monitor the radiation exposure from waste storage areas within the facility and to perform required handling operations, such as packaging and transfer, and the preparation of proper documentation associated with shipment.

11.2.2 Liquid Waste

Section 11.1.1.2 of the SAR states that normal operation of the MSTR does not produce significant liquid radioactive waste. Small quantities of liquid waste are periodically generated by minor leakages, the demineralizer, and sampling of the reactor pool and the primary coolant system equipment. According to the SAR, these liquid wastes are collected and stored in a holdup tank until the determination of their final disposition. The holdup tank is sampled and tested for radioactive materials on a monthly basis. The SAR states that liquid radioactive waste from the holdup tank is released to the sanitary system in accordance with written procedures to ensure that they are within the limits stated in 10 CFR Part 20, Appendix B, Table 3.

The NRC staff examined the annual radioactive water wastes released to the sanitary sewer for the time period 1999–2007. The quantities and activities of water released over that period were less than the Federal limits. Based on a review of MSTR water releases, which are significantly below regulatory limits, the NRC staff concludes that the radiation protection and control program is adequate in ensuring safety.

11.2.3 Radioactive Waste Management Conclusions

The NRC staff has reviewed the facility radioactive waste management program and radioactivity waste data from the SAR and concludes that there is reasonable assurance that radioactive waste released from the facility will neither exceed applicable regulations nor pose unacceptable radiation risk to the environment and the public.

11.3 Conclusions

On the basis of its evaluation of the information presented in the licensee's SAR, observations of the licensee's operations, and results of the NRC inspection program, the NRC staff concludes the following:

- The MSTR radiation protection program complies with the requirements in 10 CFR 20.1101(a), is acceptably implemented, and provides reasonable assurance that the NRC staff, the environment, and the public are protected from unacceptable radiation exposures. The radiation protection program is acceptably staffed and equipped. The radiation protection staff has adequate lines of authority and communication to carry out the program.

- The MSTR ALARA program complies with the requirements of 10 CFR 20.1101(b). The NRC's review of controls for radioactive material in the MSTR provides reasonable assurance that radiation doses to the environment, the public, and facility personnel will be ALARA.

- The results of radiation surveys carried out at the MSTR, doses to the persons issued dosimetry, and results of the environmental monitoring program help verify that the radiation protection and ALARA programs are effective.

- The licensee has adequately identified and described potential radiation sources. The licensee sufficiently controls radiation sources.

- Facility design and procedures limit the production of argon-41 and nitrogen-16 and control the potential for facility staff exposures. Conservative calculations of the quantities of these gases released into restricted and unrestricted areas give reasonable assurance that doses to the MSTR staff and public will be below applicable 10 CFR Part 20 limits.

- The facility radioactive waste management program provides reasonable assurance that radioactive waste released from the facility will neither exceed applicable regulations nor pose unacceptable radiation risk to the environment and the public.

12. CONDUCT OF OPERATIONS

The conduct of operations for the MSTR involves the administrative aspects of facility operation and the facility emergency and security plans. The administrative aspects of facility operation are the facility organization, training, operational review and audits, procedures, required actions, reports, and records. This section of the SER addresses Chapter 12 of the SAR and Section 6, "Administrative Controls," of the TS.

The primary guidance for the development of administrative control TS for research reactor operation is ANSI/ANS-15.1. The licensee's TS are based on the 1990 and 2007 versions of the standard. The NRC staff supports the use of ANSI/ANS-15.1 for TS Section 6, as discussed in Chapter 14 of NUREG-1537. The NRC staff used the 1990 and 2007 versions of ANSI/ANS-15.1 in its review of the licensee's administrative controls to determine whether the licensee's proposed TS meet the intent of the guidance and are acceptable.

12.1 Overall Organization

Responsibility for the safe operation of the reactor facility is vested within the chain of command, described in SAR Sections 12.1.1 and 12.1.2 and TS 6.1.1 and 6.1.2. Figure 12-1 shows the organization chart from SAR Figure 12.1 and TS Figure 6-1.

TS 6.1.1 describes the overall structure of the organization as follows:

> The Nuclear Reactor Facility is a part of the Department of Mining and Nuclear Engineering of the Missouri University of Science and Technology.

TS 6.1.2 describes the responsibilities of each of the organizational levels given in the organizational chart as follows:

> The Chair of Mining and Nuclear Engineering is the individual responsible for the reactor facility's licenses (Level 1).
>
> The Director Nuclear Reactor (DNR) is the contact person for the NRC and has overall responsibility for management of the facility (Level 2). The DNR shall have a minimum of 6 years of nuclear experience. The DNR shall have a Bachelor's (or higher) degree in engineering or science. Equivalent education or experience may be substituted for a degree. The degree may fulfill 4 years of the 6 years of nuclear experience required.
>
> The Reactor Manager (Level 3) shall be responsible for the day-to-day operations and for ensuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license and the provisions of the Radiation Safety Committee. During periods when the Reactor Manager is absent, his responsibilities may be delegated to a senior reactor operator (Level 4).
>
> The Reactor Manager shall have 3 years of nuclear related experience. A maximum of 2 years of equivalent full-time academic training may be substituted

for 2 of the 3 years of nuclear-related experience required. As soon as reasonably possible after being assigned to the position, the Reactor Manager shall obtain and maintain an NRC senior reactor operator's license.

The health physicist who is organizationally independent of the Reactor Facility operations group, as shown in Figure 6.1, shall be responsible for radiological safety at the facility. The health physicist may also be the Radiation Safety Officer.

Based on the licensee's organizational chart, the NRC staff concludes that the MST Chancellor has overall responsibility for the entire campus and the Chair of Mining and Nuclear Engineering, Level 1, has responsibility for the NRC license for the reactor facility. The Director Nuclear Reactor is directly responsible for the safe operation and maintenance of the reactor facility. The Director Nuclear Reactor reports to the Chair. The reactor manager is responsible for day-to-day operation of the reactor and reports to the Director Nuclear Reactor. The Radiation Safety Committee provides the safety review and audit functions for the reactor and reports to the Chancellor.

The NRC staff concludes that the RSO, the health physicist, and the health physics staff provide radiation safety and radiation protection services. This function is separate from the reactor operations organization, and the RSO reports up a parallel chain of command through the Vice Chancellor of Administrative Services to the university's Chancellor. The health physics staff reports to the RSO. The RSO has direct access to the Director Nuclear Reactor and is a member of the Radiation Safety Committee. The licensee stated that the RSO can suspend facility operations when necessary. The licensee clarified that the MSTR follows the guidelines of the 1990 version of the ANSI/ANS-15.1 for its full radiation protection program.

Individuals at all levels are defined as being responsible for safeguarding the health and safety of the public and adhering to all requirements of the license, TS, and NRC regulations.
The organization, as described in TS 6.1.1 and Figure 6-1 of the TS, and the MSTR staff responsibilities, as described in TS 6.1.2 and in SAR Section 12.1.2, are consistent with the guidance in ANSI/ANS-15.1 and NUREG-1537 and are, therefore, acceptable to the NRC staff.

