Safety Evaluation Report

Renewal of the Facility Operating License for the University of Missouri-Columbia Research Reactor

License No. R-103
Docket No. 50-186

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

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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 Curators of the University of Missouri (the licensee) for a 20-year renewal of Facility Operating License No. R-103 to continue to operate the University of Missouri-Columbia Research Reactor (MURR). In its safety review, the NRC staff considered information submitted by the licensee, past operating history recorded in the licensee's annual reports to the NRC, inspection reports prepared by NRC personnel, and firsthand observations. Based on its review, the NRC staff concludes that the licensee can continue to operate the facility for the term of the renewed facility license, in accordance with the license, without endangering public health and safety, MURR staff, or the environment.

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ABBREVIATIONS AND ACRONYMS

ADAMS	Agencywide Documents Access and Management System
AEA	Atomic Energy Act of 1954
AEC	air effluent concentration
ALARA	as low as is reasonably achievable
ANL	Argonne National Laboratory
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOR	analysis of record
Ar	argon
ARMS	area radiation monitoring system
ATR	Advanced Test Reactor
CAS	containment actuation system
CEDE	committed effective dose equivalent
CFR	*Code of Federal Regulations*
CHF	critical heat flux
Cs	cesium
DAC	derived air concentration
DCF	dose conversion factor
DDE	deep dose equivalent
DNB	departure from nucleate boiling
DOE	U.S. Department of Energy
ECP	estimated critical position
EEPS	emergency electrical power system
EP	emergency plan
EPFS	emergency pool fill system
EPZ	emergency planning zone
ESF	engineering safety feature
FFA	fuel failure accident
FGR	Federal Guidance Report
FHA	fuel-handling accident
FIR	flow instability ratio
FIRST	flux-trap irradiations reactivity safety trip
FY	fiscal year

GB	glove box
GDC	general design criteria
HC	hot cell
HEPA	high-efficiency particulate air
HEU	highly enriched uranium
HVAC	heating, ventilation, and air conditioning
HX	heat exchanger
I&C	instrumentation and control
INL	Idaho National Laboratory
IR	inspection report
ISG	interim staff guidance
Kr	krypton
LA	license amendment
LC	license condition
LCO	limiting condition for operation
LEU	low-enriched uranium
LOCA	loss-of-coolant accident
LOFA	loss of flow accident
LRA	license renewal application
LSSS	limiting safety system settings
MCNP	Monte Carlo N-Particle Transport Code
MCNP5	Monte Carlo N-Particle Version 5
MHA	maximum hypothetical accident
MKT	Missouri-Kansas-Texas (Nature and Fitness Trail)
MTC	moderator temperature coefficient
MU	Missouri University
MURR	University of Missouri-Columbia Research Reactor
MVC	moderator void coefficient
N	nitrogen
NAA	neutron activation analysis
NCLU	noncoincidence logic circuits
NFPA	National Fire Protection Association
NI	nuclear instrumentation
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation

OSL	optically stimulated luminescence
PCS	primary coolant system
PCMWS	primary coolant makeup water system
PDR	Public Document Room
PF	peaking factor
PoolCS	pool coolant system
PSP	physical security plan
p-tube	pneumatic-tube (system)
QA	quality assurance
QC	quality control
RAC	Reactor Advisory Committee
RAI	request for additional information
RCB	reactor containment building
RCCS	reactor convective cooling system
RCS	reactor control system
RG	regulatory guide
RO	reactor operator
RSC	Radiation Safety Committee
RSO	radiation safety officer
RSS	reactor safety system
RTR	research and rest reactor
RUR	reactor utilization request
SAFSTOR	safe storage
SAR	safety analysis report
SCS	secondary coolant system
SDM	shutdown margin
SER	safety evaluation report
SHE	specialty exhaust hood
SL	safety limit
SNM	special nuclear material
SOI	statement of intent
Sr	strontium
SR	surveillance requirement
SRM	staff requirements memorandum
SRO	senior reactor operator

TEDE	total effective dose equivalent
T-H	thermal and hydraulic
TNT	trinitrotoluene
TS	technical specification
U	uranium
UAl_x	uranium-aluminide
UPS	uninterruptable power supply
USGS	U.S. Geological Survey
Xe	xenon

TECHNICAL PARAMETERS AND UNITS

\$	relative reactivity compared to the delayed neutron fraction
$\Delta k/k$	absolute reactivity
β_{eff}	average delayed neutron fraction
°C	degrees Celsius
cc	cubic centimeters
Ci	curies
Ci/m^3	curies per cubic meter
cm	centimeters
°F	degrees Fahrenheit
ft	feet
ft^3	cubic feet
ft/s	feet per second
ft^3/s	cubic feet per second
g	gram
gal	gallon
gpm	gallon(s) per minute
H_2	hydrogen
in	inch(es)
k_{eff}	k-effective or eigenvalue of a nuclear system (e.g., reactor core or fuel storage)
kg	kilogram
km	kilometer
km/h	kilometer per hour
kPa	kilopascal
kW	kilowatt
kWt	kilowatt thermal
l	liter
lpm	liter(s) per minute
μCi/cc	microcuries per cubic centimeter
μCi/ml	microcurie per milliliter
μmho	measure of conductivity
μmho/cm	μmho per centimeter
μS/cm	microsiemens per centimeter
m	meter

m/s	meter(s) per second
m^3	cubic meter(s)
MeV	million electron volt
mg	milligram
mi	mile
mil	one-thousandth of an inch
mm	millimeters
m/h	miles per hour
mrem	millirem
mrem/hr	millirem per hour
mrem/yr	millirem per year
MW	megawatt
MWt	megawatt thermal
MWt/m^2	megawatt thermal per square meter
n	neutron
n/cm^2	neutrons per square centimeter
Nu	Nusselt
O_2	oxygen
Pe	Peclet
pH	potential of hydrogen
ppm	parts per million
psia	pounds per square inch, absolute
psig	pounds per square inch, gauge
rem	roentgen equivalent man
S	siemens (measure of conductivity)
scf	standard cubic foot (feet)
scfm	standard cubic foot (feet) per minute
St	Stanton
S	Siemens (measure of conductivity)
scf	standard cubic feet
scfm	standard cubic feet per minute
W	watts
W/m-K	watts per meter-degree Kelvin
yr	year

1 INTRODUCTION

1.1 Overview

By letter dated August 31, 2006, as supplemented on January 29, July 16, August 31, September 3, September 30, October 29 (two letters), and November 30, 2010; March 11, and September 8, 2011; January 6 and June 28, 2012; January 4, 2013; January 28, July 31, September 15, and October 1, 2015; and February 8, April 8, April 15, May 31, July 25, August 31, November 7, November 15 (two letters), and December 14, 2016, the Curators of the University of Missouri (the licensee) submitted a license renewal application (LRA) to the U.S. Nuclear Regulatory Commission (NRC or the Commission) for a 20-year renewal of the Class 104c Facility Operating License No. R-103, Docket No. 50-186, for the University of Missouri-Columbia Research Reactor (MURR or the reactor) (Ref. 1). MURR is a multidisciplinary research and education facility providing a broad range of analytic, radiographic, and irradiation services to the research community and commercial sector. MURR is located in the University Research Park, an extension of the campus of the University of Missouri-Columbia in Columbia, MO. A Notice of Opportunity for Hearing was published in the *Federal Register* (FR) on May 20, 2013 (78 FR 29393).

Title 10 of the *Code of Federal Regulations* (10 CFR) 50.51(a) states “[e]ach license will be issued for a period of time to be specified in the license, but in no case to exceed 40 years from the date-of-issuance.” The licensee submitted a Preliminary Hazards Summary Report (Ref. 2) in support of Construction Permit No. CPRR-68, which was issued on November 21, 1961. The licensee submitted the Hazards Summary Report (Ref. 3) and Addendums 1–5 (Ref. 4) in July 1965 in support of the application for a research reactor facility operating license. The NRC issued Facility Operating License No. R-103 (the license) to the Curators of the University of Missouri on October 11, 1966. The term of the license was for a period of 40 years from the issuance of Construction Permit No. CPRR-68 on November 21, 1961. By letter dated October 19, 2001, the NRC issued License Amendment (LA) No. 32 (Ref. 5), which extended the facility operating license expiration date from November 21, 2001, to October 11, 2006, by recapturing the construction time between the issuance of Construction Permit No. CPRR-68 and the issuance of the facility operating license. Because the licensee submitted the LRA to the NRC 30 days before the expiration of the facility operating license, the timely renewal provision provided in 10 CFR 2.109(a) authorizes the licensee to continue operating the MURR facility under the terms and conditions of the current license until the NRC staff completes action on the renewal request. A renewal would authorize continued operation of the MURR facility for an additional 20 years from the issuance of the renewed license.

Construction Permit No. CPRR-68 authorized the construction of a 10-megawatt-thermal (MWt) reactor facility; however, MURR was originally licensed to operate at a power level of 5 MWt until sufficient operating experience was gained to justify power operation at 10 MWt. The NRC staff issued LA No. 2 on July 9, 1974 (Ref. 107), which authorized operation of the MURR facility at a power level of 10 MWt.

The NRC staff based its review of the request to renew the MURR facility operating license on the information contained in the LRA, as well as supporting supplements and the licensee’s responses to the NRC staff’s request for additional information (RAI). Specifically, the LRA includes the safety analysis report (SAR) as supplemented, an environmental report, financial qualifications, and proposed technical specifications (TS). The LRA indicates that there were no requested changes to the physical security plan (PSP), emergency plan (EP), and operator requalification program as a result of the renewal request. The NRC staff conducted site visits on September 3, 2009; March 24 and 25, 2010; November 14, 2012; May 7 and 8, 2013, May 12 and 13, 2015, and December 6 and 7, 2015, to observe facility conditions and to discuss RAIs and RAI responses. The NRC staff issued RAI letters dated May 6, 2010 (Ref. 6); June 1, 2010 (Ref. 7); December 3, 2014 (Ref. 8); April 17 (Ref. 9), June 18 (Ref. 10), October 28 (Ref. 11), and December 17, 2015 (Ref. 12); March 23, 2016 (Ref. 13); August 24 (Ref. 93), and September 7, 2016 (Ref. 100). In addition, the NRC staff conducted telephone conference calls with the licensee on several occasions.

The licensee provided responses to the RAI in letters dated September 14, 2009 (Ref. 14); January 15 (Ref. 15), January 29 (Ref. 16), July 16 (Ref. 17), August 31 (Ref. 18), September 3 (Ref. 19), September 30 (Ref. 20), October 29 (two letters) (Refs. 21 and 22), and November 30, 2010 (Ref. 23); March 11 and September 8, 2011 (Refs. 24 and 25); January 6 and June 28, 2012 (Refs. 26 and 27); January 4 and March 12, 2013 (Refs. 28 and 29); January 27, 2014 (Ref. 30); January 28 (Ref. 31), July 31 (Ref. 32), and October 1, 2015 (Ref. 33); February 8 (Ref. 34), April 8 (Ref. 35), April 15 (Ref. 36), May 31 (Ref. 37), July 25 (Ref. 38), August 31 (Ref. 94), November 7 (Ref. 103), November 15 (two letters) (Refs. 104 and 105), and December 14, 2016 (Ref. 108).

Although the LRA did not request changes to the PSP, EP, and operator requalification program as part of the proposed license renewal, the NRC staff reviewed these plans to ensure they were consistent with current NRC regulations and guidance. The results of the NRC staff's review of the PSP, EP, and operator requalification program are discussed in Sections 12.8, 12.7 and 12.10, respectively, of this SER. As part of the review, the NRC staff also reviewed annual reports of facility operation submitted by the licensee and inspection reports (IRs) prepared by NRC personnel. Information from MURR annual reports cover the period from 2010 to 2015 (Refs. 39, 40, 41, 42, 43, and 44) and the NRC IRs cover the period from 2010 to 2016 (Refs. 45, 46, 47, 48, 49, 50, and 84).

With the exception of the PSP, portions of the SAR, and RAI responses that contain security-related information, material pertaining to this review may be examined or copied for a fee at the NRC's Public Document Room, Room 01 F 21, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852. The NRC staff maintains an Agencywide Documents Access and Management System (ADAMS), which provides text and image files of NRC's public documents. Publicly available documents related to this license renewal may be accessed online through the NRC's Public Library, ADAMS Public Documents collection at http://www.nrc.gov/reading-rm/adams.html. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC's Public Document Room staff by telephone at 1-800-397-4209 or 301-415-4737, or send an e-mail to PDR at Resources@nrc.gov. The PSP and material containing security-related information are protected from public disclosure under 10 CFR 73.21, "Protection of Safeguards Information: Performance Requirements." Since portions of the SAR and RAI responses contain security-related information and are protected from public disclosure, redacted versions are provided to the public in ADAMS.

Section 18, "References," of this safety evaluation report (SER) contains the dates and associated ADAMS accession numbers of the licensee's renewal application, associated supplements, NRC staff's RAI, MURR staff's responses to RAI, and other associated technical information used by the NRC staff during its review.

In conducting its safety review, the NRC staff evaluated the facility against the requirements in the regulations, including 10 CFR Part 20, "Standards for Protection against Radiation," 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,” and 10 CFR Part 70, “Domestic Licensing of Special Nuclear Material;” the recommendations of applicable regulatory guides; and relevant accepted industry standards, such as the American National Standards Institute/American Nuclear Society (ANSI/ANS)-15 series. The NRC staff also considered the recommendations contained in NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,” issued February 1996 (Ref. 51). The NRC staff compared calculated dose values for accidents against the requirements in 10 CFR Part 20.

In SECY-08-0161, “Review of Research and Test Reactor License Renewal Applications,” dated October 24, 2008 (Ref. 52), the NRC staff provided the Commission with information on plans to streamline the review of LRAs for research and test reactors (RTRs). The Commission issued its staff requirements memorandum (SRM) for SECY-08-0161, dated March 26, 2009 (Ref. 53). The SRM directed the NRC staff to streamline the renewal process for such reactors, using some combination of the options presented in SECY-08-0161. The SRM also directs the NRC staff to implement a graded approach whose scope is commensurate with the risk posed by each facility and to incorporate elements of SECY-08-0161 Enclosure 1’s alternative safety review approach. A basic requirement, as contained in the SRM, is that licensees must comply with applicable regulatory requirements.

The NRC staff developed the RTR Interim Staff Guidance (ISG)-2009-001, “Interim Staff Guidance on the Streamlined Review Process for License Renewal for Research Reactors,” dated October 15, 2009 (Ref. 54), to assist in the review of LRAs. The streamlined review process is a graded approach based on licensed power level and divides the RTR facilities into two tiers. Facilities with a licensed power level of 2 MWt and greater, or those requesting a power level increase, undergo a full review using NUREG-1537. Facilities with a licensed power level less than 2 MWt undergo a focused review that centers on the most safety-significant aspects of the renewal application and relies on past NRC reviews for certain findings.

The NRC staff conducted the MURR LRA review using the guidance in the final RTR-ISG-2009-001 (Ref. 54), and because MURR’s licensed power level is greater than 2 MWt, the NRC staff performed a full review of the licensee’s LRA in accordance with the guidance in in RTR-ISG-2009-001 using NUREG-1537.

The NRC staff separately evaluated the environmental impacts of the renewal of the license for the MURR in accordance with 10 CFR Part 51. The NRC staff published an Environmental Assessment and Finding of No Significant Impact in the *Federal Register* on November 29, 2016 (81 FR 86024), which concluded that renewal of the MURR license will not have a significant effect on the quality of the human environment.

The purpose this SER is to summarize the findings of the NRC staff safety review and to delineate the technical details considered in evaluating the radiological safety aspects for continued operation of MURR. This SER provides the technical basis for renewing the MURR license for operation at thermal power levels up to 10 MWt.

This SER was prepared by Geoffrey Wertz, Project Manager in the NRC’s Office of Nuclear Reactor Regulation (NRR), Division of Policy and Rulemaking, Research and Test Reactors Licensing Branch, and Lois Kosmas, a Financial Analyst in the NRR, Division of Inspection and

Regional Support, Financial and International Projects Branch. Energy Research, Inc., an NRC contractor, also provided input to this SER.

1.2 Summary and Conclusions Regarding the Principal Safety Considerations

In its evaluation, the NRC staff considered the information submitted by the licensee, including past operating history recorded in the licensee's annual reports to the NRC, as well as IRs prepared by NRC personnel and firsthand onsite observations. Based on this evaluation and resolution of the principal issues reviewed for MURR, the NRC staff concludes the following:

- The design, use, testing, and performance of MURR structures, systems, and components important to safety during normal operation, discussed in SAR Chapter 4, in accordance with the TSs, are safe, and safe operation of the facility can reasonably be expected to continue.

- The licensee will continue to be useful in the conduct research, education, and isotope production activities, as described in SAR Section 1.1.

- The licensee considered the expected consequences of a broad spectrum of postulated credible accidents and a maximum hypothetical accident (MHA). The licensee performed analysis of the most serious credible accidents and the MHA and determined that the calculated potential radiation doses meet regulatory requirements for unrestricted areas.

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

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

- The TSs, which provide limits controlling operation of the facility, offer a high degree of assurance that the facility will be operated safely and reliably. No significant degradation of the reactor has occurred, as discussed in Chapter 4 of the SAR, as supplemented, and the TSs will continue to help ensure that no significant degradation of safety-related equipment will occur.

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

- The licensee maintains a PSP for the facility and its special nuclear material (SNM) in accordance with the requirements of 10 CFR Part 73, "Physical Protection of Plants and Materials," which provides reasonable assurance that the licensee will continue to provide the physical protection of the facility and its SNM.

- The licensee maintains an EP in compliance with 10 CFR 50.54(q) and Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," to 10 CFR Part 50, which provides reasonable assurance that the licensee will continue to be prepared to assess and respond to emergency events.

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

- Operation of the facility and the handling of radioactive material under the control of the MURR Radiation Protection Program are not expected to result in doses to personnel in excess of 10 CFR Part 20 dose limits and are expected to be consistent with ALARA principles.

On the basis of these findings, the NRC staff concludes that the University of Missouri-Columbia can continue to operate the MURR facility in accordance with the Atomic Energy Act (AEA) of 1954, as amended; NRC regulations; and Renewed Facility License No. R-103 without endangering public health and safety, facility personnel, or the environment. The NRC staff further concludes that the issuance of the renewed license will not be inimical to the common defense and security.

1.3 General Facility Description

The MURR is housed in a dedicated facility located at the edge of the campus of the University of Missouri-Columbia as described in SAR Section 1.1.1. The facility is located in the University Research Park. The overall facility contains the reactor building and associated laboratories, classrooms, and offices.

SAR Section 1.2.2 describes the core assembly within the pool where the core is surrounded by a solid beryllium reflector and graphite reflector assemblies that are clad in aluminum. The pressure vessel and the reflectors are located in a concrete wet-well that is maintained flooded with pool coolant (the water level in the well can be lowered if needed). The pool coolant also acts as a biological shield. The inner surface of the wet-well is lined with aluminum that is bonded to the concrete (see SER Figure 5-4). The reactor is cylindrical with a central region that accommodates experiments. The core comprises eight wedge-shaped fuel assemblies (the terms fuel assemblies and fuel elements are used interchangeably throughout this SER), each with 24 curved plates of highly enriched uranium in an uranium-aluminide (UAl_x) dispersion fuel system. Each plate is clad in aluminum and has a fueled length of 64.77 centimeters (cm) (25.5 inches (in)). The outer radius of the core is 14.694 cm (5.785 in). The reactor is licensed for 10-MWt steady-state thermal power.

The pool is housed inside the RCB, which has concrete walls that are 30 cm (12 in) thick and has a dedicated ventilation system that maintains the reactor area at a negative pressure relative to the outside. The ventilation system also filters air exhausted from the building and routes it to a dedicated stack on the roof. Access to the RCB is through airlocks.

SAR Section 1.2.2 also describes the reactor as using three coolant systems—primary, pool, and secondary. The primary coolant system (PCS), described in SAR Section 5.2, provides pressurized coolant down through the reactor core, which then flows through the primary side of the two heat exchangers (HX), HX 503A and HX 503B, that are located in a mechanical equipment room (Room 114) external to the reactor containment building (RCB). A pressurizer is used to maintain the desired PCS pressure (see SER Figure 5-1). The pool coolant system (PoolCS), described in SAR Section 5.3, removes heat produced in the beryllium and graphite reflectors, the control blades, and the center test hole (flux trap) by drawing PoolCS coolant down through these regions and into a holdup tank, which is then is pumped into the PoolCS side of HX 521 located in Room 114. The coolant is then returned to the pool through the diffuser (see SER Figure 5-2). The secondary coolant system (SCS), described in SAR Section 5.4, removes heat from the PCS and PoolCS through secondary side connections to HX 503A, HX 503B, and HX 521. The heat from the PCS and PoolCS is thus transferred from these three HXs to the three-cell cooling tower, which dissipates the heat to the atmosphere—the ultimate heat sink (see SER Figure 5-3).

Control of the reactor is accomplished with five curved control blades that are described in SAR Section 4.2.2. These blades are positioned external to the pressure vessel. The four shim blades are a combination of boron carbide and aluminum, whereas the regulating blade is constructed of stainless steel. The four shim blades are used for large-scale changes in reactivity, whereas the regulating blade is used for fine control. All four shim blades will automatically insert on loss of power and shut down the reactor; the regulating blade will not.

The control room for MURR is located within the RCB at the same level as the top of the reactor pool. The control room is described in SAR Section 7.3. Area radiation monitors, air particulate monitors, and effluent monitors provide radiation detection. A smoke and fire alarm system and emergency lighting system also serve the RCB. The reactor instrument and control system operates off an uninterruptable power supply.

SAR Chapter 11 describes the MURR Radiation Protection Program. Most of the radioactivity produced by the operation of the reactor is in the form of fission products retained within the fuel elements by the integrity of the fuel cladding. The MURR Radiation Protection Organization handles spent demineralizer resins and normal radioactive laboratory waste. An ALARA Program is in effect at MURR.

SAR Chapter 10 describes the major experimental features particular to the MURR facility. Major experimental facilities consist of a flux trap, a pneumatic transfer system, in-pool irradiation locations, a thermal column, and neutron beam port facilities. The flux trap can have a large effect on core reactivity, and its use is strictly controlled by TSs and is the point of maximum neutron flux in the reactor. The pneumatic transfer system allows samples to be sent from a laboratory to a position adjacent to the reactor core.

1.4 Shared Facilities and Equipment

The MURR facility is located in the University Research Park, but it is an independent facility and does not share utilities with any other facility. University of Missouri-Columbia campus facilities directly supply water and electrical supplies to the MURR facility.

1.5 Comparison with Similar Facilities

The type of UAl_x dispersion fuel used in MURR was used previously at the Idaho National Laboratory beginning in 1952 in the Materials Testing Reactor and the Engineering Test Reactor. This fuel design is also now used in the Advanced Test Reactor (ATR). The ATR uses a similar fuel element design but in a 250-MWt serpentine or cloverleaf arrangement rather than the circular pattern used at MURR. To date, over 3,950 fuel elements similar to MURR fuel have been irradiated at the ATR from 1967 to the present.

1.6 Summary of Operation

The licensee has operated the MURR in accordance with Facility Operating License No. R-103 and has established procedures to facilitate experiments and research, as well as neutron activation analyses. The MURR is also used for student laboratory exercises and student operator training. The MURR annual reports state that the licensee typically operates the reactor over 7,500 full-power hours per year. According to SAR Section 1.5, since September 1, 1977, MURR has had an operating schedule of 24 hours a day, 6½ days per week (averaging approximately 150 hours per week). The weekly half-day shutdown is required for refueling the reactor, changing samples in the flux trap, and performing any corrective or preventative maintenance. This operating schedule equates to operating at full power, 10 MWt, for 90 percent of the available hours in the year. The licensee plans to continue operating at this level during the period of the renewed license.

1.7 Compliance with the Nuclear Waste Policy Act of 1982

Section 302(b)(1)(B) of the Nuclear Waste Policy Act of 1982, 42 U.S.C.§10222(b)(1)(B), specifies that the NRC may require, as a precondition to issuing or renewing a facility operating license for a research or test reactor, that the licensee enter into an agreement with the U.S. Department of Energy (DOE) for the disposal of high-level radioactive wastes and spent nuclear fuel. In a letter dated May 3, 1983, R.L. Morgan of DOE informed H. Denton of the NRC that DOE has determined that universities and other Government agencies operating non-power reactors have entered into contracts with DOE providing that DOE retains title to the fuel and is obligated to store or reprocess the spent fuel and high-level waste. An e-mail, dated January 15, 2014, (Ref. 56), sent from K. Osborne (DOE) to D. Hardesty (NRC) confirms this contractual obligation with respect to the fuel at MURR (DOE Contract No. 72718) continues and is valid from August 1, 2008, to December 31, 2017. Additionally, DOE states that it renews these contracts before their expiration to ensure that they remain valid. By entering into such an agreement with DOE, the licensee has satisfied the requirements of the Nuclear Waste Policy Act of 1982.

1.8 Facility Modifications and History

As detailed in SAR Section 1.7.1, on November 21, 1961, the U.S. Atomic Energy Commission issued Construction Permit No. CPRR-68. Construction commenced on July 16, 1963. On October 11, 1966, the U.S. Atomic Energy Commission issued Facility Operating License No. R-103 for the operation of MURR at power levels up to 5 MWt. The initial fuel type was of uranium-aluminum alloy. SAR Section 1.7.1 explains that the reactor achieved first criticality on October 13, 1966, and reached 5 MWt on June 30, 1967. In order to reduce the fuel cycle cost and the amount of uranium (U)-235 needed per MWD of energy produced at the MURR, a conversion was performed in 1971 to switch to a uranium-aluminide (UAl_x) dispersion-type fuel material with a maximum loading of 775 grams of U-235 per assembly. LA No. 2, issued on July 9, 1974 (Ref. 107), increased the licensed steady-state power level to 10 MWt. According to the SAR, MURR commenced full-power operation, at 150 hours per week, in 1977.

The LRA review considered all of the major changes to the MURR facility as discussed in SAR Section 1.7.4 since initial construction and operation. In addition, in its response to RAI No. 1.a (Ref. 31), the licensee provided a comprehensive list of the facility modifications that were made since original startup. Major equipment upgrades include the following:

- Modification Record 75-16, Addendum 4, “Reactor Safety System Monitoring Circuit (“White Rat”) Panel - Revision to Panel Overlays in Support of Flux-Trap Irradiations Reactivity Safety Trip (FIRST)”

- Modification Record 88-07, Addendum 3, “Exhaust Ventilation in MIB [MURR Industrial Building] Eastward Expansion”

- Modification Record 90-01, Addendum 3, “Evacuation System Changes in Support of the MURR Industrial Building Eastward Expansion”

- Modification Record 94-04, Addendum 1, “Replacement of Pool Coolant System Heat Exchanger HX52-1”

- Modification Record 01-02, Addendum 11, “Intercom and Paging System Changes in Support of the MURR Industrial Building Eastward Expansion”

- Modification Record 01-09, Addendum 5, “Emergency Electrical in MIB Eastward Expansion”

- Modification Record 14-01, “Interfacing MURR System Changes in Support of the MURR Industrial Building Eastward Expansion”

- Modification 14-01, Addendum 1, “Construction of Usable Laboratory Spaces within the Expanded MURR Industrial Building”

- Modification Record 14-02, “Normal Electrical Distribution in MIB Eastward Expansion”

- Modification Record 03-03, Addendum 5, “Fire Protection System Changes in Support of the MURR Industrial Building Eastward Expansion”

- Modification Record 03-03, Addendum 6, “Fire Protection System Changes in Support of the MIB Eastward Expansion Fit-Out”

- Modification Record 04-03, Addendum 2, “Liquid Radioactive Waste in MIB Eastward Expansion”

- Modification Record 14-04, “Lab Impex Iodine Duct Monitor”

- Modification Record 05-06, “Replace Pool Pump Piping”

- Modification Record 05-08, “Replacement of Primary Coolant System Heat Exchangers HX 503A and HX 503B”

- Modification Record 05-08, Addendum 1, “Replacement of Primary Coolant System Heat Exchangers HX 503A and HX 503B - Instrumentation Portion”

- Modification Record 05-08, Addendum 2, “Replacement of Primary Coolant System Heat Exchangers HX 503A and HX 503B - Heat Exchanger and Piping Portion”

- Modification Record 06-03, Addendum 2, “Replace Flux Trap Holder Wear Ring”

- Modification Record 09-01, “Interfacing MURR Systems with the Shipping and Receiving Building”

- Modification Record 11-01, “Flux-Trap Irradiations Reactivity Safety Trip (FIRST) Instrument Channels”

- Modification Record 11-02, “Replace Cooling Tower”

- Project RL-76, “Production of 1-131 Radiochemical Sodium Iodide Solution,” and 10 CFR 50.59 Screen No. 12-07, “Iodine-131 Processing Laboratory”

The NRC staff reviewed NRC IRs covering the period from years 2010 through 2016 (Refs. 45, 46, 47, 48, 49, 50, and 84). The NRC IRs document that the changes were implemented by the licensee in accordance with the requirements in 10 CFR 50.59, “Changes, Tests and Experiments.”

Subsequently, NRC staff identified the need to request additional information specific to the sample processing units (hot cells, fume hoods, and glove boxes), which the licensee provided its responses by letter dated July 31, 2015 (Ref. 35). The NRC staff reviewed the information and, based on its site visits and the information provided above, concludes that the licensee has accurately described the facility changes implemented since original construction of the MURR facility.

1.9 Facility Operating License Changes

The NRC staff reviewed the current MURR facility operating license and finds that the license contains conditions that control the receipt, possession, and use of byproduct material and SNM in accordance with 10 CFR Part 30, “Rules of General Applicability to Domestic Licensing of Byproduct Material,” and 10 CFR Part 70, “Domestic Licensing of Special Nuclear Material.” However, the format and organization used in some license conditions (LCs) were not consistent with that in recently issued licenses. The NRC staff reformatted and reorganized the LCs to make them easier to read and understand.

The renewed facility operating license for the MURR authorizes the receipt, possession, and use of special nuclear material (SNM) and byproduct materials. SNM consists of such material as the uranium-235 in the reactor fuel, plutonium in the plutonium-beryllium neutron source, plutonium in the form of sheets of enclosed in aluminum, and plutonium enriched in plutonium-242 in the form of a rod sealed in a stainless steel can. Byproduct material consists of such material as activation products produced by operation of the reactor in the fuel, experiments, and reactor structure and the sealed antimony-beryllium neutron source.

A description of the MURR facility operating license LCs that were revised are as follows:

Current LC 2.B.(2) authorizes the receive, possess, and use of up to 60 kilograms (kg) of contained uranium-235 of any enrichment, providing that less than 5 kg of this amount be unirradiated. The license requested that 5 kg of the 60 kg of the uranium-235 be uranium-235 of an enrichment less than 20 percent to use in experiments. In the renewed license, the provisions in current LC 2.B.(2) are restates as two license conditions, LC 2.B.2.a and LC 2.B.2.b.

Renewed LC 2.B.2.a authorizes the licensee to the receive, possess, and use, but not separate, in connection with the operation of the facility, up to 55 kg of contained uranium-235 of an enrichment of 20 percent or greater in the isotope uranium-235, providing that less than 5 kg of this amount be unirradiated.

Renewed LC 2.B.2.b authorizes the licensee the receive, possess, and use, in connection with the operation of the facility, up to 5 kg of uranium-235 of an enrichment less than 20 percent in the isotope of uranium-235, for use in experiments.

Current LC 2.B.(2) also authorizes the receive, possess, and use up to 80 grams of plutonium-beryllium neutron source; up to 20 grams of plutonium-239 in the form of sheets enclosed in aluminum for use in connection with operation of the reactor; up to 40 grams of plutonium enriched to 90% plutonium-242 in the form of a rod sealed in a stainless steel can for use in connection with operation of the reactor; and to possess, but not separate, such special nuclear material as may be produced by the operation of the facility. Current LC 2.B.(2) was reformatted as renewed LC 2.B.2.c, LC2.B.2.d, LC 2.B.2.e, and LC 2.B.2.f to facilitate readability.

Renewed LC 2.B.2.c, LC 2.B.2.d, LC 2.B.2.e, and LC 2.B.2.f state:

c. to receive, possess, and use, but not separate, in connection with the operation of the facility, up to 80 grams of plutonium-beryllium in the form of a neutron source;

d. to receive, possess, and use, but not separate, in connection with the operation of the facility, up to 20 grams of plutonium-239 in the form of sheets enclosed in aluminum;

e. to receive, possess, and use, but not separate, in connection with the operation of the facility, up to 40 grams of plutonium enriched to 90 percent in plutonium-242 in the form of a rod sealed in a stainless steel can, and

f. to receive, possess, and use, but not separate, in connection with the operation of the facility, such special nuclear material as may be produced by the operation of the facility.

Current LC 2.B.(2)(a) and LC 2.B.(2)(b) were deleted as these LCs expired on May 31, 1997, and are no longer needed.

Current LC 2.B.(3) was reformatted to renewed LC 2.B.3.a and LC 2.B.3.b to facilitate readability and to prevent the separation of byproduct material, except for byproduct material produced in experiments.

Current LC 2.B.(3) states:

> Pursuant to the Act and 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," to receive, possess, and use in connection with operation of the reactor a source of 100 curies of antimony-beryllium; and to possess, use, but not separate except for byproduct material produced in reactor experiments, such byproduct material as may be produced by operation of the Facility.

Renewed LC 2.B.3 states:

3. Pursuant to the Act and 10 CFR Part 30, the following activities are included:

    a. to receive, possess, and use, in connection with the operation of the facility, up to 100 curies of a sealed antimony-beryllium neutron source; and

    b. to receive, possess, and use, but not to separate, in connection with operation of the facility, such byproduct material as may be produced by operation of the reactor, which cannot be separated except for byproduct material produced in reactor experiments.

Current LC 2.B.(4) was reformatted as renewed LC 2.B.4.a and LC 2.B.4.b to facilitate readability.

The current LC 2.B.(4) states:

Pursuant to the Act and 10 CFR Part 40, "Domestic Licensing of Source Material," to receive, possess, and use in connection with the operation of the reactor up to 20 kilograms each of natural uranium and thorium; and up to 50 kilograms of depleted uranium for instructional and experimental purposes.

The renewed LC 2.B.4 states:

4. Pursuant to the Act and 10 CFR, Part 40, “Domestic Licensing of Source Material,” the following activities are included:

    a. to receive, possess, and use, in connection with operation of the facility, up to 20 kilograms each of natural uranium and thorium; and

    b. to receive, possess, and use, in connection with the operation of the facility, up to 50 kilograms of depleted uranium for instruction and experimental purposes.

The NRC staff discussed the changes with the licensee and the licensee agreed to the changes by letter date December 14, 2016 (Ref. 108).

2 SITE CHARACTERISTICS

2.1 Geography and Demography

2.1.1 Geography

Chapter 2 of the safety analysis report (SAR), as supplemented in the licensee's response to Request for Additional Information No. 2.1 (Ref. 17), indicates that the University of Missouri-Columbia Research Reactor (MURR or the reactor) is located in the University Research Park, which is located approximately 1 mile southeast of the main campus of the University of Missouri-Columbia in Columbia, MO. Columbia, MO, is located in Boone County in the central portion of the State. The reactor facility is located immediately to the east of Gustin Golf Course, which is owned and controlled by the University, and to the west of State Route 163.

The three areas concerning the normal operation, safety, and emergency actions associated with the reactor facility are (1) the area within the operations boundary, (2) the area within the site boundary, and (3) the emergency planning zone (EPZ).

The operations boundary comprises the outer walls of the MURR facility (the laboratory and reactor containment building (RCB)). The area within this boundary is a "restricted access" area for which the MURR facility Director has direct authority and control over all activities, normal and emergency. The adjacent reactor cooling tower building is also included within the restricted access area. A restricted access tunnel connects the cooling tower building to the laboratory building basement. There are pre-established evacuation routes and procedures known to personnel who frequent this area. The operations boundary is within the site boundary.

The site boundary consists of Stadium Boulevard, Providence Road (Route K), the Missouri University (MU) Recreational Trail, and the Missouri-Kansas-Texas (MKT) Nature and Fitness Trail. The area within these boundaries is owned and controlled by the University of Missouri and may be frequented by people unacquainted with the operation of the reactor. The MURR facility Director has authority to initiate emergency actions in this area, if required, in accordance with the provisions in the emergency plan.

In addition, an EPZ has been established for which emergency plans have been developed to ensure that prompt and effective actions can be taken to protect the public in the event of an accident. The MURR EPZ is the area bounded by a 150-meter (m) (492 feet (ft)) radius from the reactor facility ventilation exhaust stack and lies completely within the site boundary.

Important aspects of the site description are included in the design features section of the MURR technical specifications (TSs).

TS 5.1 Site Description

TS 5.1, Specification a, states:

Specification:

a. The MURR facility is situated on a 7.5-acre lot in the central portion of Research Commons, an 84-acre tract of land approximately one mile southwest of the

University of Missouri (MU) at Columbia's main campus. This campus is located in the southern portion of Columbia, the county seat and largest city in Boone County, Missouri.

Approximate distances to the University property lines from the reactor facility are 2,400 feet (732 m) to the north, 4,800 feet (1,463 m) to the east, 2,400 feet (732 m) to the south, and 3,600 feet (1,097 m) to the west.

The restricted, or licensed, area is that area inside the fenced 7.5-acre lot surrounding the MURR facility itself. Within the restricted area the Reactor Facility Director has direct authority and control over all activities, normal and emergency. There are pre-established evacuation routes and procedures known to personnel frequenting this area.

For emergency planning purposes, the site boundaries consist of the following: Stadium Boulevard; Providence Road (Route K); the MU Recreational Trail; and the MKT Nature and Fitness Trail. The area within these boundaries is owned and controlled by MU and may be frequented by people unacquainted with the operation of the reactor. The Reactor Facility Director has authority to initiate emergency actions in this area, if required.

TS 5.1, Specification a, discusses the MURR site features and provides a general description of terms of surface area and location. TS 5.1, Specification a, also establishes approximate distances to the University property lines from the reactor facility.

The U.S. Nuclear Regulatory Commission (NRC) staff finds that TS 5.1, Specification a, provides the important features of the location and boundaries of the MURR facility. This specification contains information used in the dose calculations for the postulated accidents considered by the licensee in Chapter 13 of the SAR and evaluated in Section 13 of this safety evaluation report (SER). The results of these dose calculations helps to ensure that affect radiological safety, such as public doses from the routine operation of the facility as well as the results of the SAR Chapter 13 accident analyses comply with the requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, "Standards for Protection against Radiation." The NRC staff finds that TS 5.1, Specification a, clearly describes the NRC-licensed, or licensed area, and the boundaries of the facility and are consistent with the guidance in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," issued February 1996 (Ref. 51), and the American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.1 2007, "The Development of Technical Specifications for Research Reactors" (Ref. 57). Based on the information provided, the NRC staff concludes that TS 5.1, Specification a, is acceptable.

2.1.2 Demography

As described in SAR Section 2.1.3, the population of Columbia, MO, was 91,885, based on 2000 U.S. Census Bureau data. SAR Table 2-1 reports a low population (255) out to a distance of 1 kilometer (km) (0.6 mile (mi)), with increasing population at farther distances. The license states that this is consistent with land uses because the area within 1 km (0.6 mi) primarily contains sports and academic facilities and venues.

According to the MURR SAR, the University has nearly 30,000 students and expects that number to remain relatively constant. Faurot Field at Memorial Stadium is home to the

University of Missouri football team, and is located 1 km (0.6 mi) northeast of the reactor. The stadium seats 62,000 and it may be occupied during games that occur up to eight times during the fall for periods of up to 4 hours. Adjacent to the football stadium are two indoor sporting facilities: the 15,061 seat Mizzou Arena, which opened in 2004 and the 13,611 seat Hearnes Center, which opened in 1972, and which are also occupied during their respective sporting events.

The NRC staff reviewed U.S. Census Bureau data for the 2010 U.S. Census (Ref. 58) and finds the population of Columbia, MO, is to 108,500, which increased from the estimated data provided in the SAR. The NRC staff finds that the land and surrounding the MURR facility is part of the University of Missouri at Columbia Research Park, and is occupied by various research staff in office buildings during normal business hours of the day. In addition, University of Missouri at Columbia Research Park is not available for development or construction of residential buildings so population growth is not expected in the immediate vicinity of the facility. Furthermore, the NRC staff finds that the licensee has control over the buildings in the Research Park and can implement evacuation of members of the public if necessary, in accordance with the MURR Emergency Plan (see SER Section 12.7). On the basis of its review, the NRC staff concludes that the demographic information from the U.S. Census Bureau is sufficient to allow accurate assessments of the potential radiological impact on the public resulting from operation of the facility.

2.2 Nearby Industrial, Transportation, and Military Facilities

As described in SAR Section 2.2.1, the area surrounding MURR does not contain heavy industry. The major employers are in education; government; retail; services; and, at farther distances, agriculture. Limestone quarries are located in the area, but none are closer than 3 mi (4.8 km) from MURR.

As described in SAR Section 2.2.2, the closest major transportation routes are Interstate 70 and U.S. Route 63. The intersection of these transportation routes is 3.5 mi (5.6 km) from MURR. The nearest rail line to MURR is located 1.7 mi (2.7 km) to the northeast. Also, the Columbia Regional Airport is located 10 mi (16 km) southeast of MURR and serves small and commercial aircraft. There are two runways, the main runway, which is used to support commercial aircraft and a shorter crosswind runway, which is used to support privately operated aircraft.

As described in SAR Section 2.2.3, there are no military facilities in the immediate vicinity of Columbia, MO. The nearest major base is Whiteman Air Force Base, which is located 67 mi (108 km) west of MURR. Whiteman currently hosts the 509th Bomb Wing.

In SAR Section 2.2.4, the licensee states that there are no nearby industrial, transportation, or military facilities with the potential to cause a credible accident (which could prevent a safe reactor shutdown or result in a release of radioactive material from the reactor facility) that would exceed the general public exposure limits of 10 CFR 20. Although shipments of hazardous materials may occur by highway and rail, accidents involving these materials would have no impact on the safe operation of the MURR, because they would occur to far away to have an immediate effect, and the MURR can be placed in a safe condition, by the operators, in a timely manner, if an evacuation of the facility is required.

The NRC staff confirmed the locations of nearby industrial, transportation, and military facilities through review of area maps and site visits. The NRC staff finds that the main runway of the regional airport is oriented north-northwest/south-southeast and is not in line with MURR; but the shorter crosswind runway is aligned in the general direction of MURR. However, due to the 10-mile distance, the smaller aircraft use an approach pattern that does not bring the aircraft within vicinity of the MURR facility. The NRC staff also finds that there are no major industries; transportation routes, or military facilities in the vicinity of the reactor site. Based on its review these facilities, the NRC staff concludes that there is reasonable assurance that normal operations at these facilities will not affect the continued safe operation of MURR.

2.3 Meteorology

SAR Section 2.3 describes the meteorology near MURR. Columbia is in the central part of Missouri, which is in the Great Plains. There are no mountain ranges in the immediate vicinity that affect the weather at MURR. The prevailing wind direction in Missouri is west to east, with local surface winds at MURR primarily from the south. Weather at the site varies by season with snow common in the winter. Storm systems predominantly come from the west.

According to information provided by the licensee and confirmed by the NRC staff through review of National Weather Service data (Ref. 59), monthly average temperatures range from a low of -2.44 degrees Celsius (°C) (27.6 degrees Fahrenheit (°F)) in January to a high of 25.2 °C (77.4 °F) in July. Temperature extremes are a low of -29 °C (-20 °F) and a high of 43.9 °C (111 °F). Average daily humidity ranges between a low of 55 percent in the spring to approximately 85 percent in the summer. Precipitation is distributed throughout the year with an annual average of 99 centimeters (cm) (39 inches (in)). The maximum rainfall in a 24-hour period recorded to date is 15.1 cm (5.95 in) in July 1989. The average snowfall is 68 cm (26.7 in) annually. The maximum recorded 24-hour snowfall was 50.0 cm (19.7 in) in 1979. Snow and ice are routine meteorological conditions that do not affect the operation of the MURR facility. State and local government maintain roads. In addition, construction of the Reactor Containment Building was built to local building codes, which helps ensure that any significant snow and ice events would not adversely impact safe operation of the facility.

Average wind speeds are in the 13- to 19-kilometer-per-hour (km/h) (8- to 12-mile-per-hour (m/h)) range. The most common wind direction recorded is generally from the south, with the least frequent winds from the northeast. The highest recorded 2-minute wind speed was 75 km/h (47 m/h) in 1996. The highest 5-second wind speed was 94 km/h (59 m/h), also in 1996.

SAR Section 2.3.1 describes severe weather in the region, which indicates that the rate of thunderstorm incidence in central Missouri is approximately 52 per year. Because Missouri is inland, hurricanes dissipate prior to reaching the region. The main region for tornadoes is further to the west. The annual average number of days with tornadoes in Missouri is 12. SAR Section 2.3.1 indicates that Boone County has recorded 32 tornadoes over the 55-year period from 1950 to 2005, none of which affected the MURR facility. The containment building is a robust structure composed of 12 in thick reinforced concrete walls. The design features of the containment building are maintained by the specifications provided in proposed TS 5.5 (see SER Section 6.2). Maintenance of the containment building ensures its condition is maintained adequately to protect the MURR from high winds (see SER Section 16.1.5).

The NRC staff reviewed the historical meteorological information provided by the licensee and finds the meteorological information provided for the region around the MURR facility is sufficiently documented. The NRC staff concludes that there are no meteorological-related events or consequences at the site that would pose unacceptable risk to the continued safe operation of the MURR facility.

2.4 Hydrology

SAR Section 2.4 describes the hydrology in the vicinity of the MURR facility. Columbia, MO, is located in the drainage of Perche Creek, which discharges to the Missouri River to the south. The nearest surface water to the MURR facility is Hinkson Creek approximately 300 m (1,000 ft) south of the reactor building. SAR Table 2-11 presents flow data for Hinkson Creek showing an average flow of 67.9 cubic feet per second (ft^3/s) with a peak monthly flow of 301 ft^3/s in late spring and an almost nonexistent flow at other times of the year. SAR Section 2.4.5 provides information on the 100-year flood plain, which extends to a point 61 m (200 ft) from the reactor building but does not include the facility.

SAR Section 2.2.4 describes the MU well-water system, which maintains five wells for water supply in the vicinity of MURR. All of the wells are deep, and the shallowest is at a depth of 366 m (1,200 ft). As discussed in SAR Chapter 5, MURR is a pool reactor with sufficient surrounding structural material to effectively eliminate the neutron flux at the edge of the underground structure and is well above ground water depth. SER Section 4.3 provides a description of the reactor pool and the associated barriers to prevent the migration of reactor coolant from MURR to the environment. Thus, the NRC staff finds that the potential for creation of direct activation products, or the release of radioactive contaminants, outside the pool structure is insignificant.

The NRC staff reviewed the relevant local U.S. Geological Survey (USGS) 7.5-minute series topographic map (Ref. 60) and finds that there is no probable maximum surge and seiche flooding considerations for this site because no large bodies of water are near the site where significant storm surges and seiche can form. In addition, because no existing or proposed dams are upstream of the site, the potential for seismically induced dam failure is not a consideration for the site.

The NRC staff finds that the design of the facility minimizes the potential for contamination of the ground water. Based on the information described above, the NRC staff concludes that the local hydrology does not pose a significant risk to the continued safe operation of the MURR facility.

2.5 Geology, Seismology, and Geotechnical Engineering

SAR Section 2.5 describes the geology in the vicinity of the facility. The ground underneath the MURR facility is various groupings of limestone. Soil above the limestone is primarily glacial deposits.

SAR Section 2.5.2.1 states that there are no known active faults at the site nor any that project to the site from the surrounding area. The nearest known fault is approximately 10 mi (16 km) south of the facility. The State of Missouri has had a history of significant seismic activity. The New Madrid earthquakes of calendar years 1811 and 1812 are among the largest seismic events in the past 300 years in the Continental United States. The calendar year 1811 and 1812 events are estimated to have reach an intensity of XI to XII on the modified Mercalli intensity scale. There have been numerous seismic events associated with the New Madrid faults in the last 200 years but none near the scale of the early large events. The New Madrid region is located approximately 320 km (200 mi) southeast of the facility.