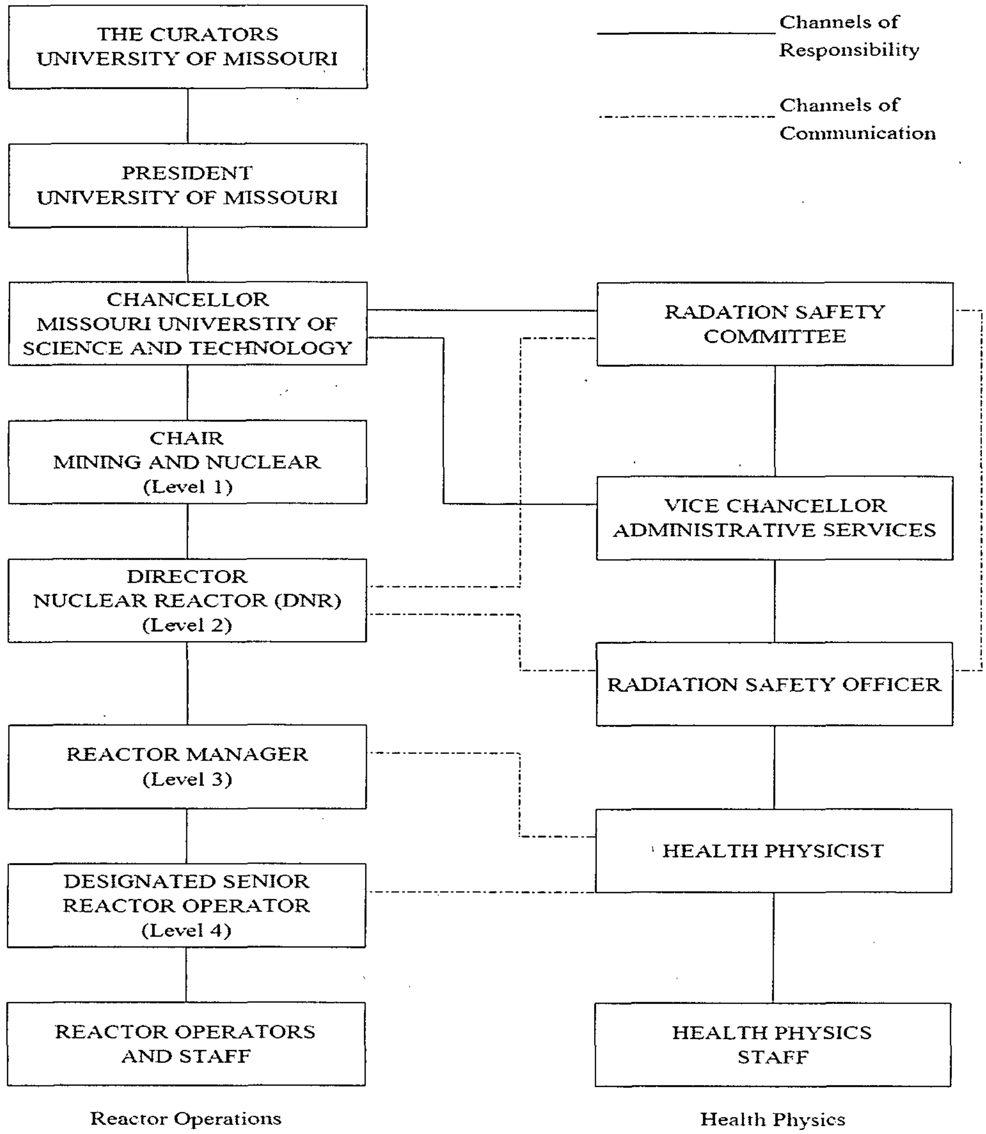

Figure 6.1 Organizational structure of the University of Missouri related to the Missouri S&T Nuclear Reactor Facility

Figure 12-1 Organizational Structure for the MSTR Facility

The licensee discussed the minimum staffing necessary to safely operate the MSTR in Section 12.1.3 of the SAR and in TS 6.1.3 as follows:

1) The minimum staffing when the reactor is not secured shall be:

   a) A certified reactor operator in the control room.

   b) A second designated person present at the reactor facility able to carry out prescribed written instructions. Unexpected absence for as long as 2 hours to accommodate a personal emergency may be acceptable, provided immediate action is taken to obtain a replacement.

   c) A designated senior reactor operator shall be readily available on call. "Readily available on call" means an individual who

      i) has been specifically designated and the designation is known to the operator on duty,

      ii) keeps the operator on duty informed of where he or she may be rapidly contacted and the phone number, and

      iii) is capable of getting to the reactor facility within a reasonable time under normal conditions (e.g., 30 minutes or within a 15-mile radius).

2) A list of reactor facility personnel by name and telephone number shall be readily available in the control room for use by the operator. The list shall include:

   a) management personnel
   b) radiation safety personnel
   c) other operations personnel

3) Events requiring the presence at the reactor facility of senior reactor operator are:

   a) initial startup and approach to power

   b) all fuel or control-rod relocations within the reactor core region

   c) relocation of any in-core experiment with a reactivity worth greater than one dollar

   d) recovery from unplanned or unscheduled shutdown or significant power reduction

The regulations in 10 CFR 50.54(i) through m(1) specify the minimum staffing necessary to safely operate the reactor, including the conditions for the presence of a senior reactor operator. The NRC staff reviewed the requirements of TS 6.1.3 and finds that they are consistent with the guidance in ANSI/ANS-15.1 and NUREG-1537 and that they satisfy the requirements of 10 CFR 50.54(k) and 10 CFR 50.54(m)(1). Accordingly, TS 6.1.3 is acceptable to the NRC staff.

12.2 Training

The licensee discusses the selection and training of personnel for key positions in Sections 12.1.4 and 12.9 of the SAR and in TS 6.1.4 and TS 6.1.5.

TS 6.1.4 specifies the requirements for operation of the reactor for training purposes as follows:

> When the reactor is being used for training purposes, the following conditions shall be met:
>
> 1) Students and trainees may operate the reactor under the direct supervision of a licensed reactor operator provided the excess reactivity is less than 0.7% delta k/k.
>
> 2) Trainees may operate the reactor under the direct supervision of a senior reactor operator when the excess reactivity is equal to or greater than 0.7% delta k/k and less than 1.5% delta k/k.

TS 6.1.4 ensures that training activities involving the reactor are conducted under the direct supervision of a licensed reactor operator with a specified level of experience and with the reactor operating within specified limits of excess reactivity appropriate for a training situation per the guidelines in ANSI/ANS-15.1 and ANSI/ANS-15.4, "Selection and Training of Personnel for Research Reactors," issued 1988. Based on the licensee's use of ANSI/ANS-15.4 and the licensee's requalification program meeting the requirements of 10 CFR Part 55, the NRC staff concludes that TS 6.1.4 is acceptable.

TS 6.1.5, "Selection and Training of Personnel," addresses the selection, training, and requalification of personnel as follows:

> The selection, training, and requalification of operations personnel shall meet or exceed the requirements of American National Standard for Selection and Training of Personnel for Research Reactors, ANSI/ANS-15.4 (1988), Sections 4–6. [6]

The Director Nuclear Reactor is responsible for ensuring the training and qualification of all operators per the guidelines in ANSI/ANS-15.1 and ANSI/ANS-15.4. Based on the licensee's use of ANSI/ANS-15.4 and because the licensee's requalification program meets the requirements of 10 CFR Part 55, the NRC staff concludes that TS 6.1.5 is acceptable.

12.3 Review and Audit Activities

The Radiation Safety Committee exists to review matters relating to the safe operation of the facility in accordance with the license, use of radioisotopes, and matters relating to the health and safety of the public and the environment. SAR Sections 12.2.1 and 12.2.2 and TS 6.2, 6.2.1, 6.2.2, and 6.2.3 describe the overall responsibility, composition, and qualification of the Radiation Safety Committee, as well as its charter and rules. The TS read as follows:

> 6.2 Review and Audit
>
> A committee shall review and audit reactor operations to ensure that the facility is operated in a manner consistent with public safety and within the terms of the facility license. The committee shall be referred to as the Radiation Safety Committee (Committee); it shall report to the Chancellor of the campus and advise the Chair of Mining and Nuclear Engineering and the DNR on those areas of responsibility specified below.
>
> 6.2.1 Composition and Qualifications
>
> The Committee shall be composed of at least five members, one of whom shall be the Radiation Safety Officer of the campus. No more than two members will be from the organization responsible for reactor operations. At least three members of the Committee shall collectively represent a broad spectrum of expertise in areas relating to reactor safety and research using radioisotopes. Qualified approved alternates may serve in the absence of regular members.
>
> 6.2.2 Charter and Rules
>
> 1) A quorum of the Committee shall consist of at least one-half of the voting members where the operating staff does not constitute a majority.
>
> 2) The Committee shall meet at least once each calendar year. Minutes of all meetings shall be disseminated to Committee members and to other responsible personnel, as designated by the Committee Chairman.
>
> 3) The Committee shall have a written statement, or charter, defining such matters as the authority of the Committee, the subjects within its purview, and other such administrative provisions as are required for the effective functioning of the Committee.

According to the SAR, the Radiation Safety Committee reports to the MST Chancellor. The Radiation Safety Committee conducts its review and audit functions in accordance with a written charter, which includes provisions for meeting frequency, voting rules, quorums, method of submission, content of presentations to the Radiation Safety Committee, and minutes. The NRC staff reviewed the charter and finds it to be in agreement with the SAR, NUREG-1537, and the recommendations of ANSI/ANS-15.1.

NUREG-1537 and ANSI/ANS-15.1 specify that the purpose of the review committee is to provide independent oversight, and that the operating staff should not constitute the majority of a quorum. The Radiation Safety Committee charter establishes a quorum of five members, which guarantees that operations staff will not be a majority. Because the Radiation Safety Committee serves as review and audit committees, at least two persons employed outside the Radiation Center must become members of the Radiation Safety Committee. The rules of the Radiation Safety Committee, as outlined in TS 6.2, the Radiation Safety Committee charter, and the SAR, are consistent with the guidelines of ANSI/ANS-15.1. These aspects of the Radiation Safety Committee are acceptable to the NRC staff.