USGS updated its seismic hazard maps (Ref. 61) in 2008 for the United States based on new seismological, geophysical, and geological information (see SER Figure 2-1). The USGS analysis results show a risk of intense seismic activity centered around the New Madrid region. The MURR facility is on the outskirts of the effects of the New Madrid region. The NRC staff reviewed the 2008 National Seismic Hazard Map produced by USGS, which shows only a 2-percent probability that in 50 years the peak lateral ground acceleration will exceed 0.10 times the acceleration due to gravity (Ref. 61).

The NRC staff finds that the licensee has provided sufficient information about geologic features and potential seismic activity at the MURR site. Based on a review of the seismic information, the NRC staff concludes that the geology of the MURR facility is suitable for supporting the reactor building, structure, and systems. The NRC staff also reviewed the accident scenarios described in Chapter 13 of this safety evaluation report and concludes that it is highly unlikely that a seismic event would cause damage to the facility that would result in the release of fission products greater than the maximum hypothetical accident.

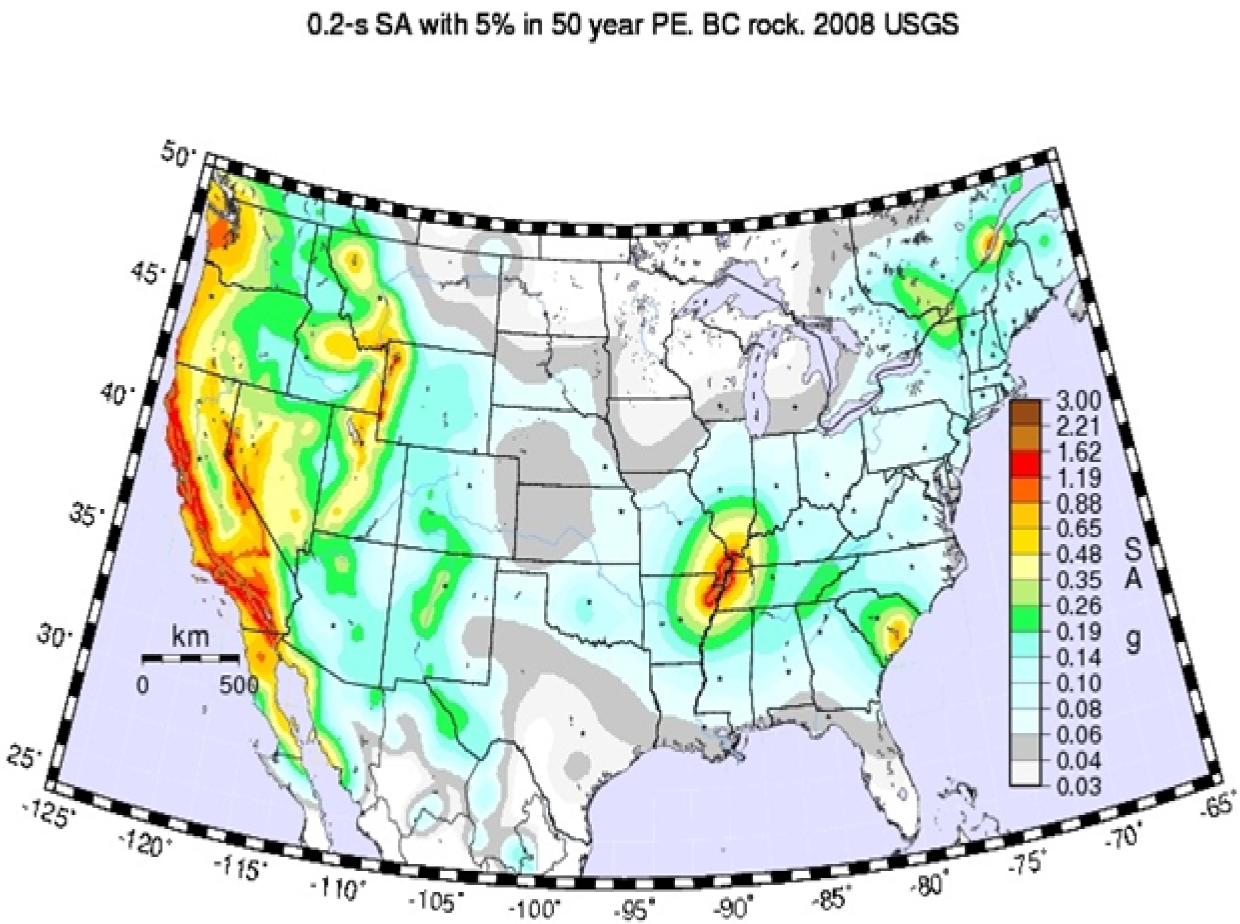

Figure 2-1 USGS 2008 National Seismic Hazard Map

Fukushima Lessons Learned Review

In response to the Fukushima accident in Japan, the NRC staff performed a preliminary assessment for research and test reactors documented in "Draft White Paper Applicability of Fukushima Lessons Learned to Facilities other than Operating Power Reactors," dated March 2, 2015 (ADAMS Accession No. ML15042A367). This assessment was further updated, finalized, and provided to the Commission in SECY-15-0081, "Staff Evaluation of Applicability of Lessons Learned from the Fukushima Dai-ichi Accident to Facilities other than Operating Power Reactors," (ADAMS Accession No. ML15050A066). The assessment identified the need for the NRC staff to perform additional evaluations for the MURR facility, specifically, MURR's ability to address scenarios where extreme external events could possibly result in loss of coolant inventory that could cause inadequate decay heat removal and fuel damage. By letter dated June 1, 2015, (ADAMS Accession No. ML15112A094), the NRC staff informed the licensee of its intention to perform an audit of the MURR to determine if additional regulatory action at MURR was necessary based on Fukushima lessons learned. By letter dated December 8, 2016 (Ref. 101), the NRC staff provided the licensee details of the results of its assessment and concluded that no additional assessments would be needed for seismic-related hazards and no other assessments would be needed for seismic-induced sloshing and high-wind-related hazards. The basis for the NRC staff's assessment can be found in Ref. 101. The NRC staff assessment also concludes that current regulatory requirements for the MURR serve as a basis for reasonable assurance of adequate protection of public health and safety and that no additional regulatory actions are necessary.

2.6 Conclusions

Based on the information provided above, the NRC staff concludes that the MURR facility has experienced no significant geographical, meteorological, or geological change since the issuance of the initial facility operating license in 1967; therefore, the site remains suitable for continued operation of the reactor. The demographics of the area surrounding the reactor have not significantly changed, nor is any significant change projected at this time, that could increase the risk to public health and safety from continued operation of MURR for the 20-year period of the license renewal. Hazards related to industrial, transportation, and military facilities will not pose a significant risk to the continued safe operation of the facility. Infrequency of the occurrence of tornadoes and earthquakes and the robustness of the facility continue to make the site suitable for operation of the reactor.

3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

Chapter 3 of the safety analysis report (SAR) describes the principal architectural and engineering design criteria for the structures, systems, and components that are required to ensure reactor facility safety and protection of the public. SAR Section 3.5 states that the University of Missouri-Columbia Research Reactor (MURR or the reactor) safety-related structures, systems, and components are the fuel, control rod drive mechanisms, reactor core assembly support structure, reactor safety system (RSS), containment system, and the anti-siphon valves.

3.1 Design Criteria

SAR Section 3.1 describes MURR's conformance to General Design Criteria (GDC) 1-57 and GDC 60-64 in Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities." The licensee described the design of the MURR facility components related to the GDC, that the licensee states it follows. Additionally, the SAR describes the GDC as they relate to other elements of the MURR design or processes, such as fire protection, sharing of facilities, control room habitation, and other such elements. The GDC are applicable to nuclear power plants. Research reactors such as the MURR are not required to follow the GDC. The licensee has chosen to consider certain GDC's in its design.

The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the information in SAR Section 3.1. Some of the more important criteria for normal operation and credible accident scenarios include the following:

- General Design Criterion 10, "Reactor Design": The fuel must prevent the release of fission products.

- General Design Criterion 2, "Design Bases for Protection against Natural Phenomena," and General Design Criterion 4, "Environmental and Dynamic Effects Design Bases": The core support structure must maintain its orientation, geometry, and structural integrity.

- General Design Criterion 26, "Reactivity Control System Redundancy and Capability": The RSS must be able to shut down the reactor.

- General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," and "Design Bases for Protection against Natural Phenomena," 15, "Reactor Coolant System Design": The reactor pool must provide adequate shielding of radiation emitted from the reactor core and provide for heat removal from reactor components.

- General Design Criterion 2, "Design Bases for Protection against Natural Phenomena": The reactor building must provide a controllable environment and protect the reactor from external environmental conditions.

With the exception of the fuel, MURR was designed and constructed in accordance with the license issued by the Atomic Energy Act of 1954, as amended, in 1965. Fuel design was changed in 1971 to an aluminide fuel, where the uranium fuel is dispersed throughout an aluminum matrix within an aluminum clad. Section 4.2 of this safety evaluation report (SER) discusses the design of the fuel, control elements, and the core support structure; SER Section 7.2 discusses the design of the RSS; and SER Section 6.2 discusses the design of the reactor building.

MURR has been maintained and/or changed through license amendments or licensee review processes, including maintenance and special procedures under 10 CFR 50.59, "Changes, Tests and Experiments," as appropriate, in accordance with the Commission's rules and regulations and Facility Operating License No. R-103, as amended. The NRC staff previously evaluated all facility license amendments, and the NRC inspection program verified that the licensee conducted the proper reviews. The application for license renewal under review includes changes made to the facility since initial licensing. SER Section 16 discusses age-related issues.

Based on the discussion above, the NRC staff concludes that the design and construction of MURR provides reasonable assurance that the reactor components and systems will continue to meet the design criteria throughout the license renewal period. The design criteria applied to MURR are based on appropriate standards, codes, and criteria and provide reasonable assurance that the facility structures, systems, and components have been built and will function as designed and required by the analyses in the SAR. The licensee has implemented acceptable technical specifications (TSs) to control important aspects of the facility design.

3.2 Meteorological Damage

SER Section 2.3 describes the meteorology in the vicinity of MURR. Meteorological data demonstrates that extreme weather conditions that could affect the structure of the MURR facility occur infrequently. The average snowfall is 68 cm (26.7 in) annually and the maximum recorded 24-hour snowfall was 50.0 cm (19.7 in) in 1979. The highest recorded wind speed in the area was a peak wind gust of 94 kilometers per hour (km/h) (59 miles per hour (m/h)) in 1996. As discussed in the licensee's response to Request for Additional Information (RAI) No. 3.1 (Ref. 17), the reactor building is built to appropriate building codes used at the time of construction and is designed to withstand extreme wind speeds (probably 145 km/h (90 m/h)) associated with thunderstorms or extratropical cyclones possible in the area. The containment building has survived over 40 years of varying site weather without sustaining any damage. The reactor is below grade and the containment building is a robust structure composed of 12 in (312 cm) thick reinforced concrete walls. The design features of the containment building are maintained by the specifications provided in proposed TS 5.5 (see SER Section 6.2). SER Section 6.1.5 discusses maintenance of the containment building.

The NRC staff finds that MURR is located below ground level and is protected by the containment building which is a thick reinforced concrete structure. The NRC staff concludes that, given the meteorological data for the MURR facility, the location of the reactor below ground level and within the containment building helps ensure that significant meteorological damage is very unlikely. The NRC staff also reviewed the MURR facility in response to the Fukushima accident. SER Section 2.5 discusses a summary of the NRC staff's review, which included high-wind related hazards. However, no additional regulatory actions were required to ensure the protection of the public health and safety.

Based on the information provided above, the NRC staff concludes that the design of the structure protects against meteorological damage and provides reasonable assurance that the facility structures, systems, and components will enable safe operations or maintain safe reactor shutdown conditions, to protect the health and safety of the public from radioactive materials and radiation exposure during the license renewal period.

3.3 Water Damage

SER Section 2.4 describes the hydrology in the vicinity of MURR. No bodies of water exist at an elevation higher than the MURR facility in the immediate vicinity of the reactor that could flood the site. The MURR facility is located well above the water table and the nearest surface water source. Historical high levels of precipitation would not raise the water table to the point of inundating the reactor building structure. On this basis, the NRC staff concludes that MURR is not likely to be damaged by water to the extent that would interfere with the safe operation or shutdown of the reactor.

3.4 Seismic Damage

SER Section 2.5 describes the seismicity in the vicinity of MURR. SAR Section 2.5.2.5 provides a detailed review of a seismic assessment of the reactor containment building (RCB) performed in 2001 (Ref. 91). The firm, Sargent & Lundy, performed the seismic assessment analysis for the licensee. The analysis adjusted the seismic response spectra, provided by Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," to reflect the ground acceleration response consistent with the criteria applicable to the Callaway Nuclear Plant, located 30 miles (48.2 km) to the east in Fulton, MO. The analysis focused on the ability of the shear walls to withstand expected seismic acceleration and the material strength of the wall structure. The licensee states that the seismic response assessment was based on a conservative methodology that used a simple equivalent static method to account for the dynamic seismic effects. The analysis concluded that the reinforced shear wall structures had sufficient design margin to withstand expected seismic loads. Furthermore, the assessment indicated that the structural design of MURR RCB would be capable of resisting the operating-basis earthquake and the safe-shutdown earthquake. The analysis did not credit the participation of the massive biological shield in resisting lateral loads.

The NRC staff reviewed the MURR facility in response to the Fukushima accident. SER Section 2.5 discusses a summary of the NRC staff's review, which included seismic considerations. However, no additional regulatory actions were required to ensure the protection of the public health and safety.

The NRC staff observes that the biological shield is a massive concrete monolith containing the entire pool structure. It provides significant lateral protection for the reactor and limits exposure of the reactor from overhead objects. The bridge over the reactor pool and water surrounding the pressure vessel provide further protection and would dampen the effects of any material that may be dislodged during a seismic event. Based on the information provided, the NRC staff concludes that the design of the structure is sufficient to protect the public in the event of seismic activity that can reasonably be expected to occur during the period of license renewal.

3.5 Systems and Components

SAR Section 3.5 indicates that the following systems and components are required to function as designed to ensure safe operation and shutdown:

- reactor fuel
- control rod drive/latch mechanisms
- reactor core assembly support structure
- RSS
- containment system
- anti-siphon system

The licensee has identified the reactor fuel, control rod drive/latch mechanisms, reactor core assembly support structure, and RSS as components important for safe operation of the facility; the SAR and this SER discuss these components in detail. The containment and anti-siphon systems are identified in SAR Chapter 6 as engineered safety features and are discussed in detail in SER Chapter 6.

As discussed in responses to RAI No. 4.5, RAI No. 5.4, RAI No. 10.5, and RAI No. A.41 (Ref. 18), the licensee has a preventive maintenance and surveillance program in place to provide reasonable assurance that the systems and components important to safety meet the performance requirements TS which the NRC staff finds acceptable.

SER Section 4.2.1 discusses the fuel design requirements. SER Chapter 13 evaluates accident scenarios, and SER Chapter 16 considers aging issues associated with the fuel. These discussions show that the fuel cladding design basis and related TSs are adequate to ensure fuel cladding integrity under all credible conditions. SER Chapter 7 discusses the design of the instrument and control systems, including the reactor control system and the RSS. SER Section 4.2.2 discusses the design of the control rods.

Based on its review of the information provided in SAR Section 3.5 and responses to the RAIs, the NRC staff concludes that the reactor fuel, control rod drive/latch mechanisms, reactor core assembly support structure, RSS, containment system, and anti-siphon systems, their design bases and related TSs, provide reasonable assurance that the RSS will function as designed to ensure safe operation and safe shutdown of the reactor.

3.6 Conclusions

Based on the above findings, the NRC staff concludes that the design bases of the electromechanical systems and components give reasonable assurance that the facility systems and components will function as designed to ensure safe operation and safe shutdown of the reactor. The NRC staff also concludes that MURR is adequately designed and built to withstand all credible and probable wind, water, and seismic events associated with the site.

4 REACTOR DESCRIPTION

4.1 Summary Description

Chapter 4 of the safety analysis report (SAR) describes the University of Missouri-Columbia Research Reactor (MURR or the reactor). MURR is a pressurized light-water research reactor that is licensed to operate at a maximum steady-state power level of 10 megawatts thermal (MWt). Figure 4-1 of this safety evaluation report (SER) illustrates the reactor core. The reactor consists of a double-walled pipe; the flux trap is the inner pipe, and the fuel is loaded in as many as eight circular wedges into the outer pipe. Water flows downward through the outer pipe cooling the fuel. Control blades and reflectors are circumferential to the outer pipe.

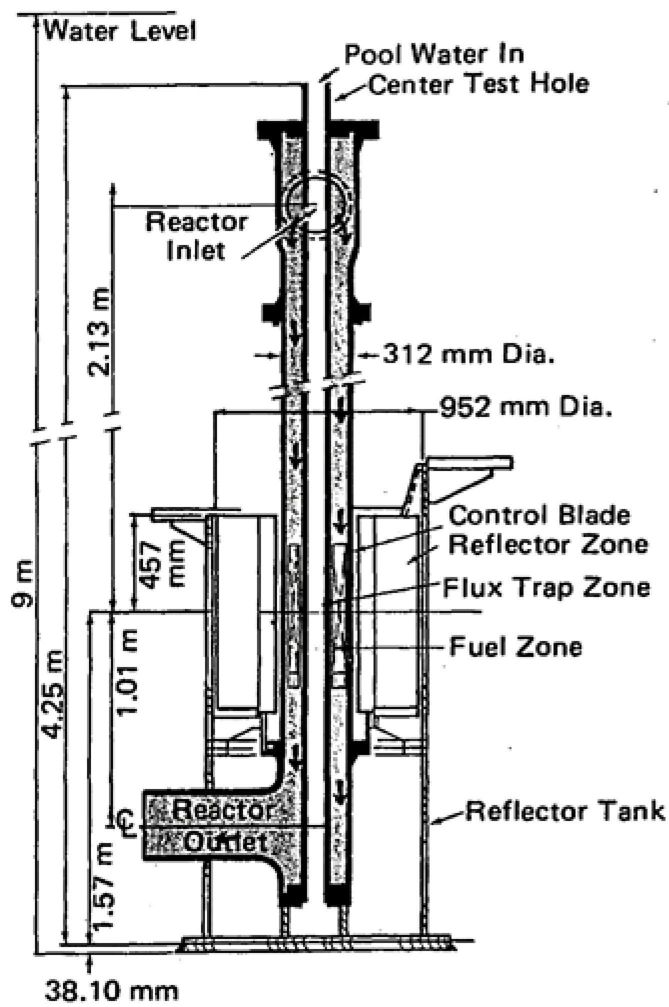

Figure 4-1 Reactor core

The reactor core, which is located in a pressure vessel, formed by the inner and outer pipe walls, is contained in an aluminum-lined pool surrounded by a concrete biological shield.

SAR Section 5.2 describes the core as being cooled by forced convection with water purified by a filtration and demineralizer system. The core can also be cooled by natural convection at low power. The water of the pressurized primary coolant system (PCS) provides moderation, neutron reflection, and is the principle heat sink during forced convection operation. The pool water (coolant) provides shielding, moderation, and neutron reflection and serves the local heat sink for the flux trap, reflector region, and control blades. The pool coolant system (PoolCS) also serves as the ultimate heat sink for the decay heat removal heat exchanger and the heat sink during natural convection cooling at low power.

SAR Section 4.2.5 describes the reactor core support structure, which is welded to the bottom of the aluminum-lined pool. Vertical alignment is maintained by tie rods attached between the pool liner and reactor pressure vessel. Experimental facilities such as the flux trap, graphite reflector irradiation positions, and bulk pool irradiation positions are accessed from bridge assemblies extending over the pool surface.

4.2 <u>Reactor Core</u>

4.2.1 Reactor Fuel

SAR Section 4.2.1.1 describes the reactor fuel. MURR uses highly enriched uranium (HEU) fuel. SAR Section 18.2 states that the MURR core, because of its compact design, requires a high loading density of uranium-235 and that the MURR core cannot perform its intended function without the use of HEU fuel. The use of low-enriched uranium (LEU) fuel requires even higher uranium-235 densities because of the non-fissioning absorption effect of uranium-238. There is no qualified LEU fuel types that can provide the uranium loading densities needed for MURR to operate. The licensee is an active participant in the Department of Energy effort to convert high-performance reactors like MURR from HEU to LEU fuel.

The uranium-aluminide (UAl_x) fuel used at MURR was developed by Idaho National Laboratory and has been used extensively in the high-flux, high-power Advanced Test Reactor (ATR). The “x” in UAl_x represents the stoichiometry, or the ratio of the aluminum to uranium atoms in the fuel mixture, which is 3 for the MURR HEU fuel. The SAR states that UAl_x fuel has unique safety features, including high fission product retention, chemical stability, and dimensional stability. The NRC staff reviewed SAR references for the fuel fabrication and development of MURR fuel (Refs. 62 and 63) and finds the typical void fraction contained in the fuel to be approximately 4 percent. These voids retain the fission products created over the entire fuel lifetime in the core. The NRC staff finds that the references detail testing to burnup levels beyond that allowed by MURR technical specifications (TSs) with no detrimental effects on fuel plate performance.

In its response to Request for Additional Information (RAI) No. 4.1 (Ref. 18), the licensee indicates that the fuel reliability of the UAl_x fuel used at MURR was high. The licensee indicates that similar fuel used at the ATR had not experienced a failure since 1993. The licensee reported one suspected fuel leak at MURR, due to a slight elevation in the iodine-131 activity detected in the PCS in 1997, over the entire lifetime of MURR facility operation. This fuel element was removed from the core early (at 84 percent of its intended usage) and retired. Based on the information provided above, the NRC staff finds that the MURR fuel is highly reliable, and, if a potential fuel defect should occur, it is identifiable by monitoring the PCS radioactivity.

In its response to RAI No. 4.2 (Ref. 17), the licensee states that MURR uses 24 UAl_x fuel plates per element that are 0.050 inch (in) (1.27 millimeters (mm)) thick. Therefore, the center radius of the innermost plate is 2.795 in (7.099 centimeters (cm)), whereas the center radius of the outermost plate is 5.785 in (14.694 cm). The fuel element is 82.55 cm (32.5 in) long.

License Amendment No. 36

In the development of its response to RAI No. 16.1.a (Ref. 25) and RAI No. 16.1.b (Ref. 19), the licensee identifies a 1973 calculation error where the Bernath correlation was incorrectly applied to the calculation of operational limits. The licensee notified the NRC and submitted an application for a license amendment (LA) to correct this by letter dated August 24, 2011 (Ref. 65). The licensee also provides an updated neutronic and thermal-hydraulic (T-H) analyses of the MURR core as part of the LA request and these analyses are referenced in the appropriate sections of this SER. The NRC staff reviewed and approved the LA request and issued LA No. 36 by letter dated July 8, 2013 (Ref. 66).

TS 5.3 Reactor Core and Fuel

TS 5.3, Specifications e, f, g, and h, state:

Specification:

(…)

e. Each reactor fuel element shall contain 24 fuel-bearing plates with a nominal active length of 24 inches and a nominal plate thickness of 0.050 inches. The nominal distance between the fuel plates shall be 0.080 inches. Plate nominal cladding thickness shall be 0.015 inches.

f. The fuel material shall be aluminide dispersion UAl_X nominally enriched to 93% in the isotope uranium-235.

g. Each reactor fuel element shall have a maximum uranium-235 loading of 775 grams.

h. The reactor fuel element cladding material shall be aluminum alloy.

(…)

TS 5.3, Specification e, establishes the fuel geometry requirements for MURR usage. The NRC staff finds that these requirements are shown to be acceptable by the T-H analysis approved in LA No. 36 (Ref. 66). That analysis demonstrates that using this geometry and the limiting safety system setting (LSSS) conditions results in fuel temperatures that are well below the safety limit (SL) temperature. Maintaining this geometry through this specification helps to ensure that the operation of the fuel is consistent with that analysis. These values are identical to those defined in SAR Table 4-3, which were used in the analyses submitted with request for LA No. 36 (Ref. 65), and are acceptable to the NRC staff.

TS 5.3, Specification f, establishes the fuel-enrichment requirement for MURR fuel. The NRC staff finds that this requirement is consistent with the neutronic analysis approved in LA No. 36 (Ref. 66). That analysis demonstrates that the resulting peaking factors (PFs), when used in the T-H analysis, provide acceptable fuel temperatures. Maintaining this enrichment limit helps to ensure that operation of the fuel is consistent with that analysis. This value is identical to the enrichment defined in SAR Table 4-3, which were used in the analysis that forms the basis for LA No. 36 (Ref. 65), and is acceptable to the NRC staff.

TS 5.3, Specification g, establishes the maximum allowable loading of uranium-235 for MURR fuel. The NRC staff finds that this limit is consistent with the analysis documented in LA No. 36 (Ref. 66). That analysis demonstrates that the resulting PFs, when used in the T-H analysis, provide acceptable fuel temperatures. Maintaining this fuel loading helps to ensure that the fuel is operated in a manner consistent with that analysis. This value is identical to the fuel loading defined in SAR Table 4-3 and used in the analysis submitted with the request for LA No. 36 (Ref. 65), and is acceptable to the NRC staff.

TS 5.3, Specification h, requires that the fuel cladding be composed of aluminum alloy. The NRC staff finds that this specification is consistent with the description provided in SAR Section 4.2.1.2, and used in the thermal-hydraulic and neutronic analyses supporting the operation of the MURR. Based on the information provided above, the NRC staff concludes that TS 5.3, Specification h, is acceptable.

The NRC staff reviewed the information provided above describing the design, testing and qualification of MURR fuel and TS 5.3, Specifications e, f and g, regarding fuel design parameters. The NRC staff finds that TS 5.3, Specifications e, f, and g, describe acceptable design features of the fuel and are consistent with descriptions provided in the SAR and LA No. 36 (Ref. 66). Based on the information provided above, NRC staff concludes that TS 5.3, Specifications e, f, g, and h, are acceptable.

TS 3.1 Reactor Core Parameters

TS 3.1, Specifications c and d, state:

Specification:

(…)

c. The reactor core shall consist of eight (8) fuel elements.
Exception: The reactor may be operated to 100 watts on less than eight (8) fuel elements with natural convection cooling (natural convection flange and pressure vessel cover removed) for the purposes of reactor calibration or multiplication measurement studies.

d. The reactor shall not be operated using fuel in which anomalies have been detected or in which the dimensional changes of any coolant channel between the fuel plates exceeds ten (10) mils.

TS 3.1, Specification c, requires that the core may only be operated with a full complement of eight assemblies installed, unless the operation of the core is limited to a power of 100 watts (W) above shutdown power, and natural convection cooling is used. Natural convection cooling requires that the natural convection flange and pressure vessel cover be removed in order to establish the natural convection cooling flow path. The T-H analysis in provide in request for LA No. 36 (Ref. 65), demonstrates that the eight fuel element configuration results in acceptable operating parameters. The NRC staff has considered the consequences of operation of less than eight assemblies at a power of 100 W and concluded that the resulting fuel temperatures would not challenge the SL. The NRC staff also finds that the licensee's response to RAI

No. A.34 (Ref. 17) states that this configuration is only used to evaluate the approach to criticality and, therefore, helps to ensure that operation is consistent with conditions that are analyzed.

TS 3.1, Specification d, requires that the reactor not be operated if fuel anomalies or dimensional changes are detected. In its response to RAI No. 16.1.b (Ref. 19), the licensee described the processes that it uses to inspect for dimensional changes. In addition, in its response to RAI No. 3.1.c (Ref. 37), the licensee further explained the potential for anomalies and their detection methods. The NRC staff reviewed the responses and finds that these responses are acceptable and that the fuel inspection processes cited help to ensure that the reactor is operated with fuel that is consistent with the reactor design, and the supporting analysis submitted with the request for LA No. 36 (Ref. 65).

SAR Section 4.2 provides the basis for the burnup limit. The licensee states that experience with the ATR fuel has shown that this burnup level results in less than 10-percent swelling of the fuel plates and no detrimental effects on the fuel performance. The swelling limit correlates to a value that will result in less than 10-percent swelling of the fuel plates, which ensures no detrimental effect on the fuel plate performance. Furthermore, it conforms to the recommendations from representative fuel irradiation studies (Ref. 63).

In its response to RAI No. 4.3 (Ref. 23), the licensee states that hydraulic testing of MURR-plate-type fuel is conducted during the qualification and testing stage and that the fuel plates are subject to higher flow velocities, temperatures, and pressure drops than typical MURR operating parameters. The licensee states in SAR Section 4.2.1.1 that the burnup limit is based on data that indicate that fuel plate swelling is less than 10 percent and has no detrimental effect on fuel plate performance.

The response to RAI No. 1 (Ref. 15) includes fuel design information, and states that the fuel elements contain aluminum alloy 6061-T6 handling and guide fixtures at each end of the fuel and 304 stainless steel rollers to facilitate configuration within the core support structure.
The NRC staff reviewed TS 3.1, Specifications c and d, regarding fuel burnup and operational limits. The NRC staff also reviewed licensee references regarding fuel testing and composition. The NRC staff finds that TS 3.1, Specifications c and d, help to ensure that the reactor is only operated in configurations and within operational limits that have been analyzed in the SAR, and minimize the potential for damage to the fuel. Based on the information provided above, the NRC staff concludes that TS 3.1, Specifications c and d, are acceptable.

SAR Chapter 4 states that the primary design objective of MURR fuel is the maintenance of fuel integrity under any anticipated operating conditions. The regulation, 10 CFR 50.36, "Technical Specifications," requires licensees to propose TSs that include SLs to reasonably protect the integrity of barriers that guard against the release of radioactivity. The NRC staff finds that SAR and various RAI responses accurately characterize the fuel elements to be used in MURR. These discussions include the design limits of the fuel elements and clearly give the technological and safety-related bases for these limits. The license renewal application refers to the fuel development program under which all fuel characteristics and parameters that are important to the safe operation of the reactor were determined. The design limits are clearly identified for use in design bases and support the limits stated in the TSs consistent with 10 CFR 50.36. Information on the design and development program for this fuel offers reasonable assurance that the fabricated fuel can function safely in the reactor without adversely affecting public health and safety. Based on the information provided above, the

NRC staff concludes that the MURR descriptions of the fuel in the SAR, the responses to RAIs, and the associated TSs, described above, are acceptable.

4.2.2 Control Rods

SAR Section 4.2.2 states that the reactivity of the reactor is controlled by five control elements in the form of curved blades that travel in the annular space between the reactor pressure vessel and the beryllium reflector. These control elements are typically referred to as control blades, but the SAR uses the terms blades and rods interchangeably (see SAR Section 1.2.2). The compliment of control blades includes four shim control blades and a single regulating control blade.

In its responses to RAI No. 4.5.a (Ref. 18) and RAI No. 4.7 (Ref. 27), the licensee provides additional details regarding the control blade design. The licensee states that the boron carbide-aluminum plates are clad with 0.9525 mm (0.0375 in) of aluminum alloy 1,100 for a nominal blade thickness of 4.445 ± 0.178 mm (0.175 ± 0.007 in). The shim control blades use BORAL® plate for absorbing material, which is, by weight, 52-percent boron carbide and 48-percent aluminum (± 2 percent). The regulating blade is constructed of stainless steel, and the neutron absorbing material is also stainless steel.

Each of the four shim blades occupies approximately 72 degrees of a circular arc around the pressure vessel. The regulating blade occupies approximately 18 degrees of the arc. The radial shape of the control blades provides additional stiffness that aids the hydraulic stability of the control blades in the operating gap between the outer reactor pressure vessel and the beryllium reflector. During normal operation, only a portion of the shim blade operates in this gap, resulting in a minimized downward force because of flow.

In its response to RAI No. 1.i (Ref. 37), the licensee clarified control blade operation by explaining that the shim blades have scram and run-in capability and that the regulating blade does not have scram or run-in capability. The TS definition of the regulating blade (TS 1.29) in the proposed TSs more fully describes the regulating blade.

TS 5.3 Reactor Core and Fuel

> TS 5.3, Specification k, states:
>
> Specification:
>
> (…)
>
> k. The reactor shall have five (5) control blades between the pressure vessel and beryllium reflector. Four (4) of the control blades shall be made of boron and aluminum for coarse control (shim blades) of reactor power. One (1) control blade shall be made of stainless steel for fine control (regulating blade) of reactor power.
>
> (…)

TS 5.3, Specification k, requires that five control blades be installed between the pressure vessel and the beryllium reflector. The NRC staff finds that this specification is consistent with the description in SAR Section 4.2.2.1 for the purpose established in General Design

Criteria 26, “Reactivity Control System Redundancy and Capability,” in Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” as stated in SAR Section 3.1.4. SAR Section 4.2.2.2 states that the control blade system is designed to ensure safe reactor control and shutdown under all operating conditions. The NRC staff finds that the reactor shutdown capability is maintained from the most reactive state with the most reactive control rod stuck in the fully withdrawn position. Based on its review of the information provided above, the NRC staff concludes that TS 5.3, Specification k, is consistent with the SAR, and design features required by 10 CFR 50.36, and is acceptable.

TS 3.2 Reactor Control and Reactor Safety Systems

TS 3.2, Specifications a through e, state:

Specification:

a. All control blades, including the regulating blade, shall be operable during reactor operation.

b. Above 100 kilowatts, the reactor shall be operated so that the maximum distance between the highest and lowest shim blade shall not exceed one inch.

c. The shim blades shall be capable of insertion to the 20% withdrawn position in less than 0.7 seconds.

d. The maximum rate of reactivity insertion for the regulating blade shall not exceed 1.5×10^{-4} Δk/k/sec.

e. The maximum rate of reactivity insertion for the four (4) shim blades operating simultaneously shall not exceed 3.0×10^{-4} Δk/k/sec.

(…)

TS 3.2, Specification a, requires that the reactor only be operated when all control blades are operable. SAR Section 4.2.2 explains that the control blades are necessary to adequately control the reactor power level and safely and rapidly shut down the reactor, if necessary. The NRC staff finds that operability of the control blades is required to satisfy the assumptions used in the MURR safety analysis discussed in SAR Chapter 13 and submitted with the request for LA No. 36 (Ref. 65). Based on its review of the information provided above, the NRC staff concludes that TS 3.2, Specification a, is acceptable.

TS 3.2, Specification b, requires the licensee to ensure that control blade height differences are maintained within one inch above a reactor power level of 100 kilowatts, so that the PFs during operation are consistent with the SAR, and supported by the analysis submitted with the request for LA No. 36 (Ref. 65). In its response to RAI No. 4.14.d (Ref. 25), the licensee provides states that excessive control blade height differences can result in flux tilts that can then lead to perturbations in the predicted power peaking. Maintaining a maximum 1-in height distance limit between the highest and lowest control blades helps to ensure that the power peaking conforms to the analysis submitted with the request for LA No. 36 (Ref. 65). Additional discussion on the PFs is provided in SER Section 4.5.1. Based on the information provided above, the NRC staff concludes that TS 3.2, Specification b, is acceptable.

TS 3.2, Specification c, requires the licensee to ensure that the shim blades can insert to 80 percent within 0.7 seconds. In its response to RAI No. 1.a.i (Ref. 37), the licensee provide the description of control rod insertion and drop time indicating that the control rods were designed to move from their fully inserted position to their 20-percent withdrawn position (or 80-percent inserted) in less than 0.7 seconds. The licensee states that this ensures prompt shutdown of the reactor in the event that a reactor scram signal, manual or automatic, occurs. The 20-percent withdrawn position is defined as 20 percent of the control blade full travel of 26 in (5.2 cm) measured from the fully inserted position. Below the 20-percent withdrawn position, the control blade insertion travel is cushioned (and slowed) by a dashpot assembly to minimize wear on the drive mechanism. The licensee indicated that approximately 91 percent of the control blade total reactivity worth is inserted at the 20-percent position. Thus, the remaining 20 percent insertion only provides 9 percent of the control blade reactivity worth. The 80 percent insertion time of the control blade is an original design feature of the reactor, and has not been altered over the life of the facility. As such, the TS remains the same for license renewal.

The NRC staff finds that the licensee's accident analyses used the 91 percent reactivity worth of the control blade, associated with the 80 percent inserted position. Since the facility currently has no capability to measure the full in control blade drop time, the licensee proposes to continue to credit only 91 percent of the total blade worth in accident analyses using the insertion time of 80 percent of the blade. The NRC staff finds that the licensee uses this assumption consistently in its responses to RAI No. 1.a.ii and RAI No. 1.a.iii (Ref. 37), thus helping to ensure that the reactivity insertion values used are consistent with the TS. The 0.7-second insertion time to the 20-percent withdrawn position is an assumption in the control system response as analyzed and found acceptable in SER Section 13.2.2. Based on the information provided above, the NRC staff concludes that TS 3.2, Specification c, is acceptable.

TS 3.2, Specification d, limits the reactivity insertion rate of the regulating blade. According to the TS basis statement, this limit is imposed to ensure that the rate of reactivity change provides a reasonable response from operator-initiated manual control blade insertions and withdrawals. The NRC staff finds that the reactivity insertion analysis, described in SER Section 13.21, indicates that the value cited in this specification is bounded by the analysis, which demonstrates that the fuel temperature results of the reactivity insertion transient are bounded by the SL. The NRC staff also finds that acceptable control rod dynamic characteristics are ensured by TS 3.2, Specification d, which limits the reactivity addition rate to 0.00015 absolute reactivity ($\Delta k/k$) per second. Based on the information provided above, the NRC staff concludes that TS 3.2, Specification d, is acceptable.

TS 3.2, Specification e, limits the rate of reactivity change associated with the four shim blades moving simultaneously. This specification helps to ensure that the maximum rate of reactivity insertion from the four shim control rod blades is limited to values that do not challenge the SL limits associated with the fuel cladding. As evaluated in SER Section 13.2.2, the analysis provided by the licensee demonstrates that the value cited in this specification is bounded by the Ramp Reactivity Insertion Accident analysis, and the results of the transient fuel temperatures are below the SL. The NRC staff finds that the TS 3.2, Specification e, reactivity insertion value is bounded by the analysis evaluated in SER Section 13.2.2 and found acceptable. Based on the information provided above, the NRC staff concludes that TS 3.2, Specification e, is acceptable.

The NRC staff reviewed TS 3.2, Specifications a through e, and finds that they provide key performance criteria for ensuring that the control blades are operable and can perform their function to control power distribution and reactivity. The NRC staff finds that the analysis provided by the licensee in the SAR and in RAI responses confirms that the values analyzed demonstrate acceptable reactor response to the uncontrolled rod (blade) withdrawal event and that the resulting fuel temperatures are less than the SL. Based on the information provided above, the NRC staff finds that TS 3.2, Specifications a through e, require LCOs consistent with 10 CFR 50.36, are acceptable.

TS 4.2 Reactor Control and Reactor Safety Systems

TS 4.2, Specifications a through f, state:

Specification:

a. All control blades, including the regulating blade, shall be verified operable within a shift.

b. The drop time of each of the four (4) shim blades shall be measured at quarterly intervals.

c. A different one of the four (4) shim blades shall be inspected semiannually so that every blade is inspected biennially. The reactor shall not be operated with a control blade that exhibits abnormal swelling or abnormalities that affect performance.

d. Above 100 kilowatts, the distance between the highest and lowest shim blade shall be verified within a shift.

e. The reactivity insertion rate of the regulating blade shall be verified on an annual basis by measuring the withdrawal and insertion speeds.

f. The reactivity insertion rate of each shim blade shall be verified on an annual basis by measuring the withdrawal and insertion speeds.

g. The total reactivity worth of each shim blade shall be measured annually or following any significant core configuration change from reference core condition. A significant core configuration change is defined as a change in reactivity greater than 0.002 $\Delta k/k$.

(…)

TS 4.2, Specification a, requires the licensee to periodically verify control blade operability. The NRC staff finds that this specification helps to ensure that the control blades are operable, consistent with the safety analysis provided in SAR Chapter 13. Furthermore, the NRC staff finds that TS 4.2, Specification a, provides confidence that the control blades will perform their control (movement) function when required. The NRC staff also finds that the surveillance interval is consistent with the guidance in NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,” issued February 1996 (Ref. 51), and American National Standards Institute/American Nuclear Society (ANSI/ANS)-15.1-2007, “The Development of Technical Specifications for Research Reactors,” issued 2007 (Ref. 57). Based on the information provided above, the NRC staff concludes that TS 4.2, Specification a, is acceptable.

TS 4.2, Specification b, requires the licensee to measure the drop time of the scrammable shim blades quarterly. The NRC staff finds that this specification helps to ensure that the scram capability of the shim control blades is ensured and is consistent with the analysis supporting operation such as shutdown margin (SDM). The NRC staff also finds that the surveillance interval stated is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 4.2, Specification b, is acceptable.

TS 4.2, Specification c, requires that the scrammable shim blades be inspected to prevent their use if swelling or abnormalities could affect their performance. Specifically, one of the four shim blades must be inspected semiannually such that every bladed is inspected biennially. In its response to RAI No. 4.5.c (Ref. 18), the licensee states that removal and inspection of the control blades is conducted semiannually to visually inspect and use geometric templates to detect swelling, distortion, blistering, rubbing, or other abnormalities. The NRC staff finds that periodic inspection of the shim blades enables detection of blade abnormalities, and any potential generic blade design deficiencies, and prevents the reactor from operating with blade defects. In response to RAI 4.5.a (Ref. 18), the licensee confirms that the results of inspections of shim control blades have shown no signs of operational limitations or flow erosion and minimal dimensional changes caused by the hydraulic and radiation environment. Based on the operational history of MURR, the NRC staff finds that the current surveillance frequency is adequate to detect developing abnormalities in the control blade. Furthermore, the NRC staff finds that the surveillance interval stated is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 4.2, Specification c, is acceptable.

TS 4.2, Specification d, requires a verification that, above a power level of 100 kilowatt (kW), the distance (height difference) between the shim blades is within 1 in (2.54 cm). The NRC staff finds that this specification helps to ensure that operation above 100 kW does not invalidate assumptions regarding the calculated PFs that were used as assumptions in the licensee's analysis of the operating T-H conditions, fuel and cladding temperatures, and reactor response to control blade movement, as provided in the licensee's response to RAI No. 4.14.d (Ref. 25). The NRC staff also finds that the surveillance interval is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 4.2, Specification d, is acceptable.

TS 4.2, Specification e, requires that the reactivity insertion rate of the regulating blade be verified on an annual basis by measuring the withdrawal and insertion speeds. The NRC staff finds that verification of the regulating blade reactivity insertion rate by measuring the withdrawal and insertion speeds helps ensure that the reactivity assumption used in the analysis provided in SAR Section 13.2.2.1.2 and in the response to RAI No. 13.3 (Ref. 25) is maintained. The NRC staff's evaluation of the analysis is documented in SER Section 13.2.2 and found acceptable. The NRC staff finds that the resulting fuel temperatures are less than the SL. In addition, the NRC staff finds that the surveillance interval is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 4.2, Specification e, is acceptable.

TS 4.2, Specification f, requires the reactivity insertion rate of each shim blade shall be verified on an annual basis by measuring the withdrawal and insertion speeds. The NRC staff finds verification of the shim blade reactivity insertion rate helps ensure the assumption used with the analysis in SAR Section 13.2.2.1.1 is maintained, and the resulting fuel temperatures are demonstrated by the analysis to be less than the SL. The NRC staff also finds that the surveillance interval is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 4.2, Specification f, is acceptable.

TS 4.2, Specification g, requires that the total reactivity worth of each shim blade be measured annually or following a change in reactivity as a result of a core configuration change of greater than 0.002 Δk/k. The NRC staff finds that TS 4.2, Specification g, is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 4.2, Specification f, is acceptable.

The NRC staff finds that TS 4.2, Specifications a through g, are consistent with the safety analysis in SAR Chapter 13 and the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 4.2, Specifications a through g, contain necessary SRs consistent with 10 CFR 50.36 and are acceptable.

SAR Section 7.5 discusses control blade drive controls, including independence about operation. These discussions include descriptions of general control independence in the manual and automatic control modes, as well as circuit features such as rod withdrawal prohibit, rod run-in, and automatic shim control. In its response to RAI No. 4.5.b (Ref. 18), the licensee states that there are no automatic circuits that could withdraw one or more shim blade drives and that there are two automatic control circuits, rod run-in and automatic shim control, that could insert the shim blade drives. SAR Section 7.5.4, describes the RCS automatic mode of reactor control, which modulates the regulating blade, for both insertion and withdrawal, based on the desired power level. The operator performs all other blade drive operations manually.

In response to RAI No. 4.7 (Ref. 27), the licensee provides an analysis that calculated the thermal conductivity of Boral® control blades (lower bound of 98 watts per meter-degree Kelvin (W/m-K), upper bound of 132 W/m-K, and the maximum radial distortion (0.0004 in) (0.001 cm)). The NRC staff reviewed the clearances provided in the control blade gaps and finds that they are adequate to accommodate, without significant interference, the maximum anticipated degree of control blade distortion due to thermal gradients produced during full-power operation.

Based on the information provided above, the NRC staff concludes that the reactor control system and reactor safety systems are effective to ensure reactor operability and shutdown capability, and will protect the fuel integrity. The NRC staff concludes the referenced TSs, are consistent with 10 CFR 50.36, and will provide assurance that public health and safety will be protected.

4.2.3 Neutron Moderator and Reflector

SAR Section 4.2.3 states that the light water surrounding the reactor core provides the principle moderator and coolant. It also states that a beryllium annulus forming the outer wall of the control blade gap serves as the principal reflector for the reactor core. Also, vertical graphite elements canned in aluminum surround the beryllium reflector to provide additional reflection.

TS 5.3 Reactor Core and Fuel

TS 5.3, Specification j, states:

Specification:

(…)

j. The reactor shall have a beryllium and graphite reflector.

(…)

TS 5.3, Specification j, requires that the operational core must have beryllium and graphite reflectors to help ensure that the flux gradients in the neutronic models are consistent with those experienced during actual operation. The NRC staff finds that TS 5.3, Specification j helps to ensure that the resulting power distribution PFs are consistent with the T-H and safety analyses. The T-H analysis submitted with the request for LA No. 36 (Ref. 65) indicates that the MURR neutronic models used both beryllium and graphite; and the results were acceptable. Based on the information provided above, the NRC staff concludes that TS 5.3, Specification j, is acceptable.

SAR Section 4.2.3 states that the licensee has the ability to replace the beryllium reflector materials on a scheduled or as-needed basis based on operation of the core. Replacement of the beryllium reflector is performed to prevent cracking, and is operational decision as the potential for beryllium cracking does not constitute a safety issue, as discussed below. Accordingly, the licensee has determined that a TS requirement to replace the beryllium is not required. In its response to RAI No. 4.8.c (Ref. 18), the licensee states that the first beryllium reflector cracked while in service in May 1981. After verification that the control rods were free to travel and met the control rod drop time TS, the reactor operated with the cracked beryllium until it was replaced in October 1981. The licensee states that the beryllium reflector is scheduled to be replaced approximately every 26,000 megawatt-days, which is an interval less than the previous reflector cracking incident, in order to avoid potential operation with a beryllium reflector with a crack again. The NRC staff reviewed the licensee's RAI response and concludes that operation with a potential beryllium crack does not constitute a safety issue because the operating with the beryllium crack does not compromise control rods movement. On this basis, the NRC concludes that a TS is not necessary.

In its response to RAI No. 4.8.a (Ref. 18), the licensee states that graphite reflector elements can be replaced as needed. Some are replaced periodically because of reconfiguration of their respective irradiation positions. The licensee also states that the aluminum used to fabricate these reflector elements has a long service history with regard to radiation fields and is particularly well suited for a neutron rich environment. Thermal expansion in the graphite reflector is accommodated by the generous design tolerances within and between the reflector elements.

In its response to RAI No. 4.8.d (Ref. 18), the licensee states that no potential experiment facility malfunctions in the reflector region could affect reactor core components. The most likely experiment facility malfunction in the reflector region is sample failure by over-pressurization and a discharge of the sample's contents and possible damage to adjacent sample cans.

In addition to the reactivity-worth limitations placed on experiments (see TS 3.8, Experiments), Table 4-1 below lists the required safety analyses and TSs necessary to support a reactor utilization request, which is the administrative process by which the reactor manager and health physics managers approve facility experiments, and that is required to demonstrate that the experiment meets all applicable TS requirements.

Table 4-1 Required Safety Analysis for Experiments

Experiment Safety Analysis	TS
Thermal Analysis	TS 3.8.r and TS 3.8.t
Sample Decomposition-Pressure Analysis	TS 3.8.s
Experiment Failure Analysis	TS 3.8.n
Loss-of-Coolant Analysis	TS 3.8.o
Failure of Other Experiments Analysis	TS 3.8.p
Corrosion Analysis	TS 3.8.j and TS 3.8.l
Explosive Analysis	TS 3.8.i

In accordance with the requirements in TS 6.2, Specification a.(3), the Reactor Advisory Committee shall review any experiment that is significantly different from any previous experiment or that involves a question pursuant to 10 CFR 50.59, “Changes, Tests and Experiments,” as defined in SAR Section 10.4.4.

In its response to RAI No. 4.8.d (Ref. 18), the licensee provides an analysis of the possible hazards associated with graphite-stored energy (Wigner energy) and the credibility of a graphite fire in research reactor applications. The analysis concludes that Wigner energy generation in MURR graphite is small; the MURR approach to canning graphite in a thermally insulated helium gap promotes thermal annealing to reduce the storage of Wigner energy, and the MURR graphite is enclosed in several noncontiguous masses under several feet of water. The energy available to ignite combustion is insufficient, or the air or other oxidizer available to sustain combustion is inadequate.

The NRC staff evaluated the neutron moderator and reflector systems in the SAR. The NRC staff has also reviewed the Wigner energy analysis and finds that Wigner energy in the MURR graphite reflector poses no potential hazard to the safe operation of the reactor because the Wigner energy generation in MURR graphite is small, the graphite is contained in a thermally insulated helium gap which reduces the storage of Wigner energy, and the graphite is located under several feet of water. Based on its evaluation, the NRC staff concludes that continued operation, within the requirements of the TS, provides reasonable assurance that the moderator and reflector systems designed for this reactor will perform as necessary and will not adversely affect safe reactor operation, prevent safe reactor shutdown, or cause uncontrolled release of radioactive material to the unrestricted environment.

4.2.4 Neutron Startup Source

SAR Section 4.2.4 states that an antimony-beryllium neutron source is used to provide source neutrons to support reactor instrumentation response checks and subcritical measurements of fuel storage racks and shipping casks. The source material is double encapsulated in stainless steel. A tagged handling line identifies the source and allows the source to be relocated from its storage position in the deep section of the reactor pool to the irradiation location in the graphite reflector region. An administrative control prevents over-irradiation of the source above the strength of 100 curies (Ci) (1 Ci equals 3.7×10^{10} disintegrations per second) as limited by renewed License Condition (LC) 2.B.3.a, as described in SER Section 1.9. The antimony-beryllium source is not typically used for routine reactor startup operations because the source neutron strength from activation of the beryllium reflector is sufficient.

In its response to RAI No. 4.9 (Ref. 18), the licensee provides calculations to show that an irradiation time of 5.0 hours per 30 days would approach the licensee activity limit of 100 Ci. In order to maintain the source strength less than the license activity limit, the antimony-beryllium source is irradiated for no greater than 2 hours every 30 days. The activity achieved with this monthly 2.0-hour irradiation at the peak reflector irradiation position is approximately 36 Ci. This periodic irradiation is tracked through the preventive maintenance tracking program that the licensee uses to schedule and record all surveillance requirements (SRs) and preventive maintenance associated with operation of the reactor.

The NRC staff evaluated the use of the antimony-beryllium neutron source and finds it is comparable to that of other licensed non-power reactors of similar design. The NRC staff reviewed the operational history and the design and finds it adequate for source range indication and subcritical measurements. On this basis, the NRC staff concludes that continued use of the antimony-beryllium source in accordance with the renewed LC 2.B.3.a, and applicable procedures, provides reasonable assurance that the source can perform the required functions safely and reliably.

4.2.5 Core Support Structures

SAR Section 4.2.5 states that the MURR core is supported by an outer reactor pressure vessel, an inner reactor pressure vessel, and the reflector tank. The overall structure length is 13 feet (ft) (3.96 meters (m)) and constructed of aluminum alloy 6061-T6. The design of the structure keeps the tops of the pressure vessels from deflecting and interfering with control blade movements. A vessel support ring welded to the pool liner floor supports the entire structure.

In its response to RAI No. 4.10 (Ref. 18), the licensee states that the greatest effect of radiation on the core support structural materials occurs in the inner reactor pressure vessel just below core centerline where fast and thermal fluxes are both at their peaks. The effect is a slight strengthening of the aluminum alloy 6061-T6 inner pressure vessel in this area with a slight reduction in ductility. This effect is primarily a result of the transmutation of some of the aluminum into silicon, which SAR Section 16.1.2 discusses and references a report on the subject. The core pressure vessel is designed to have a greater than 60-year service life, which is also discussed in SAR Section 16.1.2.

In addition, in its response to RAI No. 4.10 (Ref. 18), the licensee states that conductivity and potential of hydrogen (pH) are monitored and maintained in an appropriate range to minimize corrosion and degradation of core support structure materials. Both the PCS and PoolCS have a cleanup system for demineralization and corrosion control. These ion-exchange systems maintain a low conductivity and maintain the pH in a range between 5 and 6. The slightly acidic pH maintained in the PCS and PoolCS maximizes aluminum aqueous corrosion resistance. In addition to routine monitoring, water samples are taken weekly from the PCS and PoolCS and are analyzed for pH, conductivity, and contained radioisotopes. The small negative reactivity of the aluminum alloy 6061-T6 core support structure is slightly reduced with increased neutron fluence. This is a result of the transmutation of the aluminum and the partial burnout of the trace alloy elements and impurities in the aluminum.

The NRC staff reviewed SAR Section 4.2.5 and finds that the core support structure section adequately describes the design for providing structural support for the core and ensuring a stable and reproducible core configuration for all anticipated conditions throughout the reactor life cycle. The NRC staff also finds that the core support structure is conducive to sufficient coolant flow and is compatible with the coolant and radiation environment. On this basis, the NRC staff concludes that the core support structure is acceptable for continued operation of MURR during the license renewal period.

4.3 Reactor Pool

SAR Section 4.3 describes the reactor pool at MURR. The reactor pool and is approximately 3.0 m (10 ft) in diameter and 9.1 m (30 ft) deep. SAR Section 4.11 states that the pool liner is constructed of a 5,086-series aluminum alloy material anchored to steel-reinforced concrete with a volume of approximately 28,000 gallons (gal) (105,991 liters (l)). The concrete walls are coated with an epoxy-tar compound to prevent any chemical interactions between the aluminum liner and concrete. The relative position of the reactor within the pool prevents radiation damage to the pool liner and thus ensures that sufficient shielding exists between the reflector and the pool liner to reduce the neutron fluence seen by the pool liner.

In its response to RAI No. 4.11.a (Ref. 23), the licensee provides a calculation of the total neutron fluence of 5.82×10^{21} neutrons per square centimeter (n/cm^2) at the peak point in the pool liner (at the thermal column), from initial criticality in 1966 to 2010 (44 years). The total neutron fluence calculated by the licensee is approximately an order of magnitude (10 times) less than the exposure that resulted in the cracking experienced at other high-flux reactors using 5,000-series aluminum alloy material (8 to 9×10^{22} n/cm^2). The licensee states that another 44 years of irradiation at this rate would l result in a fluence of 1.16×10^{22} n/cm^2, still below the fluence at which cracking is seen.

SAR Section 13.2.9.1 explains that a loss-of-coolant accident (LOCA) caused by a worst-case malfunction in the PoolCS barrier would not result in a loss of fuel integrity. The reactor pool wall is penetrated by six neutron beam ports. A worst-case loss of pool water caused by a break in the neutron beam port liner could only siphon pool water to approximately 3 ft (0.91 m) above the reactor core because of the unlimited amount of raw makeup water provided by the emergency PoolCS fill system. The PoolCS low-water-level alarm would alert the operator of the break and to initiate corrective action to mitigate the water loss. Additionally, a reactor scram would occur if the PoolCS water level drops below the TS 3.2, Specification g.14, setpoint of 23 ft (7 m).

In its response to RAI No. 4.11.c (Ref. 18), the licensee provides a summary of the minimum detection level of PoolCS leakage. The drain trenches surrounding the pool structure would accept approximately 30 gal (114 l) of PoolCS water leakage before an audible alarm is initiated and displayed in the control room. Additionally, after approximately 70 gal (265 l) of accumulated leakage into the mechanical equipment room sump, an audible alarm would be initiated and displayed in the control room. Routine visual inspection of the mechanical equipment room would also detect visible leaks in the PoolCS. The large capacity of the PoolCS is sufficient to ensure that a leak in the PoolCS would be detected before activation of the PoolCS low-level alarm.

In addition, in its response to RAI No. 4.11.c (Ref. 18), the licensee states that the risk of PoolCS water entering the environment is negligible because of the thickness and width of the caisson upon which the pool and pool liner sit and because neither one directly contacts any soil. And, all probable leak paths have detection capability to alert the operators, and would provide for retention of the PoolCS water, as described above.

In its response to RAI No. 4.11.b (Ref. 18), the licensee states that the pool liner was inspected in 2001 by an independent engineering firm, which affirmed the excellent condition of the pool liner and provided additional confidence in the performance of the liner for the license renewal period. Additionally, the licensee examines the pool liner during the scheduled beryllium reflector replacements, which occur approximately every 8 years. The beryllium reflector replacement includes lowering of the PoolCS water level, which enhances visual access to conduct a more effective examination of the pool liner.

The NRC staff reviewed the licensee's RAI responses and finds that the licensee provides reasonable justification for continued operation of the reactor PoolCS during the license renewal period. The NRC staff has reviews the SAR Section 4.3 and finds that it adequately describes the reactor PoolCS design features and that acceptable detection measures and preventive maintenance procedures provide reasonable assurance that the PoolCS is capable of withstanding the corrosion and radiation environment for the period of the license renewal. The NRC staff also finds that the reactor system and experiment facility penetrations and piping are designed to prevent siphoning to minimize the potential for PoolCS boundary integrity failure, which could lead to a loss of coolant or other type of malfunction. On this basis, The NRC staff concludes that the reactor PoolCS is acceptable for continued operation of MURR during the renewal period.

4.4 <u>Biological Shield</u>

SAR Section 4.4 states that the reactor pool water and steel-reinforced concrete pool walls provide biological shielding for MURR. The concrete is magnetite. The pool walls are thick steel-reinforced concrete. The external surface is protected by ¼-in-thick (0.635 cm) steel plates. The biological shield shape and construction are similar to other licensed non-power reactors with similar designs.

In its response to RAI No. 4.12 (Ref. 21), the licensee states that the heat generated in the magnetite concrete of the biological shield is predominantly as a result of the gamma rays emitted from the core as fission (prompt and delayed) and activation gamma rays of reactor materials, including the reactor core, pressure vessels, reflector components, and water. The licensee calculated this gamma-ray heating using detailed models of MURR and using the Monte Carlo N-Particle Version 5 (MCNP5) transport code, with the gamma ray and neutron fluxes tallied in regions of the biological shield where the peak fluxes would occur (i.e., regions whose radial distances are closest to the core). Three feet (0.91 m) is the closest the core is to the magnetite concrete portion of the biological shield, and water is in the space between the reactor reflector and the biological shield.

In its response to RAI No. 4.12 (Ref. 21), the licensee illustrates that most of the gamma-ray energy is deposited within the first few inches of the shield, with insignificant heating occurring in the rest of the approximately 6.7-ft-thick biological shield. Because the pool water is essentially in contact with the inner surface of the concrete (from a heat transfer perspective), it is the major heat sink for the heating within the biological shield. The licensee developed a heat-transfer model with the pool water/concrete surface having one boundary condition and with the concrete surface/air on the other side being adiabatic.

In its response to RAI No. 4.12 (Ref. 21), the licensee provides a study entitled, "Thermal Degradation of Concrete in the Temperature Range from Ambient to 315 degrees °C," conducted by Kassir and Bandyopadhyay (Ref. 71). The study shows that the long-term heating effect on concrete at ambient temperatures and up to 250 degrees Fahrenheit (°F) (121 degrees Celsius (°C)) serves to increase its mechanical strength (Ref. 67). However, for long-term exposure to temperatures at or greater than 300 °F, the mechanical properties begin to fail. In the previous 1962 study provided by the licensee, and which is referenced in the same RAI response, the limiting temperature is conservatively set at 150 °F (65.6 °C) or a ΔT = 50 °F (10 °C), which is half of the limiting value reported by Kassir and Bandyopadhyay (Ref. 71). The worst-case analysis using the heat-transfer conductance coefficient for baryte concrete, which is half the value for magnetite concrete, shows a maximum ΔT over the 6.5-ft (1.98-m) thickness of the MURR biological shield (assuming no heat transfer to the air) of 17.7 °F (-8 °C). As such, the potential for degradation from long-term gamma-ray heating is minimal.

SAR Section 11.1.5.1, states, except for the beam ports, radiation levels around the biological shield during full-power operation of the reactor are below 2.5 mrem per hour with shielding, provided in part by the PoolCS water. If the PoolCS water level were to decrease, personnel working in the experiment facilities would be alerted by an alarm, which would provide an audible sound when the water level dropped to 23 ft (7 m) above the bottom of the pool. The reactor would also be required to shut down, thus helping to reduce the radiation source.

The NRC staff reviewed the MURR biological shield design and finds that the licensee's calculated thermal temperature gradient is sufficient to prevent concrete degradation such that the biological shield will continue to operate consistent within the design specifications set forth in the original construction. On the basis of the information provided above, the NRC staff concludes that the biological shield is able to maintain its safety function during the license renewal period and that continued operation within the requirements of the TSs and the MURR radiation protection program will provide reasonable assurance that the shielding design will continue to limit radiation exposures to the workers.

4.5 <u>Nuclear Design</u>

SAR Section 4.5, discusses the design bases for the MURR safety analysis and portions of the TSs. The SAR Section 4.5.1 states that this pressurized light-water plate-type reactor nominally operates in Mode I, a steady-state thermal power level of 10 MWt with forced convection. The reactor may also be operated in Mode II, a steady-state thermal power level of 5 MWt or less with forced convection, or in Mode III, a steady-state thermal power level of 50 kWt or less with natural circulation.

The application for LA No. 36 (Ref. 65) includes the nuclear design analysis provided by the licensee to support this license renewal. Argonne National Laboratory (ANL) performed the neutron analysis of MURR, using several programs identified in the licensee response to RAI No. 3.a (Ref. 33). Figure 4-2, illustrates the relationship between the various codes for the processing of cross-sections, core solutions, depletion of materials, and determination of critical blade position.

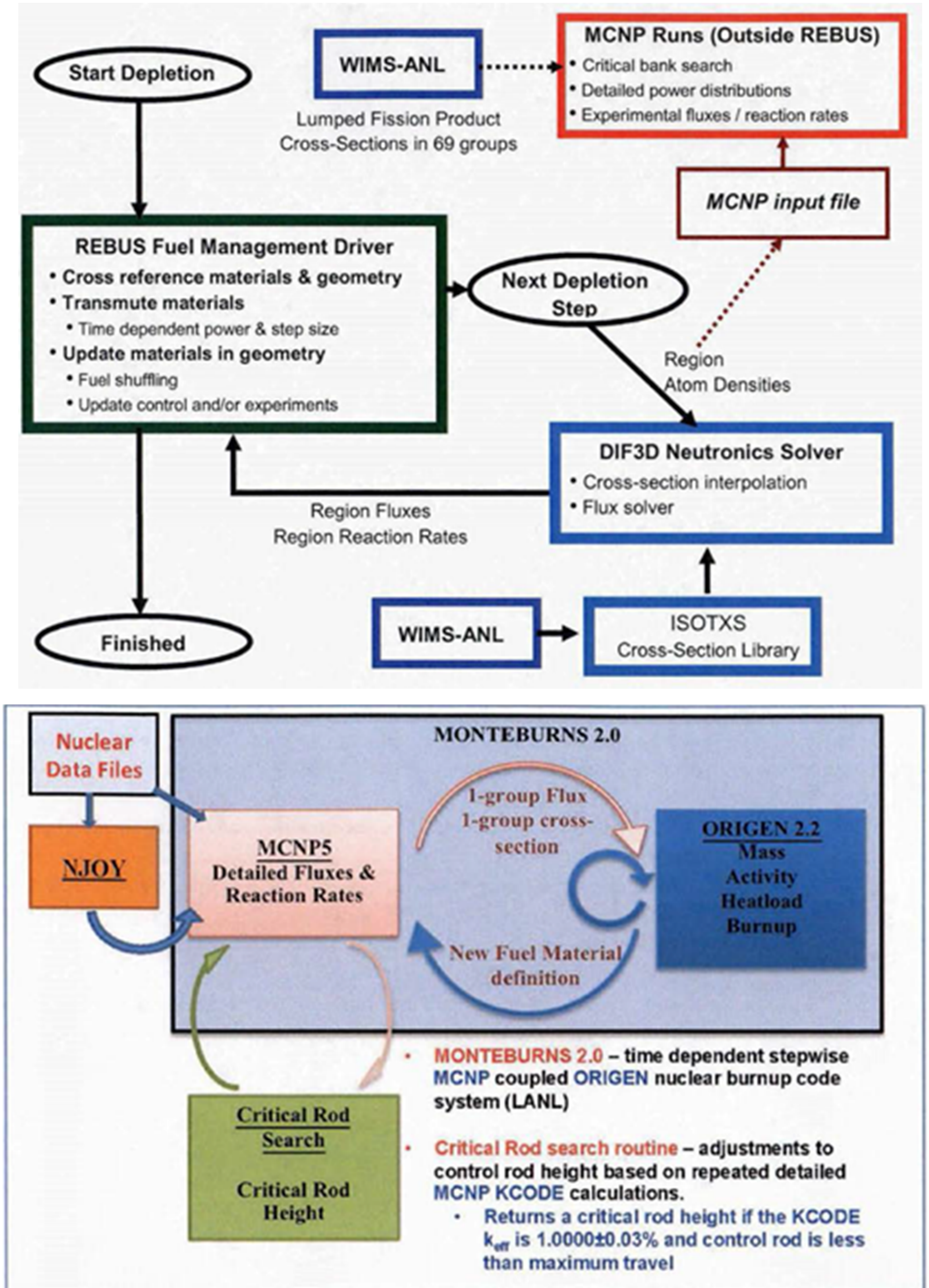

Figure 4-2 Neutronic Analysis of Record Methods

In its response to RAI No. 3.a (Ref. 33), the licensee describes what each code accomplishes, as follows:

- WIMS-ANL. WIMS-ANL is a one-dimensional lattice physics code used to generate burnup-dependent, multigroup cross-sections. The code uses either 69- or 172-group libraries of cross-section data for 123 isotopes generated from

ENDF-6. ANL developed a customized 10-group structure based on the neutron spectrum that exists in the MURR core. This multigroup data can be used in Monte Carlo N-Particle (MCNP) transport code and REBUS-MCNP analyses of depleted cores.

- REBUS-DIF3D. DIF3D is a multidimensional, multigroup neutron diffusion code that can model systems in a number of geometries. REBUS is a depletion code that uses neutron fluxes from a neutronics solver and cross-section data to solve isotopic transmutation calculations. A detailed Θ-R-Z diffusion MURR model was developed for DIF3D. The depleted core characteristics (plate-by-plate and axially segmented atom densities) can be saved and passed on to MCNP for more detailed neutronics analyses.

- MCNP. MCNP is a continuous-energy Monte Carlo neutron transport code. MCNP is capable of modeling the heterogeneous details of the MURR fuel elements, core structures, and experimental facilities while capturing the rapidly changing spectra across these various regions. Using the 69-group lumped fission product library generated by WIMS-ANL, the code can be used to model cores of depleted and fresh elements.

To speed up routine neutronics calculations for which such detailed axial, radial, and azimuthal fuel composition is not necessary, the licensee uses the MONTEBURNS program. The MONTEBURNS code is a coupled MCNP-ORIGEN 2 code system developed by Los Alamos National Laboratory. It uses the capabilities of ORIGEN 2 (Ref. 110) for isotope generation and depletion calculations and that of MCNP5 for continuous energy, flux, and reaction rate, as well as criticality calculations.

4.5.1 Normal Operating Conditions

Table 4-2 provides the conditions that pertain to full-power operation of MURR.

Table 4-2 MURR Normal Operating Parameters and Conditions

Parameter or Condition	Value and Units
License Reactor Power—Maximum	10 MW
Number of Fuel Elements in the Core	8 with 24 plates in each
Number of Control Blades	4 shim and 1 regulating
Maximum Fuel Temperature at full power	311.3 °F (155.2 °C)
Departure from Nucleate Boiling Ratio	>2
Moderator Temperature Coefficient	-6.0×10^{-5} Δk/k/°F (-$$8.1 \times 10^{-3}$)*
Moderator Void Coefficient	-2.0×10^{-3} Δk/k/% void (-$0.27)*
Worth of the 4 shim blades	-0.16550 Δk/k (-$22)*
Worth of regulating blade	-0.00230 Δk/k (-$0.31)*
prompt neutron lifetime (Λ)	57 μs
delayed neutron fraction (K_{eff} β)	0.00738

* Values are converted to $ by dividing by the delayed neutron fraction K_{eff} β = 0.00738.

In Section 3.4 of its application for LA No. 36 (Ref. 65), the licensee states that MURR typically operates in Mode I at 10 MWt with weekly shutdowns for either fuel shuffling, refueling, or both. Annually, 22 fuel assemblies (see Figure 4-4 below for an illustration of a MURR fuel assembly) are replaced and the typical annual inventory is 32 fuel assemblies (the core contains 8 fuel assemblies for normal operation). The licensee indicated that this operating history is planned to continue into the future. Normal operations begin after the completion of the weekly refueling. The reactor is restarted and operates at full power until the end of the week. Consequently, MURR conducts refueling and power ascension on a much more frequent basis than is typical for NRC licensed RTRs. Similarly, offsite shipments of fuel are more frequent to accommodate spent fuel in the limited storage facilities.

In the applicant for LA No. 36 (Ref. 65) request, the licensee prepared a comprehensive report, TRD-0125, as Attachment 10 (Ref. 65), that characterized the neutronic and T-H behavior of MURR. Although the long-term objective was to demonstrate the capability to change MURR over to using a LEU fuel design, the NRC staff finds that the report is useful for documenting the operating characteristics of MURR using the current HEU fuel. Several features of the TRD-0125 report are discussed below.

Figure 4-3 illustrates the reactor core major components. The flux trap is a water-filled pipe that is capable of hosting experiments. That pipe is surrounded by 8 fuel elements. Each fuel element contains 24 curved fuel plates. These fuel elements are within the outer pressure vessel which could be considered as a pipe within a pipe. PCS water flows through the fueled region. PoolCS water flows through the flux trap. Outside of the outer pressure vessel is a region containing the four shim blades and the regulating blade. Further out radially are the beryllium reflectors (not shown) and the graphite reflectors (not shown). This outer region is also cooled by PoolCS water. The reactor core neutronic analysis uses a Θ-R-Z geometry, which can precisely representation the curvature of the fuel plates.

As required by TS 3.2.b, the four scrammable shim control blades are required to be operated within a 1-in (2.54 cm) band of each other (i.e., banked position), whereas the regulating blade, which has a significantly lower worth, is used for fine reactivity control and moves independently of the bank. There is sufficient excess reactivity and control rod worth to operate MURR at up to 10 MWt, including the reactivity associated with fission product poisons and temperature changes. MURR uses normal, or light, water in all water-cooled regions. Water is used as both a coolant and a moderator in MURR.

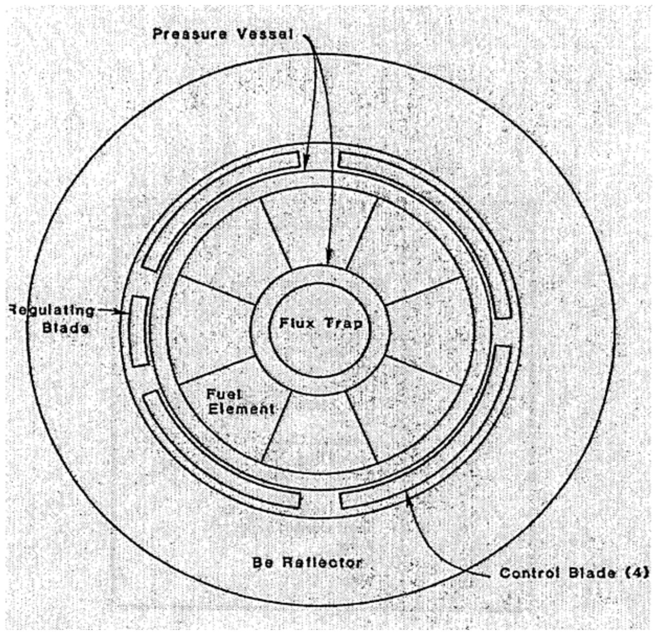

Figure 4-3 Core Components from SAR Figure 4.10

Estimated Critical Positions

Table 4-3 presents several estimate critical positions (ECP), as extracted from the licensee's response to RAI No. 3.b (Ref. 32).

Table 4-3 Comparison of Calculated versus Measured ECPs

Core Configuration	Measured Critical Control Blade Height (in)	MCNP Predicted Control Blade Height (in)	Effective k_{eff} at Actual Height
1/28/2013	16.79	16.67	0.99993
2/4/2013	16.52	16.27	0.99975
4/29/2013	15.98	15.78	1.00017
6/10/2013	15.44	15.42	0.99995
8/5/2013	16.74	16.74	0.99985
8/12/2013	15.71	15.61	0.99985
8/19/2013	15.84	15.84	1.00016
8/26/2013	15.64	15.69	1.00029

The NRC staff reviewed the licensee's documentation of ECP calculations for MURR. The NRC staff finds that the degree of agreement between MCNP predictions of the ECP and the actual criticality measurements acceptable. Based on the information provided above, the NRC staff concludes that the licensee's ECP calculation methodology is acceptable.

Control Blade Worth

As provided in its response to RAI No. 4.13.f (Ref. 18), the licensee used the BOLD-VENTURE code to calculate the control blades worth in the past. Using this method, the calculated total worth of control blades had been within approximately 3 percent of measured values. In its response to RAI No. 3.c (Ref. 33), the licensee provided a comparison of the measured and predicted control blade worth, in Table 4-4 below, which indicates agreement within 2.5 percent.

Table 4-4 Measured Rod Worth

Control Blade	Total Measured Reactivity Worth ($\Delta k/k$)	Total Calculated Reactivity Worth ($\Delta k/k$)	% difference
D	-0.03550	-0.03640	2.5%

In response to RAI No. 4.14.b (Ref. 18), the licensee clarifies that the cold (ambient temperature), clean (no appreciable xenon 135) core keff values listed in SAR Table 4.12 were taken from an internal document entitled, "Low Power Testing Program for the Missouri University Research Reactor 6.2 Kilogram Core," dated October 20, 1971. It documents how the total control blades worth are determined, including the cold, clean critical rod heights.

The NRC staff reviewed the AOR methodology and the calculated and measured rod worth data for MURR. The NRC staff finds the methodology appropriate and consistent with the methodology used at other reactor facilities. Because the calculated values are in agreement with the measured values, the NRC staff concludes that the values for control blade worth calculations using the AOR methodology are acceptable.

Peaking Factors (PFs)

In its response to RAI No. 7.a (Ref. 33), the licensee states that it uses the same PFs that was provided with its application for LA No. 36 (Ref. 65). The PFs are restated as follows:

> *From Table F.4, "SUMMARY OF MURR HOT CHANNEL FACTORS," of Appendix F of Addendum 4 to the MURR Hazards Summary Report (as revised by Amendment No. 36):*
>
> *On Heat Flux from Plate-1*
>
> *Power-related Factors*

Nuclear Peaking Factors	
Fuel Plate (Hot Plate Average)	*2.215*
Azimuthal Within Plate	*1.070*
Axial Peak	*1.3805*
Additional Allowable Factor	*1.062*
Overall	*3.4747*
Engineering Hot Channels Factors on Flux	
Fuel Content Variation	*1.030*
Fuel Thickness I/Width Variation	*1.150*
Overall Product:	4.116

> *This peak heat flux point is at axial mesh interval 14 (13 to 14 inches down the fuel plate meat) where the enthalpy rise at that interval is 52.3%. The SL is based on mesh interval 18, which has an overall peaking factor of 3.863 and an enthalpy rise of 74.8%; thus producing the most limiting combination of heat flux and enthalpy rise.*

The NRC staff reviewed the information provided in the licensee's response to RAI No. 7.a (Ref. 33), which were also submitted with the request for LA No. 36 (Ref. 65). The NRC staff finds PFs were appropriately determined, and are conservative. Based on the information provided above, the NRC staff concludes that the PFs used by the licensee are acceptable.

The Effect of Control Blade Height on Peaking Factors

SAR Section 4.5.3 indicates that differences in control blade heights can affect nuclear PFs and result in flux tilting. Localized hot spots can be created that could result in overheating of the fuel cladding. In its response to RAI No. 4.14.d (Ref. 25), the licensee provides an analysis of the effect of blade position on PFs for eight different core configurations, using a combination of new fuel and used fuel (58-week fuel). Some of these cases have a 4-in difference in the control blade heights in order to evaluate the sensitivity of the different core configurations to various PFs. The analyses include examples, which had a 4-in height difference between the highest and lowest control blade locations in the core. The licensee's results are shown in Table 4-5 below in which the calculated PFs are bounded by the assumptions used in the analysis reviewed and approved by the NRC staff in LA No. 36 (Ref. 66).

Table 4-5 Peaking Factors of Eight MURR Core Configurations

Core Configuration	Hot Spot Heat Flux		Hot Channel Enthalpy Rise	
	Hot Spot PF	With Engr. PF	Enthalpy Rise PF	With Engr. PF
1	3.08	3.65	1.97	2.09
2	3.17	3.75	2.03	2.15
3	3.06	3.62	2.01	2.14
4	3.06	3.62	2.05	2.18
5	3.08	3.64	1.95	2.07
6	3.12	3.70	1.99	2.11
7	3.01	3.57	1.99	2.11
8	3.06	3.62	2.03	2.15
Safety Limits	3.475	4.116	2.301	2.4416

The NRC staff reviewed the control blade axial height effect on PFs and finds that the 1-in height limit established by TS 3.2, Specification b, ensures that excessive flux tilting is not created (see SER Section 4.2.2 for additional information on TS 3.2, Specification b.) Based on the information provided above, the NRC staff concludes that the 1-in height limit on control blade positions established by TS 3.2, Specification b, is acceptable.

Fuel Burnup

SAR Section 4.5 and the licensee's responses to RAI No. 4.3 (Ref. 23), RAI No. 4.13.b (Ref. 18), and RAI No. 4.15 (Ref. 26) provide detailed information on the effect of uranium fission (i.e., burnup) on the MURR fuel. The licensee states in SAR Section 4.2.1.1 that the burnup limit used is based on data that indicate that fuel-plate swelling is less than 10 percent and has no detrimental effect on fuel-plate performance. Table 4-6 below provides information that is extracted from Table 3 in the licensee's response to RAI No. 4.15 (Ref. 26) and provides a comparison of key attributes between fresh fuel and high-burnup (i.e., discharged) fuel. The limiting axial level indicates the location of the highest fuel temperature.

Table 4-6 Comparison of Fresh Fuel to High-Burnup Fuel

Parameter	Fresh Fuel	High-Burnup Fuel
Limiting Axial Level*	16	17
Bulk Coolant Temperature (°F)	164.7	163.7
Hot Channel Local Velocity (ft/s)	22.19	20.02
Temperature at Oxide-Cladding Interface (°F)	287.6	285.6
Temperature at Cladding-Fuel Interface (°F)	297.9	292.1
Temperature at Fuel Center (°F)	311.3	303.6
Fuel Cladding (Blister) Temperature Limit (°F)	900	

* Fuel height in inches above the bottom of fuel meat (0 in) to top of fuel (24 in)

In SAR Section 4.2, the licensee provides the basis for the burnup limit of 2.3×10^{21} fission/cubic centimeter (cc) in TS 5.3, Specification c. The NRC staff finds that experience with the ATR fuel has shown that this burnup level results in less than 10-percent swelling of the fuel plates and thus no detrimental effects on the fuel performance. Furthermore, peak values for the burnup of MURR fuel were provided in response to RAI No. 1, Enclosure 3 (Ref. 15), and are 9.2×10^{20} fissions/cc. The NRC staff finds this value is well below the limit cited. Based on the information provided above, the NRC staff concludes that the burnup limit is acceptable.

Conclusions

The licensee has described its typical core configuration, which envelops all planned configurations for this fuel design. Based on its review of the information above, the NRC staff concludes:

- The assumptions and methods are justified and their validity is demonstrated acceptably. These comparisons of measured and calculated ECP and control blade worth indicates an acceptable level of agreement (within a few percent); and, therefore, the licensee used acceptable predictive models for MURR analysis.

- The analyses include changes resulting from burnup, plutonium buildup, and the accumulation of fission products.

- The criticality analysis establishes the ability of the licensee to predict core excess reactivity and control rods worth.

- The analysis addresses the steady power operation and kinetic behavior of the reactor and shows that the dynamic response of the control blades and instrumentation is designed to prevent uncontrolled reactor transients.

- The analysis includes consideration of those parameters that ensure that a limiting core analysis is provided. Because this core configuration has the highest power density, the licensee uses it as described in SER Section 4.6 to determine the limiting T-H characteristics for the reactor.

- The analysis and information in this section describes a reactor core system that was designed, constructed, and operated without unacceptable risk to public health and safety.

- The licensee provided justification for the Limiting Conditions for Operation and SRs required by the TSs and consistent with 10 CFR 50.36.

4.5.2 Reactor Core Physics Parameters

Reactor Kinetics Parameters

Table 4-2 in SAR Section 4.5.1 provides a number of neutronics parameters used in the neutronics analysis. Among these are the kinetics parameters: the prompt neutron lifetime (Λ) and the average delayed neutron fraction (β_{eff}). The NRC staff reviewed these parameters and finds that they are sensitive to fuel content, burnup, and core neutron leakage, which are core parameters that have been unchanged for MURR for many years. The NRC staff also finds that the parameters are relatively insensitive to the analytical methods employed as long as the methods have acceptably demonstrated the ability to model MURR behavior. The NRC staff finds that since: (1) the core parameters have not changed and the core is essentially in an equilibrium condition; (2) the analytical methods used are demonstrated by the licensee to suitably represent MURR core behavior; and (3), these same methods are the industry reference methodology for performing such analysis, that the kinetics parameters provided by the licensee are acceptable.

Based on its review of the information provided above, the NRC staff concludes that the predicted values of the parameters, Λ = 57 microseconds (μs) and β_{eff} = 0.00738, are acceptable for use in the MURR neutronic analyses.

Temperature and Void Coefficients of Reactivity

SAR Section 4.5.2 states that the neutronics analysis of MURR uses two coefficients of reactivity: (1) the moderator temperature coefficient (MTC); and (2) the moderator void coefficient (MVC). Table 4-12 in SAR Section 4.5.2 provides the calculated values for these coefficients. Based on its review of the neutronics analyses, the NRC staff finds that these parameters are sensitive to fuel content, burnup, and core leakage, and are parameters that have been unchanged for MURR for many years. The NRC staff also finds that these parameters are insensitive to the analytical methods employed as long as the methods have acceptably demonstrated the ability to model MURR behavior. Using the analytic methods provided in the analysis of record (AOR), the licensee revised the MTC in its response to RAI No. 3.d (33). Table 4-7 below provides these MTC and MVC, which act to mitigate the consequences of any sudden reactor power excursions.

Table 4-7 MURR Core Reactivity Coefficients

Parameter	SAR Table 4-12	Measured Value (Ref. 38)	RAI No. 3.d Response	TS Value (minimum)
MTC (Δk/k/°F)	-7.0×10^{-5}	-7.0×10^{-5}	-11.8×10^{-5}	-6.0×10^{-5}
MVC (Δk/k/% void)	-2.0×10^{-3}	-2.87 to -3.19×10^{-3}	N/A	-2.0×10^{-3}

The NRC staff finds that reactivity characteristics of the MURR core would not change without major modification of the reactor fuel (as the reactivity characteristics are inherent in the design of the fuel). The NRC staff also finds that the licensee's SAR analysis and its responses to RAIs validate the reactivity coefficient values and that these values are suitable to support the analysis of transients and accidents. Based on its review of the information provided above, the NRC staff concludes that the predicted values for the MTC and MVC, as provided in the Table 4-7, are acceptable and that the corresponding values used in the TSs are acceptable for use in the analysis of the MURR accident analysis.

TS 5.3 Reactor Core and Fuel

TS 5.3, Specifications a through d, state:

Specification:

The following design features apply to the reactor core and fuel:

a. The average reactor core temperature coefficient of reactivity shall be more negative than $-6.0x10^{-5}$ Δk/k/°F.

b. The average reactor core void coefficient of reactivity shall be more negative than $-2.0x10^{-3}$ Δk/k/% void.

c. The peak burnup for UAl_X dispersion fuel shall not exceed a calculated $2.3x10^{21}$ fissions per cubic centimeter.

d. The regulating blade total reactivity worth shall be a maximum of $6.0x10^{-3}$ Δk/k.

(…)

TS 5.3, Specification a, requires that the MTC be more negative than -6.0x10-5 Δk/k/°F. This specification helps to ensure that the response of the reactor to an increase in moderator temperature, such as power changes, is accompanied by a predictable negative reactivity response. The NRC staff finds that TS 5.3, Specification a, is consistent with the assumptions used in the analysis of the insertion of excess reactivity evaluated in in SER Section 13.2, which demonstrates that the postulated reactivity insertion events at MURR are sufficiently limited to protect the integrity of the fuel clad barrier. The response to RAI No. 3.d (Ref. 32) provides calculated values of the primary coolant isothermal temperature coefficient for several conditions. The most negative value is for an all-fresh core at the beginning of the cycle ($-13.2x10^{-5}$ absolute reactivity (Δk/k)), and the least negative value is for a burned core with equilibrium xenon ($-11.8x10^{-5}$ Δk/k). On this basis, the NRC staff concludes that TS 5.3, Specification a, is acceptable.

TS 5.3, Specification b, requires the MVC to be more negative than 2.0x10-3 Δk/k/% void. This specification helps to ensure that the response to the reduction in density, or voiding, in the reactor core is accompanied by a predictable negative reactivity response. The NRC staff finds that MURR's void coefficient of reactivity is consistent with the assumption used in the analysis provided in SAR Section 13.2.2. The NRC staff also finds this analysis demonstrates that the postulated reactivity insertion events at MURR are sufficiently limit the fuel temperature to protect the integrity of the fuel-clad barrier. Based on the information provided above, the NRC staff concludes that TS 5.3, Specification b, is acceptable.

TS 5.3, Specification c, requires the fuel peak accumulated burnup, as stated in terms of fission density, to be limited to a calculated $2.3x10^{21}$ fissions per cubic centimeter, which is discussed in SER 4.5.1, Fuel Burnup. This specification helps to ensure that the fuel clad barrier integrity is protected and remains intact. The NRC staff finds that TS 5.3, Specification c, is consistent with guidance in NUREG-1313, "Safety Evaluation Report Related to the Evaluation of Low-Enriched Uranium Silicide-Aluminum Dispersion Fuel for Use in Non-Power Reactors," issued July 1988 (Ref. 68). Based on the information provided above, the NRC staff concludes that TS 5.3, Specification c, is acceptable.

TS 5.3, Specification d, requires that the worth of the regulating blade be limited to 6.0×10^{-3} Δk/k. The NRC staff finds that TS 5.3, Specification d helps to ensure that the response to an uncontrolled withdrawal of the regulating blade is bounded by the analysis evaluated in SER Sections 13.2.1 and 13.2.2 and found acceptable. On this basis, the NRC staff concludes that TS 5.3, Specification d, is acceptable.

The NRC staff finds TS 5.3, Specifications a, b, c, and d provide acceptable neutronic design limitations for MURR. The NRC staff concludes that the licensee completed the analyses of neutron lifetime, effective delayed neutron fraction, and coefficients of reactivity using methods that are appropriate, and the numerical values for the reactor core physics parameters depend on features of the reactor design that are included in applicable models, and the information provided is acceptable for use in the analyses of the MURR operation.

4.5.3 Operating Limits

Safety Limits and Limiting Safety System Settings

The regulation, 10 CFR 50.36(c)(1), requires TSs that include SLs and LSSSs. This regulation defines SLs as "limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity."

TS 2.1 Safety Limits

TS 2.1, Specification a, states:

Specification:

a. The temperature of a reactor fuel element shall not exceed 986 °F (530 °C) for any operating condition.

TS 2.1, Specification a, requires that the maximum fuel temperature of a reactor fuel element not exceed 986 °F (530 °C) for any operating condition. The NRC staff reviewed TS 2.1, Specification a, and finds that it is consistent with the guidance in NUREG-1313 (Ref. 68) and in NUREG-1537, which indicates that the appropriate SL for plate-type dispersion fuels is 530 °C (986 °F). Based on the information above, the NRC staff concludes that TS 2.1, Specification a, is acceptable.

The regulation in 10 CFR 50.36(c)(1)(ii)(A) states LSSSs are "settings for automatic protective devices related to those variables having significant safety functions. Where a LSSS is specified for a variable on which a SL has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a SL is exceeded." The proposed LSSSs for the MURR TSs are provided below.

TS 2.2 Limiting Safety System Settings

TS 2.2 states:

Specification:

a. Mode I Operation

Reactor Power Level (10 MW)	125% of full power (Maximum)
Primary Coolant Flow Rate	1,625 gpm each loop(1) (Minimum)
Reactor Inlet Water Temperature	155 °F (Maximum)
Pressurizer Pressure	75 Psia (Minimum)

(1) Both primary coolant system loops are required to be in operation for Mode I.

b. Mode II Operation

Reactor Power Level (5 MW)	125% of full power (Maximum)
Primary Coolant Flow Rate	1,625 gpm either loop (1) (Minimum)
Reactor Inlet Water Temperature	155 °F (Maximum)
Pressurizer Pressure	75 Psia (Minimum)

(1) Either primary coolant system loop is required to be in operation for Mode II.

c. Mode III Operation

Reactor Power Level (50 kW)	125% of full power (Maximum)

TS 2.2, Specification a, establishes the setpoints for operation of MURR at 10 MWt. It requires that the setpoints for the: 1) Reactor Power (Channel 4, 5, and 6 High-Power Scram); 2) Primary Coolant Flow Rate; 3) Reactor Inlet Water Temperature; and, 4) the Pressurizer Pressure (as provided in SAR Table 7-8) are set to the values and conditions indicated in the table in TS 2.2 a., for operation in Mode I. The NRC staff finds that the analysis submitted with the request for LA No. 36 (Ref. 65), as described in SER Chapter 13, indicates that under normal operating conditions or accident conditions, the use of these setpoints will result in fuel temperatures that are acceptable (less than 986 °F (530 °C)). In its response to RAI No. 4.15 (Ref. 26), the licensee indicates that the maximum fuel temperature in Mode 1 operation was 311.3 °F (155 °C). In the insertion of excess reactivity analysis discussed in SER Section 13.2.1, the maximum fuel temperature was 441 °F (227 °C). The NRC staff finds that these LSSSs meets 10 CFR 50.36 (c)(1). The NRC staff also finds that operating in Mode 1 at these setpoints will result in the maximum fuel temperatures that are substantially less that the TS SL. On these basis, the NRC staff concludes that TS 2.2, Specification a, is acceptable.

TS 2.2, Specification b, establishes the setpoints for operation of MURR at 5 MWt. It requires that the setpoints for the: 1) Reactor Power (Channel 4, 5, and 6 High Power Scram); 2) Primary Coolant Flow Rate; 3) Reactor Inlet Water Temperature; and, 4) the Pressurizer Pressure (as provided in SAR Table 7 8) are set to the values and conditions indicated in the table in TS 2.2 b., for operation in Mode II. The NRC staff's review finds that the analysis submitted with the request for LA No. 36 (Ref. 65), as described in SER Chapter 13, indicates that under normal operating conditions or accident conditions, the use of these setpoints will result in fuel temperatures that are less than the temperatures evaluated from operation in Mode I and, therefore, acceptable. The NRC staff also finds that establishing these LSSSs meets 10 CFR 50.36(c)(1). On these bases, the NRC staff concludes that TS 2.2, Specification b, is acceptable.

TS 2.2, Specification c, establishes the acceptable setpoints for operation of MURR at 50 kWt. It requires that the setpoint for the Reactor Power (Channel 4, 5, and 6 High Power Scram) are set to the value indicated in the table for operation in Mode III. The NRC staff's review finds that the analysis submitted with the request for LA No. 36 (Ref. 65), as described in SER Section 13, indicates that under normal operating conditions or accident conditions, the use of this setpoint will result in fuel temperatures that are less than the fuel temperatures evaluated from Operation in Mode I and, therefore, acceptable. Based on the information above, the NRC staff concludes that TS 2.2, Specification c, is acceptable.

TS 2.2, provides settings for the automatic protective devices related to those variables that have significant safety functions. The NRC staff finds that TS 2.2, Specifications a, b, and c, provide sufficient margin to ensure that automatic protective actions occur before reaching the SL. The safety margin between the LSSSs and the SL allows for measurement and analytical uncertainties, as discussed in the licensee's response to RAI No. 4.18.c (Ref. 20). The NRC staff finds that the LSSS values provide reasonable assurance that SLs will not be exceeded as long as operation is in accordance with the TS. The NRC staff also finds that TS 2.2, Specifications a, b, and c, are consistent with the analysis submitted with the request for LA No. 36 (Ref. 65), and the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 2.2, Specifications a, b, and c, are acceptable.

Based on the information above, the NRC staff concludes that continued operation MURR, in accordance with the limits in TS 2.1 and TS 2.2, will help ensure that the fuel continues to maintain its cladding and fission product barrier integrity, and without adversely affecting public health and safety.

Excess Reactivity and Shutdown Margin

Excess reactivity is a core parameter that is important to maintaining SDM. It is used in the evaluation of experiment safety. Monitoring excess reactivity by the licensee helps in detecting core reactivity anomalies such as misaligned control blades, disconnected control blades, fuel misloading, burnable poison misloading, and fuel failures. SDM is important because it demonstrates the ability to make the reactor subcritical even if a scrammable control blade fails to insert. TSs limits help ensure that these two important reactivity limits are properly maintained. The NRC staff review of excess reactivity and SDM follow below.

TS 3.1 Reactor Core Parameters

TS 3.1 states the following:

Specification:

a. When the reactor is operated, the reactor core excess reactivity above reference core condition shall not exceed 0.098 Δk/k.

b. When the reactor is operated, the reactor shall have a shutdown margin of at least 0.02 Δk/k with:

(1) The most reactive shim blade and the regulating blade in their fully withdrawn positions;
(2) Irradiation facilities and experiments in place and the total worth of all experiments in their most reactive state; and
(3) The reactor in the reference core condition.

(…)

TS 3.1, Specification a, limits the excess reactivity of the core, which is necessary to help ensure that the SDM requirement can be maintained under any operating conditions. The licensee provided its analysis for the excess reactivity and SDM in its responses to RAI No. 4.14.c (Ref. 21) and RAI No. 4 (Ref. 32). The NRC staff finds TS 3.1, Specification a helps ensure that the reactor can be shut down with the negative reactivity of the control blades under any circumstance consistent with TS 3.1. Based on the information provided above, the NRC staff finds concludes that TS 3.1, Specifications a, is acceptable.