TS 6.2.3 lists the following functions of the Radiation Safety Committee:

> As a minimum, the Committee shall:
>
> 1) Review, in accordance with 10 CFR 50.59, "Changes, Tests, and Experiments," untried experiments and tests that are significantly different from those previously used or tested in the reactor, as determined by the DNR.
>
> 2) Review, in accordance with 10 CFR 50.59, changes to the reactor core, reactor systems, design features, or procedures that may affect the safety of the reactor.
>
> 3) Review new procedures.
>
> 4) Review all proposed amendments to the facility license and TS.
>
> 5) Review reportable occurrences and the actions taken to identify and correct the cause of the occurrences.
>
> 6) Review significant operating abnormalities or deviations from normal performance of facility equipment that affect reactor safety.
>
> This same Committee may have other responsibilities, for example oversight of the campus byproduct materials license. The Committee may assign subcommittees to act on its behalf, provided that said subcommittees report all actions in writing.

The NRC staff reviewed the above items specified for review by the Radiation Safety Committee and concludes that the review function of the MSTR Radiation Safety Committee specified in TS 6.2.3 is consistent with the guidelines of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," issued March 2007; NUREG-1537; and ANSI/ANS-15.1 and is acceptable to the NRC staff.

TS 6.2.4 requires the performance of an annual audit of reactor operations by the Radiation Safety Committee. TS 6.2.4 reads as follows:

> The Committee will arrange for a knowledgeable and impartial individual (or individuals) to review reactor operations and audit the operational records for the

following:

1) compliance with reactor procedures, TS, and license provisions

2) training and the requalification program

3) emergency plan

4) health physics

5) results of actions taken to correct deficiencies

6) experiments

7) security procedures

An impartial individual is one who is not directly affected by the findings or recommendations of the audit and has no reason to be biased concerning the review. These audits shall be performed annually.

Audits of various aspects of the health physics program are specified in Chapter 11 of the SAR and the Radiation Safety Committee charter and are a requirement of the regulations in 10 CFR 20.1102(c). TS 6.6 requires that deficiencies that are uncovered by an audit that affect reactor safety are immediately reported to Level 1 management as a reportable event. The licensee states that written reports of audit findings are submitted to Level 1 and Level 2 management within 90 days after completion of the audit.

The NRC staff reviewed TS 6.2 and SAR Section 12.2 and concludes that the MSTR review and audit functions are consistent with NUREG-1537 and ANSI/ANS-15.1. The NRC staff finds that the licensee's review and audit functions are acceptable, the committee members appear to be well qualified and have a wide spectrum of expertise, the charter and rules are acceptable, and the items that the committee will review and audit are comprehensive and acceptable.

12.4 Radiation Protection

Radiation safety and radiation protection services are the responsibility of the health physicist (Level 3 management), who reports to the RSO. The RSO has a line of communication to the reactor manager. The health physicist and staff provide the radiation protection function for the MSTR. They have the authority and responsibility to halt any perceived unsafe practices. TS 6.3 contains the administrative control related to radiation safety and provides for the following:

> The health physicist shall be responsible for implementing the radiation protection program at the reactor facility.

The university's Radiation Safety Committee, which reports to the Chancellor, oversees radiation safety. The health physicist may bring concerns to the RSO or the reactor manager, or directly to the Radiation Safety Committee. Chapter 11 of this SER presents additional discussion of radiation safety. The administrative control TS for radiation safety are consistent with ANSI/ANS-15.1 and NUREG-1537 and are therefore acceptable to the NRC staff.

12.5 Procedures

The licensee has specified in SAR Section 12.3 and in TS 6.4 the types of written procedures that the Radiation Safety Committee must review and approve before their use. The areas to be covered by such procedures include startup, operation, and shutdown of the reactor; installation and removal of fuel, control rods, and experiments; maintenance; surveillance and calibration; actions to correct malfunctions, including response to alarms; EP activities; administrative control of experiments; security activities; and radiation protection program activities. Additionally, the MSTR SOPs contain administrative controls for operations and irradiations.

TS 6.4 specifies the following types of written procedures that must be reviewed and approved by the Radiation Safety Committee before use:

> The reactor staff shall prepare and use written procedures for at least the items listed below. These procedures shall be adequate to ensure the safe operation of the reactor, but should not preclude the use of independent judgment and action should the situation require it.
>
> 1) startup, operation, and shutdown of the reactor
>
> 2) installation or removal of fuel elements, control rods, experiments, and experimental facilities
>
> 3) actions to be taken to correct specific and foreseen potential malfunctions of systems or components, including responses to alarms, suspected coolant system leaks, and abnormal reactivity changes
>
> 4) emergency conditions involving anpotential or actual release of radioactivity, including provisions for evacuation, re-entry, recovery, and medical support
>
> 5) preventive and corrective maintenance operations that could have an effect on reactor safety
>
> 6) periodic surveillance (including testing and calibration) of reactor instrumentation and safety systems
>
> 7) radiation control procedures, which shall be maintained and made available to all operations personne.
>
> 8) implementation of emergency and physical security procedures
>
> Substantive changes to the previous procedures shall be approved by the

> Committee, and the DNR (Level 2) or designated alternates. Minor modifications to the original procedures that do not change their original intent can be made by the Facility Manager (Level 3) or higher but the modifications must be approved by the DNR (Level 2) or designated alternates within 14 days.

As shown above, TS 6.4 specifies the areas to be covered by procedures. TS 6.2.3 and 6.4 specify that the Radiation Safety Committee must approve new procedures and substantive changes to procedures. TS 6.4 outlines the process for unsubstantial and temporary changes to procedures.

The NRC staff finds that the TS for procedures are consistent with the guidance of ANSI/ANS-15.1 and NUREG-1537 and that the process and method for procedures established in TS 6.4 ensure adequate management control and proper review of procedures. The NRC staff concludes that the procedural requirements in TS 6.4 provide reasonable assurance of the safe operation of the reactor and proper administration of the facility.

12.6 Experiments

TS 6.5 gives the following administrative controls for the review and approval of experiments:

> The reactor staff shall perform a thorough review of all proposed experiments to ensure that they meet the requirements of Section 3.7 of these specifications.
>
> Following the reactor staff review and approval, any proposed untried experiments will be forwarded to the Committee for its review. The DNR or designated alternate shall give approval in writing before the proposed untried experiment is initiated.
>
> Substantive changes to previously approved experiments shall be made only after review by the Committee and approval in writing by the DNR or designated alternates.

The purpose of TS 6.5 is to ensure that all experiments initially have a high level of review and that changes to existing approved experiments undergo a level of review appropriate to the significance of the change. TS 6.5 is consistent with the guidance of ANSI/ANS-15.1 and NUREG-1537 and is therefore acceptable to the NRC staff.

12.7 Required Actions

In Section 12.5.2 of the SAR and in TS 6.7.2, "Special Reports," the licensee has defined a group of incidents as reportable events. The licensee has defined the required actions for events in TS 6.6, "Required Actions."

TS 6.6.1, "Action To Be Taken in the Case of Safety Limit Violation," identifies the following actions:

> 1) The reactor shall be shut down, and reactor operations shall not be

resumed until authorized by the NRC.

2) The safety limit violation shall be promptly reported to the DNR.

3) The safety limit violation shall be reported to the NRC (see Section 6.7.2).

4) A safety limit violation report shall be prepared. The report shall describe the following:

   a) applicable circumstances leading to the violation including, when known, the cause and contributing factors

   b) effect of the violation upon reactor facility components, systems, or structures and on the health and safety of personnel and the public

   c) corrective action to be taken to prevent recurrence

5) The report shall be reviewed by the Committee, and any followup report shall be submitted to the NRC (see Section 6.7.2) when authorization is sought to resume operation of the reactor.

TS 6.7.2(1)(a) contains the requirement to report the violation of the safety limit to the NRC not later than the next working day. The actions proposed by the licensee are consistent with the guidance of ANSI/ANS-15.1 and NUREG-1537 and meet the requirements given in 10 CFR 50.36(d)(1) for actions to be taken if a safety limit is exceeded. Therefore, TS 6.6.1 is acceptable to the NRC staff.

TS 6.6.2 describes the actions to be taken in case of reportable occurrences other than those that involve exceeding a safety limit. TS 6.7.2 specifies reportable occurrences other than violations of safety limits. TS 6.6.2 specifies the following actions to be taken in case of these reportable occurrences:

> The following actions shall be taken if an event of the type identified in Section 6.7.2.(1).b or 6.7.2.(1).c occurs:
>
> 1) Reactor conditions shall be returned to normal or the reactor shall be shut down. If it is necessary to shut down the reactor to correct the occurrence, operations shall not be resumed unless authorized by Level 2 or designated alternates.
>
> 2) The occurrence shall be reported to the DNR and to the NRC (see Section 6.7.2).
>
> 3) The occurrence shall be reviewed by the Committee at its next scheduled meeting.

The TS requires that the licensee will notify the NRC of important events in a timely manner, and the actions required to be taken in case of reportable occurrences are consistent with the guidance of ANSI/ANS-15.1 and NUREG-1537 and are therefore acceptable to the NRC staff. Because these actions are consistent with those recommended in NUREG-1537 and ANSI/ANS-15.1, the NRC staff concludes that the licensee will take appropriate action in case a safety limit is exceeded or in case of other reportable actions.

12.8 Reports

Section 12.5 of the SAR and TS 6.7 list the required reports. TS 6.7.1 details the requirements for the annual operating report as follows:

> An annual progress report will be made by May 30 of each year to the NRC Document Control Desk that provides the following information:
>
> 1) A narrative summary of reactor operating experience including the energy produced by the reactor or the hours the reactor was critical, or both.
>
> 2) The unscheduled shutdowns including, where applicable, corrective action taken to preclude recurrence.
>
> 3) Tabulation of major preventive and corrective maintenance operations having safety significance.
>
> 4) A summary of changes to the facility or procedures that affect reactor safety, and performance of tests or experiments carried out under the conditions of 10 CFR 50.59 [6].
>
> 5) A summary of the nature and amount of radioactive effluents released or discharged to environs beyond the site boundary. The summary shall include to the extent practicable an estimate of individual radionuclides present in the effluent. If the estimated average release after dilution or diffusion is less than 25 percent of the concentration allowed, a statement to this effect is sufficient.
>
> 6) A summarized result of environmental surveys performed outside the facility.
>
> 7) A summary of exposures received by facility personnel and visitors where such exposures are greater than 25 percent of those allowed.

TS 6.7.2 specifies the 24-hour reportable occurrence and the followup 14-day written reports and the unusual event reports to be submitted within 30 days. This TS also defines those events that constitute a reportable occurrence. TS 6.7.2 specifies the following requirements for special reports:

> 1) If any one of the following events occurs, the licensee shall make a report describing the circumstances of the event by telephone to the NRC

Headquarters Operations Center no later than the following working day, followed by a written report, submitted to the NRC Document Control Desk, within 14 days:

a) violation of safety limits (see Section 6.6.1)

b) release of radioactivity from the site above allowed limits (see Section 6.6.2)

c) any of the following: (see Section 6.6.2)

   i) operation with actual safety-system settings for required systems less conservative than the LSSS specified in the TS

   ii) operation in violation of the LCOs established in the TS unless prompt remedial action is taken

   iii) a reactor safety system component malfunction that renders or could render the reactor safety system incapable of performing its intended safety function unless the malfunction or condition is discovered during maintenance tests or periods of reactor shutdowns

   NOTE: Where components or systems are provided in addition to those required by the TS, the failure of the extra components or systems is not considered reportable provided that the minimum number of components or systems (specified or required) perform the intended reactor safety functions.

   iv) an unanticipated or uncontrolled change in reactivity greater than one dollar(excluding trips resulting from a known cause are)

   v) abnormal and significant degradation in reactor fuel or cladding, or both; coolant boundary; or containment boundary (excluding minor leaks), where applicable, that could result in exceeding prescribed radiation exposure limits of personnel or environment, or both

   vi) an observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence or development of an unsafe condition with regard to reactor operations

2) A written report of the following shall be submitted within 30 days to the NRC Document Control Desk:

   a) significant changes in the transient or accident analyses as described in the SAR

   b) permanent changes in facility organization involving Level 1, 2 or 3 personnel

These reporting requirements are consistent with the guidance of ANSI/ANS-15.1 and NUREG-1537 and are therefore acceptable to the NRC staff.

12.9 Records

Required records are listed in Section 12.6 of the SAR and in TS 6.8 as follows:

> Records may be logs, data sheets, or other suitable forms. The required information may be contained in single or multiple records, or a combination thereof.
>
> TS 6.8.1 Records To Be Retained for a Period of at Least Five Years
>
> 1) normal reactor facility operation (but not including supporting documents such as checklists and log sheets, which shall be maintained for a period of at least one year)
>
> 2) principal maintenance operations
>
> 3) reportable occurrences
>
> 4) surveillance activities required by the TS
>
> 5) reactor facility radiation and contamination surveys where required by applicable regulations
>
> 6) experiments performed with the reactor
>
> 7) fuel inventories, receipts, and shipments
>
> 8) approved changes in operating procedures
>
> 9) records of meeting minutes and audit reports of the Committee
>
> TS 6.8.2 Records To Be Retained for at Least One Requalification Cycle
>
> Regarding retraining and requalification of licensed operations personnel, the records of the most recent complete requalification cycle shall be maintained.

TS 6.8.3 Records To Be Retained for the Life of the Facility

1) gaseous and liquid radioactive effluents released to the environment
2) radiation exposures for all personnel monitored
3) updated, corrected, and as-built drawings of the facility

The NRC staff compared TS 6.8 with NUREG-1537 and ANSI/ANS-15.1 and concludes that the requirements in TS 6.8 are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1and as such are acceptable to the NRC staff.

12.10 Emergency Planning

The licensee requested that the current EP be considered as part of the license renewal application. The NRC staff reviewed the EP against NUREG-0849, "Standard Review Plan for the Review and Evaluation of Emergency Plans for Research and Test Reactors," issued October 1983; Regulatory Guide 2.6, "Emergency Planning for Research and Test Reactors," Revision 1, issued March 1983; ANSI/ANS-15.16, "Emergency Planning for Research Reactors," issued 1982; and NRC Information Notice 97-34, "Deficiencies in Licensee Submittals Regarding Terminology for Radiological Emergency Action Levels in Accordance with the New Part 20," issued June 1997. The NRC staff concluded that the MSTR EP is in accordance with that guidance and those regulations. A letter dated October 3, 2007, from the NRC Office of Nuclear Security and Incident Response to the NRC Office of Nuclear Reactor Regulation (ADAMS Accession No. ML072750774) documents this review. 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 MSTR EP provides reasonable assurance that the licensee can respond appropriately to a variety of emergency situations and that the MSTR EP will be adequately maintained during the period of the renewed license.

12.11 Security Planning

Because the facility license authorizes possession of SNM of low strategic significance of uranium-235 enriched to less than 20 percent specified in 10 CFR 73.2 , the licensee must maintain security measures that satisfy the requirements of 10 CFR 73.67(f). The licensee has met these requirements since Amendment No. 17, issued November 23, 1999, removed the license condition to have an approved security plan. The NRC routinely inspects the licensee's measures for physical security procedures and the protection of SNM. The NRC inspection dated January 16, 2007 verified that the licensee's security measures satisfy all applicable regulations and are acceptable.

12.12 Quality Assurance

The MSTR SAR states that the licensee maintains a quality assurance program that is consistent with the guidance found in ANSI/ANS-15.8, "Quality Assurance Program Requirements for Research Reactors," issued 1995. The licensee states that the Director of the MSTR has responsibility for the quality assurance program. Normally, the reactor manager is responsible for the daily implementation of the program. The Radiation Safety Committee has responsibility for independent review and audit functions associated with the program. Quality Assurance records include inspection and test results, reviews by the Radiation Safety Committee, and analyses of modifications and design changes.

12.13 Operator Training and Requalification

The reactor manager also serves as the training coordinator and is responsible for the implementation, coordination, and operation of the Requalification Program, including the training of new operators. The licensee's Requalification Program and training program provide reasonable assurance that the licensee will have technically qualified reactor operators. The licensee submitted a revised operator requalification program with the application for license renewal. The NRC staff reviewed the program and found that it meets all applicable regulations (10 CFR 50.54(i)–(l) and 10 CFR Part 55) and is consistent with guidance contained in ANSI/ANS-15.4.

12.14 Conclusions

On the basis of the preceding discussions, the NRC staff concludes that the licensee has sufficient oversight personnel, management structure, and procedures to provide reasonable assurance that the reactor will continue to be managed in a way that will cause no significant risk to the health and safety of the public. The NRC staff has reviewed SAR Chapter 12 and TS Section 6, which discuss the licensee's proposed organization, training including operator requalification, review and audit activities, administration of radiation protection activities, procedures, experiment review, required actions, and records and reports, against the guidance in the 1990 and 2007 versions of ANSI/ANS-15.1, which the NRC staff supports for the conduct of operations, and NUREG-1537. The licensee's proposed conduct of operations in the areas reviewed is consistent with the guidance of the ANS standard and NUREG-1537. The NRC staff also reviewed Section 6 of the TS against 10 CFR 50.36, "Technical Specifications," including 10 CFR 50.36(d)(5) and (7), and concludes that the TS meets the requirements of the regulations. In addition, the NRC staff has reviewed the inspection reports prepared by the NRC inspectors and concludes that the conduct of operations at the MSTR meets the requirements of 10 CFR Part 50.