TS 3.1, Specification b, requires that the reactor can be subcritical by 0.02 Δk/k (-$2.71) with the most reactive shim blade and regulating blade in their fully withdrawn positions, the irradiation facilities and experiments in place and the total worth of all experiments in their most reactive state, and the reactor core in the reference core condition. The licensee provided their analysis for the excess reactivity and SDM in their responses to RAI No. 4.14.c (Ref. 21) and RAI No. 4 (Ref. 32). The NRC staff reviewed the licensee's reactivity values, summarized the reactivity values in Table 4-8 below, and finds that the SDM value can be achieved by the scrammable blades, with the maximum reactivity shim blade removed from the core (withdrawn or fully out), and the regulating blade fully withdrawn. The NRC staff also finds that TS 3.1, Specification b, is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 3.1, Specification b, is acceptable.

The NRC staff finds that TS 3.1, Specifications a and b, require acceptable limitations on excess reactivity and SDM. The reactivity values presented in Table 4-8 demonstrates that, by using these values and typical values for shim blade worth, MURR can achieve a safe shutdown condition. Based on the information provided above, the NRC staff concludes that TS 3.1, Specifications a and b, are acceptable.

In its response to RAI No. 1.a, (Ref. 34), the licensee provides the information, reproduced in Table 4-8 below, in order to demonstrates that, when the core is at maximum reactivity, and both the regulating blade and highest worth shim blade are withdrawn, the shutdown reactivity of the core is still far more negative than the SDM requirement.

The maximum core excess reactivity, TS 3.1, Specification a, is limited not to exceed 0.098 Δk/k ($13.27). In its response to RAI No. 4.14.c (Ref. 21), the licensee states that the excess reactivity is verified after any changes to the core are made. The verification is performed during the reactor startup when the cold, clean critical rod height is measured (reference core condition). This critical control rod position, along with the known integral rod worth data, is used to estimate the core excess reactivity (i.e., the difference between the total rod worth and the worth of the rods at the measured critical rod height). The licensee also states that after this verification, the only change to this overall excess reactivity would be from the addition or removal of movable experiments. Fuel burnup is not considered.

Table 4-8 Excess Reactivity-SDM Evaluation

Condition	TS Limits (Δk/k)
Maximum Core Excess Reactivity (TS 3.1.a)	+0.09800
Worth of the Regulating blade (SAR Table 4-12)	+0.00230
Net Core Reactivity Worth	+0.10030
Worth of Regulating blades at 80% position minus one blade (RAI No. 3.c, Ref. 34)	-0.12600
Net Core Reactivity minus Worth of regulating blades at 80% position minus one blade (calculated SDM)	-0.02570
Required SDM (TS 3.1.b)	-0.02000

The NRC staff reviewed the information provided in Table 4-8, and finds that it demonstrates the acceptability of the licensee's SDM, as provided in TS 3.1, Specification b, when the additional considerations of experiment worth and evaluated control blades worth are included. Based on the information above, the NRC staff concludes that the SDM and excess reactivity, as limited by TS 3.1, are consistent with the licensee's analysis, and acceptable.

TS 4.1 Reactor Core Parameters

TS 4.1 states the following:

Specification:

a. The reactor core excess reactivity above reference core condition shall be verified annually and following any significant core configuration and/or control blade change. A significant core configuration change is defined as a change in reactivity greater than 0.002 Δk/k.

b. The shutdown margin shall be verified annually and following any significant core configuration and/or control blade change. A significant core configuration change is defined as a change in reactivity greater than 0.002 Δk/k.

(…)

TS 4.1, Specification a, requires performance of a surveillance annually and following any significant core configuration and/or control blade change to verify excess reactivity. The NRC staff finds that TS 4.1, Specification a, helps to ensure that changes in excess reactivity are monitored and controlled. The NRC staff finds that excess reactivity is a core parameter that is important to maintaining SDM. It is used in the evaluation of experiment safety. It is also used as an input parameter to some elements of the safety analysis. Monitoring this parameter also serves the purpose of detecting core reactivity anomalies such as misaligned control blades, disconnected control blades, fuel misloading, and burnable poison misloading. It also serves to detect model inaccuracies. The NRC staff also finds that the surveillance related to TS 3.1, Specification a, and the interval is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.1, Specification a, is acceptable.

TS 4.1, Specification b, requires performance of a surveillance annually and following any significant core configuration and/or control blade change to verify SDM. The NRC staff finds that TS 4.1, Specification b, helps to ensure that the SDM is maintained. The SDM is important because it demonstrates the ability to make the reactor subcritical by the amount defined even if a scrammable control blade fails to insert; the MURR regulating blade is not scrammable. The NRC staff also finds that the surveillance related to TS 3.1, Specification b, and the interval is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information d above, the NRC staff concludes that TS 4.1, Specification b, is acceptable.

The NRC staff finds TS 4.1, Specifications a and b, are consistent with the surveillance frequency guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 4.1, Specifications a and b, are acceptable.

Conclusions

The NRC staff reviewed key parameters of the MURR operating limits, including SLs and LSSSs, excess reactivity, and SDM provided in the SAR and RAI responses referenced above and concludes the following:

- The licensee has discussed and justified all excess reactivity factors needed to ensure a operable reactor. The licensee has also considered the design features of the control systems that ensure that this amount of excess reactivity is fully controlled under normal operating conditions.

- The definition of the SDM is negative reactivity obtainable by control rods to ensure reactor shutdown from any reactor condition. Based on the assumptions that the most reactive control rod is inadvertently stuck in its fully withdrawn position and that nonscrammable control rods are in the position of maximum reactivity addition, the analysis derives the minimum negative reactivity necessary to ensure safe reactor shutdown. The licensee conservatively proposes an SDM of 0.02 $\Delta k/k$ in the TSs. This value is readily measurable and is acceptable.

Based on the information described above, the NRC staff concludes that the nuclear design and TS Safety Limits, Limiting Conditions for Operation, and Surveillance Requirements evaluated above, are adequate for continued operation of MURR during the renewal period.

4.6 Thermal-Hydraulic Design

As described in SAR Section 4.6, MURR has a steady-state licensed power of 10 MWt and has pressurized, closed-loop, forced water-cooling. Separate cooling loops remove heat from the system through the HXs to the atmosphere by means of a cooling tower (see SER Figure 5-1, Figure 5-2, and Figure 5-3). The reactor can be operated in three different operational modes involving combinations of reactor power and coolant flow: (Operational Mode I) reactor power up to 10 MWt with forced cooling; (Operational Mode II) reactor power up to 5 MWt with forced cooling; and (Operational Mode III) reactor power below 50 kWt with natural convective cooling (no forced coolant flow).

TS 1.0 Definitions, define the proposed MURR operational modes:

> 1.17 **Operational Modes** - The reactor may be operated in any of three operating modes, depending upon the configuration of the reactor coolant systems and the protective system set points.
>
> a. Operational Mode I - Reactor can be operated at a thermal power level of ten megawatts or less.
>
> b. Operational Mode II - Reactor can be operated at a thermal power level of five megawatts or less.
>
> c. Operational Mode III - Reactor can be operated at a thermal power level of fifty kilowatts or less.

The NRC staff evaluation of the licensee's T-H analysis, including the NRC staff's confirmatory analysis is set forth below and demonstrates the acceptability of these TS 1.17 Operational Modes.

Operational Modes I and II: Forced Flow

SAR Sections 4.6.2 provides a description of the analysis supporting Operational Modes I and II. The design criteria are chosen to ensure that no subcooled boiling occurs, no departure from nucleate boiling (DNB) occurs, and no flow instabilities occur which could lead to DNB. To avoid DNB, the heat flux at each local section in the core is maintained at value less than the locally evaluated DNB heat flux. The MURR fuel geometry is similar to the ATR, so the significant testing performed to establish the ATR fuel DNB can be applied to the MURR. The MURR fuel is half as long as the ATR fuel, which provides conservatism for both the DNB heat flux and flow instabilities, associated with bulk boiling. The reactor has two primary cooling loops: Operational Mode I has both loops in operation; whereas, Operational Mode II has one loop in operation.

Operational Mode III: Natural Convective Cooling

SAR Section 4.6.1 provides a description of the analysis supporting natural convection cooling of the MURR core. The reactor pressure vessel head and cooling flow flange are removed thereby providing a direct natural circulation coolant flow path to the reactor pool. The flow rate is proportional to the direct heating of the coolant by the core. The T-H design analysis is performed to ensure that the coolant remained subcooled (i.e., suppression of boiling). At a Mode III (natural convection cooling) reactor power level of 150 kWt, the analysis indicates that the maximum surface temperature of the MURR fuel is 230.2 °F (110 °C). This analysis supports the SL of fuel temperature not exceeding 986 °F (530 °C) and LSSS (62.5 kWt) as provided in TS 2.1 and TS 2.2, Specification c, respectively (reference Section 4.5.3 of this SER for additional details on the TS limits).

NRC Staff Confirmatory TRACE Calculations of MURR Flow Stability and CHF Margin

MURR is a plate fuel reactor, which has 8 fuel elements that are arranged to form a cylindrical reactor core. Each 45-degree arc fuel element contains 24 plates with parallel flow channels between the plates. The fuel elements are shown in Figure 4-4. The fuel elements contain 24 fuel plates that are arranged in concentric arcs.

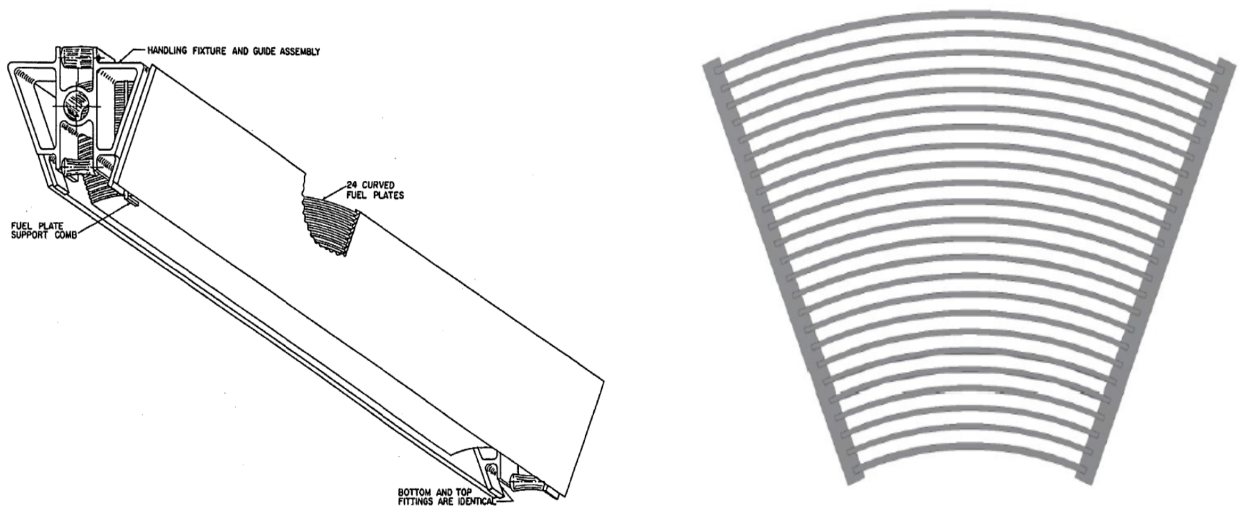

Figure 4-4 MURR fuel element

MURR Steady-State Operation Limits

Plate fuel reactors have thermal limits that are determined by flow stability. The point of vapor generation in the limiting channel determines the flow stability point. The reduction in flow from the increased pressure drop due to vapor generation in the limiting channel can lead to critical heat flux (CHF). The onset of the point of vapor generation is more limiting than the CHF since it occurs prior to the CHF in the flow channel. The limiting flow channel will be the channel with the maximum power to flow ratio. Among channels with equal power to flow ratios the channel with the highest heat flux at the exit to the channel will be the most limiting because it will have the smallest margin to net vapor generation. The eight fuel elements are peaked relative to each other because of the different burnup states of the fuel elements. Within a fuel element, the peak power fuel plates are those near the inner and outer radial limits of the fuel element because of the additional moderator outside of the fuel element. There is also a power variation along the plates with the peak power at the ends of the plates. This is because the moderator to fuel ratio increases at the ends of the plates. The MURR safety analysis determined that the limiting fuel plate and channel is at the inner edge of the fuel element. The operating limits for MURR are shown in Table 4-9.

Table 4-9 Operating Limits

Parameter	Value
Power	12.5 MW
Minimum Pressurizer Pressure	75 psia
Core Exit Pressure	54 psia
Core Inlet Temperature	155 °F
Core Inlet Velocity	23.1 ft/s

MURR calculated the limiting conditions based on taking the most limiting of three criteria. The three criteria are listed in attachment 1 of the response to RAI No. 4.18.a, which is the relevant part of the “MURR Upgrade Safety Limit Analysis”:

(1) 0.5 of the CHF predicted by the Bernath CHF correlation.
(2) The Flow Instability Ratio (FIR) calculated, using a value of $\eta = 25$, must be greater than 1.
(3) The coolant exit temperature in the hot channel must be less than the saturation temperature at the coolant exit pressure.

The three criteria are used to define the onset of flow instability in the core. Flow instability occurs when a significant void fraction is generated in the limiting channel. This increases the pressure drop across the channel and starves it of flow since there is an equal plenum-to-plenum pressure drop across all of the parallel flow channels in the core and the flow want to follow the path of least resistance. The reduction in flow causes the channel to go into CHF and may lead to significant fuel damage in the high power density, low melting point aluminum fuel.

Criterion 1 is based on the evaluation of Bernath CHF correlation but it is not being used to calculate the CHF limit. The criterion is actually based on correlating a multiplicative constant on the Bernath correlation with the onset of significant void generation in the test channel. The Croft test report (Ref. 85) shows that the ratio of the Bernath CHF to the measured burnout conditions varied from 1.73 to 2.82 for the test section with the width closest to MURR with more than half of the measurements exceeding a value of 2. The range of multipliers that would have to be applied to predict the burnout would range from 0.35 to 0.57 with more than half of the tests needing a value of less than 0.5. The value of 0.5 used in the MURR calculations are in the range of the measurements but it does not conservatively bound the data. A value of 0.35 would be needed to bound all of the test section data. Waters commented on the comparison of his ATR stability data to the Bernath correlation and stated, “A comparison of the present data with correlations for stable boiling burnout should not be meaningful. Such a comparison only shows that hydraulic instability generally occur at power levels of about 40 to 60 percent of those predicted for burnout by the Bernath correlation for experimental local conditions” (Ref. 86). Although the method used by Croft is not generally applicable to determining stability in the general case for plate reactors it does appear to be adequate to use in MURR because the flow geometry of the fuel in MURR is very close to the test facility used to measure the data. NRC staff confirmatory TRACE calculations show that a DNBR of 2.0 calculated using the Bernath correlation is close to the stability limit calculated using the Saha-Zuber correlation for the onset of significant void generation.

Criterion 2 is based on the flow stability methodology described by Bowring and uses data measured by Whittle and Forgan (Ref. 87). The parameters R in the Bowring methodology and η correlated by Whittle and Forgan are a measure of the subcooling where significant vapor generation can occur. The parameter η is used in calculating R determined from data measurements. The measured η for the test section with the width closest to MURR ranges from 23.4 to 28.2 with an average value of 26. Using higher values of η in calculating the FIR makes the correlation more conservative so using a value of η of 25 is not conservative since it is in the middle of the range of measurements. The recommended value of η for plate fuel reactor calculations is 32.5. Using 25 instead of 32.5 increases the predicted power for the onset of a flow instability by a factor of 1.038 (Ref. 88). Use of conservative peaking factors can make the overall calculations acceptable.

Criterion 3 is the condition for the onset of saturated boiling in the exit region hot channel. The Croft test data that forms the foundation of the MURR thermal limits shows that the channel flow instability occurred when the test section power level was 85-92% of the power required to reach saturation conditions for the test section that was closest to the MURR channel dimensions. The onset of significant voids in the MURR flow conditions occurs before the fluid exit temperature reaches the saturation boiling temperature, and therefore, it is never a limiting temperature and criterion 3 is not further evaluated by the NRC staff.

Of equal importance in calculating the onset of flow instability is how the methodology is applied. Most of the Whittle and Forgan data was from a test section that had a uniform heat flux profile across the plate in the direction perpendicular to flow and along the axial extent of the heated section. The Croft data had a chopped cosine axial shape and the heated section did not fully extend across the flow channel (Ref. 85). Data taken by Waters had a heat flux profile that varied in axial direction and across the lateral extent flow channel (Ref. 86). Waters' analysis of the data found that there was very little mixing in the lateral direction and that the lateral heat flux peaking was important in determining the onset of flow instability (Ref. 89). Feldman also found that the lateral heat flux profile was important for calculating the onset of flow instability when he was using the FIR method in PLTEMP to analyze the Croft and Waters data (Ref. 88). Feldman used a value of $\eta = 32.5$ in his calculations but the same conclusion applies when other values of η are used. Using the peak heat flux in the lateral direction and an axial heat flux profile that maximizes the heat flux at the channel exit as a hot channel heat flux profile is required for calculating conservative values for the onset of flow instability. MURR does apply the 1.15 lateral PF calculating the heat flux in Bernath CHF calculations but only uses a 1.03 factor in the COBRA FIR calculations.

The licensee determined that the limiting fuel plate for MURR was plate 1 of a new fuel bundle. MURR used a fuel channel width that was reduced by 10 percent from the nominal value of 0.08 in (0.20 cm) to 0.072 in (0.18 cm) to account for fuel swelling. In its application for LA No. 36 (Ref. 65), the licensee provided a set of curves showing the calculated limiting power as a function of core inlet temperature, core flow rate, and pressurizer pressure using conservative channel dimensions and power peaking factors. The tables indicated if the power was limited by the Bernath correlation (criterion 1) or the FIR (criterion 2). At the point on the curve corresponding to Pressure = 75 psia, Temperature = 160 °F (71 °C), and Flow = 3,200 gpm (12,113 lpm) (pressure is at the TS 2.2 limit, temperature and flow are conservative as compared to the TS 2.2 limit), the limiting power was shown to be ~12.5 MWt and the limit was due to application of the Bernath criterion. This is equal to the LSSS limit of 12.5 MWt. The analysis is performed at the LSSS limit of 12.5 MWt (not the licensed limit of 10 MWt) to demonstrate that a reactor transient terminated at the LSSS limit is analyzed and the results demonstrate acceptable fuel performance. As was discussed above, the limits derived from the application of the Bernath criterion is close to the stability limit calculated using the Saha-Zuber correlation for the onset of significant void generation confirming that the Bernath correlation method gives a reasonable prediction of flow stability due to the onset of significant void generation.

Based on the discussion above, the NRC staff finds that the analysis performed by the licensee used qualified calculation methods and conservative (or if not conservative, justifiable) assumptions. Therefore, the NRC staff concludes that the T-H analysis in the MURR license renewal SAR, as supplemented, demonstrates that the MURR has acceptable safety margins with regard to T-H conditions, for Operational Modes I and II, as defined in TS 1.17, Specifications a and b.

NRC Staff TRACE Confirmatory Calculations

The NRC staff used the TRACE thermal-hydraulic system safety analysis code to perform confirmatory flow stability and CHF calculations for the MURR reactor to compare against the results submitted by the licensee (Ref. 89). The criteria for flow stability in TRACE is the point of significant void generation in subcooled boiling. TRACE uses the Saha-Zuber correlation for the onset of significant void generation (Ref. 89). The Saha-Zuber correlation defines the onset of significant void generation using the Stanton (St) number as a function of Peclet (Pe) number. There are two regions defined in the correlation. The thermally controlled low flow region is defined by a constant Nusselt (Nu) number of 455. Since St is Nu/Pe the Stanton number limit is St = 455/Pe. The Stanton number criterion for the onset of significant voids in the flow-controlled region is 0.0065. This is the region that covers the range of operating conditions for MURR. Lower Stanton numbers mean an earlier onset of significant voids. The Whittle and Forgan data can be analyzed in terms of a Stanton number criterion for the onset of flow instability. The test section 3 data gives Stanton numbers in the range of 0.00886 to 0.0107 for the onset of flow instability so the Saha-Zuber correlation should produce conservative results for the Whittle and Forgan data. This was confirmed by comparing TRACE calculations of the Whittle and Forgan data to test results.

The TRACE calculations modeled a single limiting flow channel. The average heat flux used for the limiting channel was 2.22 MWt/per square meter (m^2). This was based on a core power of 12.5 MWt and values for the heat flux PFs given in Table 4-15 of the MURR SAR. It corresponds to a channel to core average PFs of ~3.04. The value for the heat flux assumes that all power is deposited in the fuel. The MURR SAR estimates that 93 percent of the power is deposited in the fuel. The channel flow area is based on a minimum plate spacing of 0.072 in (0.18 cm) which is 10 percent below the nominal value of 0.08 in (0.2 cm). The reactor exit pressure of 54 psia was based on the minimum pressurizer pressure of 75 psia. The axial discretization in the 24 in (60.9 cm) long active core region is 1 inch per node. The axial nodalization in the core region is consistent with the axial noding used in TRACE assessments of plate fuel flow stability and CHF data. A uniform axial heat flux profile was used because it is conservative compared to using the actual axial profile for flow stability limits. The flow stability calculated using a uniform heat flux should bound the flow stability limit for any realistic axial power profile since the core exit heat flux for the uniform heat flux will be higher that the heat flux for a realistic power shape and the core exit heat flux determines the margin to the onset of significant void generation. Summaries of the important inputs used in the TRACE calculations are listed in Table 4-10.

Table 4-10 TRACE Nominal Model Values

Parameter	Value
Power	12.5 MW
Core Exit Pressure	~ 54 psia
Core Inlet Temperature	~ 158 °F (70 °C)
Core Inlet Velocity	~ 23.1 ft/s (7.04 m/s)
Heat Flux	2.22 MW/m^2
Flow Channel Width	0.072 in (0.0018288 m)

Steady-State Flow Stability and CHF Calculations

The flow stability limit can be approached by reducing flow, reducing pressure, or increasing power from the nominal operating point. All three of these methods approach the point of net vapor generation by reducing the amount of liquid subcooling at the channel exit. TRACE calculations were performed that reduced flow, reduced pressure, and increased power in a stepwise fashion compared to the base case to find the flow stability limits. The results are shown in Table 4-11.

Table 4-11 TRACE Calculated Stability Limits

Parameter	Value
Core Power	13.75 MWt for 75 psia and 23.1 ft/s
Pressure	66.3 psia for 12.5 MWt and 23.1 ft/s
Core Inlet Velocity	20.3 ft/s (6.2 m/s, 3164 gpm) for 12.5 MWt and 75 psia

The stability limits calculated using conservative TRACE calculations show that there is margin in the operating limits when operating at the LSSS limits. The TRACE calculated CHF values at the flow stability limit show a minimum DNBR value greater than 2.0 using either the Bernath or Groeneveld correlation. For the 13.75 MWt core power limit calculation in Table 4-11 the calculated DNBR is 2.13 using the Bernath correlation and 3.13 using the 1995 Groeneveld correlation. The plots provided by the licensee in its application LA No. 36 (Ref. 65), which the NRC staff evaluated and found acceptable, indicate a power of ~13.8 MWt for the same conditions which corresponds to a DNBR vale of 2.0 using the Bernath correlation. The TRACE result shows that a DNBR of 2.0 calculated using the Bernath correlation is close to the stability limit of 13.75 MWt calculated using the Saha-Zuber correlation for the onset of significant void generation. Flow stability is more limiting than CHF for the MURR plate fuel as expected. A TRACE calculation was also performed to confirm that the flat axial power profile gave more conservative results than a realistic power profile. The results of the calculation showed that the flat axial power profile is more conservative because it has a higher exit heat flux.

The NRC staff finds that the confirmatory analysis demonstrates that the LSSS setpoints in the MURR TSs help to ensure that MURR, when operated in accordance with the TS, will exhibit T-H conditions that are acceptably bounded by the guidance provided in NUREG-1537 and the established SL for fuel temperature at all steady state conditions.

The NRC staff has reviewed the thermal-hydraulic data and analyses presented by the licensee, and finds the MURR thermal-hydraulic characteristics are acceptable and sufficient to ensure fuel integrity will not be lost under all analyzed conditions. The DNBR is 2.0 or greater. The limits provided by the TSs provide reasonable assurance that CHF will not be exceeded, thereby maintaining fuel plate temperatures within the values specified. The NRC staff finds that the T-H design has acceptable safety margins with regard to T-H conditions, for Operational Modes I and II, as defined in TS 1.17, Specifications a and b. The NRC staff concludes that the thermal-hydraulic design, as demonstrated in the safety analysis, is adequate for continued operation of MURR when operated within limits of the Technical Specifications.

MURR Steady State Natural Convective Cooling Mode Operation

MURR has a low power operating mode that relies on natural circulation cooling. The low power mode has a nominal maximum power level of 50 kWt and an LSSS scram set point of 62.5 kWt. The safety analysis supporting this mode of operation was performed for a power level of 150 kWt and the safety criterion is prevention of boiling in the core. The flow loop hardware configuration for this mode of operation has the flange in the invert loop and the reactor pressure vessel cover removed. Buoyancy driven natural circulation provides cooling of the reactor core in this mode of operation. Water enters the loop from the pool through the open flange in the invert loop and returns to the pool through the opening created with the reactor cover removed. The cooling flow flows up through the core in this mode of operation. This is the opposite direction of flow compared to the forced flow cooling mode. The limiting flow channel is the channel with the maximum power to flow ratio. Among channels with equal power to flow ratios, the channel with the highest heat flux at the channel exit is the most limiting because it has the smallest margin to net vapor generation. The eight fuel elements are peaked relative to each other because of the different burnup states of the fuel elements. Within a fuel element the peak power fuel plates are those near the inner and outer radial limits of the fuel element because of the additional moderator outside of the fuel element. There is also a power variation along the plates with the peak power at the ends of the plates because the moderator to fuel ratio increases at the ends of the plates.

MURR met the safety limit of fuel temperature not exceeding 986 °F (530 °C) by prevention of boiling in the core. Hazards Summary Report, Section 5.5.3 (Ref. 3), provides a detailed safety analysis of natural convection cooling of the reactor core for initial low power operation. This analysis shows that the reactor can safely be operated at a power level of 150 kWt in the natural convection mode when the inlet coolant to the core being provided from the reactor pool water at a temperature of 100 °F (38 °C).

Table 4-12 Parameters for Natural Convective Cooling Analysis

Parameter	Value
Power	150 kWt
Top of Pool Pressure	14.11 psia
Pool Temperature	100 °F (38 °C)

The MURR analysis in the Hazards Summary Report consists of three parts.

1) The analysis calculates a natural circulation core flow rate based on average core conditions.

2) The analysis uses the calculated core flow rate to determine an average bulk temperature rise in the flow through the core. The average liquid bulk temperature rise and some peaking factor parameters are used to calculate the maximum fuel surface temperature in the core.

3) The analysis determines what pressure and corresponding depth of water is needed in the pool to suppress boiling at the maximum fuel surface temperature.

The MURR analysis calculated the core flow rate using a method that assumes single-phase natural circulation flow. The assumption is validated by the calculated safety limit criteria which is prevention of boiling in the core. This analysis calculated a core flow rate of 5.42 kg/s (11.96 lb/s). This corresponds to a fluid velocity in the core of ~0.167 m/s.

To determine the temperature rise through the core and from the fluid to the cladding, the MURR analysis in the Hazards Summary Report calculates the peak fuel surface temperature using equation (9) provided in Section 5.5 of the MURR Hazards Summary Report (Ref. 3). This equation calculates the temperature rise in a core average channel as a base value and adjusts that base value by using peaking factors to estimate the temperature rise at the limiting location in the core. The analysis appears based on an earlier core design that used 5 kg of Uranium in the core and lower peaking factors than the current core design that uses 6.2 kg of Uranium. The NRC staff concludes that the analysis is still conservative since it is performed at a core power (150 kWt) that is more than 2 times the value of the trip set point (62.5 kWt). The increased power used in the analysis exceeds any effect caused by the changes in peaking factors due to the different core fuel loading. The report states that the maximum wall temperature is 230.2 °F (110 °C) but it is not clear how this value was obtained. Using the peaking factor values presented in the report to calculate the maximum wall temperature using equation (9), the NRC staff calculated a value of 194.7 °F (90.4 °C). Using the peaking factor values from the 6.2 kg core loading in the SAR gives a value of 204.8 F (96 °C). These values are both below the boiling temperature of water in the core.

To determine the margin to boiling the MURR analysis uses the Jens-Lottes correlation to determine what pressure is needed to suppress boiling in the core. The MURR analysis determined that the saturation temperature in the core needs to be less than 227.2 °F (108.4 °C) and requires 11.7 ft (3.56 m) of water above the top of the heated core to meet this limit. The MURR calculation states that the pool depth required for shielding is 17 feet of water above the core and the limit, TS 3.2, Specification g.13, is 23 ft (3.96 m). These limits require more water in the pool than what is required to prevent boiling in the core.

Confirmatory Calculation

The NRC staff used TRACE to perform an independent calculation to confirm that equation (9) was conservative for calculating peak core conditions in the MURR reactor. A single heated channel calculation was performed that used a peaking factor of ~3.4 compared to an average channel. This bounded the peak to average channel factor and the channel lateral power shape peaking. The NRC staff calculated the peak fuel temperature to be ~88 °C (192 °F). This value is less than the peak fuel surface temperature of 230.2 °F (110 °C) calculated by MURR using equation (9) of the Hazards Summary Report, and demonstrates acceptable fuel temperatures for natural convective cooling mode of MURR operation, Operational Mode 3, as defined in TS 1.17, Specification c.

Conclusions

Based on the discussions above, the NRC staff finds that the analysis performed by the licensee used calculation methods that are conservative with justifiable assumptions, as demonstrated by the results of the NRC staff's confirmatory calculations. In addition, the results of the thermal-hydraulic analysis demonstrate the acceptability of the Operational Modes in TS 1.17. The NRC staff also finds the modes described in TS 1.17 acceptable. Therefore, the NRC staff concludes that the thermal-hydraulic analysis in the MURR license renewal SAR, as supplemented, demonstrates that the MURR has acceptable safety margins with regard to thermal-hydraulic conditions in the low power natural convection operating mode.

4.7 Conclusions

Based on the above findings and conclusions, the NRC staff concludes that the licensee has adequately described the bases and functions of the reactor design to demonstrate that MURR can be safely operated and shut down from any operating condition or accident assumed in the safety analyses in SAR Chapter 13. The systems provide adequate control of reactivity, containment of coolant, barriers to the release of radioactive material, and sufficient radiation shielding for the protection of facility personnel. Nuclear and T-H design, as described in the SAR, and safety limits, as required by the TSs, adequately provide for the protection of fuel integrity. The NRC staff concludes that the reactor design is acceptable to support continued operation of the MURR facility during the renewal period.

5 REACTOR COOLANT SYSTEMS

5.1 Summary Description

Chapter 5 of the safety analysis report (SAR) states that the University of Missouri-Columbia Research Reactor (MURR or the reactor) has two coolant systems that use light water. The pressurized primary coolant system (PCS) cools the reactor core during forced convection operation. The reactor is located in an open concrete pool whose coolant system (PoolCS) provides cooling for the pool and for the reactor during natural convection operation. Both the PCS and PoolCS water are filtered and demineralized through cleanup systems designed to maintain conductivity less than 3 micro-ohms. Two heat exchangers (HXs) in the PCS and the one in the PoolCS are capable of removing the heat from full reactor power operation to the secondary coolant system and the cooling tower, which is located adjacent to the building. The PoolCS maintains the pool temperature sufficiently cool to minimize evaporative losses and to prevent thermal degradation of the demineralizer system.

SAR Section 5.1 describes the three modes of operation supported by the reactor coolant systems as follows:

> Mode I—At power levels of up to 10 MWt with the PCS pressurized and at a flow rate of approximately 3,750 gpm (14,195 liters per minute (lpm), and a pool coolant flow rate of approximately 1,100 gpm (4,164 lpm); used when all heat exchange and pumping capacity is available;
>
> Mode II—At power levels of up to 5 MWt with the PCS pressurized and at a flow rate of approximately 1,875 gpm (7,098 lpm), and a pool coolant flow rate of approximately 600 gpm (2,271 lpm); utilizing only half the design heat exchange and pumping capacity available; and
>
> Mode III—At power levels of up to 50 kWt with the PCS open to the reactor pool because the reactor pressure vessel head removed, the flanged port open, and the pool water level as the elevation of either the upper or lower reactor bridge; used for core flux calibrations following the loading of a new core, or after fuel rearrangement.

Mode I exists because the reactor was originally operated with a maximum power level of 5 MWt. When the licensed power level increased to 10 MWt, additional cooling system equipment was added to the facility.

Technical Specification (TS) 1.17 defines the operational modes for MURR, which the NRC staff evaluated and found acceptable in SER Section 14.1.

5.2 MURR Coolant Systems

5.2.1 Primary Coolant System

SAR Section 5.2 describes the PCS, which consists of the reactor pressure vessel, two main circulating pumps, two HXs, two automatic isolation valves, a pressurizer, a closed in-pool convective cooling system (decay heat removal system), an in-pool invert loop and anti-siphon system, a fuel element failure monitoring system, and a bypass loop for water cleanup (see Figure 5-1, below). SAR Section 13.2.9.4 describes the pressure relief valves, which are located on the PCS lines and the pressurizer tank. The drawing in SAR Figure 5.1 indicates the locations of a ½-inch (in) (1.27-centimeter (cm)) valve for the pressurizer tank and a 2-in (5.08-cm) valve for the PCS cold leg.

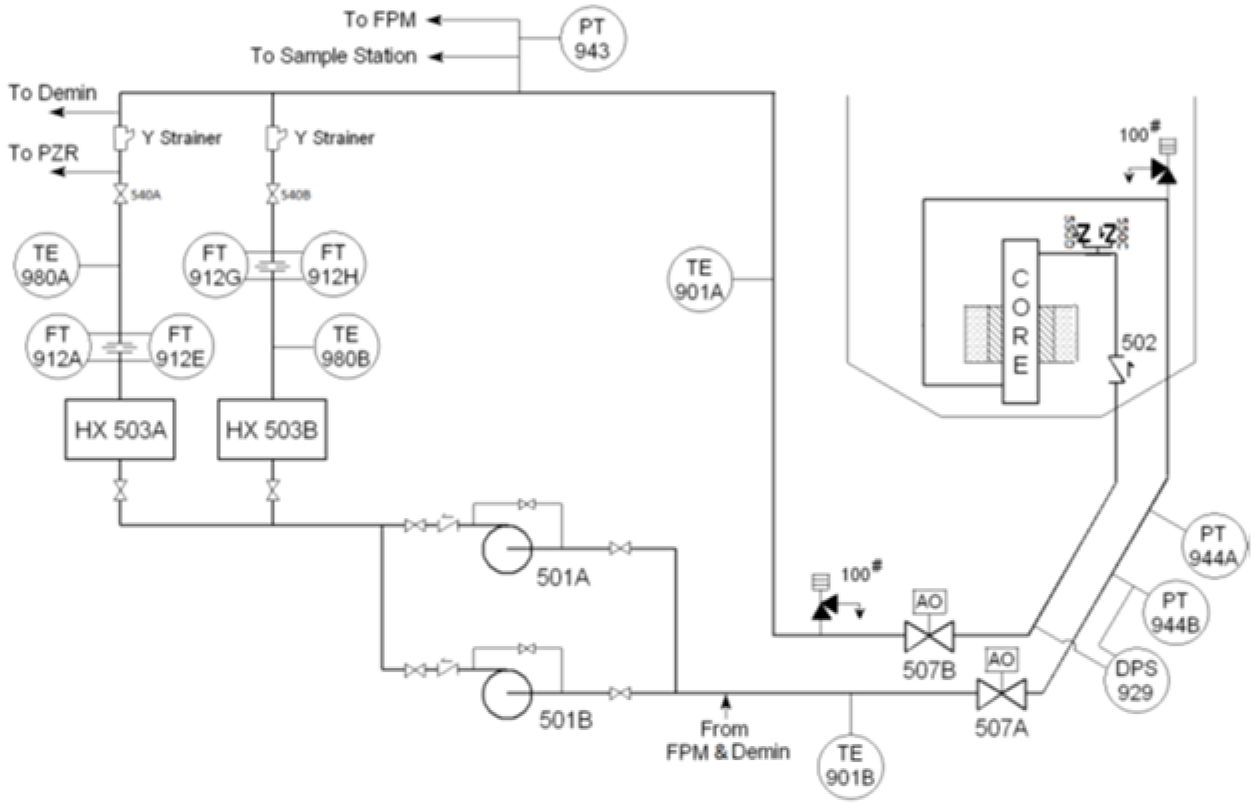

Figure 5-1 Simplified schematic of the primary coolant system

As described in SAR Section 5.2, the MURR PCS allows continuous full-power operation (Mode I) of the reactor while maintaining the PCS coolant within an acceptable temperature and flow rate, as specified in TS 2.2, "Limiting Safety System Settings." In its response to Request for Additional Information (RAI) No. 5.1 (Ref. 18), the licensee states that the principal purpose of the limiting safety system setting (LSSS) inlet temperature and flow-rate limits of 155 degrees Fahrenheit (°F) (68.3 degrees Celsius (°C)) and 3,650 gpm (13, 817 lpm) (two loop operation) in TS 2.2, Specification a, is to preserve the integrity of the fuel plates and cladding and to protect the safety limit provided in TS 2.1, SER Section 4.5.3. The normal (Mode I) operating inlet temperature is 120 °F.

SAR Section 4.6.1 and the licensee's response to RAI No. 4.16 (Ref. 26) also state that low-power, natural convection operation (Mode III) of MURR is possible without the heat removal capability of the PCS. In Mode III operation, the head of the pressure vessel is removed, thereby connecting the PCS with the large heat sink capacity of the 28,000 gallons (gal) (105,991 liters (l)) of coolant of the PoolCS. Operation in the natural convection cooling alone is allowed up to a power level of 50 kWt in accordance with the limits in TS 2.2, Specification c (see Section 4.2.1 of this safety evaluation report (SER)). Adequate PoolCS level, required by TS 3.2, Specification f.3, provides cooling capacity and adequate shielding from direct core radiation (see SER Section 7.2).

SAR Section 13.2.1 describes the potential for a loss-of-coolant accident (LOCA) in the PCS or SCS, which is minimized by certain design features. Isolation valves allow isolation of the PCS coolant near the penetrations in the biological shield wall. As described in SAR Section 6.3, a worst-case siphoning of the PCS coolant caused by a break in the PCS could only siphon water to approximately 5 feet (ft) (1.5 meters (m)) above the reactor core. If that were to occur, a reactor shutdown, required by TS 3.3, Specification a (see SER Section 5.2.7) and continued coverage of the core by the PCS coolant makeup water system and the anti-siphon system would prevent loss of fuel integrity.

SAR Section 3.1.5 states that the PCS is protected from overpressure by relief valves installed on the pressurizer and primary coolant piping. The relief valves are set lower than the TS 3.5, Specification b limit of 110 pounds per square inch gauge (psig), (see SER Section 7.2), thus providing a sufficient margin to assure that the PCS design pressure of 125 psig will not be exceeded.

SAR Section 5.5 describes the chemical environment of the PCS, which is maintained to inhibit corrosion of the fuel cladding, core components, and the PCS structures constructed of aluminum and stainless steel. The maintenance of key chemical properties within acceptable levels minimizes the potential for fuel-clad-corrosion-induced failure and activated contaminants becoming a radiological hazard. TS 4.3 (SER Section 5.2.7) requires surveillance to be performed to test the reactor coolant systems, and PCS, and PoolCS.

SAR Section 5.2.11 states that PCS coolant is sampled as stated in TS 3.3. Additionally, the licensee's response to RAI No. 5.2.a (Ref. 24) states that the PCS is sampled in accordance with preventative maintenance surveillance check procedure R4-W1, "Primary Water Analysis," and operating procedure OP-RO-531, "Primary and Pool Sample Station," on a weekly basis. This includes pH, radioisotopes, and conductivity testing. In response to RAI No. 5.2.a (Ref. 24), the licensee also states that, in addition to the weekly sampling, the PCS coolant conductivity is also measured continuously and display in the control room from which the operators log the value every 2 hours. In SAR Section 5.4.9, the licensee states that it also performs a monthly analysis of PCS coolant for the presence of tritium.

In its response to RAI No. 5.3 (Ref. 18), the licensee states that the PoolCS is sampled weekly in accordance with preventative maintenance surveillance check P5-W1, "Pool Water Analysis," and operating procedure OP-RO-531.

In its response to RAI No. 5.2.b (Ref. 17), the licensee states that MURR operates at 10 MWt approximately 150 hours per week. Consequently, the total gamma-ray activity resulting from neutron capture, fission products, and activation products is substantial. Any resulting gamma-ray radiolysis would be maximized in the reactor core and fuel storage areas in the reactor pool. In both locations, the water is continuously purified through ion exchange, maintaining a quality standard for purified water that meets ISO 3696 Grades 1 and 2. Table 5-1 below gives the water quality data for the MURR PoolCS and PCS, as reproduced from the licensee's response to RAI No. 5.2.b (Ref. 17).

Table 5-1 Pool and Primary Coolant Systems Water Quality

System	pH	Conductivity (μS/cm)	Total Dissolved Solids (ppm)
Pool	5.5 ± 0.3	1.9 ± 0.9	1.2 ± 0.6
Primary	5.6 ± 0.3	1.2 ± 0.5	0.8 ± 0.3

The licensee states that during reactor operation, the vent tank contains water, water vapor, air, and any collected gases resulting from radiolysis. In its response to RAI No. 5.2.b (Ref. 17), the licensee states that under routine MURR operating conditions, radiolysis of water in the PCS will occur; however, given the high water quality, recombination of the initial radiolysis species (hydrogen (H_2) atoms, hydroxyl radicals, and hydrated electrons) occurs on a picosecond timescale, resulting in negligible production of the stable radiolysis products H_2, hydrogen peroxide, and oxygen (O_2)). The licensee indicates that observation of H_2 or O_2 gases in the PCS vent tank requires the presence of these gases at concentrations exceeding their solubility at the primary coolant temperature (normally 54 °C (129 °F)). Table 5-2 below summarizes the solubility values, as provided in the licensee's response to RAI No. 5.2.b (Ref. 17).

Table 5-2 Hydrogen and Oxygen Gas Solubility in the PoolCS and PCS Water

Gas	PoolCS Water (38 °C)		PCS Water (54 °C)	
	Concentration (gal/L)	Grams in the PoolCS (28,000 gal)	Concentration (gal/L)	Grams in the PCS (2,000 gal)
H_2	0.0014	148.4	0.0012	9.1
O_2	0.0066	699.6	0.0054	40.9

In its response to RAI No. 4.10 (Ref. 18), the licensee states that the conductivity and pH are maintained at 2.0 micro-Siemens (μS) and 5.0 to 6.0, respectively. Radioiodine concentrations in the PCS coolant are monitored. This requirement provides further assurance of adequate early detection of a loss of fuel cladding integrity.

Furthermore, in its response to RAI No. 5.2.b (Ref. 17), the licensee indicates that if radiolysis without recombination, in excess of negligible quantities, were to occur in the PCS, hydrogen gas would be constantly produced beyond its solubility limit and ultimately be collected in the primary coolant system vent tank causing it to pressurize and sporadically off-gas to the pool and be exhausted from the containment building by the pool-sweep system. The pool sweep system provides ventilation at the top of the pool, and any gases are exhausted through the hot exhaust line and released by the main exhaust stack (SAR Section 6.2.3.8), without recirculation in the containment building, and with significant dilution provided by the main exhaust system.

Based on its review of the information in the SAR and responses to RAIs as described above, the NRC staff finds that the licensee has shown that neither H_2 nor O_2 gas emanation is observed in the irradiated fuel storage areas in the MURR pool nor from the PCS vent tank, indicating that radiolysis, without recombination, of MURR PoolCS and PCS water is negligible. Furthermore, any gases emanating from the PoolCS and PCS are captured by the PoolCS sweep system and exhausted from the MURR containment building without being recirculated in the containment building and with large dilution factors. The NRC staff reviewed the PCS and finds that sufficient heat can be removed from the fuel under all possible operating conditions to preclude loss of fuel integrity from thermal-stress-related failure. Accordingly, the NRC staff concludes that the PCS is acceptable for continued operation during the renewal period.

5.2.2 Primary Coolant Makeup Water System

SAR Section 5.6 describes the primary coolant makeup water system (PCMWS), stating that it consists of the pressurizer tank, primary coolant charging pump, automatic controls, a nitrogen supply system, and two 7,000-gal (26,498-L) water storage tanks. The purpose of the PCMWS is to ensure that the PCS pressure is maintained within the LSSSs (TS 2.2) for both 5 MWt and 10 MWt operation, and to replenish PCS coolant lost due to evaporation and routine sampling. Two 7,000-gal (26,498 l) demineralized water storage tanks provide the PCS make-up water. Under normal circumstances, make-up water is provided by the PCS pressurizer, which is sized to provide coolant inventory for changes in temperature in the PCS. If additional coolant make-up water is required, the positive displacement belt driven charging pump to the PCS automatically adds water, at a conductivity of less than 2.0 µmho. The PMWCS maintains the pressure in the PCS by injecting N into the pressurizer. An automatic switch maintains the desired pressure. Primary coolant water inventory and level are automatically controlled by a charging pump. TS 3.9, Specification a (see SER Section 5.2.7), requires the PCMWS to be connected to a source of at least 2,000 gallons of primary grade water.

The NRC staff finds that the operation and the design capacity of the PCWMS appears sufficient to provide makeup water to replace coolant lost due to minor leaks in order to maintain an acceptable primary coolant loop inventory. The NRC staff also finds that the PCWMS design helps to ensure the PCS pressure is maintained at the pressurizer pressure of the LSSS setpoint (TS 2.2). Based on its review, the NRC staff concludes that the PCMWS is acceptable for continued operation during the renewal period.

5.2.3 Reactor Convective Cooling System

Section 5.2.7 of the “Hazards Summary Report,” dated July 1, 1965 (Ref. 3), describes the reactor convective cooling system (RCCS), which includes two parallel, redundant valves, the PoolCS HX, and the necessary piping and headers to provide a flow path for the removal of decay heat following a shutdown that is accompanied by PCS loop isolation or in the event of the loss of normal coolant flow (see SER Figure 5-4, SER Sections 13.3 and 13.4).

The NRC staff finds that the RCCS appears capable of cooling the reactor following a shutdown accompanied by a loss of PCS or in the event of loss of normal coolant flow. On this basis, the NRC staff concludes that the RCCS is acceptable.

5.2.4 Pool Coolant System

SAR Section 5.3 states that the MURR PoolCS is designed to transfer a heat load of at least 10 MW from the PoolCS to the SCS through a water-to-water plate-type HX. The PoolCS cooling water is pumped from the reactor pool through the holdup tank, the circulating pumps, the PoolCS-to-SCS HX and is returned to the reactor pool through the pool diffuser spool. The PoolCS comprises two main circulating pumps, an HX, an automatic isolation valve, a reflector plenum natural convection valve, a holdup tank, a return diffusor, and a bypass loop for water purification. Figure 5-2 below provides additional details.

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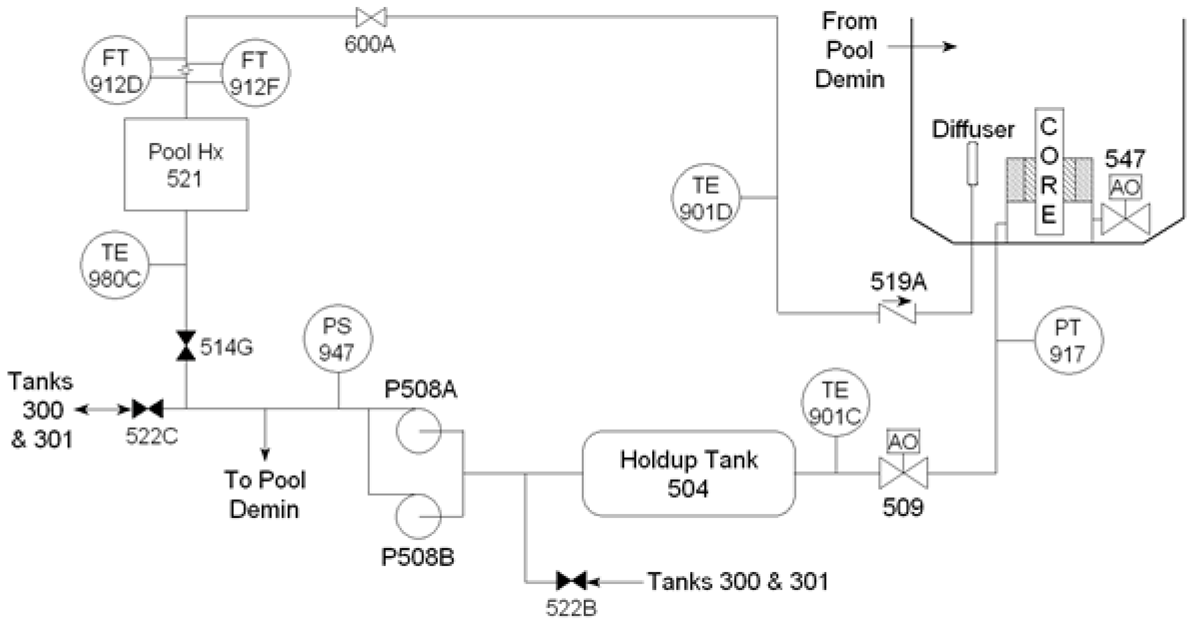

Figure 5-2 Simplified schematic of the pool coolant system

SAR Section 13.2.9.1 and the licensee's response to RAI No. 5.6 (Ref. 18) provide a description of the consequences of a leak from the PoolCS. A small leak from the pool coolant hold-up tank would easily be detected before the water level in the reactor pool had lowered significantly. The leak would also be visible during observation of the mechanical equipment room during periodic routine Reactor Operator (RO) patrols. Any leakage would be collected in sumps that would activate float switches that would automatically pump the water to the Liquid Waste Disposal System (as described in SAR Section 9.11.4). Activation of these switches will also cause an audible and visual alarm in the control room, and sufficient time would be available for the RO to shut down the reactor, and secure the pool coolant circulation pumps and close isolation valve V509 before a significant loss of pool water could occur. A large leak could potentially cause a major loss of water from the pool. In the event of a large leak in the pool coolant hold-up tank, an automatic reactor scram would occur from either a reduction in coolant flow, pressure, or pool water level. Regardless, a large or small leak in the pool coolant hold-up tank would not interfere with reactor cooling, cause an uncontrolled loss or release of primary coolant, or prevent a safe reactor shutdown. In the worst-case scenario, the leak would occur in an unisolable section of the PoolCS, and the emergency pool fill system (EPFS) would provide make-up water in excess of 1,000 gpm (3,785 lpm) to ensure that the reactor pressure vessel remained covered by coolant. The licensee also states that TS 3.9, Specification b, helps to ensure a source of emergency pool water should a leak occur (see SAR Section 9.7.1).

SAR Section 5.5.4 describes the holdup tank, which provides decay time for any Oxygen and Nitrogen isotope radioactivity to decay before returning to the pool. PoolCS coolant is sampled weekly per licensee procedure requirements for gamma-ray-emitting isotopes to detect any significant increase in leakage from the PCS, particularly around the pressure vessel head flange gasket.

SAR Section 11.1.5.1 discusses radiation levels above the pool during full-power operation (Mode I). The licensee states that these radiation levels are considered acceptable and in conjunction with the Radiation Exposure Control Program discussed in SAR Section 11.1.5, and the As Low As Is Reasonably Achievable (ALARA) Program should ensure personnel exposures remain below the limits in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, “Standards for Protection against Radiation.”

The NRC staff reviewed the MURR PoolCS and finds that the PoolCS design and HX capacity, as described in the MURR SAR, are adequate for removing any heat load created by reactor operation. The NRC staff also finds that the PoolCS system design minimizes the potential for leakage of PoolCS water to the environment and that differential pressure (between the PoolCS and SCS, See SER Section 5.2.5) and radioactivity monitoring should allow sufficient time for corrective action to mitigate any leakage. On this basis, the NRC staff concludes that the PoolCS is acceptable for continued operation during the renewal period.

5.2.5 Secondary Coolant System

SAR Section 5.4 describes the secondary coolant system (SCS), which is designed to transfer the heat from the PCS and PoolCS to the environment through the plate-type HXs and a cooling tower. The SCS also provides a heat sink for the laboratory building air conditioning loads. The SCS consists of four circulation pumps, a PoolCS coolant HX, two PCS coolant HXs, two automatic temperature control valves, a water treatment system, a cooling tower, and a radiation monitoring system. The coolant temperature at the inlet and outlet of the cooling tower are displayed in the control room. Secondary coolant temperatures are controlled automatically by butterfly valves to maintain a constant cold-leg temperature in the PCS and PoolCS. Conductivity and pH are monitored automatically and adjusted by the water quality control system located in the cooling tower. Other chemical additions are performed manually to help control water hardness and microbial growth (see Figures 5-3 and 5-4, below).

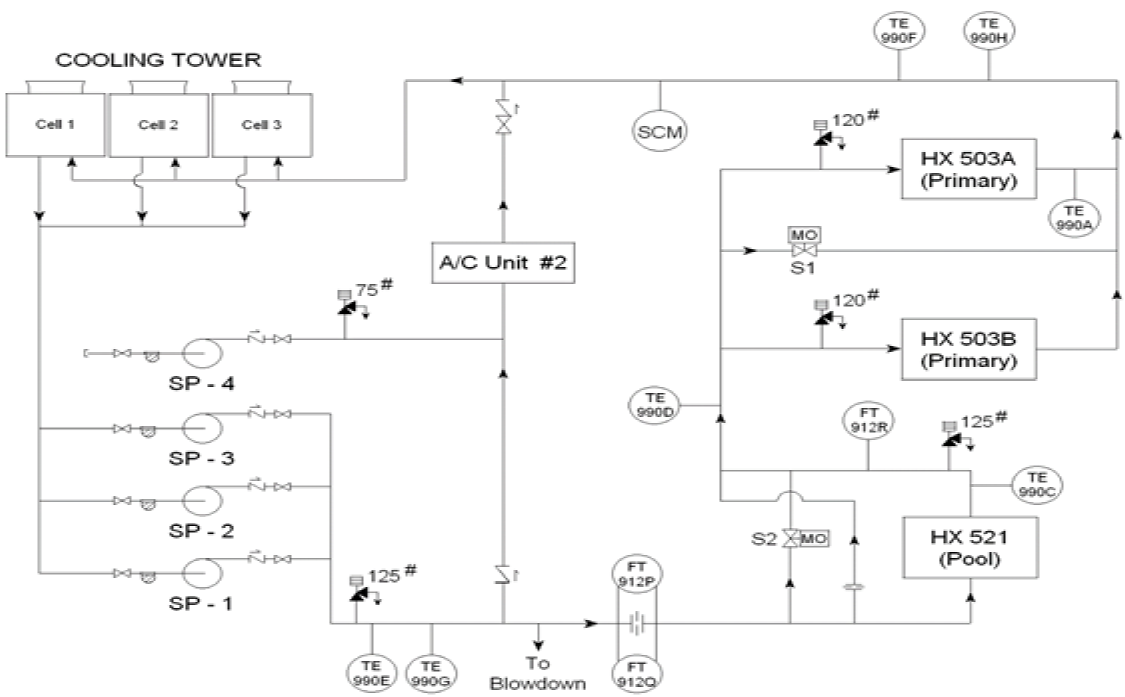

Figure 5-3 Simplified schematic of the secondary coolant system

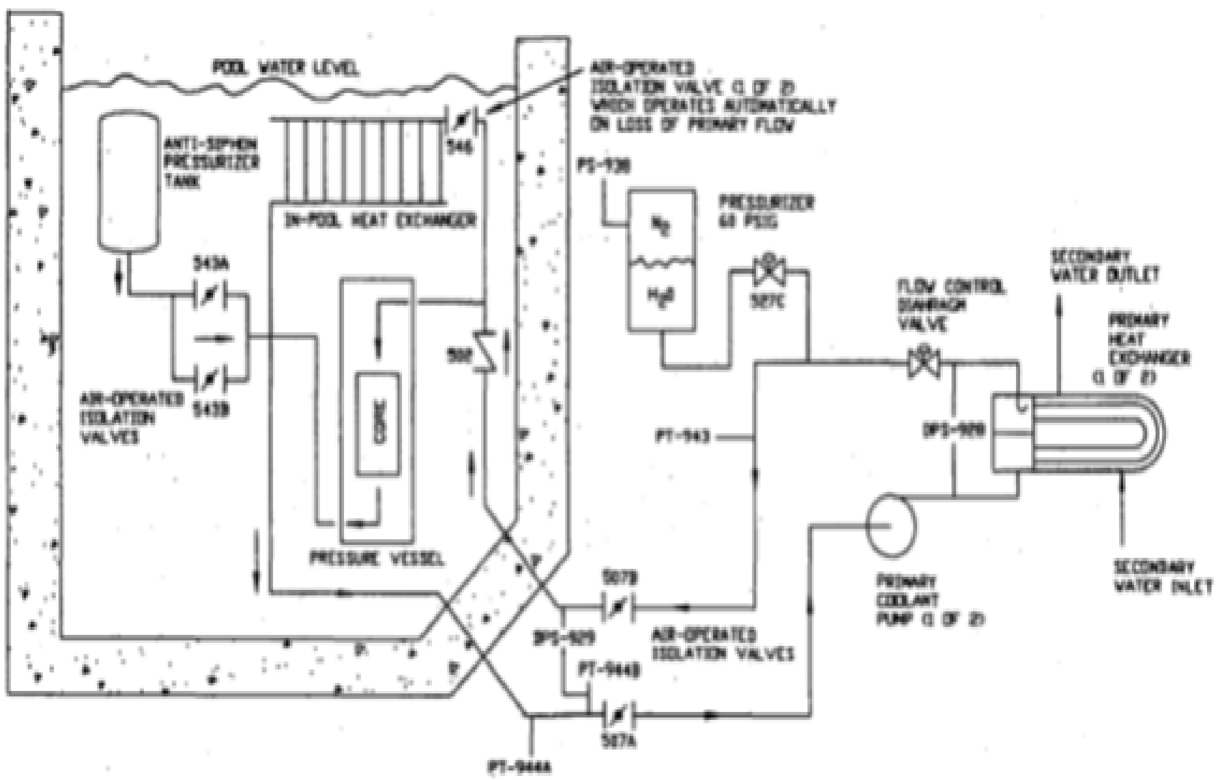

Figure 5-4 Schematic showing the reactor vessel location within the pool

In its response to RAI No. 5.3 (Ref. 18), the licensee states that the design of the plate-type HXs makes a PCS-to-SCS or a PoolCS-to-SCS leak extremely unlikely. The plate design is such that the most likely leak path is past a plate-to-plate gasket and into the mechanical equipment room (Room 114). For a leak to occur into the adjacent coolant system, a leak path would have to be created through one of the plates, which is minimized by a design that has no flow stagnation points. In the event of this unlikely leak path in the PoolCS coolant HX, pressures are higher on the SCS side of the HX than on the pool side under all operating conditions. This would result in a SCS-to-PoolCS leak. In the event of this leak path occurring in one of the PCS HXs, pressures vary on the SCS side such that the PCS coolant could conceivably leak into the SCS. Radioactivity is monitored in the SCS by an in-line sodium iodine detector with sufficient range and responsiveness to detect an increase in activity. In its response to RAI No. 5.3 (Ref. 18), the licensee states that the radiation monitor readings are recorded every 2 hours in accordance with its procedures.

In its response to RAI No. 5.3 (Ref. 18), the licensee states that, in regards to environmental consequences to this hypothetical leak at an overly conservative leak rate of 2.2 liters per hour (0.53 gallons per hour), the activity concentration for the three most significant isotopes associated with PCS coolant intruding into the SCS would be less than 50 percent of the effluent limits in Table 2, “Effluent Concentrations,” of Appendix B, “Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage,” to 10 CFR Part 20 at the time of the longest interval between logs (2 hours). Upon the discovery of the leak with the SCS alarm, the activity concentrations of the individual constituents of the contaminated SCS water would be below the uncontrolled release limits for those isotopes and would not pose a dose threat to the public.

The NRC staff reviewed the MURR SCS and finds that the SCS design and HX capacity, as described in the MURR SAR, are adequate for removing any heat load required by reactor operation. The NRC staff also finds that the system design should minimize the potential for leakage of PoolCS water to the environment and that differential pressure and radioactivity monitoring should allow sufficient time for corrective action to mitigate any leakage. The NRC staff concludes that the SCS design is acceptable for continued operation during the renewal period.

5.2.6 Reactor Coolant System Cleanup System

SAR Section 5.5 describes the PCS cleanup system, which is designed to help preclude corrosion of the fuel and other core and PCS components and to minimize the presence of activated contaminants in the coolant systems. There are two demineralizer loops associated with the reactor: one serving the PCS and one serving the PoolCS. Each loop is independent of the other with the exception of the demineralizer tanks, which are interchangeable from one loop to another by means of a pipe and valve arrangement. This arrangement allows a depleted demineralizer bed to be removed from service and a new bed placed on-line without an interruption in reactor operation. The two main purposes of the cleanup system are (1) to reduce the inventory of radioactive nuclides present in the coolant, and (2) to help maintain a primary-grade level of water quality, which limits chemical corrosion to essential components. Both PCS and PoolCS coolant is processed by the system. The PCS cleanup system uses particulate filters and a demineralizer to control the conductivity below 3.0 μmho (measure of electrical conductance). Conductance probes monitor demineralizer inlet and outlet conductivity and provide a high conductivity alarm in the control room. The entire purification system for both the PCS and the PoolCS are contained behind concrete-shielded work cells (fully enclosed rooms). System operations are performed remotely by use of reach rods that connect valve handles to the valves through a 2-ft thick concrete wall.

In its response to RAI No. 5.5 (Ref. 17), the licensee states that any release of water containing radionuclides from the reactor coolant system cleanup system would drain into sumps located in the floors of the coolant cleanup rooms. The presence of water in the sumps would activate switches that would automatically pump the water to the liquid waste disposal system, as described in SAR Section 9.11.4. Pool water quality is normally very high such that radiation levels on the demineralizer are not significant. However, over time, the demineralizer resin beds become depleted and are replaced with new resin material. Additionally, the demineralizer room is accessible only to authorized reactor staff members who are knowledgeable concerning the potential radiation levels. ALARA practices help to ensure that the occupancy time is minimal and could be further restricted if unusual radiation levels are present. System operation is performed by using extensions on valve handles from the valve body through a 2-ft (0.61-m) concrete shield wall. As such, the licensee indicates that leaks in the reactor coolant cleanup system should not result in any additional exposure to personnel or release to the environment in excess of the limits in 10 CFR Part 20. The licensee also states that dosimetry records and routine monitoring results have indicated that no unusual exposures to either the operating staff or to the public have occurred during the normal performance of maintenance of this system.

SAR Section 5.5.7, indicates that the PCS or PoolCS flow may be remotely diverted to any of the three mixed bed demineralizer tanks by the use of reach rods. When a resin bed is depleted, the standby bed is placed on service, the depleted resin bed is removed from service and the resins transferred to the resin storage tank. A depleted resin bed that has undergone decay is then transferred by the use of a water carrier to the resin storage tank. The depleted bed is then dumped into resin drying barrels. A new resin bed is transferred to the empty demineralizer tank and placed in standby. All transfer water, including effluent wastes, is directed to either the radioactive liquid waste retention disposal system or the drain collection tank. Resin beds are removed from service when conductivity can no longer be maintained at less than 3.0 µmho. Operational history has shown that resin beds for the pool coolant system generally last five months. Primary coolant system resin beds generally last 5 years. The spent resin is placed in a storage tank to reduce the total activity sent to the radioactive liquid waste system. Previously decayed resin is then transferred, through a water carrier to the regenerator. The regenerated resin is then placed back in standby within the reactor coolant cleanup system.

In its response to RAI No. WG13 (Ref. 103), the licensee provided clarification that the regeneration station described in SAR Section 5.5.7 has not been used in over 30 years, as only provides a central transfer point, allowing resin to be transferred to or from any of the demineralizer tanks, including the resin storage tank. The regeneration station is also used for dumping depleted resin and loading new resin.

The NRC staff reviewed the MURR PCS cleanup system and finds that it is of similar design to, and consistent with coolant chemistry limits at, other similar licensed non-power reactors. The NRC staff also finds that the system design includes consideration of ALARA principles to ensure doses to the workers are minimized, and any leaks of the system can be mitigated with minimal exposure to the workers. Based on the information provided above, the NRC staff concludes that the PCS cleanup system will continue to minimize the potential corrosion of the fuel, core components, and will help to maintain the chemistry limits of the PCS water within acceptable limits during the renewal period.

5.2.7 Coolant Systems Technical Specifications

The NRC staff finds that the SAR and RAI responses comprehensively describe the MURR coolant systems. TSs control design requirements (TS 5.2), Limiting Conditions for Operation (TS 3.3 and TS 3.9), and Surveillance Requirements (SRs) (TS 4.3 and TS 4.9). These TSs are discussed below:

TS 5.2 Reactor Coolant Systems

TS 5.2, Specifications a through g, and i, and Exception a and b, state:

Specification:

The MURR utilizes three (3) reactor coolant systems: primary, pool, and secondary. The following design features shall apply to these coolant systems:

a. The reactor coolant systems shall consist of not less than a reactor pressure vessel, a primary pressurizer, two (2) primary coolant circulation pumps, two (2) primary coolant heat exchangers, two (2) pool coolant circulation pumps, one (1) pool coolant heat exchanger, and one (1) pool water hold-up tank, plus all associated piping and valves.

b. The secondary coolant system shall be capable of continuous discharge of heat generated at the operating power of the reactor.

c. The circulation pumps and heat exchangers of the primary coolant system shall constitute two (2) parallel systems separately instrumented to permit safe operation at five megawatts on either system or ten megawatts with both systems operating simultaneously.

d. The pool coolant circulation pumps shall be instrumented and connected so as to permit safe operation at five or ten megawatts on either pump or both pumps operating simultaneously.

e. All major components of the reactor coolant systems in contact with pool or primary water shall be constructed principally of aluminum alloys or stainless steel.

f. The pool and primary coolant systems shall have a water clean-up system.

g. The pool and primary coolant piping shall have isolation valves between the reactor and mechanical equipment room.

(…)

i. The reactor shall have a natural convection coolant flow path for Mode III operation.

(…)

<u>Exceptions</u>:

a. The reactor may be operated in Mode II with any component removed from the shutdown leg of the system for emergency repairs.

b. Some materials in off-the-shelf commercial components may be excepted from Specification 5.2.e.

TS 5.2, Specification a, requires the reactor coolant systems to consist of not less than a reactor pressure vessel, a primary pressurizer, two (2) primary coolant circulation pumps, two (2) primary coolant heat exchangers, two (2) pool coolant circulation pumps, one (1) pool coolant heat exchanger, and one (1) pool water hold-up tank, plus all associated piping and valves. The NRC staff finds that these components are integral to the analysis of thermal-hydraulic behavior and some of the accident and transient analysis. The NRC staff also finds that providing these components helps to ensure that important design features of the reactor control systems are maintained, as described in SAR Sections 1.2, 5.1, and 5.2. The components listed are the main components of the PCS and portions of the PoolCS. Based on the information above, the NRC staff concludes that TS 5.2, Specification a, is acceptable.

TS 5.2, Specification b, requires a secondary coolant system capable of continuously discharging heat generated by the operation of the reactor. The NRC staff finds that the SCS has a heat-rejection capability to the ultimate heat sink (the atmosphere) to reject the heat produced by the operation of the reactor, as described in the SAR Section 5.4. Based on the information provided above, the NRC staff concludes that TS 5.2, Specification b, is acceptable.

TS 5.2, Specification c, requires that the circulation pumps and heat exchangers of the primary coolant system constitute two parallel systems, separately instrumented to permit safe operation at five megawatts on either system, or ten megawatts with both systems operating simultaneously. The NRC staff finds that having this capability helps to ensure that the circulation pumps and HXs of the PCS are maintained and separately instrumented, as described in SAR Sections 5.2.3 and 5.2.4. The NRC staff also finds that TS 5.2, Specification c, also helps to ensure redundancy in the design of the heat removal system. Based on the information provided above, the NRC staff concludes that TS 5.2, Specification c, is acceptable.

TS 5.2, Specification d, requires the licensee to have PoolCS circulation pumps that are instrumented and available to support reactor operation at 5 MWt or 10 MWt, as described in SAR Section 5.3. The NRC staff finds that this specification helps to ensure that the PoolCS pumps are capable of removing the heat from reactor operation deposited in the pool and transferring it to the SCS, based on the design information provided in SAR Section 5.3. Based on the information above, the NRC staff concludes that TS 5.2, Specification d, is acceptable.

TS 5.2, Specification e, requires that the major reactor components that are in contact with the PoolCS or PCS water are constructed principally of aluminum or stainless steel to minimize the effect of corrosion on those components and the potential activation of corrosion products by the reactor, as described in SAR Section 16.1. The NRC staff finds that this specification helps to minimize the effect of corrosion and corrosion products in the reactor (TS 5.2, Exception b, provides exceptions and is evaluated below). Based on the information above, the NRC staff concludes that TS 5.2, Specification e, is acceptable.

TS 5.2, Specification f, requires the licensee to have a water cleanup system that is available to remove potentially radioactive contaminants in reactor PoolCS or PCS water and to protect reactor components against corrosion, as described in SAR Section 5.5. The NRC staff finds that this specification helps to ensure that the water cleanup system is available to remove corrosion products from the reactor PoolCS or PCS water and to protect reactor components against corrosion, consistent with the description in SAR Section 5.5. Based on the information above, the NRC staff concludes that TS 5.2, Specification f, is acceptable.

TS 5.2, Specification g, requires the licensee to have isolation valves for the PCS and PoolCS so that a line break can be isolated to minimize the water inventory lost from their respective systems. The NRC staff finds that these valves, which are discussed in SAR Sections 5.2.5 and 5.3.4, serve to isolate the PCS for cases in which the primary pressure decreases to the setpoint in TS 3.2. The NRC staff also finds that the isolation valves help to ensure that coolant inventory is maintained in the reactor pressure vessel. Based on the information above, the NRC staff concludes that TS 5.2, Specification g, is acceptable.

TS 5.2, Specification i, requires the licensee to have passive (natural) convective cooling flowpath to support the removal of heat generated during low-power operation in Mode III, as described in SAR Section 4.6.1 and in the licensee's responses to RAI No. A.25 and RAI No. A.50 (Ref. 17). The NRC staff finds that TS 5.2, Specification i, helps to ensure that the conditions required for heat removal are maintained during Mode III operation. Based on the information above, the NRC staff concludes that TS 5.2, Specification i, is acceptable.

TS 5.2, Exception a, allows any component of a shutdown loop to be removed as long as operation is limited to Mode II. The NRC staff finds that TS 5.2, Exception a, is consistent with the analysis provided with the application and issued License Amendment No. 36 (Ref. 66), which allows single loop operation in Mode II. The NRC staff also finds that each loop can be operated independently in accordance with TS 5.2, Specification c. Based on the information above, the NRC staff concludes that TS 5.2, Exception a, is acceptable.

TS 5.2, Exception b, allows some materials to be exempt from TS 5.2, Specification e, which requires that all major components of the reactor coolant systems in contact with pool or primary water be constructed principally of aluminum alloys or stainless steel. The licensee states, in its response to RAI No. A.49 (Ref. 22), that the use of TS 5.2, Exception b, which provides that some materials in off-the-shelf commercial components may be excepted, is intended primarily to apply to instrumentation components that are not commercially available in the materials specified in TS 5.2, Specification e. The licensee also acknowledges that some components may have improved corrosion resistance and perform in the PCS water more reliably using materials other than aluminum alloys and stainless steels. Additionally, the licensee states that these materials are evaluated with regard to corrosion potential, both individually and in galvanic potential with their surroundings, fatigue or cycle lifetime, temperature and pressure service reliability, and potential for dissolution, erosion, and activation in the coolant. The results of the evaluation are documented under the 10 CFR 50.59 process.

The NRC staff finds that the exception in TS 5.2, Exception b, allows instrumentation with some components that are not commercially available in aluminum alloys or stainless steels to be used in the reactor. The NRC staff also finds that changes to components are evaluated in accordance with the requirements in 10 CFR 50.59, to ensure the materials can withstand corrosion and can perform in the PCS water, and audited in accordance with TS 6.2, Specification a. Based on the information above, the NRC staff concludes that TS 5.2, Exception b, is acceptable.

The NRC staff reviewed TS 5.2, Specifications a through g, i, and Exceptions a and b. The NRC staff finds that TS 5.2, Specifications a through g, i, and Exceptions a and b describe key design features of the reactor coolant system at MURR and are consistent with the SAR. The NRC staff also finds these design features help to define the reactor system configuration as it is described in the models used to perform the accident analyses described in SAR Chapter 13. Based on the information above, the NRC staff concludes that TS 5.2, Specifications a through g, i, and Exception a and b, specify design requirements and are acceptable.

<u>TS 3.3 Reactor Coolant Systems</u>

TS 3.3 states:

<u>Specification</u>:

a. The reactor shall not be operated in Modes I or II unless the following components or systems are operable:

    (1) Anti-siphon system;
    (2) Primary coolant isolation valves V507A/B; and
    (3) In-pool convective cooling system.

b. The reactor shall not be operated with forced circulation unless:

    (1) The continuous primary coolant system fuel element failure monitor is operating,

OR

(2) The primary coolant system is sampled and analyzed at least once every four (4) hours for evidence of fuel element failure.

c. The reactor shall not be operated if a radiochemical analysis of the primary coolant system indicates an iodine-131 concentration of greater than $5x10^{-3}$ µCi/ml.

d. The reactor shall not be operated if a radiochemical analysis of the pool coolant system indicates gross radioactivity twice the historical average.

e. The reactor shall not be operated with forced circulation unless:

(1) The continuous secondary coolant system monitor is operating,

OR

(2) The secondary coolant system is sampled and analyzed for gross radioactivity at least daily.

f. The reactor shall not be operated if a radiochemical analysis of the secondary coolant system exceeds the limits of 10 CFR 20, Appendix B, Table 2, Column 2.

g. The conductivity of the water in the primary coolant system shall be maintained at less than 5 µmho/cm when averaged over a period of one (1) quarter.

h. The pH of the water in the primary coolant system shall be maintained between 5.0 and 7.0 when averaged over a period of one (1) quarter.

i. The conductivity of the water in the pool coolant system shall be maintained at less than 5 µmho/cm when averaged over a period of one (1) quarter.

TS 3.3, Specification a, requires that, for operation in Mode I or Mode II, the anti-siphon system, isolation valves, and in-pool convective cooling system be operable. The NRC staff finds that this is required because it is an assumption in the safety analysis for LOCA and loss of flow accidents (SAR Sections 13.2.3 and 13.2.4). The NRC staff also finds these systems and components help ensure that the reactor core would continue to be cooled by available reactor coolant water even if a postulated double-ended PCS pipe break event occurred by admitting a fixed volume of air to the highest point of the invert loop, thus eliminating any potential for siphoning. In addition, having these components operable helps to ensure that the conditions encountered during operation are consistent with those postulated in the safety analysis. Based on the information above, the NRC staff concludes that TS 3.3, Specification a, is acceptable.

TS 3.3, Specification b, requires that the reactor shall not be operated with forced circulation unless the PCS monitor is operable or, if not, the PCS is regularly sampled and analyzed at least once every 4 hours for evidence of fuel failure. In its response to RAI No. A.37 (Ref. 18), the licensee indicates that the PCS monitor setpoint was 12,000 counts per minute, which corresponds to an iodine-131 (I-131) concentration of approximately $1x10^{-4}$ microcuries per milliliter (µCi/ml). The licensee states that this value is below the TS 3.3, Specification c, limit of $5x10^{-3}$ µCi/ml, thus providing early detection of an I-131 presence in the PCS. The NRC staff finds that the fuel element failure monitor provides a continuous indication (readout) of activity in the reactor coolant system. The NRC staff also finds that allowing operation with the fuel element monitor out of service, which requires PCS sampling every 4 hours, to be acceptable based on the potential leakage path from the PCS to the PoolCS limits any activity to the PoolCS. In addition, the NRC staff finds that TS 3.3, Specification b, helps to ensure that systems and actions are maintained to detect the onset of fuel failure. Based on the information above, the NRC staff concludes that TS 3.3, Specification b, is acceptable.

TS 3.3, Specification c, limits the primary coolant system Iodine activity to 5×10^{-3} μCi/ml. In its response to RAI No. A.38 (Ref. 17), the licensee states that the limit of 5×10^{-3} μCi/ml represents a small percentage of the total I-131 activity (2.23×10^{-7} μCi/ml) in the reactor core and, as such, provides an effective indicator of a potential fuel element cladding failure. The NRC staff finds that TS 3.3, Specification c, helps to ensure that operation with potential fuel element cladding problems is avoided. Based on the information above, the NRC staff concludes that TS 3.3, Specification c, is acceptable.

TS 3.3, Specification d, prohibits operation of the reactor if a radiochemical analysis of the PCS indicates a gross radioactivity twice the historical average. The NRC staff finds that this specification helps ensure that any change in radioactivity in the PCS is monitored and limited to twice the historical average to ensure that any potential for radioactivity to be transferred to the PoolCS or SCS is minimized. Based on the information above, the NRC staff concludes that TS 3.3, Specification d, is acceptable.

TS 3.3, Specifications e, requires that the reactor is not operated with forced circulation unless the SCS monitor is operating or the SCS is sampled and analyzed for gross activity at least daily. SAR Section 5.4.8 describes the SCS monitoring system, which uses a scintillation detector to measure gross activity of the coolant and provide the results to an analog meter located in the control room. The meter contains an adjustable alarm setting that actuates a control room annunciator. The NRC staff finds that this TS helps ensure that radioactive isotopes detected in the SCS are limited so that the potential for dose to the public from the release of that inventory is controlled. The NRC staff also finds that TS 3.3, Specification e, helps to ensure that the SCS is routinely monitored for radioactivity and that an increase in activity that could be the result of a PCS or PoolCS to SCS leak is provided to the control room operators. Based on the information above, the NRC staff concludes that TS 3.3, Specification e, is acceptable.

TS 3.3, Specification f, prohibits reactor operation if the radiochemical analysis of the SCS exceeds the limits in 10 CFR 20, Appendix B, Table 2, Column 2, which are the effluent release limits for water (limit public dose to 50 mrem). The NRC staff finds that compliance with the limits in 10 CFR 20, Appendix B, Table 2, Column 2, helps ensure that any potential doses to a member of the public are compliant to the limits in the regulations in 10 CFR 20. This specification is also consistent with the guidance in American National Standards Institute/American Nuclear Society (ANSI/ANS)-15.1-2007, “The Development of Technical Specifications for Research Reactors,” issued 2007 (Ref. 57), and NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non Power Reactors,” issued February 1996 (Ref. 51). Based on the information above, the NRC staff concludes that TS 3.3, Specification f, is acceptable.

TS 3.3, Specification g, limits the conductivity in the PCS water to a value of less than 5 μmho/cm averaged over 1 quarter of a year. The NRC staff finds that TS 3.3, Specification g, helps to ensure that the corrosion rate on fuel and other core components in the PCS is monitored and controlled. The NRC staff also finds that the limit of less than 5 μmho per centimeter (μmho/cm) is consistent with the guidance in NUREG-1537. Based on the information above, the NRC staff concludes that TS 3.3, Specification g, is acceptable.

TS 3.3, Specification h, limits the pH in the PCS coolant to between 5.0 and 7.0 when averaged over 1 quarter of a year. The NRC staff finds that TS 3.3, Specification h, helps to ensure that the corrosion rate on fuel and other core components in the PCS is controlled. The proposed pH range of 5 to 7, averaged over 1 quarter of a year, is consistent with the guidance in NUREG-1537. This specification is also consistent with the guidance in NUREG-1537. Based on the information provided above, the NRC staff concludes that TS 3.3, Specification h, is acceptable.

TS 3.3, Specification i, limits the conductivity in the PoolCS to less than 5 μmho/cm when averaged over 1 quarter of a year. The NRC staff finds that TS 3.3 Specification i, helps to ensure that the corrosion rate on fuel and other core components in the PoolCS is controlled. The NRC staff also finds that the limit of less than 5 μmho/cm is consistent with the guidance in NUREG-1537. Based on the information above, the NRC staff concludes that TS 3.3, Specification i, is acceptable.

The NRC staff reviewed TS 3.3, Specifications a through i, and finds that they are consistent with the design basis of the facility, the MURR safety analysis, the effluent release limits in 10 CFR 20, and the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 3.3, Specifications a through i, are acceptable.

TS 3.9 Auxiliary Systems

TS 3.9, Specification a, states:

Specification:

a. The reactor shall not be operated unless the primary coolant make-up water system is operable and connected to a source of at least 2,000 gallons of primary grade water.

(…)

TS 3.9, Specification a, prohibits reactor operation unless the primary coolant make-up water system is operable and connected to a source of at least 2,000 gal (7,570 l) of primary grade water. SAR Section 5.6 states that the purpose of the PCMWS is to ensure the PCS pressure is maintained within the LSSSs for reactor operation. The NRC staff finds that TS 3.9, Specification a, helps to ensure that the primary coolant make-up water system is operable with a minimum quantity of makeup water (2,000 gal (7,570 l)) available during all modes of operation. The NRC staff also finds that the primary coolant make-up water system is capable of providing makeup water lost during normal reactor. Based on the information above, the NRC staff concludes that TS 3.9, Specification a, is acceptable.

TS 4.3 Reactor Coolant Systems

TS 4.3, Specifications b through h, state:

Specification:

(…)

b. The primary coolant system fuel element failure monitor shall be channel-checked on a monthly basis and channel-calibrated on a semiannual basis.

c. A primary coolant sample shall be taken during each week of reactor operation and a radiochemical analysis performed to determine the concentration of iodine-131.

d. A pool coolant sample shall be taken monthly and a radiochemical analysis performed to determine gross radioactivity.

e. A secondary coolant sample shall be taken quarterly and a radiochemical analysis performed to determine gross radioactivity.

f. The conductivity and pH of the water in the primary coolant system shall be measured on a monthly basis.

g. The conductivity of the water in the pool coolant system shall be measured on a monthly basis.

h. The primary coolant system relief valves shall be tested for operability biennially, with at least one of the valves tested on an annual basis.

TS 4.3, Specification b, requires the licensee to channel check the primary coolant system fuel element failure monitor monthly and calibrate the channel on a semiannual period. The NRC staff finds that this specification helps to ensure that the PCS fuel element failure monitor is channel operable and capable of detecting a possible fuel cladding defect or fuel failure. The NRC staff finds the surveillance interval for TS 4.3, Specification b, is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.3, Specification b, is acceptable.

TS 4.3, Specification c, requires the licensee to sample the PCS each week of reactor operation and to perform a radiochemical analysis to determine the concentration of I-131, which will provide an indication of a possible fuel cladding defect or failure. The NRC staff finds that the surveillance interval for TS 4.3, Specification c, is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.3, Specification c, is acceptable.

TS 4.3, Specification d, requires the licensee to sample and analyze the PoolCS coolant monthly for gross radioactivity to ensure a suitable water quality that protects the fuel cladding and reactor components from activated corrosion products. The NRC staff finds that the surveillance interval for TS 4.3, Specification d, is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.3, Specification d, is acceptable.

TS 4.3, Specification e, requires the licensee to sample and analyze the SCS for gross radioactivity quarterly to help ensure the detection of PCS or PoolCS to SCS leak. The NRC staff finds the surveillance interval for TS 4.3, Specification e, is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 4.3, Specification e, is acceptable.

TS 4.3, Specification f, requires the licensee to measure the PCS conductivity and pH monthly. The NRC staff finds that TS 4.3, Specification f, helps to maintain a suitable chemical environment for the use of core components. The surveillance interval is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.3, Specification f, is acceptable.

TS 4.3, Specification g, requires the licensee to measure the conductivity of the PoolCS water monthly. The NRC staff finds that TS 4.3, Specification g, helps to maintain a suitable chemical environment for the use of core components. The NRC staff also finds that the surveillance interval is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.3, Specification g, is acceptable.

TS 4.3, Specification h, requires the licensee to test the operability of the PCS relief valves biennially, with at least one of the valves tested on an annual basis, in order to help ensure the valves are able to protect the PCS from an overpressure condition. As describes in SAR Section 3.1.5, the primary coolant system is protected from overpressure by relief valves installed on the pressurizer and the primary coolant piping. The relief valves are set lower than the TS 3.5, Specification b, limit of 110 psig, thus providing a sufficient margin to ensure that the primary coolant system design pressure of 125 psig will not be exceeded. The NRC staff finds that the biennially test period, with each valve being tested in alternate years, is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.3, Specification h, is acceptable.

The NRC staff finds that TS 4.3, Specifications b through h, are consistent with the analysis presented in SAR Chapter 13 and the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.3, Specifications b through h, are acceptable.

TS 4.9 Auxiliary Systems

TS 4.9, Specification a, states:

Specification:

a. The operability of the primary coolant make-up water system shall be tested on a semiannual basis.

(…)

TS 4.9, Specification a, requires the licensee to test the operability of the PCMWS semiannually. The NRC staff finds that the surveillance interval is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 4.9, Specification a, is acceptable.

The NRC staff reviewed the MURR coolant systems TSs and concludes the following:

- The PCS is designed in accordance with the design bases derived from the analyses in the SAR.

- Design features of the PCS and components provide assurance of fuel integrity under all possible reactor conditions. The system is designed to remove sufficient fission heat from the fuel to allow all licensed operations without exceeding the established LSSSs that are included in the TS.

- Designs and locations of PCS components have been specifically selected to avoid coolant loss that could lead to fuel failure, uncontrolled release of excessive radioactivity, or damage to safety systems or experiments.

- The chemical quality of the primary coolant will limit corrosion of the fuel cladding and coolant system components for the duration of the license and for the projected utilization time of the fuel.

- The TSs, including testing and surveillance requirements, provide reasonable assurance of necessary PCS operability for reactor operations as analyzed in the SAR.

- The design bases of the PCS provide reasonable assurance that radioactive materials will be contained, and the environment and public health and safety will be protected.

- The licensee has demonstrated through an analysis in SAR Section 13.2.3 and in its response to RAI No. 13.4.b (Ref. 27) that the transition into natural circulation after flow reversal through the core during a LOCA or loss of flow accident will not compromise fuel integrity.

The NRC staff finds that continued operation during the renewal period, in accordance with the TSs provides reasonable assurance that the PCS, PoolCS, SCS, and their support systems can perform all the intended functions as described in the SAR. Based on the information above, the NRC staff concludes that these MURR systems are acceptable.

5.3 Nitrogen-16 Control System

SAR Section 5.7 describes the nitrogen-16 (N-16) control system, which states that radiation exposure from N-16 is mitigated by the use of holdup tanks in both the PCS and PoolCS before coolant enters the demineralizer system. Other parts of the PCS and PoolCS with the potential for N-16 are in shielded areas or shielded areas with limited access during reactor operation. The SAR also indicates that the primary holdup tank has a 100-gal (378-L) capacity and is constructed of aluminum with a design pressure of 150 pounds per square inch, gauge. The flowpath within the tank is composed of several tortuous turns around a set of five alternating aluminum baffles. In its response to RAI No. 5.6 (Ref. 18), the licensee states that the only credible system failure is a leak in either the holdup tank, which is essentially no different than a leak anywhere else in the PCS, or the PoolCS, as analyzed in SAR Chapter 13.

The NRC staff finds that the N-16 control system, as described in the SAR, appears effective to promote a flowpath that provides a delay of approximately 2 minutes, which allows additional time for the N-16 (7-second half-life) to decay before reaching the demineralizer system, and helps to minimize personnel exposure from the decay of N-16. Based on the information above, the NRC staff concludes that the MURR N-16 control system is acceptable.

5.4 Decay Heat Removal System

SAR Section 5.8 describes the decay heat removal system, which consists of an in-pool HX, automatic isolation valves, and associated piping. The system is designed to remove decay heat load generated following a reactor scram after 30 days of continuous 10 MWt operation, with a primary coolant inlet coolant temperature of 10 °F (60 °C), and a pool temperature of 100 °F (38 °C) without the net formation of steam in the PCS. In its response to RAI No. AA22 (Ref. 103), the licensee clarified the SAR description and states that the 30 days of continuous operation represented an upper limit for the design of the decay heat removal system because MURR never operates greater than 7 days. The NRC staff finds that the design-estimated heat load of 30 days of continuous operation is conservative because MURR can only operate for 6 days before needing to shut down to refuel. In SAR Section 5.8, the licensee also states that removal of decay heat is achieved by the automatic opening of two 6-in (15-cm) isolation valves (V546A and V546B) following a low PCS flow or pressure signal, thus allowing the PCS coolant to enter the in-pool HX. These valves may also be opened manually from the reactor bridge assembly. Primary coolant flow undergoes a flow reversal in the core and reactor core cooling will start by natural circulation. Heat from the core will be transferred to the pool through the pressure vessel walls and the in-pool HX.

The NRC staff reviewed the MURR decay heat removal system and finds it acceptable to provide decay heat removal in accordance with the design description from SAR Section 5.8, as supplemented by the licensee's RAI response.

TS 5.2 Reactor Coolant Systems

TS 5.2, Specifications j and k, state:

Specification:

(…)

j. The reactor shall have a decay heat removal system.

k. The primary coolant system shall contain at least two (2) operable pressure relief valves.

TS 5.2, Specification j, requires a decay heat removal system, as referenced in SAR Section 5.8. This NRC staff finds that TS 5.2, Specification j, is consistent with the analysis in SAR Section 13.2.9.3, which assumes that the isolation valves fail to open, and the decay heat removal system is required to support reactor core cooling. Based on the information above, the NRC staff concludes that TS 5.2, Specification j, is acceptable.

TS 5.2, Specification k, establishes the requirement to have two operable pressure relief valves on the PCS, as discussed in SAR Sections 3.1.5 and 13.2.9.4 and depicted in SAR Figure 5.1. The NRC staff finds that TS 5.2, Specification k, is consistent with the criterion in SAR

Section 3.1.5 and the analysis in SAR Section 13.2.9.4, and helps to ensure that a flow path is available for PCS pressure relief. Based on the information above, the NRC staff concludes that TS 5.2, Specification k, is acceptable.

The NRC staff has reviewed the design of the decay heat removal system and finds that the decay heat removal system is capable of remove heat load and PCS pressure relief valves and finds that TS 5.2, Specifications j and k, are consistent with MURR design objectives and the safety analysis. Based on the information above, the NRC staff concludes that TS 5.2, Specifications j and k, are acceptable.

5.5 Conclusions

The NRC staff reviewed the design of the MURR cooling systems, as described in SAR Chapter 5 and in responses to RAIs, and finds that the PCS, SCS, and PoolCS have sufficient capacity for the removal of heat generated during continuous full-power reactor operation. The NRC staff also finds that the systems contain sufficient features to minimize corrosion of components and fuel, prevent or detect losses of coolant, and provide one of the barriers to prevent fission product release to the environment. The NRC staff concludes the following:

- The licensee described and analyzed the MURR coolant systems, has derived the design bases from other chapters of the SAR, and provided acceptable methods to remove sufficient heat to ensure the integrity of the components.

- TSs, including design, testing and surveillance requirements as required by 10 CFR 50.36, provide reasonable assurance of necessary auxiliary cooling system operability for all modes of operation.

Based on the information above, the NRC staff concludes that the MURR coolant systems are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007, and sufficient for continued reactor operation during the renewal period.

6 ENGINEERED SAFETY FEATURES

6.1 Summary Description

Chapter 6 of the safety analysis report (SAR) describes the engineered safety features (ESFs) credited with mitigating the consequences of an accident and with helping to maintain any potential radiological doses below the limits allowed in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, "Standards for Protection against Radiation." The two systems designated ESFs at the University of Missouri-Columbia Research Reactor (MURR or the reactor) are the containment system and the anti-siphon system. In the event of an accident, the containment system would mitigate any inadvertent release of radioactivity to the environment. The anti-siphon system helps to ensure that the core will not become uncovered during a loss-of-coolant accident (LOCA). The SAR description indicates that redundancy is incorporated in both systems to ensure that no single failure of either ESF will cause any part of the containment system and the anti-siphon system to become inoperative.

6.2 Containment System

In SAR Chapter 5, the licensee states that the MURR containment system consists of several components. The primary passive component is the reactor containment building (RCB) and associated penetrations. The active components include the following:

- utility entry water seal
- primary and backup plenum doors
- hot exhaust line isolation valves
- sealing gaskets
- pressurized air supplies

SAR Section 6.2.2.2 states that the RCB is a five-level, poured-concrete structure with an internal volume of 240,000 cubic feet (6,796 cubic meters). The concrete walls of the RCB have been designed to withstand peak internal pressure of a value of 2.0 pounds per square inch gauge (psig) (13.8 kilopascal (kPa) above atmosphere). The maximum temperature of the reactor coolant is given as 160 degrees Fahrenheit (°F) (71 degrees Celsius (°C)), which if released into the RCB would not lead to a pressure build up near 2.0 psig. Additionally, the licensee calculated that the 0.3-meter (m) -diameter (1-foot (ft) -diameter)) column of water that is located above the pressure vessel can absorb 108 megawatt-seconds of energy before reaching boiling. This indicates that any large positive reactivity insertion that may lead to an uncontrolled power excursion and a pressure increase that could rupture the primary coolant system (PCS) within the pool and cause an increase in the RCB pressure, which would quickly be quenched by the pool water. The licensee evaluated a severe reactor meltdown causing an aluminum water reaction, with a conservative assumption of 1.3 percent of the aluminum reacting with water, and there was not enough energy released to exceed the 2.0-psig (13.8-kPa) design pressure on the reactor containment.

SAR Section 6.2.3.1 describes the utility entry water seal (seal trench). It is a water-filled trap that provides 2.0-psig (13.8-kPa (above atmosphere)) overpressure relief protection for the RCB. SAR Section 6.2.3 describes the primary containment isolation features shown in SAR

Table 6-1, which consists of personnel entry, equipment entry, and ventilation ducting doors, and pneumatic tubing, and electrical cable penetrations. SAR Section 6.2.3.4 describes the sliding doors as being electrically driven and horizontally operated. Door 504 isolates the supply, and Door 505 isolates the exhaust ventilation ducting. The SAR states that when a door is in the fully closed position, a rotary limit switch energizes a solenoid-operated three-way valve, inflating a gasket mounted in the door facing and thus sealing the door. Actuation of the reactor isolation or facility evacuation switches located in the reactor control room or the facility evacuation switch located in the facility lobby (Room 202) will close Door 504 and Door 505. A radiation level greater than the setpoint of either of the reactor bridge radiation monitors or either of the exhaust plenum radiation monitors will also close both doors automatically.

SAR 6.2.3.8 describes the hot exhaust line, which is a 16-inch (in) (40.64-Centimeter (cm)) pipe which discharges potentially contaminated gases from the reactor containment building. Exhaust air from areas which produce radioactive gases or airborne contamination is ducted to this 16-in (40.64-cm) line which penetrates the west wall of the containment building just below the ceiling level and discharges to the facility exhaust plenum located in the west tower. The hot exhaust line has isolation valve 16A, which is an air-operated-to-open, spring-to-close, butterfly valve, and isolation valve 16B is an air-operated-to-open, air-operated-to-close, butterfly valve.