13. ACCIDENT ANALYSIS

The accident analyses presented in the MSTR SAR establish safety limits and limiting conditions that are imposed on the MSTR through the TS. In the SAR, the licensee analyzed potential reactor transients and other hypothetical accidents. The licensee's analysis includes the potential effects of natural hazards, as well as potential accidents involving the operation of the reactor. The NRC staff evaluated the licensee's analytical assumptions, methods, and results.

TS 2.1 designates a maximum temperature limit to ensure that the integrity of the fuel cladding is maintained to guard against an uncontrolled release of fission products. TS 2.1 is given as follows:

> The safety limit shall be on the temperature of fuel element cladding, which shall be 510 °C (950 °F).

NUREG-1313 states that the initial releases of fission products will occur at the blister cladding temperature of about 527 degrees C (981 degrees F). In protecting the fuel element integrity, TS 2.1 provides that the cladding temperature shall not exceed 510 degrees C (950 degrees F), which is 17 degrees C lower than the blister cladding temperature. The maximum cladding temperature associated with full-power (200-kW) operations is only about 90 degrees C (194 degrees F), with a normal operating inlet pool water temperature of 20 degrees C (68 degrees F). Because the licensee provides the safety limit at less than the blister cladding temperature, TS 2.1 is acceptable to the NRC staff.

TS 2.2 designates setpoints (LSSS) for the safety channels that will initiate an automatic action (scram) to prevent exceeding the safety limit. TS 2.2 is given as follows:

> The limiting safety system setting shall be on reactor thermal power, P, which shall be no greater than 300 kW, or 150% of full power.

Natural convection in the reactor pool provides reactor cooling. Accordingly, the only parameter that can be used to limit the fuel cladding temperature is the reactor power. Section 4.6 of the SAR shows that at a reactor power of 300 kW, the maximum cladding temperature is well below 105 degrees C (221 degrees F). This temperature is much lower than the temperature at which fuel element damage could occur. The NRC staff concludes that an extremely large safety margin exists between the LSSS (reactor scram at 300 kW or equivalent to 105 degrees C) and the safety limit of 510 degrees C, and as such TS 2.2 is acceptable to the NRC staff.

In the following subsections, the various accident scenarios have been categorized according to their corresponding accident type as defined by the NRC staff in NUREG-1537, Part 1. The licensee has analyzed the most limiting accident scenario for each accident type for the potential hazards posed to the health and safety of the public and the MSTR staff.

The following accidents were analyzed for the MSTR:

- failure of a fueled experiment—designated as the MHA
- insertion of excess reactivity

- loss of coolant
- loss of coolant flow
- mishandling or malfunction of fuel
- experiment malfunction
- flooding of an irradiation facility
- failure of a movable experiment
- loss of normal electrical power
- external events
- mishandling or malfunction of equipment—reactor startup accident

The failure of a fuel element outside of the reactor pool is not considered credible, since none of the fuel is removed from the pool. All fuel elements are stored in the storage pit inside the pool. Removal of a fuel element would require the development and approval of a new procedure to perform this function.

13.1 Maximum Hypothetical Accident

The licensee presents the MHA, the accident with the greatest potential impact on the health and safety of the public, in Section 13.1.1 of the SAR. This accident assumes that an irradiated experiment containing fissile material fails and releases all gaseous fission products in the reactor building. The MSTR license allows for two different types of fueled experiments, one of which generates less than 1 W of power and the other generates between 1 W and 25 W of power. The second type of fuel experiment has an added restriction requiring it to be located beneath at least 4.88 m (16 ft) of water.

For the first experiment failure, the licensee calculates a maximum dose (TEDE) to a member of the public of 7 mrem, and to MSTR personnel of 64 mrem, respectively. For the second experiment failure, the licensee calculates a maximum dose (TEDE) to a member of the public of 46 mrem, and to MSTR personnel of 410 mrem. Because the potential impact of a failed 25-W experiment is the greatest of all reviewed accidents, the license declares that the failure of a 25-W fuel experiment is the MHA for the MSTR.

The values presented are well within the regulatory requirements of 10 CFR Part 20. In addition, while performing the calculations the licensee applied the following conservative assumptions, resulting in highly conservative or noncredible consequence estimates. The NRC staff evaluates each of these assumptions as follows:

(1) Experiments running for an infinite amount of time and all fission products, including long-lived ones, are saturated.

This is a conservative assumption, since less than one-half of the fission products would be at saturation after a 1-day (8-hour) irradiation, and as such is acceptable to the NRC staff.

(2) All (100 percent) noble gases and 50 percent of the halogens would be released from the experiment on total failure.

This consumption is consistent with NUREG-0772, and as such is acceptable to the NRC staff.

(3) Pool water does not provide any scrubbing of the noble gases.

This assumption does not account for any physical size or form of the material in the fueled experiment. Noble gases can be trapped in the material matrices, but this assumption does not provide any credit for trapping. Because this assumption is conservative, it is acceptable to the NRC staff.

(4) Pool water removes 90 percent of the iodine isotopes.

The licensee states that this assumption is also conservative, since 95 percent of iodine isotopes are removed by the pool water, and as such it is acceptable to the NRC staff.

(5) Isotopes are instantaneously released to and uniformly distributed in the reactor room air.

There will be a delay between the time of capsule failure and fission product release under 4.88 m (16 ft) of water to the building environment. The evacuation time is assumed to be 5 minutes, so any evacuation would be started before the activity in the air would saturate. In addition, this assumption ignores radioactive decay during the finite mixing time. Because it is a conservative assumption, it is acceptable to the NRC staff

In addition, the licensee also assumes that the exhaust fans are not in operation. The free air volume of the reactor bay is 1,700 m^3. An average breathing rate of the reactor personnel is 1.25×10^6 cubic centimeters per hour, and that person would stay in the reactor room full of airborne radioactive gases and particulates for 5 minutes. Those assumptions are consistent with NUREG-0772 and the facility description. In addition, the licensee states that an evacuation time of 5 minutes is conservative based on the licensee's experience in previous emergencies of 3 minutes evacuation time. Therefore, those assumptions are acceptable to the NRC staff.

In its results, the licensee calculates a maximum thyroid dose of 28.3 millisieverts (mSv) (2.83 rem) and a maximum dose (TEDE) of 4.10 mSv (0.410 rem) to MSTR personnel. The thyroid dose value is more than a factor of 10 below the regulatory limit for an individual organ of 500 mSv (50 rem) specified in 10 CFR 20.1201, and the TEDE value is a factor of 10 below the regulatory limit of 50 mSv (5 rem) specified in 10 CFR 20.1201.

For calculation of a maximum dose to an individual outside the reactor building, the licensee assumes that the exhaust fans are operating and that the most exposed individual remains in place throughout the time required to remove essentially all of the contaminated air from the reactor room. It also assumes that there is additional dispersion of the exhausted air before the dose recipient is immersed in it, with an equivalent of 2.0×10^{-2} seconds per cubic meter. The licensee also assumed that all radioisotopes released in the reactor building leak out within 24 hours. There was no radioactive decay and, hence, no decrease in the source strength. Because those assumptions are conservative in that they do not account for the removal of any radioisotope as the result of radioactive decay and/or from plating out of the air, or they do not allow for any sealing of the building. Because these assumptions are conservative, they are acceptable to the NRC staff.

The licensee calculates the TEDE for an individual located just outside of the reactor building as 0.46 mSv (0.046 rem), which is below the regulatory limit of 1 mSV (0.1 rem) specified in 10 CFR 20.1301, "Radiation Dose Limits for Individual Members of the Public."

The NRC staff verified the licensee's analysis and assumptions and concludes that the MHA has been adequately analyzed.

TS 3.7.2(3) provides limits to the fueled experiment as follows:

> Fueled experiments shall not be allowed in or near the reactor unless specifically approved by the Radiation Safety Committee. Fueled experiments in the amount which would generate a power greater than 25 W shall not be irradiated at the MSTR facility. Fueled experiments which generate more than 1 W power shall be irradiated in the reactor pool at least 4.88 m (16 ft) deep under the pool water surface. Fueled experiments which generate less than 1 W power may be irradiated anywhere in the facility. Fueled experiments shall be encapsulated to contain all fission products during irradiation. The encapsulation device shall be designed to prevent degrading of the device due to pressure and temperature of the fueled experiment.

The licensee proposed two changes to TS 3.7.2(3), one of which is to reduce power from 100 W to 25 W and the other is to ensure the integrity of the fueled container (encapsulation). Those changes ensure that a fueled experiment will not result in undue radioactivity release to the environment and ensure compliance with 10 CFR Part 20. Therefore, TS 3.7.2(3) is acceptable to the NRC staff.

The licensee states that fueled experiments have not been used for the past 15 years. Their future use is unlikely. In addition, the licensee states that the MSTR SOPs require a review by the Radiation Safety Committee and also include administrative steps to prevent a failure of a fueled experiment.

Base on the above considerations, the NRC staff concludes that the licensee's assumptions and calculational methods are consistent with guidance contained in regulatory guides, NUREG series documents, and previous NRC licensing actions. The NRC staff reviewed the licensee's assumptions and calculation method and concludes that the licensee is capable of calculating conservative doses for the limiting MHA. The NRC staff reviewed the associated TS and frequency of an experiment and concludes that a fueled experiment failure is unlikely, and the resulting doses from the limiting MHA would be below the applicable regulatory limits.

13.2 Insertion of Excess Reactivity

The licensee discusses the potential impact of an insertion of the maximum excess reactivity in Section 13.1.2 of the MSTR SAR. According to the SAR, a sufficient excess reactivity is needed to provide for temperature effect override, xenon override, and operational and experimental flexibility. The licensee assumes that inadvertently inserting a fuel element into a vacancy at the periphery of the core will result in a reactivity insertion of 1.5% $\Delta k/k$.

The SAR explains that the potential significant consequences associated with the rapid insertion of reactivity accident are damage to the fuel or cladding material and/or direct radiation exposure to operations personnel. However, an accident due to step insertion in the MSTR is unlikely due to its limit of a reactivity insertion to 1.5% $\Delta k/k$. The SAR provides a result from a test conducted by the Idaho National Engineering Laboratory on the SPERT-I reactor, which contains fuel elements similar to those in the MSTR. The results indicated that an instantaneous 1.6% $\Delta k/k$ reactivity addition produces approximately a 10 megawatt per second energy release. The SPERT-I tests demonstrated that no fuel melting or fission product release occurred under these conditions.

TS 3.1(1) limits the maximum excess reactivity to ensure that the reactor can be operated safely and can be shut down at all times. This TS is given as follows:

> The maximum excess reactivity for reference core conditions with secured experiments and experimental facilities in place shall be no more than 1.5% $\Delta k/k$.

Based on the SPERT-I results and the limit of secured experiments to 1.5% $\Delta k/k$, the NRC staff concludes that a stepwise reactivity insertion would not adversely affect the health and safety of the public and the reactor personnel.

To preclude the possibility of having an internal vacancy into which a fuel element could be inadvertently inserted, TS 3.1(4) states the following:

> The reactor shall be operated only when all lattice positions internal to the active fuel boundary are occupied by either a fuel element, control rod fuel element, or by an experimental facility.

The licensee analyzed the maximum credible step reactivity insertion for the MSTR reactor, 1.5% $\Delta k/k$ as specified in TS 3.1(1). The upper limit of the reactivity insertion value corresponds to the insertion of a fuel element at the periphery of the core. TS 3.1(4) requires that the MSTR be operated with all internal core positions filled to prevent inadvertent reactivity insertion into the reactor core except at peripheral positions. The only credible reactivity transient would be initiated by dropping a fuel element next to the core. The NRC staff notes that TS 3.1(4) precludes operation of the reactor with an internal vacancy into which a fuel element could be inserted; therefore, TS 3.1(4) is acceptable to the NRC staff.

No fuel damage is expected as the result of a reactivity insertion accident. The maximum fuel temperature is well below the safety limit of 510 degrees C (950 degrees F) specified in TS 2.1 for the aluminum cladding of the fuel. In addition, MSTR SOPs include administrative steps to prevent a transient such as that described in SAR Section 13.1.2. Accordingly, the NRC staff concludes that a reactivity insertion accident would not cause fuel damage or the release of fission products.

13.3 Loss of Coolant

Based on its review of the Section 13.1.3 of the SAR, the NRC staff concludes that the MSTR pool cannot be drained by any method other than by pumping the water out of the pool. The only way for a sudden loss of coolant to occur would be some catastrophic collapse of the pool. A loss-of-coolant event at the MSTR has a very low probability of occurrence because of the pool design (discussed in Chapter 3 of this SER) and the low risk of external events (discussed in Chapter 2 of this SER).

The licensee analyzed the loss-of-coolant accident with the assumptions that it occurred during full-power operation and that fission product activity is at saturation. The licensee further assumes that if the core becomes uncovered, heat transfer would occur by the natural convection of ambient air. The licensee performed a steady-state analysis showing that the amount of heat removed is proportional to the cladding temperature or heat generation in the fuel elements. For a catastrophic rupture of the pool, the licensee assumes that the water would take 1 second to empty in a "free fall" past the core. Using the decay heat generation level of 1 second, the licensee predicts that the cladding temperature would reach 410 degrees C (770 degrees F). The NRC staff notes that this is a conservative estimate since decay heat generation decreases over time along with the cladding temperature.

In addition, studies conducted at the Oak Ridge Research Reactor demonstrate that an instantaneous loss of coolant would not lead to fuel damage. In a 1967 study, "Water-Loss Tests in Water-Cooled and Moderated Research Reactors," Webster examined data from the Oak Ridge Research Reactor, as well as data from similar experiments conducted at the low-intensity testing reactor and the Livermore pool-type reactor. These reactors are light-water-moderated research reactors that use flat-plate fuel and are similar in design to the MSTR. Webster concluded that plate-type fuel can withstand a loss-of-coolant accident after infinite operation at power levels up to 3 megawatts. Given the similarity of the MSTR design to that of the low-intensity testing reactor and the Livermore pool-type reactor, the NRC staff concludes that the results of these studies are applicable to the MSTR. Accordingly, the NRC staff finds that, given that the MSTR licensed maximum power is 0.2 megawatts, a complete, instantaneous loss of coolant will not lead to fuel damage or to the release of fission products

Considering the decrease in decay heat, the licensee estimates that after 1 minute, the cladding temperature would be considerably lower. After 1 minute, the decay heat power is about a factor of 2 lower, with a corresponding cladding temperature of 200 degrees C (392 degrees F). The predicted cladding temperature is well below the safety limit of 510 degrees C (950 degrees F) specified in TS 2.1.

Based on the above consideration, the NRC staff concludes that the licensee's method for calculating expected cladding temperature after a loss-of-coolant accident event is acceptable. The NRC staff concludes that a loss of coolant to the MSTR will not lead to fuel damage or to the release of fission products.

13.4 Loss of Coolant Flow

Section 13.1.4 of the SAR states that there is no credible situation where a coolant channel could become blocked. The NRC staff finds that this is a reasonable statement since cooling is natural circulation and there is no forced flow. Furthermore, the NRC staff acknowledges that an object would have to be massive enough and would have to fall into the bottom of the reactor pool to block coolant flow. However, the licensee states that the reactor core design (see Section 4.2 of the SAR) and the operating procedures that require inspecting the reactor core before operation would prevent the blockage of the coolant channels by a large object. Even if it occurs, the NRC staff notes that numerous alarms (bulk water temperature, water level, water flow, and radiation monitors) are available to signal the need for operator action to shut down the reactor. Even if there were a loss in the ability of the primary and secondary cooling systems to remove heat from the primary coolant, and the reactor remained at full power, it would take hours for the water level to evaporate down to the top of the core. As the water level dropped past the top of the core, the negative void coefficient of reactivity would shut down the reactor. Makeup water could easily be provided from external sources by the operators. Because the reactor operators have multiple indicators of the loss of coolant flow, the NRC staff concludes that the reactor will be shut down in a timely manner if loss of coolant flow occurred.

13.5 Mishandling or Malfunction of Fuel

The licensee states in Section 13.1.5 of the SAR that the movement of fuel occurs only under water, and only one fuel element is moved at a time. The licensee states that if a fuel element were to drop from the fuel handling tool, it would dent only one of the end fittings. This is a reasonable position in view of the weight and shape of the fuel bundle. In addition, the licensee states that it has established a reactor operating procedure for fuel handling and required the presence of a senior reactor operator (TS.6.1.3).

The NRC staff acknowledges that cladding might fail if a fuel element were to be dropped underwater during transfer or during operation of the reactor due to a manufacturing defect, corrosion, or overheating of the fuel element. However, for this case most of the halogens will be scrubbed by the primary coolant in the reactor pool, so the radiation dose will be lower than for the MHA.

The NRC staff reviewed the MHA scenario for the Ohio State University research reactor (OSURR), which is similar to the MSTR. The OSURR MHA scenario assumes that one of the fuel elements experiences the complete removal of the cladding from one side of one fuel plate and that the reactor is operated at 500 kW(t) for an infinite irradiation time. Those are conservative assumptions since the maximum power level of the MSTR is only 200 kW, and the maximum run time is 8 hours. The OSURR's results show that the maximum dose (TEDE) to a member of the public is 25 mrem and the maximum dose (TEDE) to the reactor personnel in the restricted area is 180 mrem. Those values are well within the requirements specified in 10 CFR Part 20.