SAR Section 6.2.4 describes a second set of isolation doors, designated the backup doors, which are located in the reactor containment building supply and exhaust plenums, thereby providing redundancy for containment building isolation. Each door is held open against gravity by a double-acting pneumatic cylinder. When the doors are shut, the steel plenum chamber above the door becomes part of the containment system. Air is supplied to the pneumatic cylinders from the facility main air compressors and the emergency air compressor. A 0.375-in (0.95-cm) solid rubber gasket, which is installed in the door facing, creates a seal for the backup doors when in the closed position. The backup doors are normally kept open during reactor operation. A radiation level greater than the setpoint of either of the reactor bridge radiation monitors or either of the exhaust plenum radiation monitors will close both isolation doors automatically. Two solenoid-operated valves, installed in series, control the air supply to each pneumatic cylinder. A closure signal will de-energize both solenoid valves, causing air to be vented from the pneumatic cylinder and allowing gravity to close the isolation door. Actuation of either of the solenoid-operated valves will close the backup door.

SAR Section 6.2.5 describes the importance of compressed air. To ensure proper operation, all RCB closures with inflatable gaskets, air valves and pneumatic cylinders are dependent on a continuous supply of compressed air. These include the following:

- personnel airlock Door 276 sealing gasket
- personnel airlock Door 277 sealing gasket
- motorized ventilation isolation Door 504 sealing gasket
- motorized ventilation isolation Door 505 sealing gasket
- truck entry Door 101 seating gasket
- ventilation exhaust valve I6A actuator (keep open only)

- ventilation exhaust valve I6B actuator

- backup doors pneumatic cylinders

SAR Section 6.2.5 describes the requirements of the compressed air system, which supplies compressed air into equipment inside the containment structure through a pipe in the seal trench. Should the main air system become inoperative, the emergency air compressor provides an alternate supply of compressed air. In addition, SAR Section 8.2.4 indicates that the emergency electrical power system provides power to the compressor in the emergency air system.

SAR Section 9.14 describes the four interconnected compressed air systems at the MURR facility, which are:

- The Main Air System supplies the majority of the compressed air needs of MURR

- The Emergency Air System supplies compressed air to isolation valves and sealable closures of the RCB if the Main Air System becomes inoperable

- The Valve Operation Air System supplies compressed air as a backup to the Main Air System

- The Instrument Air System supplies compressed air to the pneumatically-operated temperature and humidity controls of the facility heating, ventilation, and air-conditioning systems; it may be isolated so that compressed HVAC air from the Main Air System can be utilized.

The Emergency Air System provides compressed air to components of the Reactor Building Containment System. The other three systems provide compressed air to various locations throughout the MURR facility including components of the ventilation system.

In its response to Request for Additional Information (RAI) No. 9.6 (Ref. 17), the licensee states that, in the event of a failure of the compressors or other component within the Main Air System, each subsystem that uses the Main Air System, is isolated from the Main Air System through in-line check valves and has a backup compressor to provide service for that subsystem. The emergency air system provides air to operate the hot exhaust line isolation valves and backup doors pneumatic cylinders and to supply the inflatable gaskets that maintain the RCB closures. Included in this system is a backup air compressor and associated check valve to prevent subsystem failure if the main air system is compromised. The hot exhaust line isolation valves are in series and are of different types. Valve 16A is a spring-to-close and air-to-open valve that is designed to be fail safe, whereas valve 16B is an air-to-open and air-to-close valve. Valve 16B has an additional independent compressor in the event that all two previous compressors fail. In the event that all air pressure is lost, valve 16A would spring to close and isolate containment from the hot exhaust system. Air would also be lost to all of the inflatable seal gaskets and cause a breach in containment integrity. However, it would take an additional failure of the primary system coolant boundary and a fuel cladding failure before fission products could be released. A failure of the containment system must occur simultaneously with another event for the system to fail to perform its function.

Single-Failure Analysis

SAR Section 6.2.7 describes the single-failure analysis provided by the licensee. The licensee states that it analyzed the active components and found that no single failure of an active component will lead to loss of containment. In response to RAI No. 9.6 (Ref. 17), the licensee indicates that if all compressed air is lost, it could result in possible breach in containment. The licensee states that there are solenoid controllers on each of the containment door gasket seals. These solenoids are checked for operability every 4 hours while operating. The most likely failure of consequence would be a failure of a single solenoid, or the failure of the line connecting the solenoid to the gasket seal, which results in the failure of a single gasket seal. As such, a gasket seal failure could disable the sealing function of Door 276, Door 277, Door 504, or Door 505. The closure of backup doors that use a passive rubber seal mitigates the loss of a gasket seal for Door 504 and Door 505. Door 276 and Door 277 do not have backup doors; however, these doors are redundant because they both service either end of the personnel access tunnel; therefore, the failure of one would not affect the operability of the other. However, personnel would need to break containment to exit the RCB.

SAR Table 6-1 lists all of the penetrations in the containment and their corresponding sealing methods. The utility entry uses a water seal; others use a door and inflatable or passive gasket or a valve. The electrical lines use sealed connectors.

Also, technical Specification (TS) 5.5, Specifications c, sets a maximum RCB leak rate and requires leak rate tests to be done on the containment structure and systems to verify the containment function.

The NRC staff reviewed the containment system and finds that its design is consistent with the guidance in Section 3.1 of NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,” issued February 1996 (Ref. 51), which states that the single failure of any active component will not prevent safe reactor shutdown or result in unsafe conditions. Based on the information provided above, the NRC staff finds that a single failure of the containment system would not lead to either personnel or public exposure greater than that analyzed by the licensee in the Maximum Hypothetical Accident in SAR Chapter 13.

TS 5.5 Reactor Containment Building

TS 5.5, Specifications a through c, state:

Specification:

The reactor containment building is a five-level, poured-concrete structure with 12-inch thick reinforced exterior walls configured to form the shape of a cube, with each side being approximately 60 feet long. Below grade within the containment structure is a space extending to the north that is 15 feet high by 37 feet deep by 40 feet wide. The following design features apply to the MURR reactor containment building:

a. The reactor and fuel storage facilities shall be enclosed in a containment building with a free volume of at least 225,000 cubic feet.

b. Whenever reactor containment integrity, as defined by Specification 3.4.a, is required, containment building ventilation exhaust shall be discharged at a minimum of 55 feet above containment building grade level.

c. The containment building leakage rate shall not exceed 16.3 cubic feet per minute at STP with an overpressure of one pound per square inch gauge or 10% of the contained volume over a 24-hour period from an initial overpressure of two pounds per square inch gauge. The test shall be performed by the make-up flow, pressure decay, or reference volume techniques.

(…)

TS 5.5 states that the reactor containment building is a five-level, poured-concrete structure with 12 inch thick reinforced exterior walls configured to form the shape of a cube, with each side being approximately 60 feet long. Below grade within the containment structure is a space extending to the north that is 15 feet high by 37 feet deep by 40 feet wide. The NRC staff finds that this specification is consistent with the description provided in SAR Section 6.2.2.1, and thus acceptable.

TS 5.5, Specification a, requires that the reactor and fuel storage facilities be enclosed in a containment building with a free volume of at least 225,000 cubic feet. The NRC staff finds that this minimum allowable value was used in the safety analysis to determine the concentrations of the materials released during postulated accidents, which are then used in the calculation of dose assessments for occupational workers and the public. The NRC staff also finds that TS 5.5, Specification a, helps to ensure that design volume of the RCB is maintained consistent with the safety analysis. Based on the information above, the NRC staff concludes that TS 5.5, Specification a, is acceptable.

TS 5.5, Specification b, requires that, whenever reactor containment integrity is required, as defined by Specification 3.4.a, the containment building ventilation exhaust be discharged at a minimum of 55 feet above containment building grade level. The NRC staff finds that this specification is consistent with description of the elevated stack in SAR Section 9.1.2.2, and the elevated release height was used as an assumption in the calculation of doses for occupational workers and the public in SER Chapter 13. Based on the information above, the NRC staff concludes that TS 5.5, Specification b, is acceptable.

TS 5.5, Specification c, requires that the containment building leakage rate not exceed 16.3 cubic feet per minute at standard temperature and pressure (STP) with an overpressure of one pound per square inch gauge or 10% of the contained volume over a 24 hour period from an initial overpressure of two pounds per square inch gauge (psig). The test must be performed by the make-up flow, pressure decay, or reference volume techniques. The NRC staff finds that this leak rate was used in the calculation of dose assessments for occupational workers and the public for some of the postulated accidents in SAR Chapter 13. The NRC staff also finds that the actual measured leak rate is established from performing an annual integrated leak rate test (pressure decay) in accordance with TS 4.4, Specification a. Based on the information above, the NRC staff concludes that TS 5.5, Specification c, is acceptable.

The NRC staff reviewed TS 5.5, Specifications a, b, and c, and finds that TS 5.5, Specifications a, b, and c, describe containment system design features, such as building volume, elevated release height, and leakage rate, that are relied upon in the safety analyses for the evaluations of postulated accidents. Based on the information above, the NRC staff concludes that TS 5.5, Specifications a, b, and c, are acceptable.

In its response to RAI No. 6.1 (Ref. 17), the licensee indicated that there are four radiation detectors associated with the containment building isolation system. Table 6-1 below lists the detector location, the current setpoint, and the basis for that setpoint. When the licensee conducts controlled evolutions in the pool area, the “Reactor Pool Upper Bridge” detector may be bypassed to prevent spurious activation.

Table 6-1 Radiation Monitors Required by TS 3.7

Detector Location	Current Setpoint	Setpoint Basis
Containment Building Exhaust Plenum No. 1	3.0 mR/hr	10 times the normal operating background
Reactor Pool Upper Bridge—ALARA	10,000 mR/hr	As determined by the Health Physics Branch
Reactor Pool Upper Bridge	50 mR/hr	10 times the normal operating background
Containment Building Exhaust Plenum No. 2	3.0 mR/hr	10 times the normal operating background

TS 3.4 Reactor Containment Building

TS 3.4 states:

Specification:

a. For reactor containment integrity to exist, the following conditions shall be satisfied:

(1) The truck entry door is closed and sealed;

(2) The utility entry seal trench is filled with water to a depth required to maintain a minimum water seal of 4.25 feet;

(3) All of the reactor containment building ventilation system’s automatically-closing doors and automatically-closing valves are operable or placed in the closed position;

(4) The reactor mechanical equipment room ventilation exhaust system, including the particulate and halogen filters, is operating;

(5) The personnel airlock is operable (one door shut and sealed);

(6) The reactor containment building is at a negative pressure of at least 0.25 inches of water with respect to the surrounding areas; and

(7) The most recent reactor containment building leakage rate test was satisfactory.

b. Reactor containment integrity shall be maintained at all times except when:

(1) The reactor is secured,

AND

(2) No movement of irradiated fuel with a decay time of less than sixty (60) days or experiments with the potential for a significant release of airborne radioactivity outside of containers, systems, or storage areas,

AND

(3) No movement of experiments that could cause a change of total worth greater than 0.0074 $\Delta k/k$.

c. When reactor containment integrity is required, the reactor containment building shall be automatically isolated if the activity in the ventilation exhaust plenum or at the reactor bridge indicates an increase of 10 times above previously established levels at the same operating condition. Exception: The containment isolation set point may temporarily be increased to avoid an inadvertent scram and isolation during controlled evolutions such as experiment transfers or minor maintenance in the reactor pool area. The pool area shall be continuously monitored, and, if necessary, a manual containment isolation actuated, until the automatic set point is reset to its normal value.

TS 3.4, Specification a, requires conditions for the establishment of the RCB containment integrity. These conditions are listed in TS 3.4, Specifications a.(1) through a.(7). The NRC staff finds that TS 3.4, Specifications a.(1) through a.(7) help ensure that facility equipment that is required to maintain containment integrity are identified and that their required status or condition is established. In addition, the NRC staff finds the negative pressure difference to the ambient atmospheric pressure of 0.25 in (0.64 cm) of water equivalent required on the RCB also helps ensure proper containment integrity is maintained. The NRC staff finds TS 3.4, Specifications a.(1) through a.(7) provide conditions essential to maintaining containment integrity as used in the assumptions in the accident analyses in SER Chapter 13. Based on the information above, the NRC staff concludes that TS 3.4, Specifications a.(1) through a.(7), are acceptable.

TS 3.4, Specifications b.(1) through b.(3) provide the exceptions for containment integrity. In TS 3.4 basis, the licensee states that containment integrity is required during any operational activity (reactor not secured), or with the movement of irradiated fuel with a decay time of less than 60 days, or experiments with the potential for a significant release of airborne radioactivity outside of containers, systems, or storage areas, or with the movement of an experiment that could cause a change of total reactivity worth greater than 0.0074 $\Delta k/k$. The NRC staff finds that TS 3.4, Specifications b.(1) through b.(3) help to ensure that the conditions required to allow the removal of containment integrity are properly delineated. In its response to RAI No. A.6 (Ref. 23), RAI No. A.27 (Ref. 18), and RAI No. 7.a (Ref. 33), the licensee provides radiological dose estimates for the "60-Minute Dose from Radioiodine and Noble Gases in Containment" scenario as 1.09 mrem to a worker who remained in the reactor bay for one hour. The NRC staff reviewed the dose calculations and finds that the dose estimates were performed correctly and the resulting dose demonstrates that the 60-day decay criteria is an acceptable basis for TS 4.3, Specification b.(1). Based on the information above, the NRC staff concludes that TS 3.4, Specifications b.(1) through b.(3), are acceptable.

TS 3.4, Specification c, specifies additional requirements for containment integrity. As described in SAR Section 6.2.3.8, the MURR containment has two operational modes: (1) normal operation under which the containment provides a slight negative pressure; and (2) an isolation mode that involves de-energization of the containment ventilation air supply and return fans, and isolation of ventilation plenum doors and quick closing valves 16A and 16B. The NRC staff finds that the containment isolation mode is a functional configuration associated with accident conditions. The containment can have integrity and be in either of these conditions. RCB isolation is initiated when radiation activity in the exhaust plenum or at the reactor bridge increases by a factor of 10 above established levels or the reactor operator manually isolates the RCB. The TS exception in TS 3.4, Specification c, applies to the activities that are known to cause increases in measured activity such as experiment transfers and minor maintenance. The NRC staff also finds that TS 3.4, Specification c, helps to ensure that the appropriate operating conditions that require containment integrity are appropriately described and that the TS exception, which allows a setpoint increase during minor maintenance or experiment transfer, is useful to avoid an unnecessary scram and containment isolation. Based on the information above, the NRC staff concludes that TS 3.4, Specification c, is acceptable.

The NRC staff finds that TS 3.4, Specifications a, b, and c, help to ensure that potential releases of radioactive material in the event of fuel damage or experiment failure are contained within the containment building, and any potential radioactive releases are consistent with the accident analysis in SAR Chapter 13. Based on the information above, the NRC staff concludes that TS 3.4, Specifications a, b and c, are acceptable.

TS 4.4 Reactor Containment Building

TS 4.4 states:

Specification:

a. The reactor containment building leakage rate shall be measured annually, plus or minus four (4) months. The test shall be performed by the make-up flow, pressure decay, or reference volume techniques. No repairs or modifications shall be performed just prior to the test.

b. The reactor containment building leakage rate shall be measured following any modification or repair that could affect the leak-tightness of the building.

c. The containment actuation (reactor isolation) system, including each of its radiation monitors, shall be tested for operability at monthly intervals.

d. When required by Specification 3.4.b, containment integrity shall be verified to exist within a shift.

TS 4.4, Specification a, requires measurement of the RCB leak rate annually, plus or minus four (4) months. The test shall be performed by the make-up flow, pressure decay, or reference volume techniques. No repairs or modifications shall be performed just prior to the test. The NRC staff finds that TS 4.4, Specification a, helps ensure that the RCB leak rate is measured annually, by make-up flow, pressure decay, or reference volume techniques, and prior to any maintenance. In its response to RAI No. 1 (Ref. 15), the licensee provides the results of an independent assessment report entitled, “Containment Structure Condition Assessment”

(Ref. 91). This report, prepared by the Sargent & Lundy engineering firm in 2001 for the licensee, concludes that the measured leak rate was well within the allowable leak rate (the actual value was not reported) and provides recommendations for correcting the noted leaks. The report also states that the MURR RCB was structurally adequate to resist the expected operating-basis earthquake and safe-shutdown earthquake events and, with continued maintenance, would continue to provide acceptable service. The NRC staff finds the report detailed the recommended maintenance needs to support the RCB through the license renewal period, and finds it acceptable (Ref. 91). The NRC staff also finds that the TS 4.4, Specification a, surveillance interval is consistent with the guidance in NUREG-1537 and American National Standards Institute/American Nuclear Society (ANSI/ANS)-15.1-2007, "The Development of Technical Specifications for Research Reactors," issued 2007 (Ref. 57). Based on the information above, the NRC staff concludes that TS 4.4, Specification a, is acceptable.

TS 4.4, Specification b, requires a measurement of the RCB leakage rate following any modification or repair that could affect the leak-tightness of the RCB. The NRC staff finds that this specification helps ensure the integrity of the RCB, and is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.4, Specification b, is acceptable.

TS 4.4, Specification c, requires the licensee to test the operability of the containment actuation (RCB isolation) and system radiation monitors that initiate RCB isolation monthly. The NRC staff finds that TS 4.4, Specification c, helps ensure that the containment isolation system is capable of performing one of the intended functions, which is RCB isolation. The NRC staff finds that this TS 4.4, Specification c, helps ensure that the containment integrity conditions assumed in the safety analysis described SER Chapter 13 are maintained, and the surveillance interval is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 3.4, Specification b, is acceptable.

TS 4.4, Specification d, requires the licensee to verify the containment integrity within a shift, as required by TS 3.4, Specification b. The NRC staff finds that TS 4.4, Specification d helps demonstrate that the RCB containment is performing its design function, and the surveillance interval is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.4, Specification d, is acceptable.

The NRC staff reviewed TS 4.4, Specifications a through d, and finds that TS 4.4, Specifications a through d, help ensure that integrity of the RCB is maintained, except for the exceptions noted in the specifications. Based on the information above, the NRC staff concludes that TS 4.4, Specifications a through d, are acceptable.

In its response to RAI No. 9.6 (Ref. 17), the licensee indicates that the supply and exhaust air doors close automatically when the reactor evacuation switch in the control room lobby is triggered or when the reactor bridge radiation monitor detects radiation levels greater than the setpoint. The reactor operator can also manually isolate the RCB. Isolation valves on the hot exhaust lines close within 3 seconds of being actuated. The air supply used to close the valve comes from three different independent supply sources to provide redundancy. The doors with inflatable gaskets rely on compressed air to ensure the seal. Air supply enters the containment through the seal trench and stays above 70 psig (482.6 kPa above atmosphere), with an emergency backup air supply line near the seal trench. The air supply system remains isolated from the electrical system in case of a loss of normal electrical power. The NRC staff finds that the isolation features of the RCB containment system are effective to isolate the containment and are not vulnerable to a loss of isolation caused by a single failure of any containment isolation equipment.

The NRC staff reviewed the testing, surveillance provisions, intervals, and related TSs and concludes that the confinement ESF will be available and operable when required for mitigating accident consequences, and the applicable TSs meet 10 CFR 50.36.

6.3 Anti-siphon System

As described in SAR Section 6.3, the anti-siphon system consists of a pressure tank, two automatic isolation valves, a level controller, and associated piping and valves. The system functions as a backup in case of a LOCA for the various safety instrumentation and equipment (e.g., pressure sensors, pump and valve interlocks, and other such instrumentation and equipment) to ensure that the reactor core does not become uncovered. The design criteria helps to ensure that enough air is available to break the siphon should a double-ended PCS pipe rupture occur and to minimize air introduced into the PCS should a leak-by of the isolation valve occur. This system is maintained by keeping the anti-siphon system pressure above 27 psig (207 kPa) in accordance with TS 3.5, Specification b.(2), discussed in SER Section 7.2.

TS 5.2 Reactor Coolant System

TS 5.2, Specification h, states:

Specification:

(…)

h. The primary coolant system shall have two (2) anti-siphon isolation valves.

(…)

TS 5.2, Specification h, requires that the PCS to have anti-siphon valves. In its response to RAI No. 6.2 (Ref. 27), the licensee states that the thermal-hydraulic analysis that it performed using RELAP5 specifically includes components to model the anti-siphon valves. The licensee states that the use of either value will ensure effective operation of the anti-siphon system, and provides for redundancy to ensure proper system operation even if a failure of one of the valves occurred. The NRC staff finds that this specification helps to ensure that the reactor core will not become uncovered during a LOCA, and that the decay heat removal capacity of the PoolCS is maintained. The NRC staff finds that the presence of these valves is consistent with assumptions used in the LOCA analysis described in the SAR. Based on the information above, the NRC staff concludes that TS 5.2, Specification h, is acceptable.

TS 4.3 Reactor Coolant Systems

TS 4.3, Specification a, states:

Specification:

a. The following components or systems shall be tested for operability at monthly intervals except during extended shutdown periods when the valves shall be tested prior to reactor operation:

(1) Anti-siphon system;
(2) Primary coolant isolation valves V507A/B; and
(3) In-pool convective cooling system.

TS 4.3, Specification a, requires operability testing of the anti-siphon system, the PCS isolation valves, and the PoolCS convective cooling system, monthly except during extended shutdown periods when the valves shall be tested prior to reactor operation. The NRC staff finds that this specification helps to ensure that these components are tested monthly or, if shutdown, before reactor operation to ensure high confidence that the systems and components will perform their expected safety functions, as described in the SAR Chapter 6. The NRC staff also finds that TS 4.3, Specification a, helps to ensure that equipment is operable, as provided in the assumptions in the safety analysis. The NRC staff finds that the surveillance interval is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.3, Specification a, is acceptable.

The NRC staff reviewed the LOCA RELAP5 model assumptions and the response to RAI No. 6.2 (Ref. 27), which describes, the assumptions for the anti-siphon isolation valve closure times and finds that the analysis properly evaluates the anti-siphon system response. Based on its review, the NRC staff concludes that the testing frequency of anti-siphon isolation valves has shown to be adequate over the past 40 years of continuous operation.

6.4 Conclusions

The NRC staff reviewed the ESFs for MURR described in the SAR and in responses to RAIs, and concludes the following:

- The licensee identified an MHA that could lead to unacceptable radiological consequences.

- The licensee’s analysis of this MHA in SAR Chapter 13 includes assumptions regarding the operation of the MURR RCB (an ESF), whose design function is to prevent unacceptable radiological consequences.

- The facility has an anti-siphon system (an ESF) that acts as backup to equipment used to mitigate a LOCA.

- Both ESFs will control the release of radioactive material, including contaminated primary coolant. The ESFs are designed, and TS requirements, for periodic surveillance and testing to ensure ESF operability and availability.

- The functioning of the ESF, as designed, reasonably ensures that an MHA at the reactor facility will not subject the public, the environment, or the facility staff to unacceptable radiological consequences.

- The applicable TSs limiting conditions for operation, surveillance requirements and design features meet 10 CFR 50.36.

7 INSTRUMENTATION AND CONTROL SYSTEMS

Chapter 7 of the safety analysis report (SAR) describes the instrumentation and control (I&C) systems, including the design criteria and support design bases, and the functional and safety analyses of the I&C systems. The SAR also describes independent systems that monitor reactor power provide reactor protection, control, and monitoring functions.

7.1 Summary Description

SAR Section 7.1, states that the I&C Systems at MURR comprises of the sensors, electronic circuitry, displays, and actuating devices that are available to provide the information and means to safely control the reactor and avoid or mitigate potential accidents.

The Nuclear Instrumentation (NI) System continuously monitors and displays the neutron flux from the subcritical source multiplication range, through the critical range, and through the intermediate flux range to full power while also providing reactor period information. In addition, the NI System provides input signals to the Reactor Safety and Rod Control Systems.

The Rod Control System enables manual control of reactor power from source to power range levels and automatic control after a minimum power level has been attained. This system also provides the capability of placing the reactor in a subcritical condition by a rod run-in, which initiates the automatic insertion of the control blades at a controlled rate should a monitored parameter exceed a predetermined value. Inputs which govern the rod run-in system are supplied from the neutron flux monitors, process transducers, and safety interlocks. A rod run-in may also be initiated manually by the reactor operator.

The Process Instrumentation and Control System monitors, displays, and controls the following reactor plant parameters: temperature, pressure, flow, and pressurizer liquid level. The system provides input signals to the Reactor Safety System in addition to control interlocks for the primary and pool coolant system circulation pumps and automatic isolation valves.

The Reactor Safety System is designed to prevent operation of the reactor in regions in which fuel damage may occur. This is accomplished through promptly placing the reactor in a subcritical, safe shutdown condition by a reactor scram, which initiates the instantaneous drop of the control blades by interrupting power to their electromagnets should a monitored parameter exceed a predetermined value. Inputs which govern the Reactor Safety System output are supplied from the neutron flux monitors, process transducers, and safety interlocks. A reactor scram may also be initiated manually by the reactor operator.

The Engineered Safety Features Actuation Systems receive input signals from monitoring instruments and initiate the operation of the engineered safety systems which are designed to mitigate the consequences of certain identifiable accidents, thereby keeping radiological exposures to the operating staff and the general public within the limits of 10 CFR 20.

Four Radiation Monitoring Systems detect and quantify radiation and activity levels at various locations within the facility, within various reactor systems, and within the exhaust gases released to the uncontrolled environment. One of these systems provides input signals to the Engineered Safety Features Actuation System, which initiates the Containment System – an engineered safety feature which provides a complete isolation of the reactor containment building.

The I&C logic, block, and flow diagrams of the reactor I&C systems are provided as Figures 7.4 through 7.11 in SAR Chapter 7 and include the following:

- Figure 7.4, “Annunciator Control”
- Figure 7.5, “Rod Control Systems”
- Figure 7.6, “Rod Run-In System”
- Figures 7.7, 7.8, and 7.9, “Process Instrumentation Control and Interlock”
- Figure 7.10, “Reactor Safety Systems”
- Figure 7.11, “Area Radiation Monitoring System”

7.2 <u>Design of Instrumentation and Control Systems</u>

SAR Sections 7.2.1 and 7.2.2 describe the design of I&C systems. These systems are designed to do the following:

- Provide the reactor operators (ROs) with information on the operating status of the reactor and the facility.
- Provide the means to manually insert and withdraw the control blades.
- Provide for automatic control of reactor power level.
- Provide the means to insert the control blades should a monitored parameter exceed a predetermined value.
- Provide the means to detect and measure the radiation and activity levels at the reactor facility, including the release of radioactive gases from the facility.
- Provide a means to initiate the ESF.
- Provide for the storage of operational data for later retrieval.

In the SAR, the licensee states that the principal purpose of the reactor facility is to support the use of the experimental facilities. An essential feature of the reactor core design is to maximize this potential by having high neutron leakage. Control of the reactor, during all conditions of operation, is accomplished by surrounding the reactor core with a reflector region (beryllium metal followed by canned graphite), which increases the fraction of leakage neutrons that return to the core region, and by interposing a movable shroud of a material opaque to thermal neutrons (boron carbide-aluminum mixture) between the core and the beryllium reflector. No other reactor parameter, such as primary and/or pool coolant temperature and pressure, is used to change or alter reactor power other than through inherent reactivity feedback effects. In addition, the beryllium reflector effectively decouples the reactor core from reactivity effects caused by variations to the experimental facilities. The only experimental facility which is not decoupled from the reactor in this manner is the center test hole (flux trap). The licensee states that this decoupling design feature can be considered an inherent safety feature that effectively assists the control blades in controlling reactor power.

SAR Section 7.3 describes the essential displays and control equipment that enable an RO to observe and control the operation of the reactor. They are located on two cabinets (SAR Figure 7.1)—the reactor control console and the instrument panel. Analog display devices on the control console show reactor conditions (see SAR Figure 7.2), and SAR Tables 7-1 and 7-2 describe each reactor control console display instrument monitored. Data recorders, associated coolant system information and control, and area radiation and effluent monitors are located on cabinets adjacent to the control console. In addition, a 60-point annunciator, which provides the RO with an audible and visual alarm of an abnormal condition, is mounted on the upper left side of the instrument panel. Instrument panel display and control equipment locations are listed in SAR Table 7-3 and shown in SAR Figure 7.3.

SAR Section 7.3, states that the instrumentation on these cabinets is arranged in locations that incorporate human engineering factors to facilitate the safe and efficient operation of the reactor. The output instruments and the controls in the University of Missouri-Columbia Research Reactor (MURR or the reactor) control console have been designed for checking operability, inserting test signals, performing calibrations, and verifying trip settings. Control console locking devices reasonably ensure that the facility is only operated by authorized personnel. The control console and instrument panel are located within the reactor control room—a centralized operating station located on the third level of the reactor containment building (RCB).

SAR Section 7.2.2 provides the design-basis requirements for the I&C systems with respect to response time, accuracy, and continuity of operation. The MURR RSS includes all of the sensing devices, electronic circuits and equipment, signal conditioning equipment, and electromechanical devices that serve to affect a reactor shutdown by the removal of the holding current from the four control rod drive mechanism electromagnets or to activate the ESFs.

SAR Section 7.2.2, states that the channels, which provide the input signals to the RSS, are redundant. The circuitry for the nuclear instruments and the protective equipment located in the reactor control room, including the signal cables, are not physically separated; however, these channels are accessible to the reactor operator and under continuous surveillance. The signal lines from the process sensors, transmitters, controllers, and switches, located in the mechanical equipment room, provide redundant protective functions, and are separated and identified. Redundancy is incorporated into the ESF actuation systems to ensure that no single component or circuit failure will render any portion of the system inoperative.

SAR Section 7.2.3 states that the licensee has periodically updated the MURR I&C systems to take advantage of technological improvements and state-of-the-art developments, while retaining the desirable design characteristics of the original system. I&C-related technical specifications (TSs) are derived from the analysis provided in SAR Chapter 13. Systems credited to prevent or mitigate releases of radioactive material are described. The accidents analyzed in SAR Chapter 13 include a loss of coolant, loss of flow, a failed fuel element, a failed fueled experiment, and a series of reactivity events.

TS 3.2 identifies the required channels, setpoints, and interlocks to ensure proper operation of the RCS to avoid conditions that could jeopardize the integrity of the fuel element cladding or endanger personnel health and safety and to specify the minimum number of RSS instrument channels that must be operable for safe reactor operation. TS 4.2 provides the corresponding surveillance requirements (SRs).

TS 3.2 Reactor Control and Reactor Safety Systems

TS 3.2, Specifications f, g, and h, state:

Specification:

(…)

f. The reactor shall not be operated unless the following rod run-in functions are operable. Each of the rod run-in functions shall have 1/N logic where N is the number of instrument channels required for the corresponding mode of operation.

	Rod Run-In Function	Number Required (N) Mode I	Mode II	Mode III	Trip Set Point
1.	High Power Level	3	3	3	115% of full power (Max)
2.	Reactor Period	2	2	2	10 Seconds (Min)
3.	Pool Low Water Level	1	1	0	27 feet (Min)
4.	Vent Tank Low Level	1	1	0	1 foot below centerline (Min)
5.	Rod Not-In-Contact With Magnet	4	4	4	Magnet disengaged from any rod
6.	Anti-Siphon System High Level	1	1	1[(1)]	6 inches above valves (Max)
7.	Truck Entry Door	1	1	1	Loss of entry door seal pressure
8.	Regulating Blade Position	2	2[(2)]	2[(2)]	≤ 10% withdrawn or bottomed
9.	Manual Rod Run-In	1	1	1	Push button on Control Console

(1) These Instrument Channels are not required when in Mode III operation below 50 kW in natural convection cooling (natural convection flange and pressure vessel cover removed). These Instrument Channels are required when in Mode III operation with forced cooling.

(2) Not required during calibration measurements of the regulating blade.

g. The reactor safety system and the number (N) of associated instrument channels necessary to provide the following scrams shall be operable whenever the reactor is in operation. Each of the safety system functions shall have 1/N logic where N is the number of instrument channels required for the corresponding mode of operation.

	Reactor Safety System Instrument Channel	**Number Required (N) Mode I**	**Mode II**	**Mode III**	**Trip Set Point**
1.	High Power Level	3	3	3	125% of full power (Max)
2.	Reactor Period	2	2	2	8 Seconds (Min)
3.	Primary Coolant Flow	4	2	2(1)	1,625 gpm(2) (Min)
4.	Differential Pressure Across the Core	1	0	0	3,200 gpm(3) (Min)
5.	Differential Pressure Across the Core	0	1	1(1)	1,600 gpm(3) (Min)
6.	Primary Coolant Low Pressure	4	4	4(1)	75 psia(4) (Min)
7.	Reactor Inlet Water Temperature	2	1	1(1)	155 °F (Max)
8.	Reactor Outlet Water Temperature	1	1	1(1)	175 °F (Max)
9.	Pool Coolant Flow	2	2	0	850 gpm (Min)
10.	Differential Pressure Across the Reflector	1	0	0	2.52 psi (Min) 8.00 psi (Max)
11.	Differential Pressure Across the Reflector	0	1	0	0.63 psi (Min) 2.00 psi (Max)
12.	Pressurizer High Pressure	1	1	1(1)	95 psia (Max)
13.	Pressurizer Low Water Level	1	1	1(1)	16 inches below centerline (Min)
14.	Pool Low Water Level	0	0	1	23 feet (Min)
15.	Primary Coolant Isolation Valves 507A/B Off Open Position	1	1	1(1)	Either valve off open position
16.	Pool Coolant Isolation Valve 509 Off Open Position	1	1	0	Valve 509 off open position
17.	Power Level Interlock	1	1	1	Scram as a result of incorrect

					selection of operating mode
18.	Facility Evacuation	1	1	1	Scram as a result of actuating the facility evacuation system
19.	Reactor Isolation	1	1	1	Scram as a result of actuating the reactor isolation system
20.	Manual Scram	1	1	1	Push button on Control Console
21.	Center Test Hole	2(5)	2(5)	2(5)	Scram as a result of removing the center test hole removable experiment test tubes or strainer

(1) These Instrument Channels are not required when in Mode III operation below 50 kW in natural convection cooling (natural convection flange and pressure vessel cover removed). These Instrument Channels are required when in Mode III operation with forced cooling.

(2) Flow orifice ΔP (instrumentation displayed in gpm) or heat exchanger ΔP (instrumentation displayed in psi) in each operating heat exchanger leg corresponding to the flow value in the table.

(3) Core ΔP (instrumentation displayed in psi) corresponding to the core flow value in the table.

(4) Trip pressure is that which corresponds to the pressurizer pressure indicated in the table with normal primary coolant flow.

(5) Not required if reactivity worth of the center test hole removable experiment sample canister and its contents or the strainer is less than the reactivity limit of Specification 3.8.b. This safety function shall only be bypassed with specific authorization from the Reactor Manager.

h. The following reactor control interlocks shall be operable whenever the reactor is in operation.

	Interlock	Function	Minimum Numbers Operable
1.	Rod Withdrawal Prohibit	Prevents the control rods from being withdrawn unless the control system logic functions listed in the Bases have been satisfied	1
2.	Automatic Control Prohibit	Prevents placing the reactor in automatic control unless the control system logic functions listed in the Bases have been satisfied	1

TS 3.2, Specifications f.1 through f.9, prohibit the reactor operation unless the stated rod run-in functions are operable for each mode of reactor operation. SAR Section 7.5.5, states that the rod run-in system is designed to initiate the automatic insertion of the four control blades at a controlled rate should a monitored parameter exceed a predetermined value or during a scram condition. The rod run-in conditions that will initiate a rod run-in include a short period detected in Channels 2 and 3; low reactor pool level; low vent tank level; anti-siphon system high level; a regulating rod less than or equal to 10-percent withdrawn; a regulating rod bottomed; an open truck entry door; high power detected on Channels 4, 5, and 6; and a rod that is not in contact with the magnet. In its response to Request for Additional Information (RAI) No. 4.5.b (Ref. 18), the licensee states that there are no automatic circuits that could withdraw one or more control blade drives and that there are two automatic control circuits (the rod run-in and automatic shim control) that could insert the control blade drives. In its response to RAI No. A.10 (Ref. 18), the licensee clarifies that the reference to the control blades was to the shim blades and not the regulating blade, which can be inserted and withdrawn from the reactor when the reactor control system is operated in the automatic shim control mode.

The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the basis for each of the rod run-in functions in TS 3.2, Specifications f.1 through f.9, as described in SAR Section 7.5.5 and in responses to RAIs, and finds that the rod run-in system provides a response to reactor conditions in advance of the condition becoming more adverse. SAR Section 7.7 indicates that the rod run-in system is not part of the RSSs; however, it does provide a protective function by introducing shim blade insertion to terminate a transient before actuating an RSS setpoint trip. The NRC staff reviewed the rod run-in system and confirms that it is not credited in any of the safety analyses in SAR Chapter 13. The NRC staff finds that the rod run-in system is designed to mitigate the effect of an adverse reactor condition, including a scram, if necessary. Based on the information above, the NRC staff concludes that TS 3.2, Specifications f.1 through f.9, are acceptable.

TS 3.2, Specifications g.1 through g.21, require the RSSs and associated number of instrument channels to be operable to provide certain scrams during modes of operation. SAR Section 7.7, Table 7-8, provides the RSS reactor scrams for high power level, primary coolant flow, primary coolant pressure, and reactor inlet temperature which support the assumptions used in the safety analysis in SAR Chapter 13 and as the basis to support the analysis for the limiting safety system settings in TS 2.2, Specifications a, b, and c. TS 3.2, Specification g, also requires that several key critical instrument channels are correctly functioning to ensure that the reactor can be scrammed in the event of the detection of an abnormal condition such as short reactor period, low coolant flow, pressurizer high and low pressure, low pool level, high reactor power level, and facility evacuation. In its responses to RAI No. A.18 (Ref. 17), RAI No. A.19 (Ref. 17), RAI No. A.20 (Ref. 17), RAI No. A.21 (Ref. 17), RAI No. A-26 (Ref. 21), RAI No. 7.2 (Ref. 18), and RAI No. 7.7 (Ref. 24), the licensee provides additional information on the various instrument channels.

The NRC staff reviewed TS 3.2, Specifications g.1 through g.21, and finds that the instrument channels, their required operating mode, and the setpoints are consistent with the assumptions used in the analyses in SAR Chapter 13. The NRC staff also finds that the licensee's responses to RAI No. A-18 (Ref. 17), RAI No. A-19 (Ref. 17), RAI No. A-20 (Ref. 17), RAI No. A-21 (Ref. 17), RAI No. A-26 (Ref. 21), RAI No. 7.2 (Ref. 18), and RAI No. 7.7 (Ref. 24) acceptable. Furthermore, the NRC staff finds that the instrument channels are consistent with the guidance in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," issued February 1996 (Ref. 51), and American National Standards Institute/American Nuclear Society (ANSI/ANS)-15.1-2007, "The Development of Technical Specifications for Research Reactors," issued 2007 (Ref. 57). Based on its review, the NRC staff concludes that TS 3.2, Specifications g.1 through g.21, are acceptable.

TS 3.2, Specifications h.1 and h.2, require that reactor control interlocks be operable for each mode of reactor operation. SAR Sections 7.5.3.1 and 7.5.4 describe the rod withdrawal prohibit interlock, which ensures that the control blades cannot be withdrawn unless the following conditions are satisfied:

- The master control switch is in the "on" position.
- There are no nuclear instrument anomalies.
- The shim rods are bottomed and in contact with their associated electromagnets.
- The source range indicators are functioning properly and are within designated parameters.
- The thermal column door is closed.

In addition, the reactor cannot be operated in the automatic mode unless the following conditions are satisfied:

- The reactor period is greater than 35 seconds.
- The reactor power level is greater than the "auto control prohibit" setpoint.
- The regulating blade position is greater than 60-percent withdrawn.
- The range selector switch is in the 5-kilowatt (kW) red scale position or above.

The NRC staff reviewed TS 3.2, Specifications h.1 and h.2, and finds that these specifications require the reactor control interlocks be operable for each mode of reactor operation to prevent rod withdrawal and automatic control unless the interlock criteria described in SAR Sections 7.5.3.1 and 7.5.4 are satisfied and maintained. The NRC staff reviewed TS 3.2, Specifications h.1 and h.2, and finds that they are consistent with the assumptions in the safety analysis described in SAR Chapter 13. Based on the information above, the NRC staff concludes that TS 3.2, Specifications h.1 and h.2, are acceptable.

The NRC staff finds that TS 3.2, Specifications f, g, and h, help to ensure that the assumptions used in the safety analysis in SAR Chapter 13 are maintained and that the required reactor control and RSSs are operable and functioning properly. The NRC staff finds TS 3.2, Specifications f, g, and h, are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 3.2, Specifications f, g, and h, are acceptable.

TS 4.2 Reactor Control and Reactor Safety Systems

TS 4.2, Specifications h through m, state:

Specification:

(…)

h. The rod run-in functions required by Specification 3.2.f shall be channel calibrated on a semiannual basis.

i. The reactor safety system shall be channel tested before each reactor startup involving a refueling, if the facility was unsecured and unstaffed, a shutdown greater than 24 hours, or quarterly.

j. The reactor safety system instrument channels listed in Specification 3.2.g shall be channel calibrated on a semiannual basis.

k. The reactor control interlocks listed in Specification 3.2.h shall be channel calibrated on a semiannual basis.

l. A thermal power verification of power range indication, using coolant flows and differential temperatures, shall be performed weekly when the reactor is operating above 2 MW.

m. Following any modifications or repairs on any portion of the Reactor Control and Reactor Safety Systems, the modified or repaired portion of the system shall be satisfactorily tested before the system is considered operable.

TS 4.2, Specification h, helps ensure the operability of the rod run-in system in TS 3.2, Specification f, by requiring a semiannual channel calibration of the rod run-in functions described in TS 3.2, Specification f. The NRC staff reviewed TS 4.2, Specification h, and finds that the semiannual channel calibration frequency is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.2, Specification h, is acceptable.

TS 4.2, Specification i, helps ensure the operability of the RSSs by requiring that the instrument channels are channel tested before each reactor startup involving a refueling, if the facility was unsecured and unstaffed, a shutdown greater than 24 hours or quarterly. The NRC staff reviewed TS 4.2, Specification i, and finds that the channel testing frequency is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.2, Specification i, is acceptable.

TS 4.2, Specification j, helps ensure the operability of the RSSs by requiring that the instrument channels are channel calibrated on a semiannual periodicity. The NRC staff reviewed TS 4.2, Specification j, and finds that the channel calibration frequency is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.2, Specification j, is acceptable.

TS 4.2, Specification k, helps ensure the operability of the reactor control interlock by requiring that the reactor control interlocks listed in TS 3.2, Specification h, are channel calibrated on a semiannual basis. The NRC staff reviewed TS 4.2, Specification k, and finds that the channel calibration frequency provides confidence that the reactor control interlocks will operate as described in SAR Sections 6.1, 7.1, 7.5.1, 7.5.5, 7.6.1, 7.7.2.1, and 7.8.3. The NRC staff also finds that the surveillance interval stated is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.2, Specification k, is acceptable.

TS 4.2, Specification l, helps ensure the operability of the power range indication by requiring a weekly thermal power verification when the reactor is operating above 2 megawatts thermal to provide confidence to the operators that the indicated power level is accurate. The NRC staff reviewed TS 4.2, Specification l, and finds that the surveillance periodicity is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.2, Specification l, is acceptable.

TS 4.2, Specification m, helps ensure that the Reactor Control and Reactor Safety Systems are tested following any modifications or repairs. The NRC staff reviewed TS 4.2, Specification m, and finds the specification is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 4.2, Specification m, is acceptable.

The NRC staff finds that TS 4.2, Specifications h through m help to ensure that the operability of the reactor safety, interlocks, and rod run-in systems and components that help to support facility operation are consistent with the safety analysis assumptions described in SAR Chapter 13. The NRC staff also finds that the surveillance requirements (SRs) for reactor control and safety system, described in TS 4.2, Specifications h through m, are adequate, and are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.2, Specifications h through m, are acceptable.

SAR Section 7.4 describes the nuclear instrumentation system, which consists of three neutron flux monitors, a wide-range neutron flux monitor, and a multiscaler. The neutron flux monitors provide neutron flux measurements from shutdown through 100 percent of full-power operation and will scram the reactor in the event that a nuclear instrument anomaly is detected.

The wide-range neutron flux monitor provides continuous neutron flux monitoring over a 10-decade range from shutdown to 125 percent of full-rated power and will scram the reactor if any one of the following conditions occurs:

- A module is removed from the instrument drawer.

- The drawer selector switch is placed in any position other than OPERATE.
- The high-voltage power supply output drops below a predetermined minimum voltage.

The multiscaler provides the reactor operator with a continuous indication of subcritical neutron source multiplication during a reactor startup. Operability of both the source-range nuclear instrument channel and the reactor pool temperature channel is required to satisfy the safety analysis assumptions described in SAR Chapter 13.

TS 3.5 contains required channels and setpoints for reactor instrumentation to ensure that sufficient reliable information is presented to the reactor operator (RO) to ensure safe operation of the reactor. TS 4.5 provides the corresponding SRs.

TS 3.5 Reactor Instrumentation

TS 3.5, Specifications a, b, and c, state:

Specification:

a. The reactor shall not be operated unless the following instrument channels are operable:

	Instrument Channel	Minimum Numbers Operable Mode I	Mode II	Mode III
1.	Power Range Nuclear Instrument Channel	3	3	3
2.	Intermediate Range Nuclear Instrument Channel	2	2	2
3.	Source Range Nuclear Instrument Channel	1(1)	1(1)	1(1)
4.	Reactor Pool Temperature	1	1	1

(1) Required for reactor startup only.

b. Sufficient instrumentation shall be operating to assure that the following limits are not exceeded during operation:

	Parameter	Limit
1.	Primary Coolant System Pressure	110 psig (Max)
2.	Anti-Siphon System Pressure	27 psig(1) (Min)
3.	Reactor Pool Temperature	120 °F(2) (Max)

(1) Not required for Mode III operation.
(2) Reactor Pool Temperature limit is a maximum of 100 °F when in Mode III operation below 50 kW in natural convection cooling (natural convection flange and pressure vessel cover removed).

TS 3.5, Specification a, requires that the reactor shall not be operated unless the power range, intermediate range, and source-range nuclear instrument channels and the reactor pool temperature channel are operable for each mode of reactor operation. The NRC staff finds that this specification helps to ensure that the power range, intermediate range, and source-range nuclear instrument channels and the reactor pool temperature are operable for each mode of reactor operation to provide reliable information to the RO. The NRC staff also finds that TS 3.5, Specification a, is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 3.5, Specification a, is acceptable.

TS 3.5, Specification b, requires that the instrumentation for the primary coolant system (PCS) pressure, the anti-siphon system pressure, and reactor pool temperature be sufficient to ensure that the limits are not exceeded during steady-state operations for the PCS pressure (110 pounds per square inch, gauge (psig) (maximum)), the anti-siphon system pressure (27 psig (minimum)), and the reactor pool temperature (120 degrees Fahrenheit °F (49 degrees Celsius °C) (maximum)).

SAR Sections 5.2.6 and 7.6.3, describe the pressurizer system needed for operation of the reactor consistent with the assumptions used in the safety analysis in SAR Chapter 13 for reactor pressure. The NRC staff finds that the design and function of the pressurizer system for the PCS helps ensure that exposures will be maintained below the exposure limits in Title 10 of the Code of Federal Regulations (10 CFR) Part 20, "Standards for Protection against Radiation."

SAR Sections 6.3 and 7.8 state that maintaining the anti-siphon system pressure at or above 27 psig ensures that the reactor core will not become uncovered during a loss-of-coolant accident. The design and functional features of the anti-siphon system ensure that exposures will be maintained below the exposure limits in 10 CFR Part 20.

Also, SAR Chapter 13 includes the safety analysis performed for radiological consequences for a postulated failure of the anti-siphon system. Based on the safety analysis described in SAR Chapter 13, two operational limits have been imposed on the system: (1) maximum system pressure shall not exceed 45 psig, and (2) minimum system pressure shall not fall below 27 psig.