Based on the above considerations, the NRC staff concludes that the mishandling of fuel has no expected radiological consequences, and there is no need to consider this accident scenario as the MHA.

13.6 Experiment Malfunction

Two accidents were considered: (1) flooding of an irradiation facility and (2) failure of a movable experiment.

13.6.1 Flooding of an Irradiation Facility

Section 13.1.6.2 of the SAR describes the scenario of flooding of an irradiation facility. In this scenario, the licensee assumes that the inserted irradiation element (see the description of an irradiation element in Section 10.2.5 of this SER) would develop a leak and instantaneously floods the air cavity. The licensee's analysis indicates that the reactivity insertion would be negative regardless of where the element is positioned. For the present LEU fuel, the SAR states that flooding of an irradiation facility located in the center of the core would result in a stepwise reactivity insertion of -0.5% Δk/k and, on the periphery, would result in -0.1% Δk/k. The NRC staff concludes that flooding of an irradiation facility will reduce the core reactivity; therefore, the flooding of the irradiation facility would not pose a significant risk to the reactor.

13.6.2 Failure of a Movable Experiment

Section 13.1.6.2 of the SAR describes the scenario of a failure of a movable experiment. In this scenario, the licensee assumes that an experiment worth -0.4% Δk/k instantaneously fall away from the reactor while the reactor is at full power, resulting in a stepwise insertion of 0.4% Δk/k. The licensee also assumes that the most reactive control rod is stuck and does not scram, but the other control rods insert on a period scram signal. The licensee's calculations show that the maximum power would reach 450 kW and then would decrease as the reactor scrammed. The peak heat flux of about 5 W/cm^2 would result in a cladding temperature well below the safety limit specified in TS 2.1.

The NRC staff reviewed the licensee's analysis and concludes that this analysis is sufficient and appropriate. A test result shows that a cladding temperature is less than 450 degree C, well below the safety limit specified in TS 2.1. TS 3.2.2 provides a scram function to shut down the reactor. TS 3.7 limits the maximum reactivity worth to 0.4% Δk/k for a movable experiment.

Based on the above considerations, the NRC staff concludes that the failure of a movable experiment will not lead to fuel damage or to the release of fission products.

13.7 Loss of Normal Electric Power

Section 13.1.7 of the SAR explains that a loss of normal electric power will initiate an automatic scram of the reactor by causing the control rods to drop by gravity into the core. The reactor is cooled by natural circulation, both in operation and after shutdown. Therefore, a loss of normal electric power has no effect on reactor cooling since there is sufficient coolant in the reactor pool to absorb the decay heat from the reactor without the need for the primary or secondary cooling system. As stated in Section 8.2 of this SER, the MSTR does not have a requirement for emergency power. In addition, the operator can easily verify shutdown of the reactor manually by visually inspecting the core from the reactor top.

Based on the above considerations, the NRC staff concludes that a loss of normal electric power poses little risk to the health and safety of the public and of the MSTR staff.

13.8 External Events

Chapters 2 and 3 of this SER discuss the meteorological and seismic hazards of the reactor facility. According to the SAR, the external events are a tornado and an earthquake. The NRC staff notes that the MSTR is built on bedrock below ground level and the pool is constructed of reinforced concrete. In addition, the reactor core will be protected from damage by 4.88 m (16 ft) of water in the pool, by the reactor bridge above the pool, and by the building structures. Therefore, the NRC staff concludes that a direct hit by a tornado strike would be unlikely to damage the reactor fuel elements.

The NRC staff acknowledges that an earthquake could cause a rupture of the reactor pool, which could lead to the partial or total loss of coolant. However, this scenario is bounded by the loss-of-coolant accident scenario discussed in Section 13.3 of this SER, which indicated that instantaneous loss of coolant will not lead to fuel damage or the release of fission products. Because the seismic activity in the area is also low, the building was designed and built to exceed seismic building code requirements, and the potential consequence of external events would be bounded by the MHA, the NRC staff concludes that external events at the MSTR would not pose a significant risk to the health and safety of the public.

13.9 Mishandling or Malfunction of Equipment—Reactor Startup Accident

Section 13.1.9 of the SAR describes the scenario of a mishandling or malfunction of equipment—reactor startup accident. In this scenario, the licensee assumes that the three shim/safety rods and the regulating rod are continuously withdrawn while the reactor is subcritical or critical and at zero power. The maximum reactivity insertion rate is estimated to be 0.074% $\Delta k/k$ per second.

The licensee analyzed this scenario in a coupled neutronic thermal-hydraulic computer code and predicts a maximum fuel cladding temperature of 147 degrees C (297 degrees F), which is significantly below the safety limit of 510 degrees C specified in TS 2.1, and no fuel damage is expected. Furthermore, the licensee states that the MSTR RCSs have several interlocks and automatic shutdown circuits built into them to eventually shut the reactor down before any damage to the fuel could occur. The licensee provides some protective scrams as follows:

- period scram, which shuts down the reactor automatically when the reactor period is less than 5 seconds
- reactor power exceeding 150 percent of full power

As discussed in Chapter 7 of this SER and Table TS 3-1, the NRC staff also finds that the log and linear power channel in the RCSs includes a 30-second reactor period rod withdrawal prohibition, which serves to establish a reasonable and conservative limit for normal operations, and 15-second reactor period rundown to provide an additional layer of period protection before reaching the reactor safety system 5-second scram setpoint. In addition, the RCS log and linear power channel has a 120-percent reactor power setpoint at which a reactor rundown is initiated.

Based on the above considerations, the NRC staff concludes that the licensee properly analyzed a reactor startup accident and has sufficient RCSs to shut the reactor down before any damage to the fuel could occur. The NRC staff concludes that the reactor startup accident would not cause fuel damage or the release of fission products.

13.10 Conclusions

The licensee analyzed an MHA and found the radiological consequences to be below the applicable 10 CFR Part 20 regulatory limits for occupational doses and doses to members of the general public. The NRC staff finds that the licensee's assumptions and methods used for calculating doses are conservative and appropriate. The licensee likewise analyzed a number of credible, although highly unlikely, accident scenarios and found the consequences 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 and that the MHA bounds the consequences of the credible accidents. The licensee and the NRC staff do not expect a credible accident to have any offsite radiological consequences at the MSTR. Therefore, the NRC staff concludes that accidents at the MSTR do not pose a significant risk to the health and safety of the public, the facility personnel, or the environment.

14. TECHNICAL SPECIFICATIONS

The NRC staff has evaluated the TS as part of its review of the application for renewal of Facility License No. R-79. The TS define certain features, characteristics, and conditions governing the operation of the MSTR. The TS are explicitly included in the renewed license as Appendix A. The NRC staff reviewed the format and content of the TS for consistency with the guidance found in ANSI/ANS-15.1 and NUREG-1537. Other chapters of this SER discuss the evaluations of individual specifications. The NRC staff specifically evaluated the content of the TS to determine if the specifications meet the requirements in 10 CFR 50.36. The NRC staff concludes that the MSTR TS do meet the requirements of the regulations. The NRC staff based this conclusion on the following findings:

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

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

- To satisfy the requirements of 10 CFR 50.36(d)(1), the licensee provided TS setting a safety limit on the fuel temperature and LSSSs for the RPS to preclude reaching the safety limit.

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

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

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

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

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

15. FINANCIAL QUALIFICATIONS

15.1 Financial Ability To Operate the Reactor

As stated in 10 CFR 50.33(f), "Except for an electric utility applicant for a license to operate a utilization facility of the type described in 10 CFR 50.21(b) or 10 CFR 50.22, [an application shall state] information sufficient to demonstrate to the Commission the financial qualification of the applicant to carry out, in accordance with regulations of this chapter, the activities for which the permit or license is sought."

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

The MSTR is located on the MST campus in Rolla, MO, and MST is one of four State universities under the University of Missouri. The University of Missouri is regulated under the supervision or direction of the Board of Curators of the University of Missouri. The Missouri State Governor, with the advice and approval of the State senate, appoints nine members to serve on the Board of Curators. The September 16, 2008, submittal identified all of the members of the Board of Curators, as well as the MST President and Chancellor.

According to the application, the MSTR is operated within the MST campus and is operated by the staff of the School of Mines and Nuclear Engineering. In the supplemental submittal dated September 16, 2008, the operating costs for the MSTR are estimated to range from $197,800 in fiscal year 2009 to $224,861 in fiscal year 2013. The applicant expects that the MSTR funding source will continue for the above-referenced fiscal years. The NRC staff reviewed the applicant's estimated operating costs and found them to be reasonable.

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

15.2 Financial Ability To Decommission the Facility

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

In the supplement to the application dated September 16, 2008, MST estimated decommissioning costs (fourth quarter of 2007) for the MSTR at $1,944,691. This estimate was based on escalating the 1990 decommissioning cost of $816,000 to 2007 dollars. The NRC staff reviewed the initial 1990 decommissioning cost estimate and concludes that the estimate was reasonable. In the submittal dated September 16, 2008, MST developed a methodology for escalating the decommissioning cost based on NUREG-1307, "Report on Waste Burial Charges," Revision 12. MST used the vendor disposal option with the Richland, WA, site for the disposal cost option; however, the State of Missouri is not eligible to ship waste to the Richland disposal site. In addition, as of July 1, 2008, only members of the Atlantic Compact are permitted to dispose of waste at the South Carolina disposal site. Since the South Carolina site is closed, for estimating purposes, the use of the Richland, WA, disposal rate is reasonable based on the mix of waste and available disposal options. However, when new disposal facilities become available or if the South Carolina disposal site reopens to members outside the compact, disposal rates will likely be significantly higher. Furthermore, the licensee would be under an obligation under 10 CFR 50.9 to update any changes in projected cost, including changes in costs due to increases in disposal costs.

The equation from NUREG-1307, Revision 12, was used to update the cost. Additional input into that formula was based on the U.S. Bureau of Labor Statistics data for the Midwest region for 1990 and 2007. The methodology applied resulted in an average annual escalation of 5.2 percent for 17 years, which is considerably higher than the consumer price index average of 2.91 percent for the same period.

The November 7, 2008, submittal documented that the MST Chancellor has the authority and responsibility to approve funding for future annual operating costs and decommissioning activities associated with operations authorized by NRC Reactor License No. R-79, Docket 50-123. This authority is established by Section 70.010C of the University of Missouri Board of Curators Collected Rules and Regulations. In the November 7, 2008, submittal, the Chancellor stated, "I intend to have funds made available when necessary to decommission the reactor facility belonging to Missouri S&T. I intend to request and obtain these funds sufficiently in advance of decommissioning to prevent delay of required activates." The submittal dated November 17, 2008, contained the oath and affirmation statement for the Letter of Intent that supported the November 7, 2008, submittal.

The NRC staff reviewed the applicant's information on decommissioning funding assurance as described above and finds that the applicant is a State government licensee under 10 CFR 50.75(e)(1)(iv), the statement of intent is acceptable, the decommissioning cost estimate is reasonable, and MST's means of adjusting the cost estimate and associated funding level periodically over the life of the facility is reasonable.

15.3 Foreign Ownership, Control, or Domination

Section 104d of the Atomic Energy Act, as amended, prohibits the NRC from issuing a license under Section 104 of the Act to "any corporation or other entity if the Commission knows or has reason to believe it is owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government." The NRC regulation at 10 CFR 50.38, "Ineligibility of Certain Applicants," contains language to implement this prohibition. According to the application, MST is an agency of the State of Missouri and not owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government. The NRC staff does not know or have reason to believe otherwise.

15.4 Nuclear Indemnity

The NRC staff notes that the applicant currently has an indemnity agreement with the Commission, and said agreement does not have a termination date. Therefore, MST will continue to be a party to the present indemnity agreement following issuance of the renewed license. Under 10 CFR 140.71, "Scope," MST is not required to provide nuclear liability insurance. The Commission will indemnify MST for any claims arising out of a nuclear incident under the Price-Anderson Act (Section 170 of the Atomic Energy Act, as amended) and in accordance with the provisions under its indemnity agreement pursuant to 10 CFR 140.95, "Form of Indemnity Agreement with Nonprofit Educational Institutions," up to $500 million. Also, MST 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 MSTR and, when necessary, to shut down the facility and carry out decommissioning activities. For disposal cost estimating purposes, the use of the Richland, WA, rate is reasonable based on the mix of waste and available disposal options. However, when new disposal facilities become available, or if the South Carolina disposal site reopens to members outside its compact, disposal rates will likely be significantly higher. Furthermore, the licensee would be under an obligation under 10 CFR 50.9 to update any changes in projected cost, including changes in costs due to increases in disposal costs. Finally, the NRC staff concludes that there are no problematic foreign ownership or control issues and no insurance issues that would preclude the issuance of a renewed license.

16. OTHER LICENSE CONSIDERATIONS

16.1 Prior Use of Reactor Components

As detailed in previous sections of this SER, the NRC staff concludes that continued operation of the MSTR will not pose a significant radiological risk. The bases for these conclusions include the assumption that the facility systems and components are in good working condition. However, reactor systems and components may experience chemical, mechanical, and radiation-induced degradation, especially over years of reactor operation. Systems and components that perform safety-related functions must be maintained or replaced to ensure that they continue to protect adequately against accidents. Such systems and components found at the MSTR include the fuel cladding and the reactor safety system.

Section 4.2.1 of this SER describes the reactor fuel. Degradation of the fuel cladding may occur due to (1) thermal cycling and high fuel temperature, (2) radiation damage, (3) erosion, (4) mechanical impact or fuel handling, and (5) corrosion. The following describes these degradation mechanisms:

(1) Because the MSTR does not have any pulse capability, thermal cycling occurs only because of startup and shutdown of the reactor. During a cycle, the maximum cladding surface temperature change is approximately 90 degrees C (194 degrees F). The temperature change at the cladding-fuel interface is not expected to be significantly greater. This temperature change does not have the potential to cause degradation of the fuel cladding. The licensee calculated a maximum cladding temperature of 105 degrees C (221 degrees F) during the maximum power level, while the blister temperature for this type of fuel element in accordance with NUREG-1313 is 527 degrees C (981 degrees F). The NRC staff concludes that this temperature is too low to cause degradation of the cladding.

(2) Aluminum-clad MTR-type fuel does not have a history of failure resulting from radiation damage. The Reduced Enrichment for Research and Test Reactors Fuel Development Program tested the fuel type used at the MSTR at high burnup and observed no fuel failures. The NRC staff evaluated those results in NUREG-1313. Exposure to radiation doses greater than those expected at the MSTR caused no significant degradation in similar fuel plates.

(3) Natural convection cooling does not generate the coolant velocities or pressures necessary to erode the cladding.

(4) The design of in-pool structures and components minimizes the chance of mechanical impact. The design of the standard fuel element places aluminum plates at the outside of the element, effectively shielding the cladding of the fueled plates. The design of the control rod fuel element places aluminum plates on either side of the center gap, thus effectively shielding the cladding of the fueled plates from impact with the control rod. The control rod elements do have fueled outer plates. These elements are centrally located in the core and are thus protected from external impacts. Fuel handling requires specially designed tools that do not come in contact with the cladding. The core plenum shields the fuel from tools and small objects, should they fall into the reactor pool. Based on the designs of the in-pool structures, standard fuel element, control rods, and fuel handling, the NRC staff concludes that there is reasonable assurance that fuel aging will not significantly increase the likelihood of fuel-cladding failure, or the quantity of gaseous

fission products available for release in the event of a loss of cladding integrity for TRIGA fuel operated under the conditions of the MSTR.

(5) TS 3.3 places requirements on the resistivity of the primary coolant. TS 4.3 specifies surveillance intervals for the chemical properties of the coolant. These TS adequately ensure that no significant corrosion of the cladding has occurred during prior use or will occur during the period of this renewed license.

The electrical design of the reactor safety system (safety channel circuitry, control rod magnets, etc.) precludes accidents as a result of system component failure. As discussed in Chapter 7 of this SER, failure or removal for maintenance of safety-related I&C components causes a safe reactor shutdown. TS 4.2 specifies surveillance requirements for the reactor safety system. These requirements ensure the detection of gradual degradation of system components. Additionally, the MSTR staff performs regular preventive and corrective maintenance and replaces system components as necessary. Nevertheless, some equipment malfunctions have occurred. The NRC staff's review indicates that most of these malfunctions were one-of-a-kind and typical of even industrial-quality electrical and mechanical I&C. There is no indication of significant degradation of the I&C, and there is strong evidence that the MSTR staff will remedy any future degradation with prompt corrective action. The licensee has chosen to have a flexible operations schedule that allows shutdown for maintenance when necessary.

The NRC staff did not consider prior utilization of other systems and components because degradation would occur gradually, be readily detectable, and would not affect the likelihood of accidents. Some examples include degradation of the reactor pool liner, secondary coolant pump, and chart recorders. Section 4.3 of this SER discusses pool liner degradation and repair frequency. The licensee monitors pool level and makeup water to detect any loss of pool water that exceeds what is expected from evaporation. Based on the above consideration, the NRC staff concludes that aging related to other components is not significant, and the licensee can detect them early.

16.2 Conclusions

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

17. CONCLUSIONS

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

- The application for license renewal dated August 30, 2004, as supplemented on November 16, November 27, and December 26, 2007, and on January 17, March 6, June 26, September 16, and November 7, 2008, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended, and the Commission's rules and regulations set forth in 10 CFR Chapter I.

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

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

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

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

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