The reactor pool maximum temperature-operating limit of 120 °F (49 °C) provides an operating limit to ensure that adequate cooling is available to the reactor and fuel/pool components during all modes of operation. During natural convection operation pool temperature is limited to 100 °F (38 °C) to help ensure that the reactor is operated within the assumptions of the thermal-hydraulic analysis (see Section 2.6 of the SER).

The NRC staff reviewed the SAR and TS 3.5, Specification b, and finds that the limit established for the PCS pressure of 110 psig helps to ensure that the system design pressure of 125 psig is not exceeded, for the anti-siphon system pressure at or above 27 psig helps to ensure that the reactor core will not become uncovered during a loss-of-coolant accident, and for the maximum temperature operating limit of 120 °F (49 °C) helps to ensure that adequate cooling is available to the reactor and fuel/pool components during all modes of operation. The NRC staff also finds that TS 3.5, Specification b, is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 3.5, Specification b, is acceptable.

The NRC staff reviewed TS 3.5, Specifications a and b, and finds that TS 3.5, Specifications a and b, help ensure that assumptions used in the safety analyses in SAR Chapter 13 are maintained and are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 3.5, Specifications a and b, are acceptable.

<u>TS 4.5 Reactor Instrumentation</u>

TS 4.5 states:

<u>Specification</u>:

a. The instrument channels required by Specification 3.5.a shall be channel calibrated on a semiannual basis.

b. The instrumentation required to monitor the parameters required by Specification 3.5.b shall be channel calibrated on a semiannual basis.

c. All nuclear instrumentation channels shall be channel-tested before each reactor startup. This test shall not be required prior to a restart within two (2) hours following a normal reactor shutdown or an unplanned scram where the cause of the scram is readily determined not to involve an unsafe condition or a failure of one or more nuclear instrumentation channels.

TS 4.5, Specifications a requires the instrument channels listed in TS 3.5, Specification a to be channel calibrated on a semiannual basis. The NRC staff finds TS 4.5, Specification a helps to provide confidence that these I&C systems and components will perform their expected safety functions, as described in SAR Chapter 13, and that any instrument drift is corrected. The NRC staff also finds that TS 4.5, Specifications a is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.5, Specifications a is acceptable.

TS 4.5, Specifications b requires the instrumentation required to monitor the parameters listed in TS 3.5 Specification b to be channel calibrated on a semiannual basis. The NRC staff finds TS 4.5, Specification b helps to provide confidence that these I&C systems and components will perform their expected safety functions, as described in SAR Chapter 13, and that any instrument drift is corrected. The NRC staff also finds that TS 4.5, Specifications b is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.5, Specifications b is acceptable.

TS 4.5, Specification c, requires all instrumentation channels to be channel tested before each reactor startup unless certain conditions are met. The NRC staff reviewed TS 4.5, Specification c, and finds that this specification helps to ensure that the instrumentation channels performs their expected safety functions, as described in the SAR. The NRC staff also finds that the surveillance interval is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.5, Specification c, is acceptable.

The NRC staff finds that TS 4.5, Specifications a, b, and c, help to ensure that the instrumentation identified in TS 3.5 to support operator control of the reactor is operable and is indicating accurate values. Based on the information above, the NRC staff concludes that TS 4.5, Specifications a, b, and c, are acceptable.

TS 5.3 Reactor Core and Fuel

TS 5.3, Specification m, states:

Specification:

(…)

m. A minimum of one (1) decade of overlap shall exist between adjacent ranges of nuclear instrument channels.

TS 5.3, Specification m, requires that a minimum of 1 decade of overlap between adjacent ranges of nuclear instrument channels. The NRC staff reviewed TS 5.3, Specification m, and finds that this specification helps to ensure that the NI is capable of accurately providing indication the nuclear fission activity during all power levels. The NRC staff also finds that a minimum of 1 decade of overlap is a standard and common industry practice for the operators to ensure that the source-range, intermediate-range, and wide-range neutron instrumentation is accurately indicating neutron flux activity from the reactor core during power changes. Based on the information above, the NRC staff concludes that TS 5.3, Specification m, acceptable.

The NRC staff reviewed the design of the I&C systems, as described in SAR Chapter 7 and finds that the design bases, system description, and performance objectives properly describe the design criteria. The NRC staff finds that the TSs contain limiting conditions for operation (LCOs) and SRs for components and functions of the I&C systems, which are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007, and the LCOs and SRs intervals used at other research reactors. The NRC staff also finds that the LCOs and SRs provide reasonable assurance that I&C components will be operable when needed and that any failure or degradation will be detected in a timely manner. Furthermore, the NRC staff finds that the specified calibration frequencies are adequate to prevent significant drift in instrument setpoints and detection ranges. The NRC staff also finds that the nuclear instrumentation is in accordance with the design description in SAR Section 7.4. On this basis, the NRC staff concludes that TSs meet 10 CFR 50.36 and the design of the I&C is acceptable for continued operation at MURR during the renewal period.

7.3 Reactor Control System

SAR Chapter 7.5 describes the RCS. Control of MURR is accomplished by five neutron-absorbing control blades. Each control blade is attached to a control rod drive mechanism by means of a support and guide extension (offset mechanism). Four of the control blades, referred to as the shim blades, are used for coarse adjustments to the neutron density of the reactor core. The fifth control blade is a regulating blade used for fine adjustment of reactor power level. The four shim blades are actuated by electromechanical control rod drive mechanisms that position, hold, and scram each shim blade. The control rod drive mechanisms are mounted on the upper (operating) bridge over the reactor pool surface. Each control rod drive mechanism consists of a 0.02-horsepower (14.92-watt), 115-volt, 1-ampere, single-phase, 60-cycle motor connected to a ball-bearing lead screw assembly through a reduction gear box and overload clutch.

In the SAR, the licensee also states that the reactor power ascension begins in manual control rod withdrawal mode, with the operator controlling withdrawal of the control blades. Control blades can be withdrawn singly or ganged. Automatic control of the reactor may be selected once the reactor reaches a stable power. In automatic control, the regulating blade is adjusted based on neutron flux measurements made by a compensated ion chamber.

The following indications are displayed on the reactor control console for each shim blade:

- Power On. Power is available to the electromagnets.
- Drive Full In. The control rod drive mechanism is fully inserted.
- Drive Full Out. The control rod drive mechanism is fully withdrawn.
- Magnet Engaged. The electromagnet is engaged to the anvil.
- Rod Full In. The shim blades are fully inserted.

SAR Section 7.5 describes the control circuits that include a provision to ensure that control blades are not operated in an unsafe manner. Control blades withdrawal from a shutdown condition is limited in accordance with TS 3.2, Specification h, if there are any anomalies in the NI or if the source range meter is too high. The RCS also includes a rod run-in feature that allows the automatic insertion of control blades at a controlled rate. SAR Section 7.5.5 states the rod run-in system is designed to initiate the automatic insertion of the control blades at a controlled rate should a monitored parameter exceed a predetermined value. This system is not part of the RSS; however, it does provide a protective function by introducing shim blade insertion to terminate a transient before actuating a limiting safety system setting trip and, thereby, reducing stress imposed on the reactor systems by the scram transient. The rod run-in system also inserts the control rod drive mechanisms in the event of a scram condition. Nominal blade insertion speed is 2 inches (5.05 cm) per second for the shim blades.

SAR Section 1.7.3, provides information indicating that LA No. 14, issued April 14, 1981, authorized MURR to change the definition of "Reactor Secured" to allow the use of the dummy load test connectors. In its response to RAI No. A.9 (Ref. 21), the licensee states that the basis for the use of the dummy load test connectors was to allow the Master Control switch to be placed in the “on” position (energized) so that the licensee could perform surveillance tests on reactor systems which required electrical power. Installation of the dummy load test connectors requires electrically disconnecting the control rod drive mechanisms. This ensures that power is not available to the control rod drive mechanisms or to the control rod magnets; therefore it is impossible to withdraw the control rods. The dummy load connector has the same plug interface as the control rod drive mechanism and is wired to simulate the electrical load of the control rod drive mechanism. With the reactor secured and the dummy load test connectors installed, the Master Control Switch can be positioned in the "on" position and reactor system surveillance tests can be performed, while it is impossible to withdraw the control blades from the fully inserted position.

SAR Section 7.6 describes the RCS instruments which include sensors and controls associated with power, period, control rod position, differential pressure across the core and reflector, coolant water level, cooling system temperatures, ventilation, radiation, and experimental facilities. Sensors and processors include flux level detection and control, wide-range fission detection, linear power, pool temperature, control rod permissives, control rod interlocks, and safety trips. TSs which related to the RCS provide key safety requirements and parameters based on the safety analysis. Associated surveillances provide confidence that the I&C systems will perform when required.

The NRC staff reviewed the RCS described in SAR Chapter 7, conducted multiple site visits to observe the I&C equipment, and finds the following:

- The licensee analyzed the normal operating characteristics of the reactor facility, including thermal steady-state power levels and the planned reactor uses. The licensee also analyzed the functions of the RCS and components designed to permit and support normal reactor operations, and the RCS and its subsystems and components will give all necessary information to the operator or to automatic devices to maintain planned control for the full range of anticipated reactor operations.

- The components and devices of the RCS are designed to sense all parameters necessary for facility operation with acceptable accuracy and reliability and to transmit the information with high accuracy in a timely fashion, and control devices are designed for compatibility with the analyzed dynamic characteristics of the reactor.

- The licensee employed sufficient interlocks to limit hazards to personnel and to ensure compatibility among operating subsystems and components in the event of single isolated malfunctions.

- The RCS is designed so that any single malfunction in its components will not prevent the reactor protection systems from performing their necessary functions or will not prevent the safe shutdown of the reactor.

- The TSs, including testing, checking, and calibration provisions, surveillance tests and intervals, and design as required by 10 CFR 50.36, provide assurance that the RCS will function as designed.

Based on its review, the NRC staff concludes that the RCS is acceptable.

7.4 Reactor Safety System

SAR Section 7.7 describes the RSS. SAR Section 7.4 describes the NI system, which provides information to the RSS. The RSS consists of the electronic circuitry that can initiate the instantaneous drop of the reactor control blades (reactor scram) by interrupting power to their electromagnets should a monitored parameter exceed a predetermined value. A reactor scram may also be initiated manually by depressing a pushbutton on the reactor control console. SAR Table 7-8 lists the reactor scrams. In its response to RAI No. 7.2 (Ref. 18), the licensee clarifies the available scrams for MURR; Table 7-1 in SAR Chapter 7 provides that information.

Table 7-1 SAR and TS Scram Comparison

SAR Table 7-8 Reactor Scrams	TS 3.2, Reactor Control and Reactor Safety Systems
Manual	Manual Scram
Channel 2 & 3 Short Period	Reactor Period

SAR Table 7-8 Reactor Scrams	TS 3.2, Reactor Control and Reactor Safety Systems
Reactor Loop Low Flow	Primary Coolant Flow
Reactor Loop High Temp	Reactor Outlet Water Temperature
Pressurizer Low Level	Pressurizer Low Water Level
Reactor Pool Below Refuel Level	Pool Low Water Level
Channel 4, 5, & 6 High Power	High Power Level
Power Level Interlock	Power Level Interlock
Pressurizer High Pressure	Pressurizer High Pressure
Reflector Hi-Low Diff. Pressure	Differential Pressure Across the Reflector
Pool Valve 509 Off Open	Pool Coolant Isolation Valve 509 Off Open Position
Evacuation or Isolation	Reactor Isolation and Facility Evacuation
Reactor Loop Low Pressure	Primary Coolant Low Pressure
Low Primary HX Diff. Pressure	Primary Coolant Flow
Pool Loop Low Flow	Pool Coolant Flow
Bldg. Plenum & Bridge High Activity	Reactor Bridge and Containment Building Exhaust Plenum Radiation Monitor

SAR Section 7.4 describes the NI system, which includes three neutron flux monitors, a wide-range neutron flux monitor, and associated electronics. The neutron flux monitors provide six channels of neutron flux measurements from source level (shutdown) through 100 percent of full-power operation—one source range, two intermediate ranges, and three power ranges. The neutron flux monitors are fission chambers mounted in a watertight enclosure outside the reflector region at the approximate height of the active region of the core. The neutron flux monitors are calibrated based on their location outside the core. The wide-range neutron flux monitor is a compensated ion chamber, which is also located in a drywell at the height of the core. TS 3.2 provides requirements for the number of instrument channels, which must be operable for various parameters and different reactor modes. These parameters include power level, reactor period, differential pressure across the core and reflector, coolant flow and temperature, and reactor pool level. TS 4.2 requires surveillance frequencies for reactor control and RSS instrumentation to ensure NI operability, which the NRC staff evaluated and found acceptable in SER Section 7.2.

SAR Section 7.5.5 describes the RSS, which contains parallel non-coincidence logic circuits (NCLUs) that take input from the NI system. In the event of reactor conditions that meet set parameters, the NCLUs send signals to independent trip actuator amplifiers, which interrupt power to the electro-magnets holding the shim blades. Upon interruption of power, the shim blades drop to their lower positions and shut down the reactor. The parallel NCLUs and trip actuator amplifiers are designed and constructed so that no single failure will prevent a scram signal from shutting down the reactor.

The NRC staff reviewed the RSS as describes in SAR Chapter 7 and finds the following:

- The licensee analyzed the design and operating conditions of the RSS. The protection channels and protective responses are sufficient to ensure that no safety limit, limiting safety system setting, or RSS-related limiting conditions for operation discussed and analyzed in the SAR will be exceeded.

- The RSS design is sufficient to provide for all isolation and independence from other reactor subsystems required by SAR analyses to avoid malfunctions or failures caused by the other systems.

- The design of the RSS can be readily tested and maintained.

- The RSS is designed to maintain function or to achieve safe reactor shutdown in the event of a single random malfunction within the system.

- The RSS is designed to prevent or mitigate hazards to the reactor or escape of radiation so that the full range of normal operations poses no radiological risk to public health and safety, the facility staff, or the environment.

Based on its review, the NRC staff concludes that the RSS is acceptable.

7.5 Engineered Safety Features Actuation Systems

SAR Chapter 6 describes the ESFs for MURR, and SAR Section 7.8 describes the ESF actuation systems. ESFs are designed to prevent accidents and control the release of radioactive materials to the environment should an accident occur. At MURR, the ESF systems are the containment actuation system (CAS) and the anti-siphon actuation system.

The Containment Actuation System

SAR Section 7.8.2 describes the CAS, which is designed to completely isolate the RCB, thereby preventing or mitigating any uncontrolled release of radioactive materials to the environment during an accident. Isolation of the RCB can be automatically initiated by radiation detectors located at the reactor pool upper bridge and in the containment building exhaust plenum. Radiation detectors are independently powered and periodically tested to verify function. Isolation can be manually actuated by switches in the reactor control room or the facility lobby. Manual or automatic actuation of the CAS causes the following actions to occur:

- A reactor scram will occur.

- All normally open RCB penetrations with automatic sealable closures will close.

- An audible alarm will sound throughout the containment building.

- A flashing light at the entrance to the containment building personnel airlock will illuminate.

TS 3.4, Specification b, requires that containment integrity be maintained whenever the reactor is not secured or when recently irradiated (less than 60 days) fuel is being handled. TS 4.4

provides SRs for the CAS. The SRs help to ensure that the system will function as designed when needed. The NRC staff evaluated TS 3.4 and 4.4 and finds them acceptable in SER Chapter 6.

The Anti-siphon Actuation System

SAR Section 7.8.3 describes the anti-siphon actuation system, which functions as a backup system to the various safety instrumentation and equipment, including pressure sensors and pump and valve interlocks, to ensure that the reactor core does not become uncovered during a loss-of-coolant accident. The system is designed to admit a fixed volume of air to the high point of the reactor outlet piping, or invert loop, instantaneously establishing the pressure in this area at equal to or greater than atmosphere. This prevents a siphon action from being created as a result of a rupture of the primary coolant piping. The anti-siphon actuation system is automatically actuated upon detection of PCS low pressure. Actuation of the anti-siphon system will initiate a reactor scram to shut down the reactor. The instrumentation associated with the system is redundant so that no single failure would prevent its proper operation.

TS 3.3, Specification a, requires the anti-siphon system be operable for reactor operations in Mode I or Mode II. TS 3.5, Specification b.2, contains requirements for system pressurization. TS 4.3, Specification a, includes SRs for the anti-siphon system and provides operational testing frequencies to ensure system operability. SER Section 6.3 describes the anti-siphon system, and SER Chapter 5 evaluates and found acceptable the associated TS LCOs and SRs in TS 3.3, and TS 4.3.

The NRC staff reviewed the ESF actuation systems describes in the SAR and finds the following:

- The licensee analyzed the scenarios for all postulated accidents at the facility, including all accidents for which consequence mitigation by the ESFs is required or planned. The designs of the ESF actuation systems provides assurance of reliable operation if required.

- The design considerations of the ESF actuation systems provide reasonable assurance that these systems will detect changes in measured parameters as designed and will initiate timely actuation of the applicable ESF

- The technical specifications, including surveillance tests and intervals for the ESF actuating systems, intervals provide assurance of actuation of ESFs when required.

Based on its review, the NRC staff concludes that the ESF actuation systems are acceptable.

7.6 Control Console and Display Instruments

SAR Section 7.3 discusses the essential displays and control equipment that enables an RO to observe and control the operation of the reactor. Display and control equipment are located on two cabinets—the reactor control console and the instrument panel. SAR Figures 7.1 and 7.2 show the reactor control room layout and the reactor control console layout, respectively. SAR Figure 7.3 depicts the instrument panel layout. The control console contains the entire control blade positioning indicators and controls. The console also includes power level indications for source range, intermediate range, power level range, and wide range. The control console also houses the scram button. Control console locking devices reasonably ensure that the facility is only operated by authorized personnel. As discussed in SAR Section 7.3, windows along the west wall provide the RO with an unobstructed view of the reactor pool surface and operating (upper) bridge area. One-inch-thick steel plating is positioned and supported against the lower half of the windows to provide shielding for personnel in the control room from reactor pool background radiation. At the rear of the control room is the instrument panel. This panel contains the read outs for PCS parameters, radiation monitors, containment isolation status, and other nonreactor instrumentation. The annunciator panel is also located on the instrument panel. A 60-point annunciator, which provides the RO with an audible and visual alarm of an abnormal condition, is mounted on the upper left side of the instrument panel.

In the SAR, the licensee also states that the control console and instrument panel are located within the reactor control room, which is a centralized operating station located on the third level of the RCB. 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 finds that the designs are similar.

Based on its review of the information above, the NRC staff finds the following:

- The licensee demonstrated that all nuclear and process parameters important to safe and effective operation of MURR will be displayed at the control console. The display devices for these parameters are easily understood and readily observable by an operator positioned at the reactor controls.

- The control console design and operator interface are sufficient to promote safe reactor operation. The output instruments and the controls in the control console have been designed to provide for checking operability, inserting test signals, performing calibrations, and verifying trip settings. The availability and use of these features will ensure that the console devices and subsystems will operate as designed.

- The annunciator and alarm panels on the control console provide assurance of the operability of systems important to adequate and safe reactor operation.

- The locking system on the control console ensures that the reactor facility will not be operated by unauthorized personnel.

Based on its review, the NRC staff concludes that the control console and display instruments are acceptable.

7.7 Radiation Monitoring Systems

SAR Section 7.9 describes the MURR radiation monitoring system, which includes the ARMS, the fuel element failure monitoring system, the secondary coolant monitoring system, and the off-gas radiation monitoring system. The system is used to continuously monitor gamma-ray radiation levels at various remote locations in the reactor facility. Radiation levels are displayed on 4- or 5-decade logarithmic scale meters positioned in a centrally located control chassis. The control chassis is mounted on the reactor control room instrument panel. Table 7-10 in SAR Chapter 7 presents the locations of the remote detector assemblies and associated detection ranges.

SAR Section 7.9.2 describes the 10 area radiation monitors that provide data on gamma-radiation levels through the facility (0.1 to 10,000 mrem per hour). The readouts of these area monitors are on the same console in the control room. Each area monitor has an adjustable setpoint and is tied to a visible and audible alarm. The detectors for area radiation monitors in expected lower level radiation fields use Geiger-Mueller detectors. The detectors for monitors in potentially higher radiation fields use pressurized ion chambers. Radiation detectors at four critical locations (containment building plenum exhaust 1 and 2 and the reactor pool upper bridge on the north and south side) can initiate a reactor scram if abnormal levels of radiation are detected.

TS 3.7, Specifications a and b, require that the reactor bridge, RCB exhaust plenum, and the stack radiation monitors all be operable for reactor operation to ensure that sufficient radiation monitoring information is available to the RO during reactor operations. TS 4.7, Specifications a and b, provide SRs to ensure proper operation of the radiation monitoring instrumentation. TS 3.7 and TS 4.7 are evaluated and found acceptable in SER Chapter 11.

SAR Section 7.9.3 describes the fuel element failure monitoring system that consists of a scintillation detector adjacent to the anion resin column that continuously monitors the primary coolant for fission product activity resulting from a potential fuel element failure. The instrument channel is mounted in a rack unit on the ARMS and is equipped with an adjustable setpoint trip that initiates a "Reactor Loop Coolant Hi Activity" annunciator alarm on detection of a high-radiation level. TS 3.3, Specification b, requires the operability of the fuel element failure monitor when the reactor is operated in a forced circulation mode. TS 3.3, Specification b is evaluated and found acceptable in SER Chapter 5.

SAR Section 7.9.4 describes the MURR secondary coolant monitoring system that continuously monitors the secondary coolant system for the presence of radioactive isotopes, which could indicate a leak from the primary or pool coolant systems through their respective heat exchangers (HXs). This secondary coolant monitoring system consists of a scintillation detector that measures the gross activity of the secondary water. The output from the detector is fed through an interface box that provides signal amplification to one of the electronics channels of the ARMS where the signal is processed and displayed on a logarithmic scale analog meter in counts per minute. The instrument is mounted in a rack unit of the ARMS and is equipped with an adjustable setpoint trip that initiates a "Secondary Coolant Hi Activity" annunciator alarm on detection of a high-radiation level. The secondary coolant monitor is located in the return leg of the secondary piping, downstream of the pool and primary HXs.

SAR Section 7.9.5 describes the stack off-gas radiation monitoring system that monitors release rates in the stack exhaust. The system consists of a three-channel radiation detection system designed to measure the airborne concentrations of radioactive particulate, iodine, and noble gas in the facility exhaust air. The stack air is sampled by an isokinetic probe located in the ventilation exhaust plenum. The output from each channel is displayed on a local meter in counts per minute and on a three-pen strip chart recorder mounted in the reactor control room. An audible and visual alarm alerts the operator to high activity or abnormal airflow through the radiation detection equipment. TS 3.7, Specifications a and b, require all the stack radiation monitors to be operable for reactor operation to ensure that sufficient radiation monitoring information is available to the RO during reactor operations. TS 4.7, Specifications a and b, provide SRs to ensure proper operation of the radiation monitoring instrumentation. TS 3.7 and TS 4.7 are evaluated and found acceptable in SER Chapter 11.

The NRC staff reviewed the radiation monitoring systems as describes in the SAR and finds the following:

- The designs and operating principles of the I&C of the radiation detectors and monitors are described and shown to be applicable to the anticipated sources of radiation.

- The instrument locations are appropriate for the experiments conducted and potential upset conditions at MURR.

- The monitoring system is adequately designed to provide information about the magnitude of the radiation fields of greatest interest in the reactor building and to alert personnel to the existence of any abnormally elevated radiation fields.

- The applicant analyzed the scenarios for all postulated accidents at the facility, including all accidents for which consequence mitigation by ESFs is required or planned. The staff evaluated the radiation monitoring systems and determined that the design of their actuation systems provide reasonable assurance of reliable operation, if required.

Based on its review, the NRC staff concludes that the radiation monitoring systems are acceptable.

7.8 Conclusions

The NRC staff finds that the nuclear and nonnuclear I&C systems are adequately designed and implemented to provide safe and reliable startup, operation, and shutdown of the reactor during normal operation. Furthermore, the NRC staff finds that the RSS is adequate to protect the safety limit on fuel temperature and maintain the reactor in a state as analyzed in the accident analysis. The NRC staff finds that the ESF actuation system design is adequate to ensure that the ESFs will be activated when needed. The NRC staff also finds that the radiation monitoring system is adequately designed and that detectors are appropriately located to ensure that ROs will be appropriately warned when abnormal radiation levels are detected.

Based on the information above, the NRC staff concludes that the I&C systems at MURR and the TSs discussed above meet 10 CFR 50.36 requirements for design, LCO, and SR TSs and are acceptable for continued safe reactor operation during the renewal period.

8 ELECTRICAL POWER SYSTEMS

8.1 Normal Electrical Power Systems

Section 8.1 of the safety analysis report (SAR) describes the normal electrical power system for the operation of the University of Missouri-Columbia Research Reactor (MURR or the reactor) facility. The University of Missouri-Columbia has a power plant that supplies normal electrical power to MURR. Electrical connections are also in place with the City of Columbia electrical system. To reduce spurious reactor scrams caused by voltage fluctuations or a momentary interruption in electrical power, an uninterruptable power supply (UPS) is installed to provide regulated 120-volts alternating-current electrical power to the reactor instrumentation and control system. The UPS also ensures that the reactor control console and instrument panel indications remain operable during the period from a loss of the normal electrical power supply until the diesel generator starts and loads the electrical bus. A loss of normal electrical power can cause a reactor scram; however, the UPS will ensure the continuous availability of the console and instrumentation and control system to allow the reactor operator to continuously monitor the reactor for as long as 2 hours. Reactor shutdown is passive and fail safe in that if normal electrical power is lost, the control blades automatically fall to their bottom position beside the core because of gravity, thus shutting down the reactor

The NRC staff reviewed the design of the MURR electrical power system and finds that the normal power system is adequately designed to support the normal operation of MURR. The NRC staff also finds that MURR will shut down unassisted with loss of normal electrical power and that it does not rely on normal electrical power to maintain safe shutdown conditions. Normal electrical power provides no safety-related functions.

8.2 Emergency Electrical Power Systems

SAR Section 8.2 describes the design and construction of the emergency electrical power system (EEPS), which provides power to essential reactor components to monitor systems and ensure personnel safety should the facility suffer a loss of normal electrical power. The EEPS is not required to maintain the reactor in a safe and shutdown condition. The EEPS supplies power using a 275-kilowatt (368-horsepower-electric) diesel generator. The fuel capacity is sufficient fuel to maintain operation for 10 hours under full load, and the diesel is capable of starting and providing full electrical load within 7 seconds. The major electrical loads supplied by the EEPS are reactor controls, area radiation monitor, emergency compressor, facility exhaust fans, and personnel airlock doors, emergency lighting, and are listed in SAR Section 8.2.4.

In its response to RAI No. 4.g (Ref. 37), the licensee provides supplemental information on operation and testing of the EEPS and states that the normal operation of the automatic transfer switch will detect a loss of normal electrical power, initiate operation of the emergency power diesel generator, and load the electrical buses on the EEPS. The licensee indicates that the emergency power generator has proven to be reliable since its installation in 1989. The licensee tests the emergency power generator weekly for at least 30 minutes and indicates that the weekly check has been demonstrated as sufficient to ensure that the emergency power generator will start and run when required. The licensee's EEPS is tested in conformance to its compliance procedure entitled, "CP-17, Emergency Generator Load Test." The compliance procedure tests the operation of the EEPS, including the automatic transfer switch and emergency power generator.

The NRC staff finds that the design of the EEPS suitable to provide power to essential reactor monitoring and personnel safety systems should the facility suffer a loss of normal electrical power. The EEPS is not required to support safe shutdown of the reactor or to maintain a safe shutdown condition. The operating history of the emergency diesel generator indicates the system is reliable and will provide emergency power when needed. Based on the information provided above, the NRC staff concludes that the EEPS is acceptable.

TS 5.6 Emergency Electrical Power System

TS 5.6, Specification a, states:

Specification:

The following design feature applies to the emergency electrical power system:

a. The MURR shall have an emergency power generator capable of providing emergency electrical power to the emergency lighting system, the facility ventilation exhaust system, reactor instrumentation, and the personnel air lock doors.

TS 5.6, Specification a, requires the MURR facility to have an emergency power generator capable of providing emergency electrical power to the emergency lighting system, the facility ventilation exhaust system, reactor instrumentation, and the personnel airlock doors. The NRC staff finds that TS 5.6, Specification a, helps ensure that emergency electrical power generator is available for providing electrical power to the emergency electrical systems. The NRC staff also finds that this specification is consistent with the description of the EEPS in SAR Section 8.2. Based on its review, the NRC staff concludes that TS 5.6, Specification a, is acceptable.

TS 3.6 Emergency Electrical Power System

TS 3.6 states:

Specification:

a. The reactor shall not be operated unless the emergency electrical power system is operable.

TS 3.6, Specification a, requires that the reactor not be operated unless the EEPS is operable. The NRC staff finds that this specification helps to ensure that emergency power is available to supply emergency electrical loads in the event that normal power is interrupted. Based on its review, the NRC staff concludes that TS 3.6, Specification a, is acceptable.

TS 4.6 Emergency Electrical Power System

TS 4.6, Specifications a and b, state:

Specification:

a. The operability of the emergency power generator shall be verified on a weekly basis.

b. The ability of the emergency power generator to assume the emergency electrical loads shall be verified on a semiannual basis.

TS 4.6, "Emergency Electrical Power System," requires verification of system operability.

TS 4.6, Specification a, requires the emergency power generator to be verified operable on a weekly periodicity. The NRC staff finds that TS 4.6, Specification a, requires a test of the emergency diesel generator weekly, is consistent with the guidance in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," issued February 1996 (Ref. 51), and American National Standards Institute/American Nuclear Society (ANSI/ANS)-15.1-2007, "The Development of Technical Specifications for Research Reactors," issued 2007 (Ref. 57). Based on its review, the NRC staff concludes that TS 4.6, Specification a, is acceptable.

TS 4.6, Specification b, requires the licensee to conduct an electrical load test of the emergency power generator semiannually. The NRC staff finds that TS 4.6, Specification b, surveillance requirements are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on its review, the NRC staff concludes that TS 4.6, Specification b, is acceptable.

The NRC staff finds that TS 4.6, Specifications a and b are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On this basis, the NRC staff concludes that TS 4.6, Specifications a and b, are acceptable.

8.3 Conclusions

Based on its review of SAR Chapter 8, the NRC staff concludes that the design of the normal and emergency power systems is adequate to provide necessary emergency electrical power during the renewal period and the TSs discussed above meet 10 CFR 50.36 requirements for design, LCO, and SR TSs. The NRC staff concludes that emergency electrical power is not required for reactor shutdown, evacuation of the reactor building, or prevention of an accident evaluated in the SAR.

9 AUXILIARY SYSTEMS

9.1 Heating, Ventilation, and Air-Conditioning Systems

SAR Section 9.1 describes the design and construction of the heating, ventilation, and air-conditioning (HVAC) systems at the MURR facility. Steam-supplied units in the air intake system heat the air in the reactor containment building (RCB). Air entering the containment building first passes through a dust filter, followed by heating and cooling coils. Much of the containment building airflow is recirculated, with a fraction of the total flow as fresh makeup air. The containment building is maintained at a slight negative pressure relative to outside air. Air exhausted from the containment building is mixed with exhaust air from the laboratory building, and then it passes through pre-filters and high-efficiency particulate air filters. The filtered air is then discharged to the environment through a 21-meter (m) (70-foot (ft)) stack. Two 100-percent exhaust fans are used to discharge through the stack, with one in operation and one in standby. A problem with the operating exhaust fan that results in a shutdown will initiate the startup of the standby fan. In SAR Section 9.1, the licensee provides an analysis that discusses all sources of radioactive material that could become airborne in the reactor room from the full range of reactor operations, and explains how the radioactive material is controlled by the HVAC system so that it could not inadvertently escape from the RCB.

The NRC staff finds that the MURR HVAC system is adequate to control the release of airborne radioactive effluents during the full range of reactor operations because any release is contained by the RCB containment system (SER Chapter 6.2). The NRC staff also finds that the licensee discussed all sources of radioactive material that could become airborne in the reactor room from the full range of reactor operations. The NRC staff reviewed analyses in SAR Section 9.1 and finds that the analyses indicate that operation of the HVAC system limits the distributions and concentrations of the airborne radionuclides in the reactor facility, so that during the full range of reactor operations, no potential occupational exposures would exceed the design bases. The NRC staff also finds that the licensee considered the height and flow rate of the stack that exhausts facility air to the unrestricted environment for the design-basis dose rates for the maximum exposed personnel in the unrestricted environment.

The NRC staff also finds that the HVAC system is an integral part of a containment system at the reactor facility. The design of the containment system and analysis of its operation help ensure that it will function to limit normal airborne radioactive material as analyzed in SER Chapters 9 and 11. SER Chapter 11 evaluates the occupational and public doses due to the operation of the MURR and indicates that the potential radiation doses will not exceed the limits of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, “Standards for Protection against Radiation.” The NRC staff finds that operation of the HVAC system to limit doses is also consistent with the facility’s As Low As Is Reasonably Achievable (ALARA) Program. The NRC staff finds that the licensee proposed acceptable TSs, including testing and surveillance that will provide reasonable assurance of necessary HVAC system operability for the full range of reactor operations. Technical Specifications (TSs) associated with the RCB include TS 5.5, which establishes the design features required for RCB containment, TS 3.4 which establishes the limiting conditions for operation (LCO) applicable to containment, and TS 4.4 which provides the surveillance requirements (SRs) for LCOs in TS 3.4 (see SER Chapter 6.2).

Based on its review, the NRC staff concludes that the HVAC system is sufficient to maintain acceptable conditions for personnel and equipment.

9.2 Handling and Storage of Reactor Fuel

SAR Section 9.2 describes the handling and storage of reactor fuel and states that, upon its removal from the shipping container, new fuel is physically inspected against acceptance criteria to ensure no damage or defects occurred during transport. Fuel that satisfies the acceptance criteria is stored in the storage vaults that are equipped with fuel racks designed with a criticality-safe geometry with a k_{eff} of less than 0.9 until the fuel is ready to be placed in the reactor. Specially designed tools, operated by trained and qualified operators, are used to move fuel. Fuel that has been irradiated is stored in one of the 88 in-pool fuel storage locations. The geometry and materials used for these locations are designed to ensure that the k_{eff} is less than 0.9 under all conditions of moderation. The design considerations in the storage locations also included the criteria that the natural convection of the pool coolant is sufficient to cool the fuel and provide radiation shielding sufficient to protect facility staff. The fuel is stored in the in-pool locations until radioactive decay reduces radiation and decay heat levels such that conditions for shipping the fuel to the Department of Energy are achieved.

The NRC staff finds the fuel-handling and storage described in SAR Chapter 9.2 and licensee's response to Request for Additional Information (RAI) No. 2 (Ref. 32), provides a detailed reactivity analysis using the Monte Carlo N-Particle (MCNP) transport code (see SER Section 4.5). The NRC staff finds that the licensee's MCNP analysis indicated that the k_{eff} was less than 0.56 for all potential fuel storage locations for all levels of moderation (including flooded conditions). The NRC staff finds that non-flooded conditions for irradiated fuel were either not allowed by TSs, or not credible given the physical configuration of the storage locations (pool). The NRC staff also finds that the fuel storage k_{eff} values and calculation methods are consistent with the guidance in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," issued February 1996 (Ref. 51). Based on its review, the NRC staff concludes that the k_{eff} for the fuel-handling and storage of both new and irradiated fuel at MURR is less than 0.9.

The NRC staff finds that the MCNP analyses demonstrate that the fuel storage design will ensure that criticality cannot occur and that, even under optimum neutron moderation and reflection conditions, the maximum neutron multiplication does not exceed 0.90. Based on the above information, the NRC staff concludes that fuel will be stored under appropriate control and in such manner to preclude criticality.

TS 5.3 Reactor Core and Fuel

TS 5.3, Specification i, states:

Specification:

(…)

i. The reactor fuel shall be contained in the aluminum pressure vessel, in-pool fuel storage locations, or the fuel storage vault.

(…)

TS 5.3, Specification i, requires the storage locations for fuel storage to be the aluminum pressure vessel, in-pool fuel storage locations, or the fuel storage vault. The NRC staff finds that TS 5.3, Specification i, is consistent with the description of fuel use and storage locations in

SAR Chapter 9, and NRC staff observations during site visits. Based on the information above, the NRC staff concludes that TS 5.3, Specification i, is acceptable.

TS 5.4 Fuel Storage

TS 5.4 states:

Specification:

The following design features apply to fuel storage:

a. All fuel elements or fueled devices outside the reactor core shall be stored in a geometrical array where the value of K_{eff} is less than 0.9 under all conditions of moderation and reflection.

b. Irradiated fuel elements or fueled devices shall be stored in an array which will permit sufficient natural convection cooling such that the temperature of the fuel element or fueled device will not exceed its design values.

TS 5.4, Specification a, requires that fuel elements or fueled devices that are not located in the reactor core shall be stored in a geometrical array such that the k_{eff} is less than 0.9 for all levels of moderation and reflection, which helps ensure that the fuel elements or fueled devices remain subcritical. In its response to RAI No. 2 (Ref. 32), the licensee provides an updated fuel storage MCNP analysis for both unirradiated and used (spent) fuel for all storage areas. The NRC staff finds that the MCNP analysis is acceptable because it demonstrates that the k_{eff} for all storage locations (in-pool or vault) are less than 0.635 for the most limiting configuration and fuel reactivity. The NRC staff also finds that this k_{eff} is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007, which recommends a k_{eff} that is less than 0.9. Based on the information provided above, the NRC staff concludes that TS 5.4, Specification a, is acceptable.

TS 5.4, Specification b, requires irradiated fuel elements and fueled devices be stored in an array that provides sufficient natural convection cooling to ensure that the temperature of the fuel element or fueled device will not exceed its design value. The NRC staff finds that unirradiated fuel elements are not included in this TS as they do not generate decay heat and do not need cooling requirements for storage. In its response to RAI No. 5.2.b (Ref. 17), the licensee provides information indicating that the fuel storage racks are cooled by the pool coolant system (PoolCS), which maintains the pool water temperature around 100 degrees Fahrenheit (37 degrees Celsius).

The NRC staff reviewed the description provided in SAR Chapter 9.2, as supplemented by the licensee’s RAI responses, and observed the fuel storage racks and the PoolCS during its site visits. The NRC staff finds that the licensee has appropriate cooling capacity to maintain the irradiated fuel below any design temperature limits based on the location of the fuel element storage locations and licensee’s MCNP analysis of the resulting k_{eff}. The NRC staff also finds that the fuel is fully submerged in the reactor pool. Based on its review, the NRC staff concludes that TS 5.4, Specification b, is acceptable.

The NRC staff reviewed TS 5.4, Specifications a and b, and finds that the specifications are consistent the analyses in SAR Section 9.2, and in the licensee’s RAI responses, consistent with observations made by the NRC staff during site visits, and consistent with the guidance in

NUREG-1537 and ANSI/ANS-15.1-2007. Based on its review, the NRC staff concludes that TS 5.4, Specifications a and b, are acceptable.

<u>TS 5.5 Reactor Containment Building</u>

TS 5.5, Specification d, states:

<u>Specification</u>:

(…)

d. The containment building shall have a secured fuel storage room with the key or combination under control of the Reactor Manager.

TS 5.5, Specification d, requires a secured fuel storage room with the key or combination under the control of the reactor manager. The NRC staff finds that it is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007, for control of storage of fuel. Based on its review, the NRC staff concludes that TS 5.5, Specification d, is acceptable.

<u>TS 4.1 Reactor Core Parameter</u>

TS 4.1 states:

<u>Specification</u>:

(…)

c. The reactor core shall be verified to consist of eight (8) fuel elements after a refueling for a reactor startup.

   Exception: The reactor may be operated to 100 watts above shutdown power on less than eight (8) elements for the purposes of reactor calibration or multiplication measurement studies.

d. One out of every eight (8) fuel elements that have reached their end-of-life shall be inspected for anomalies.

TS 4.1, Specification c, requires verification after refueling for a reactor startup that the core configuration consists of eight fuel elements. An Exception to TS 4.1, Specification c, allows less than eight fuel elements for the purpose of reactor calibration or multiplication measurements.

The NRC staff finds that TS 4.1, Specification c, helps to ensure that the core configuration is consistent with the design analyses in SAR Chapter 13. The NRC staff finds that operation with less than eight fuel elements is allowed in Mode III, which is limited to 100 watts reactor thermal power, and which allows the licensee the ability to perform calibrations or reactivity measurements at low power. In its License Amendment (LA) No. 36 (Ref. 66), the NRC staff concluded that that operation with seven fuel elements at this limited power level (Mode III) does not pose a challenge to the fuel temperature safety limit or any accident analyses. Based on the information above, the NRC staff concludes that TS 4.1, Specification c, is acceptable.

TS 4.1, Specification d, requires one out of every eight fuel elements that have reached their end of life to be inspected for anomalies. The inspection criteria is provided in TS 3.1, Specification d (SER Section 4.2.1), and includes an inspection for anomalies and dimensional changes of any coolant channels greater than 10 mils (0.010 in). In its response to RAI No. 3.c (Ref. 37), the licensee states that the manufacturer of MURR fuel elements provides no guidance or recommendations for detecting fuel element deterioration. TS 3.3, Specification c (see SER 5.2.7), provides limits on the I-131 concentration in the PCS of 5 x 10-3 μCi/ml, which provides for the early detection of a leaking fuel element. TS 5.3, Specification c (see SER Section 4.5.2), limits the peak fissions per cubic centimeter burn up to values that have been shown to result in less than 10 percent swelling of the fuel plates which would result in an increase in fuel plate thickness of approximately 5 mils (0.005 in). The licensee indicates that a worst-case scenario involving two adjacent fuel plates that swell towards the same coolant channel gap, would cause a decrease in the nominal coolant channel gap of 10 mils (Note: Nominal coolant channel gap is 80 mils, with a lower fabrication tolerance of 72 mils). Therefore, in accordance with the TSs described above, MURR's definition of fuel anomalies consists of an increase in I-131 activity in the primary coolant system, a coolant channel dimension change of 10 mils or greater, and failure of a visual inspection. The license performs fuel inspections in accordance with procedure, FM-152, "Fuel Element Inspection."

The NRC staff reviewed the licensee's response to RAI No. 3.c (Ref. 37) and finds that TS 4.1, Specification d, is consistent with the guidance provided in NUREG-1537 to inspect a percentage of plate-type fuel. Based on its review, the NRC staff concludes that TS 4.1, Specification d, is acceptable.

The NRC staff reviewed TS 4.1, Specifications c and d, and finds that the surveillance intervals are consistent with the guidance in NUREG-1537. Based on the information provided above, the NRC staff concludes that TS 4.1, Specifications c and d, are acceptable.

9.3 Fire Protection Systems and Program

SAR Section 9.3 describes the fire protection system, which is designed to protect the facility staff and mitigate the damage to the property. The system provides two primary functions—(1) detection, which affords an early warning of an actual or potential fire condition by a combination of heat, smoke, and remote manual devices, and (2) suppression. The fire suppression system is a combination of multiple systems, including a deluge system used in the cooling tower, a pre-action system used in areas that contain highly sensitive electronic equipment, a dry fire main system used in the RCB, and a traditional sprinkler system used throughout the rest of the laboratory building. The containment building fire suppression system consists of three firehose cabinets connected to a dry fire main. Cross connecting it to the rest of the facility's wet system by a manual isolation valve located in the laboratory basement can flood this system.

As described in SAR Chapter 4, the fire protection system is not required to accomplish a safe shutdown of the reactor or to maintain a safe shutdown condition. The reactor is controlled by four shim control blades and one regulating control blade. The four shim blades are held out of the core by electromagnets. In its response to RAI No. 7.1 (Ref. 18), the licensee indicates that, in the event of a power loss (because of fire or otherwise), the blades would drop to their lower positions because of gravity and shut down the reactor. Fire detection alarms indicate in the central control station with a repeater station in the control room. Normal and emergency power supplies power the alarms.

SAR Section 9.3 states that the fire protection system can receive a virtually unlimited supply of water from the University of Missouri-Columbia fire and domestic cold water main. Four Siamese hose fittings are connected to the MURR fire main. These fittings are located outside the facility, and they facilitate connecting a pumper truck between the fittings and fire hydrants, which are located in the vicinity of the hose fittings, thus providing an additional water supply path to the fire main. In addition, fire extinguishers are strategically located throughout the facility. The fire protection system is powered from the emergency electrical distribution system. The system also has a self-contained 24-hour battery backup.

SAR Section 9.3 states that the laboratory building is constructed of noncombustible materials such as concrete blocks with brick veneer exterior walls. Interior walls are mostly concrete block. The containment building walls are poured reinforced concrete. Exits from the facility meet the requirements of National Fire Protection Association (NFPA) 101, "Life Safety Code," (Ref. 95). All areas in the facility have access to at least two exits, with the exception of the RCB. The RCB is considered "Low Hazard Industrial Occupancy," with respect to fire safety. Only one exit is required for these spaces as long as the travel distance to the exit is adequate. Table A-5-6.1 in NFPA 101 indicates that a distance of travel to an exit for these spaces should be a maximum of 300 ft (91.4 m). All areas of the containment building are well within this distance to an exit. In its response to RAI No. 9.1 (Ref. 17), the licensee provides additional information to support the 300-ft (91.4-m) distance limit.

The NRC staff finds that the fire protection system is consistent with the description provided in SAR Section 9.3, and supplemented in the licensee's response to RAI No. 9.1 (Ref. 17). The NRC staff observations of the fire protection system, emergency egress, and flammable material storage performed during site visits of the facility confirmed the design description provided by the SAR. The NRC staff reviewed the licensee's Fire Protection Program for preventing fires and finds that the program helps ensure that the facility meets local and national fire and building codes. The NRC staff also finds that the systems designed to detect and combat fires at the facility can function as described in the SAR and limit damage and consequences, and the potential radiological consequences of a fire will not prevent safe reactor shutdown. The NRC staff reviewed the MURR emergency plan (EP) and finds that there are procedures to support immediate response and notification of a fire. The appropriate sections of the facility EP adequately address any fire-related release of radioactive material from the facility to the unrestricted environment. Additionally, an agreement is in place acknowledging the commitment from the City of Columbia Fire Department to respond to a fire emergency at MURR. Based on the information above, the NRC staff concludes that the Fire Protection Program is acceptable for license renewal.

9.4 Communication Systems

SAR Section 9.4 describes the MURR facility communications system, which indicates that there are telephones placed throughout the facility, the facility has an intercom paging system, and communication over the intercom paging system is possible using any phone. Additionally, walkie-talkie units, which are only used for equipment testing and for emergency use, are stored in a central location in the facility.

The NRC staff reviewed the MURR facility communications system and finds that it is designed to provide two-way communication between the reactor control room and all other locations necessary for safe reactor operation. The NRC staff also finds the MURR facility communications system allows the reactor operator on duty to communicate with the supervisor on duty and with health physics personnel on duty, allows a facility-wide announcement of an emergency, and has provisions for summoning emergency assistance from designated personnel as discussed in detail in the physical security and EPs. Based on its review, the NRC staff concludes that the MURR facility communications system is acceptable.

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

In addition to material authorized under the reactor license, SAR Section 9.5 describes Materials License No. 24-00513-32 held by the Curators of the University of Missouri (the licensee) for the possession and use of licensed materials throughout the reactor facility. In its responses to RAI No. 9.2 (Ref. 17), the licensee indicated that the licensed material is controlled through the use of research-specific or production-specific project authorizations as authorized by the Radiation Safety Committee. These project authorizations provide the administrative controls for the use of radioactive materials controlled in accordance with the requirements of 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," throughout the facility and provide the administrative controls necessary to segregate the radioactive materials from other materials used and stored in the facility. Transfers of materials between the two licenses are not routine but occur periodically. When a material transfer occurs, the licensee uses a dedicated transfer form to document the transfer between one project and another and is usually completed by the health physics staff in concert with the authorized supervisor of the project or projects involved with the transfer. The licensee states that it also maintain records of the licensed material transfers.

In its response to RAI No. 9.3 (Ref. 17), the licensee indicates that no changes to the current reactor license possession limits are needed because of the LRA. However, in support of future experiments, the licensee requested changes to current License Condition (LC) 2.B.(2) that 5 kilograms (kg) of the existing 60 kg SMN in the current LC be allowed for separation in the form of low enriched uranium (LEU). Specifically, the current LC 2.B.(2) authorizes the receive, possess, and use of up to 60 kilograms (kg) of contained uranium-235 of any enrichment, providing that less than 5 kg of this amount be unirradiated. The renewed license restates the current LC 2.B.(2) as two license conditions, LC 2.B.2.a and LC 2.B.2.b.

The renewed LC 2.B.2.a authorizes the receive, possess, and use, but not separate, in connection with the operation of the facility, up to 55 kg of contained uranium-235 of an enrichment of 20 percent or greater in the isotope uranium-235, providing that less than 5 kg of this amount be unirradiated.

The renewed LC 2.B.2.b authorizes the receive, possess, and use, in connection with the operation of the facility, up to 5 kg of uranium-235 of an enrichment less than 20 percent in the isotope of uranium-235, for use in experiments.

The NRC staff finds that, for experiments, the 5 kg in the form of LEU out of the current 60 kg of SNM allow for separation is acceptable. The NRC staff also finds that the cumulative limit of SNM was unchanged and remains at 60 kg. On this basis, the NRC staff concludes that the proposed change is acceptable.

The other remaining possession limits specified in the renewed LCs are as follows:

- to receive, possess, and use, but not separate, in connection with the operation of the facility, up to 80 grams of plutonium-beryllium in the form of a neutron source

- to receive, possess, and use, but not separate, in connection with the operation of the facility, up to 20 grams of plutonium-239 in the form of sheets enclosed in aluminum

- to receive, possess, and use, but not separate, in connection with the operation of the facility, up to 40 grams of plutonium enriched to 90 percent in plutonium-242 in the form of a rod sealed in a stainless steel can

- to receive, possess, and use, but not separate, in connection with the operation of the facility, such special nuclear material as may be produced by the operation of the facility

- to receive, possess, and use, in connection with the operation of the facility, up to 100 curies of a sealed antimony-beryllium neutron source

- to receive, possess, and use, but not to separate, in connection with operation of the facility, such byproduct material as may be produced by operation of the reactor, which cannot be separated except for byproduct material produced in reactor experiments

- to receive, possess, and use, in connection with operation of the facility, up to 20 kilograms each of natural uranium and thorium

- to receive, possess, and use, in connection with the operation of the facility, up to 50 kilograms of depleted uranium for instruction and experimental purposes

The NRC staff reformatted the LCs to facilitate readability and to prevent the separation of byproduct material, except for byproduct material produced in experiments. The NRC staff discussed these changes with the licensee and the licensee agreed to the changes by letter date December 14, 2016 (Ref. 108).

The NRC staff reviewed SAR Section 9.5 and responses to RAI Nos. 9.2 and 9.3 (Ref. 17), for the possession and use of byproduct, source, and SNM and finds that the licensee has facilities and processes designed for the possession and use of byproduct, source and SNM, as well as byproduct and SNM produced by operation of the reactor. The NRC staff finds that the MURR facility design bases in the SAR include limits on potential personnel exposures that are in compliance with 10 CFR Part 20 and are consistent with the facility ALARA Program, as described in SAR Chapter 11. The NRC staff also finds that the facility provides reasonable assurance that uncontrolled release of radioactive material to the unrestricted environment will not occur, and if it does, the doses would be within the limits of 10 CFR Part 20, as described in SAR Chapters 11 and 13. Based on the information above, the NRC staff finds the licensee's controls for the possession and use of byproduct, source, and SNM acceptable for the renewal period.

9.6 Cover Gas Control in Closed Primary Coolant Systems

SAR Section 9.6 describes the use of nitrogen gas to maintain PCS pressure at appropriate levels. The nitrogen is supplied from two banks of three cylinders each. One bank is used for operation, whereas the other is in standby. Pressure regulators are used to step down the pressure of the supplied nitrogen to 655 kilopascal above atmosphere (95 pounds per square inch, gauge). When the operational bank of cylinders drops below designated limits, the supply of nitrogen automatically switches to the backup cylinders, and an annunciator alarm is triggered in the control room.

The NRC staff reviewed the licensee's use of nitrogen cover gas as described in the SAR and finds that the reactor is designed to operate with a cover gas control system helps provide adequate pressure control for the PCS. The NRC staff finds that the cover gas control system is designed to prevent the uncontrolled release of radioactive material and interference with safe reactor operation or shutdown. Based on its review, the NRC staff concludes the cover gas in the closed PCS is acceptable for maintaining PCS pressure.

9.7 <u>Other Auxiliary Systems</u>

<u>Emergency Pool Fill System</u>

SAR Section 9.7 describes the emergency pool fill system (EPFS). The water line and associated supply are capable of delivering a minimum of 3,785 liters (1,000 gallons per minute) in the event of a leak of the PoolCS, as described in SAR Section 13.2.9. The water line enters the RCB through the water seal. The system includes anti-siphon features so that the system could not inadvertently drain the reactor pool.

The NRC staff reviewed the EPFS, and observed the location of the water line during its site visit. The NRC staff finds that the water line enters the RCB through the water seal, and that its design provides a defense-in-depth component to the facility to help ensure that the reactor remains flooded under any accident conditions. Based on its review, the NRC staff concludes that the EPFS is acceptable.

<u>TS 3.9 Auxiliary Systems</u>

TS 3.9 Specification b, states:

<u>Specification</u>:

(…)

b. The reactor shall not be operated unless the emergency pool fill system is operable.

TS 3.9, Specification b, requires the EPFS be available to supply water in case a PoolCS leak occurs. The NRC staff finds that TS 3.9, Specification b, helps to ensure that the EPFS is available to provide water to the reactor to mitigate the LOCA or a PCS leak due to a breach of an experiment beam port. The NRC staff also finds that the EPFS support the assumptions used in the LOCA analysis described in SAR Section 13.3, and is sufficiently designed to mitigate the consequences of a pool PoolCS leak. Based on its review, the NRC staff concludes that TS 3.9, Specification b, is acceptable.

<u>TS 4.9 Auxiliary Systems</u>

TS 4.9, Specification b, states:

<u>Specification</u>:

(…)

b. The operability of the emergency pool fill system shall be tested on a semiannual basis.

TS 4.9, Specification b, requires a semiannual test of the EPFS. The NRC staff finds that TS 4.9, Specification b, helps to ensure the operability of the EPFS when required to provide emergency pool fill water as assumed in the SAR. The NRC staff also finds that the surveillance interval is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.9, Specification b, is acceptable.

Based on a review of the EPFS as described in SAR Section 9.7, the NRC staff concludes that the EPFS is acceptable for providing a reliable water source for emergency pool fill.

Pool Skimmer System

SAR Section 9.8 describes the pool skimmer system, which indicates that the system removes particles from the surface of the reactor pool. The system can also be used as a mechanism to lower or raise the water level in the pool. In its response to RAI No. 9.5 (Ref. 17), the licensee provides an analysis that demonstrated that any water losses from malfunction of the pool skimmer system are bounded by other analyzed pool water leakage scenarios, which are discussed in SAR Section 13.3. The NRC staff finds that the pool skimmer system as described in the SAR, and that it meets the requirements and objectives identified in the SAR and RAI response for maintaining water inventory. Based on the information above, the NRC staff concludes that the pool skimmer system is acceptable for removing surface particles and supports maintaining water inventory for the reactor pool.

Demineralized Water Supply System

SAR Section 9.12 describes the demineralized water supply system, which produces and stores high-purity water for normal reactor and facility operation. Untreated water is conditioned, filtered, and then passed through a mixed bed resin column. The system also includes a separate ion exchange demineralizer as a backup.

The NRC staff finds the demineralized water supply system, as described in the SAR, consistent with the systems used at similar reactor facilities, and it meets the requirements and objectives identified in the SAR. Based on the information provided above, the NRC staff concludes that the demineralized water system is an acceptable system for maintaining high-purity water inventory for normal operation.

Reactor Loop Vent System

SAR Section 9.13 describes the reactor loop vent system. This system collects and filters gases collected from the PCS. Gases are collected in a vent tank and periodically exhausted through an absolute and charcoal filter. There are no functions or malfunctions of the reactor loop vent system that could initiate a reactor accident, prevent safe reactor shutdown, or initiate the uncontrolled release of radioactive material.

The NRC staff finds that the reactor loop vent system, as described in the SAR, meets the requirements and objectives identified in the SAR. Based on the information provided above, the NRC staff concludes that the reactor loop vent system is an acceptable system for collecting and filtering reactor-generated gases.

Compressed Air Systems

SAR Section 9.14 and licensee's response to RAI No. 9.6 (Ref. 17) describe the four interconnected compressed air systems at the MURR facility, which include the Main Air System, the Instrument Air System, the Valve Operation Air System, and the Emergency Air System. A description of each compressed air system follows:

The Main Air System supplies compressed air for the MURR facility and laboratory needs and contains the main compressors for each of the other subsystems. In the event of a failure of the air compressors or other components within the system, each air subsystem is isolated from the Main Air System by use of check valves and have a backup compressor to provide service for that subsystem.

The Instrument Air System serves to control facility HVAC. It is normally not connected to the other subsystems through isolation valves. If this system was cross connected with the other subsystems during maintenance and a failure were to occur, the associated check valve would prevent interactions between the other subsystems. No reactor safety features are associated with the facility heating and air conditions system

The Valve Operation Air System serves to provide compressed air to remotely operated pool and primary coolant valves. Each valve is air-operated to the appropriate operating position and spring-operated to the fail-safe condition upon a loss of compressed air. A loss of compressed air would cause the valve to spring operate to the fail-safe position and may initiate an associated reactor scram, but would in no case prevent the safety system from functioning as required.

The Emergency Air System provides air for operation of the hot exhaust line isolation valves 16A and 16B, the containment isolation backup doors, and to the inflatable gaskets, which maintain the reactor containment building isolation. This system also has a backup air compressor and associated check valve to prevent a subsystem failure if the Main Air System is compromised. The hot exhaust line isolation valves are in series and of different valve types: valve 16A is spring-to-close and air to open valve, which is designed fail-safe, while valve 16B is air-to-open and air-to-close. Valve 16B has an additional independent compressor should all of the three previous compressors fail.

The NRC staff reviewed the compressed air systems described in the SAR and in the licensee's response to RAI No. 9.6, and finds that the compressed air systems provide the compressed air required for the various components needed to operate the MURR facility. The NRC reviewed the potential for single failure of the compressed air system (see SER Section 6.2) and finds that the system is protected from a failure due to the loss of a single component. Based on the information above, the NRC staff concludes that the compressed air system is acceptable to support operation of the MURR facility.

Radioactive Waste Disposal System

SAR Section 9.11 describes the radioactive waste disposal system in which it states that gaseous, solid, and liquid radioactive wastes are disposed through the facility ventilation and air treatment system, the solid radioactive waste program, and the radioactive liquid waste retention and disposal system. SAR Chapter 11 discusses the radioactive waste disposal programs in detail.

The NRC staff reviewed the radioactive waste disposal system and programs, as described in the SAR, and finds that they appear effective to manage and control the disposal of radioactive waste. A detailed NRC staff's review of each type of radioactive waste generated and disposed is provided in SER Section 11.1. Based on the information provided above, the NRC staff concludes that the radioactive waste disposal system is acceptable.

Battery-Operated Emergency Lights

SAR Section 9.11 describes the battery-operated emergency lights, which are positioned throughout the reactor containment and laboratory buildings and operate in conjunction with selected lights powered by the emergency electrical power system to provide lighting during a loss of normal electrical power. Each light has a self-charging battery pack and a switching circuit that actuates upon electrical power failure.

The NRC staff finds that the battery-operated emergency lights will provide effective lighting for personnel egress should the normal lighting fail. The NRC staff observed the locations of the battery-operated emergency lights during its site visit and noted that the lights would provide emergency lighting for personnel egress if the normal lights were to lose electrical power. Based on its review, the NRC staff concludes that the battery-operated emergency lights are acceptable.

9.8 Conclusions

The NRC staff reviewed the auxiliary systems, described in SAR Chapter 9 and licensee's responses to RAIs, and finds that the systems are designed to perform the functions required by the their respective design bases. The NRC staff also finds the design of the systems considers potential malfunctions that could affect reactor operations and finds no analyzed malfunctions that could initiate a reactor accident, prevent safe reactor shutdown, or initiate an uncontrolled release of radioactive material. The NRC staff finds the TSs applicable to the auxiliary systems, as discussed above, are consistent with the SAR description, including LCOs and SRs and provide reasonable assurance that the systems will be operable when needed, as required by 10 CFR 50.36. Based on its review, the NRC staff concludes that the auxiliary systems at the MURR facility are effective to support continued operation during the renewal period.

10 EXPERIMENTAL FACILITIES AND UTILIZATIONS

10.1 Summary Description

Chapter 10 of the safety analysis report (SAR) describes the University of Missouri-Columbia Research Reactor (MURR or the reactor) experimental facilities, which are used to provide irradiation services to researchers and commercial entities, as well as for education. The main purpose of the MURR is to provide neutrons to the experimental facilities. In order to accomplish the goals of the experiment program, the design of the reactor and the experimental facilities must be rather unique, but must also emphasize safety as a paramount concern, which dictates design criteria associated with the experimental facilities and TSs. A major safety feature of the reactor can be found in the design of the beryllium reflector, which effectively decouples (from a reactivity point of view) the reactor core from experiments located in the beamports, bulk pool and graphite reflector irradiation positions, the pneumatic tube system, and the thermal column position. The only experimental facility not neutronically isolated from the reactor is the center test hole (flux trap). However, this experimental facility is subject to a high degree of administrative control as discussed in the following sections. Examples of current MURR experimental applications are neutron activation analysis, neutron and gamma-ray scattering, and neutron interferometry. Various technical specifications (TSs) provide limitations for the effect on reactivity of all experiments and means for technical and safety review of experiments.

10.2 Experimental Facilities

10.2.1 Reactor Experimental Facilities

SAR Section 10.1 describes, in general terms, the following reactor-related experimental facilities:

- six beam ports
- the thermal column
- graphite reflector irradiation positions
- pool irradiation positions
- center flux trap
- a pneumatic transfer system that transfer sample carriers, or “rabbits,” into and out of the graphite reflector region from selected laboratories

Beam Ports

SAR Section 10.3.2 describes the design and construction of the beam ports, which includes a total of six beam ports for MURR. There are three 10-centimeter (cm) (4-inch (in)) inside diameter beam ports and three such ports with an inside diameter of 15 cm (6 in). The beam ports are constructed of aluminum; they pass through the graphite reflector and terminate just outside of the beryllium reflector. The beam ports are positioned at different heights relative to the core and are spaced at 30-degree intervals with three on one side of the reactor and the other three on the opposite side of the reactor relative to the location of the control room. The beam ports are used to channel neutrons and gamma radiation from the reactor core with minimal scattering and attenuation between the beryllium reflector and the experiment equipment to facilitate the irradiation of approved experiments.

Also, the licensee indicates that beam ports that are not in use are closed using a 7.6-cm-thick (3-in-thick) lead and steel shield door. Movement of the shield door requires the use of the overhead crane. Each beam port has a connection to an off-gas vent line so that any radioactive gases are routed to the MURR exhaust stack.

The NRC staff reviewed the descriptions of the beam ports, as provided in the SAR, and finds that the design of the beam ports are consistent with the SAR description. The NRC staff observed the beam ports during its site visit, and finds that the beam ports are similar to irradiation beam ports used at other research reactor facilities, and the description is consistent with the guidance provided in NUREG-1537. Based on its review, the NRC staff concludes that the licensee adequately described design and construction of the beam ports.

Thermal Column

SAR Section 10.3.3 describes the design and construction of the thermal column, which is a column in an aluminum housing containing up to 1.5 meters (60 in) of graphite and is used for performing neutron radiographs and large sample irradiations. The column includes a removable lead shield to reduce gamma radiation. The structure includes vents tied to the facility exhaust system that allow the removal of radioactive gases generated in the thermal column. Access to the face of the thermal column requires movement of the thermal column door. The door has stepped edges to reduce radiation streaming, and the inner surface is lined with boral sheeting.

The NRC staff reviewed the description of the thermal column, as provided in the SAR, and finds that the design of the thermal column is consistent with the SAR description. The NRC staff observed the thermal column during its site visit and finds the thermal column similar to irradiation thermal columns at other research reactor facilities, and the description is consistent with the guidance in NUREG-1537. The NRC staff finds that the experiments performed with the thermal column are consistent with the SAR description. Based on its review, the NRC staff concludes that the licensee adequately described the design and construction of the thermal column, and the thermal column is acceptable for use.

Graphite Reflector Irradiation Positions

SAR Section 10.3.5 describes the design and construction of the graphite reflector irradiation positions used for the irradiation of samples in a region of relatively high thermal flux. Elements of the graphite reflector can be removed to provide locations for irradiation of samples that are larger than the pneumatic tube (p-tube) system can accommodate. Samples irradiated at these locations are encapsulated in canisters and placed in sample holders. Samples are placed in position with a line. Samples may be rotated to deliver a uniform neutron flux with an extension rod connected to a small motor above the pool surface.

The NRC staff reviewed the description of the graphite reflector irradiation positions, as provided in the SAR, and finds that the design of the graphite reflector irradiation positions is consistent with the SAR description. The NRC staff observed the graphite reflector irradiation positions during its site visit and finds the graphite reflector irradiation positions similar to in-core irradiation positions used at other research reactor facilities, and the SAR description is consistent with the guidance in NUREG-1537. The NRC staff finds that the experiments performed with the graphite reflector irradiation positions are consistent with the SAR description. Based on its review, the NRC staff concludes that the licensee adequately described the graphite reflector irradiation positions, and the graphite reflector irradiation positions are acceptable for use.

Bulk Pool Irradiation Positions

SAR Section 10.3.6 describes the bulk pool irradiation positions. The bulk pool irradiation area is the water region above and to the outside of the graphite reflector which provides an area for the placement of sample holders in a region of relatively low thermal flux (less than 5×10^{13} n/cm^2-sec). Material to be irradiated in this region is first encapsulated in an aluminum sample canister, and then inserted into an aluminum sample holder. The sample holder is lowered into a designated bulk pool irradiation position by a handling line. The sample material may or may not require rotation.

The NRC staff reviewed the description of the bulk pool irradiation positions, as provided in the SAR, and finds that the design of the bulk pool irradiation positions is consistent with the SAR description. The NRC staff observed the bulk pool irradiation positions during its site visit and finds the bulk pool irradiation positions similar to in-core irradiation positions used at other research reactor facilities, and the SAR description is consistent with the guidance in NUREG-1537. The NRC staff finds that the experiments performed with the bulk pool irradiation positions are consistent with the SAR description. Based on its review, the NRC staff concludes that the licensee adequately described the bulk pool irradiation positions, and the bulk pool irradiator positions are acceptable for use at MURR.

Center Flux Trap

SAR Section 10.3.1 describes the design and construction of the center flux trap, which allows the irradiation of samples in a region of very high thermal flux. The center hole is interior to the pressure vessel containing the fuel. The inside diameter of the center hole is 11.43 cm (4.5 in), and the irradiation region extends 38.1 cm (15 in) on either side of the core horizontal centerline. Two different canisters can be used at MURR in the central hole location. The three-hole canister includes three welded aluminum tubes with an inner diameter of 3.4 cm (1.334 in). The six-hole canister has three welded aluminum tubes similar to the three-hole canister but with three smaller tubes with an inner diameter of 1.7 cm (0.68 in). In the six-hole canister, the smaller tubes are welded into the spaces between the larger tubes.

In License Amendment (LA) No. 35 (Ref. 72), the NRC staff approved changes to the TSs to allow an additional scram channel related to the use of the center test hole flux trap known as the flux-trap irradiations reactivity safety trip (FIRST). The FIRST device contains limit switches that indicate whether the canister or strainer is secure in the center test hole. If the canister or strainer is not secure, the reactor safety system (RSS) is tripped resulting in a reactor scram. The changes to the RSS make the center test hole canister an experimental facility and a normal part of the reactor. In LA No. 35, the NRC staff concluded that the FIRST device had redundant sensors and circuitry and that no single failure of a component will render the device incapable of performing its function. Based on its review, the NRC staff concluded that the design of the FIRST device met the guidance in Institute of Electrical and Electronics Engineer Standard 603-2009, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating

Stations," issued 2009 (Ref. 96), and LA No. 35 added TS 3.2, Specification g, Scram Function No. 21, "Center Test Hole."

The NRC staff reviewed the description of the center flux trap, as provided in the SAR and documented in LA No. 35, and finds that the design of the center flux trap and experiments performed with the center flux trap are consistent with the description in SAR Section 10.3.1 and LA No. 35, and the guidance provided in NUREG-1537. Based on its review, the NRC concludes that the licensee adequately described the design and construction of the center flux trap, and the center flux trap is acceptable for use.

<u>Pneumatic Tube System</u>

SAR Section 10.3.4 describes the design and construction of the pneumatic tube system, which transfers samples into and out of the graphite reflector region of the reactor. The sample containers are commonly known as "rabbits," which are moved into and out of the neutron flux using compressed air. The system has four irradiation terminals near the reactor, and six stations that can send or receive samples. The system can simultaneously transfer four samples. The SAR states that only two terminals and three sending-receiving stations are currently in use.

The NRC staff reviewed the description of the pneumatic tube system, as provided in the SAR, and finds that the design of the pneumatic tube system is consistent with the SAR description. The NRC staff observed the pneumatic tube system during its site visit and finds the pneumatic tube system similar to in-core irradiation systems used at other research reactor facilities, and the SAR description is consistent with the guidance in NUREG-1537. The NRC staff finds that the experiments performed with the pneumatic tube system are consistent with the SAR description. Based on its review, the NRC staff concludes that the licensee adequately described the pneumatic tube system, and the pneumatic tube system is acceptable for use.

<u>Conclusions</u>

The NRC staff reviewed the description of the experimental facilities, as described in the SAR and in LA No. 35, and finds that activities done at the facility are consistent with the description in the SAR and LA No. 35, and described in accordance with the guidance in NUREG-1537. The NRC staff also finds that these facilities are typical of other research reactor facilities, and TS 5.3, which the NRC staff evaluated and found acceptable in SER Section 10.2.3, appropriately controls their use. Furthermore, the NRC staff also finds that the experimental facilities provide an acceptable means for the irradiation of samples in MURR when used in compliance with the TS.

The NRC staff also reviewed the use of the MURR experimental facilities as described in its safety analysis in SAR Chapter 13. Based on its review, the NRC staff concludes that the experimental facilities are acceptable, can be used without damaging the fuel, and do not pose a significant risk to the health and safety of the public or facility personnel.

10.2.2 Isotope Processing Units

SAR Section 1.2.4 states the laboratory building has a hot cell that is used to open and seal material irradiation canisters. The hot cell is described as being constructed to handle 10,000 curies (Ci) of cobalt or equivalent in the "Preliminary Hazards Summary Report" (Ref. 2), which was submitted with the initial licensing application for the MURR facility, During site visits, the NRC staff observed the installation and use of additional hot cells and fume hoods. As such, the NRC staff requested additional information and received the licensee's response to Request for Additional Information (RAI) No. 1.a (Ref. 32), which provided an updated description of the of processing units (i.e., hot cells, glove boxes, and fume hoods). Table 10-1 below summarizes the processing units.

Table 10-1 MURR Isotope Processing Units

Location	Designation	Typical Isotope	Location	Designation	Typical Isotope
Basement	HC-01	General Use	Room 241	HC-05	P-32/33 S-35
	HC-02A	Mo-99		GB-06	P-32/33 S-35
	HC-02B	Mo-99		GB-07	P-32/33 S-35
	HC-03	General Use		GB-08	P-32/33 S-35
	HC-04	General Use		GB-27	P-32/33 S-35
	GB-19	General Use	Room 242A	Fume Hood	P-32/33 S-35
	GB-30	General Use		GB-01	General Use
Room 111	GB-11	Pm-147	Room 242C	GB-18	General Use
	GB-24	General Use	Room 244	Fume Hood	General Use
	GB-25	Bi-210	Room 245	Fume Hood	General Use
Room 213	Fume Hood	General Use		SEH-01	General Use
Room 216	Fume Hood	General Use	Room 247	SEH-04	General Use
Room 218	Fume Hood	General Use	Room 251	Fume Hood	General Use
Room 222	Fume Hood	General Use	Room 255	Fume Hood	General Use
	SEH-02	Non-Rad	Room 257	Fume Hood	General Use
Room 224	Fume Hood	General Use	Room 259	Fume Hood	General Use
Room 225	Fume Hood	General Use	Room 299D	HC-08A	Lu-177
Room 227	Fume Hood	General Use		HC-08B	Lu-177
	GB-21	General Use	Room 299M	HC-06	Lu-177
	SEH-05	NAA	Room 299N	SEH-06	Lu-177
Room 232B	GB-03	General Use	Room 299P	HC-10	Mo-99
	GB-14	Rh-105		GB-29	Mo-99
	GB-15	Se-75	Room 299R	HC-07	Mo-99
	GB-17	Au-198		HC-09	Mo-99
Room 238	GB-01	General Use	Room 299T	HC-IIA	I-131
	GB-02	Lu-177		HC-IIB	I-131

Location	Designation	Typical Isotope	Location	Designation	Typical Isotope
	GB-04	General Use		HC-1 IC	I-131
	GB-05	Ge/As-77	Room 299V	Fume Hood	I-131 (QC)
	GB-09	General Use			
	GB-12	Sm-153			
	GB-13	Gd-159			
	GB-16	General Use			
	GB-20	Re-186/188			
	GB-22	Au-198			
	GB-23	General Use			
	GB-28	General Use			

Table Glossary:
GB glove box
HC hot cell
NAA neutron activation analysis
QC quality control
SEH specialty exhaust hood

In its response to RAI No. 1.b (Ref. 32), the licensee describes the controls needed to ensure that a postulated accident would not exceed the limits in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, "Standards for Protection against Radiation." The licensee indicates that hot cells, glove boxes, and fume hoods (i.e., processing units) at MURR are controlled in several ways with regard to the radiological aspects of their use and with respect to occupational and public dose considerations. The processing units are controlled by their authorized research or production user group at MURR in accordance with an approved reactor license project authorization. A screening or evaluation under 10 CFR 50.59, "Changes, Tests and Experiments," is performed in conjunction with completion of the reactor license project authorization. Before a MURR research or production user group is authorized the use of a hot cell or glovebox, the reactor health physics staff conducts evaluations to ensure that the proposed radioactive material will be adequately shielded by the assigned processing unit. If a new isotope is identified for use at MURR, an evaluation occurs to determine whether the existing fleet of processing units at MURR is sufficient to meet the radiation protection needs of the facility for both occupationally exposed staff and the general public. If no such facility exists, a review process occurs (either within an existing project or during the creation of a new reactor license project) as to what design characteristics are needed to provide the appropriate level of radiation protection to staff and to the general public before designing or procuring a new processing unit. If higher radioactivity levels are required for the process, existing hot cells or gloveboxes will be evaluated in relationship to the characteristics of the proposed nuclides, including the chemistry and ergonomics of the process, necessary for the safe and effective utilization of the radioisotopes.

Also, the licensee indicates that within a reactor license project authorization evaluation, consideration is given with regard to how occupational exposures will be minimized to the radiation workers based on the quantity of nuclides expected and the chemical form to be used in the hot cell, glovebox, or fume hood. The project authorization defines and lists the quantities and chemical forms appropriate for the hot cell or glovebox, depending on the specific processing unit shielding and ventilation capabilities. Historically, isotopes irradiated at the MURR facility are metals or metallic compounds that are not subject to volatilization or aerosolization. Any heating during the processing of these isotopes is much less than the melting temperature of the metals or metallic compounds supplied for irradiation. Metallic compounds are usually in the form of nitrates or oxides and are thus more prone to decomposition rather than volatilization. The heating of these compounds during irradiation is considered during the reactor utilization request (RUR) safety evaluation process before placing them in the reactor for irradiation to ensure that adverse heating conditions do not occur because of nuclear heating processes because high temperatures would destroy the compound being irradiated, thus rendering them useless for processing and further use.

The licensee also indicates that all reactor license projects are reviewed by the Isotope Use Subcommittee of the Reactor Advisory Committee (RAC) for approval after review by the reactor health physics staff, and reviewed and approved by the reactor health physics manager and the reactor manager. The reactor license projects process provides a conclusion of any analysis performed during the review process limiting the quantity of radionuclides used within any hot cell, glovebox, or fume hood with respect to the safety of workers and the public.

In its response to RAI No. 1.c (Ref. 32), the licensee provides information related to the services (e.g., electrical; heating, ventilation, and air-conditioning; plumbing; lighting, alarms; and other such services) that may be needed to support the processing units. The licensee indicates that the available services that could possibly be attached to a hot cell, glovebox, or fume hood include exhaust ventilation, electrical power, filtration, domestic cold water, vacuum, and radioactive liquid drains, but none of these services are required to mitigate the consequences of any postulated accident in a hot cell, glovebox, or fume hood. If a hot cell, glovebox, or fume hood is interfaced with any of the services that are described in the SAR, then a modification record and a 10 CFR 50.59 screening or evaluation is performed. Any filtration that is installed is in keeping with the as low as is reasonably achievable (ALARA) principles of the MURR Radiation Protection Program to maintain effluent discharges—water and air—as low as possible. The licensee provided radioactive liquid and air releases from the facility for the last 10 years, which indicated, with the exception of argon-41, that all other isotopes discharged were less than 0.6 percent of the release limit. The licensee indicated that these data demonstrate a comprehensive and effective ALARA Program.

In its response to RAI No. 1.d (Ref. 32), the licensee indicates that no changes to the TSs were needed because of the addition of the processing units (hot cells, glove boxes, fume hoods).

In addition to the information on the isotope processing units, the licensee provided in its response to RAI No.1 (Ref. 31), the modifications at the MURR facility implemented since the submittal of the LRA in calendar year 2006. Included in the list of modifications were the applicable license reviews performed as required by the regulations in 10 CFR 50.59, for the isotope processing units.

The NRC staff reviewed the changes to the facility by the addition of the processing units described in SER Table 10-1 and toured the facility on several occasions. The NRC staff finds that the changes substantially increased the licensee's ability to conduct research and produce isotopes for medical use. The NRC staff finds that the licensee evaluated these changes in accordance with the requirements of 10 CFR 50.59. In addition, the NRC staff finds that these changes are adequately described in the SAR, as supplemented by licensee's responses to RAIs, and the experimental facilities are typical of other research reactor experimental facilities. Based on the information above, the NRC staff finds the MURR isotope processing facilities acceptable.

10.2.3 Experimental Facility Controls

TS 5.3 Reactor Core and Fuel

TS 5.3, Specification I, states:

(…)

I. The reactor shall have the following experimental facilities:
   1. Six (6) beam tubes which penetrate the graphite reflector;
   2. A center test hole located in the flux trap;
   3. A portion of the graphite reflector;
   4. A bulk pool consisting of the water region above and outside the graphite reflector; and
   5. A thermal column.

(...)

TS 5.3, Specification I, allows certain experimental facilities to be used within MURR. The NRC staff finds that the experimental facilities listed in TS 5.3, Specification I are as described in SAR Section 10.3, as supplemented by the licensee's response to RAI No. 1 (Ref. 32). The NRC staff also finds this specification requires key design features of the MURR experimental facilities and is consistent with statements in the SAR and RAI responses. Based on the information above, the NRC staff concludes that TS 5.3, Specification I, is acceptable.

10.3 Experiment Review

In SAR Section 10.4 and in its responses to RAI No. 10.3.a and RAI No. 10.3.b (Ref. 24), the licensee describes the processes used for the review of experiments. All requests for irradiation at MURR must be submitted in writing and must be approved by the reactor manager. Review of the irradiation request includes an analysis of thermal and pressure effects on the sample and of the potential for corrosion and explosions. The review also must include the effects of sample failure and the effects of sample failure on other experiments in the reactor.

Further, the reactor utilization request is a mechanism used to verify that each experiment, or single experiment sample, complies with all of the applicable TS, and other limitations based on good operating, engineering, and health physics practices. This process, which is detailed in Administrative Procedure AP-RO-135, "Reactor Utilization Requests," specifically requires the reactor and reactor health managers to prepare, review, and approve a safety analysis before an experiment can be conducted. Each safety analysis includes, but is not limited to, the following major criteria:

- criticality and/or reactivity considerations

- heat generation considerations

- shielding considerations

- off-gas and/or chemical reactions

The safety analysis also includes all credible accident and transient scenarios to ensure that the experiment does not jeopardize the safe operation of the reactor or constitute a hazard to the safety of the facility staff and member of the public. The safety review process also includes the reactor health physics manager, the assistant reactor manager-physics, and the reactor manager to review the irradiation requests. If irradiation requests are determined to be a new class of experiment or to have safety significance, the review is submitted to the Reactor Safety Subcommittee, which reports to the Reactor Advisory Committee (RAC). TS 6.2 requires the establishment of the RAC and that its charter must include safety reviews (see SER Section 12.1).

In its response to RAI No. 10.4 (Ref. 24), the licensee provided information concerning the potential failure of a fueled experiment that was evaluated and accepted by the NRC staff in LA No. 34 (Ref. 67). In its responses to RAI No. 10.5.a through 10.5.g (Ref. 18), the licensee provided additional information concerning the use of the center flux hole and the associated controls for the FIRST system, including the applicable TS.

The NRC staff reviewed the experiment approval process and concludes that it implements the requirements of TS 3.8 and TS 6.2, which helps to ensure the safe use of irradiation facilities and irradiated products.

TS 3.8 Experiments

TS 3.8 states:

Reactivity Limits Specification:

a. The absolute value of the reactivity worth of each secured removable experiment shall be limited to 0.006 Δk/k.

b. The absolute value of the reactivity worth of all experiments in the center test hole shall be limited to 0.006 Δk/k.

c. Each movable experiment or the movable parts of any individual experiment shall have a maximum absolute reactivity worth of 0.001 Δk/k.

d. The absolute value of the reactivity worth of each unsecured experiment shall be limited to 0.0025 Δk/k.

e. The absolute value of the reactivity worth of all unsecured experiments which are in the reactor shall be limited to 0.006 Δk/k.

Materials Specification:

f. Each fueled experiment shall be limited such that the total inventory of iodine-131 through iodine-135 in the experiment is not greater than 150 Curies and the maximum strontium-90 inventory is no greater than 300 millicuries.

g. Fueled experiments containing inventories of iodine-131 through iodine-135 greater than 1.5 Curies or strontium-90 greater than 5 millicuries shall be in irradiation containers that satisfy the requirements of Specification 3.8.s or be vented to the facility ventilation exhaust stack through high efficiency particulate air (HEPA) and charcoal filters which are continuously monitored for an increase in radiation levels.

h. Each non-fueled experiment that is intended to produce iodine-131 shall be limited such that the inventory of iodine-131 is not greater than 150 Curies.

i. Explosive materials shall not be irradiated nor shall they be allowed to generate in any experiment in quantities over 25 milligrams of TNT-equivalent explosives. Explosive materials shall be limited to a total quantity of 100 milligrams of TNT-equivalent explosive in the reactor containment building.

j. Corrosive materials shall be doubly encapsulated in corrosion-resistant containers to prevent interaction with reactor components or pool water. Should a failure of the encapsulation occur that could damage the reactor, then the potentially damaged components shall be removed and inspected.

k. Cryogenic liquids shall not be used in any experiment within the reactor pool.

l. Fluids shall only be utilized in beamport loop experiments and shall be of types which will not chemically react in the event of leakage and shall be maintained at pressure and temperature conditions such that the integrity of the beam tube will not be impaired in the event of loop rupture.

m. The normal operating procedures shall include controls on the use or exclusion of corrosive, flammable, and toxic materials in experiments or in the reactor containment building. These procedural controls shall include a current list of those materials which shall not be used and the specific controls and procedures applicable to the use of corrosive, flammable, or toxic materials which are authorized.

Failure and Malfunctions Specification:

n. Where the possibility exists that the failure of an experiment could release radioactive gases or aerosols into the containment building atmosphere, the experiment shall be limited to that amount of material such that the airborne concentration of radioactivity when averaged over a year will not exceed the limits of 10 CFR 20. Exception: Fueled experiments that produce iodine-131 through iodine-135 and non-fueled experiments that are intended to produce iodine-131 (see Specifications 3.8.f and 3.8.h).

o. Experiments shall be designed and operated so that identifiable accidents such as a loss of primary coolant flow, loss of experiment cooling, etc., will not result in a release of fission products or radioactive materials from the experiment.

p. Experiments shall be designed such that a failure of an experiment will not lead to a direct failure of another experiment, a failure of a reactor fuel element, or to interfere with the action of the reactor safety and reactor control systems or other operating components.

q. No experiments shall be placed in the reactor pressure vessel or water annulus surrounding the center test hole other than for reactor calibration.

r. Cooling shall be provided to prevent the surface temperature of a submerged irradiated experiment from exceeding the saturation temperature of the cooling medium.

s. Irradiation containers to be used in the reactor, in which a static pressure will exist or in which a pressure buildup is predicted, shall be designed and tested for a pressure exceeding the maximum expected pressure by at least a factor of two (2).

t. The maximum temperature of a fueled experiment shall be restricted to at least a factor of two (2) below the melting temperature of any material in the experiment. First-of-a-kind fueled experiments shall be instrumented to measure temperature.

Other Specification:

u. Only movable experiments in the center test hole shall be removed or installed with the reactor operating. All other experiments in the center test hole shall be removed or installed only with the reactor shutdown. Secured experiments shall be rigidly held in place during reactor operation.

v. Non-fueled experiments that are intended to produce iodine-131 shall be processed in hot cells that are vented to the exhaust stack system through charcoal filters which are continuously monitored for an increase in radiation levels.

TS 3.8, Specification a, requires that the absolute value of the reactivity worth of each secured removable experiment be limited to 0.006 absolute reactivity (Δk/k) ($0.81). In accordance with the definition in TS 1.36, a secured experiment is any experiment that is rigidly held in place by mechanical means and cannot move while the reactor is operating. The secured experiments are removed individually once their restraints are removed. The NRC staff finds experiment removal is governed by the accident analysis described in SAR Section 13.2.1, involving the rapid insertion of reactivity into the reactor. In its response RAI (Ref. 103), the licensee indicates that a limit on the total reactivity worth of all secured experiments in not necessary since for MURR, these experiments are rigidly held in place, the accidental simultaneous removal of multiple secured experiments is not a credible accident scenario as it is controlled by procedure, and thus there is no need to limit the total reactivity worth of all secured experiments. The licensee understands that any experimental reactivity addition must satisfy the shutdown margin (SDM) requirement, which includes the reactivity for all (secured and non-secured) experiments (see TS 3.1, Specification b.2). The SDM was evaluated and found acceptable in SER Section 4.5.3. In addition, limits on the reactivity associated with experiments is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 3.8, Specification a, is acceptable.

TS 3.8, Specification b, requires the absolute value of the reactivity worth of all experiments in the center test hole to be limited to 0.006 Δk/k ($0.81). The NRC staff finds that this specification limits the reactivity worth to 0.006 Δk/k, which is bounded by the accident analysis in SAR Section 13.2.1, involving the rapid insertion of reactivity into the reactor. The NRC staff also finds that this reactivity addition is limited by the SDM requirements, which was evaluated in SER Section 4.5.3 and found acceptable. In addition, limits on the reactivity associated with experiments is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 3.8, Specification b, is acceptable.

TS 3.8, Specification c, requires that each moveable experiment or the moveable parts of any individual experiment to have a maximum absolute reactivity worth of 0.001 $\Delta k/k$ ($0.14). The NRC staff reviewed TS 3.8, Specification c, and finds that this specification helps to limit the individual reactivity worth to a very small amount (0.001 $\Delta k/k$) compared to the amount needed to achieve a prompt criticality (0.007 $\Delta k/k$, or $1.00). The NRC staff also finds that, because these types of experiments are a subset of unsecured experiments, the acceptability of their contributed individual worths is also limited by TS 3.8, Specification e, which is evaluated and found acceptable below. In addition, limits on the reactivity associated with experiments is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 3.8, Specification c, is acceptable.

TS 3.8, Specification d, requires that the reactivity of each unsecured experiment (defined in TS 1.43) be limited to 0.0025 $\Delta k/k$ ($0.34). The NRC staff reviewed TS 3.8, Specification d and finds that this specification for unsecured experiments is appropriately limited in the reactivity worth allowed and the acceptability of their contributed individual worth is limited by TS 3.8, Specification e, below. In addition, the limits on the reactivity associated with experiments is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 3.8, Specification d, is acceptable.

TS 3.8, Specification e, requires that the absolute value of the reactivity worth of all unsecured experiments be limited to 0.006 $\Delta k/k$ ($0.81). The NRC staff finds that because an unsecured experiment presents a higher risk for a reactivity insertion event, this total reactivity worth allowed was evaluated and found acceptable in SER Section 13.2.1. The NRC staff also finds that the rapid insertion of reactivity analysis demonstrates the acceptability of a reactivity change of 0.006 $\Delta k/k$ on the reactor power and concludes that the reactor control and RSSs are effective to mitigate any potential adverse consequences. In addition, the limits on the reactivity associated with experiments is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff finds that TS 3.8, Specification e, is acceptable.

TS 3.8, Specification f, requires each fueled experiment to be limited such that the total inventory of iodine-131 (I-131) through I-135 in the experiment is not greater than 150 Ci and the maximum strontium-90 inventory is no greater than 300 millicuries. The NRC staff finds that this specification helps to ensure that potential releases of radioactive material from the irradiation of fueled experiments do not exceed the analysis of the failed fueled experiment in SAR Chapter 13. In its response to RAI No. 7 (Ref. 33), the licensee provides the methodology and calculation results for a postulated release at the levels stated in TS 3.8, Specification f. The NRC staff reviewed the failed fueled experiment, which is the limiting radiological accident for MURR, and concludes that potential occupational and public doses were acceptable (see SER Section 13.1). In addition, the limits on the amount of radioactive materials associated with experiments is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 3.8, Specification f, is acceptable.

TS 3.8, Specification g, requires that fueled experiments containing inventories of I-131 through I-135 greater than 1.5 Ci or strontium-90 greater than 5 millicuries be contained in irradiation containers that satisfy the requirements of TS 3.8, Specification s, or be vented to the facility ventilation exhaust stack through high-efficiency particulate air and charcoal filters that are continuously monitored for an increase in radiation levels. The NRC staff finds that TS 3.8, Specification g, helps to limit any potential releases of a fueled experiment inventory by requiring such releases to be contained in irradiation containers that meet the requirements of TS 3.8, Specification s, or to be vented to the facility stack through appropriate filters. The NRC staff also finds that these controls help to ensure that fueled experiment malfunctions will have radiological consequences that are bounded by the accident analysis in SAR Section 13.1. The NRC staff finds that TS 3.8, Specification g, limits the fueled experiment inventory such the experiments are bounded by the failed fueled experiment analysis, as provided in the licensee's response to RAI No. 7 (Ref. 33), which demonstrates that any potential radiological doses to the MURR staff or to any member of the public is within the limits in 10 CFR 20.1201 and 10 CFR 20.1301. In addition, limits on the amount of radioactive materials associated with experiments is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 3.8, Specification g, is acceptable.

TS 3.8, Specification h, requires each nonfueled experiment that is intended to produce I-131 be limited such that the inventory of I-131 is not greater than 150 Ci. The NRC staff finds that TS 3.8, Specification h, helps ensure that the inventory of such an experiment is not greater than the inventory analyzed in the failed fueled experiment analyzed in the licensee's response to RAI No. 7 (Ref. 33) and evaluated in SER Section 13.1. In addition, limits on the amount of radioactive materials associated with experiments is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 3.8, Specification h, is acceptable.

TS 3.8, Specification i, requires that explosive materials not be irradiated, or allowed to generate in any experiment in quantities over 25 milligrams (mg) of trinitrotoluene (TNT)-equivalent explosives. The NRC staff finds that TS 3.8, Specification i, limits the effective amount of potentially explosive material that can be used in an experiment. The limit is 25 mg of TNT-equivalent explosive, which is a normalization technique for equating properties of an explosive amount of TNT. This technique requires a determination of the heat of explosion of the material in question and of the amount of that material that is equivalent to the heat of explosion of 25 mg of TNT for comparison. The NRC staff also finds that this specification helps to ensure that any MURR experiment methods apply this consideration to ensure proper usage and encapsulation. To ensure effective encapsulation, the NRC staff finds that it is important to correctly determine the yield strength of the material employed to ensure the integrity of the encapsulation. In its response to RAI No. 5, (Ref. 103), the licensee proposes a limit of 100 mg TNT-equivalent in the RCB. The NRC staff finds this limit acceptable because it will allow a reasonable amount for experiments without endangering the facility staff, and it is consistent the guidance in NUREG-1537, Part 1, Chapter 14, Section 3.8.2. Also, in its response to RAI No. 13.9.b (Ref. 19), the licensee provides the requirements for the encapsulation of explosive material and the calculation methods for ensuring proper encapsulation. The NRC staff reviewed the responses to RAIs and finds that the licensee's calculation considered the explosive potential and used the proper method to demonstrate the acceptability of the selected material for encapsulation. In addition, limits on the amount of explosive materials associated with experiments is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on its review, the NRC staff concludes that TS 3.8, Specification i, is acceptable.

TS 3.8, Specification j, requires corrosive materials to be doubly encapsulated in corrosion-resistant containers to prevent interaction with reactor components or pool water, and if a failure should occur that could damage the reactor, requires the potentially damaged components to be inspected. In its responses to RAI No. A.32 (Ref. 21), RAI No. 4.8.b (Ref. 24), and RAI No. 10.3 (Ref. 24), the licensee provides additional information pertaining to the controls used for the irradiation of potentially corrosive materials. In its response to

RAI No. 6 (Ref. No. 100), the licensee adds the provision to inspect components, which could be damaged if a failure of the encapsulation occurred. The NRC staff finds that TS 3.8, Specification j, helps to ensure that the irradiation of corrosive materials in experiments cannot lead to a failure that is chemically adverse to core components. The NRC staff also finds that this specification requires that materials that could be corrosive to MURR systems be double encapsulated to prevent interaction with reactor components or contaminate the PCS. In addition, the TS limits the amount of corrosive materials associated with experiments which is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on its review, the NRC staff concludes that TS 3.8, Specification j, is acceptable.

TS 3.8, Specification k, prohibits the use of cryogenic liquids in any experiment within the reactor pool. The NRC staff finds that TS 3.8, Specification k, restricts the use of cryogenic liquids so that they may not be used in experiments within the reactor pool which helps ensure that highly energetic (pressure-inducing) reactions do not occur from the introduction of such materials into the reactor system. Because the flux trap and in core locations are within the pool, this specification also prevents their use in the flux trap. In its response to RAI No. 10.1 (Ref. 24), the licensee indicates that experiments using cryogenic liquids are limited to external locations such as a beam port so any malfunction would not damage reactor pool components. The licensee also indicates that administrative controls would include the use of missile shielding and location control to enhance safety margins of any pressure-inducing event. The NRC staff finds the licensee's controls on cryogenic liquid to be acceptable. Based on the information above, the NRC staff concludes that TS 3.8, Specification k, is acceptable.

TS 3.8, Specification l, requires the use of fluids to be limited to loop experiments placed in the beamports, be of the type that will not chemically react in the event of leakage, and to be maintained at pressure and temperature conditions such that the integrity of the beam tube will not be impaired in the event of loop rupture. The NRC staff finds that TS 3.8, Specification l, helps to ensure that beam tubes will not be adversely affected by chemicals in the experiment by requiring chemical compatibility between fluid experiments and beamports. It also requires that the licensee evaluate the proposed loop experiment pressure and temperature conditions such that loop failure will not lead to impairment of the beam tube. Based on the information above, the NRC staff concludes that TS 3.8, Specification l, is acceptable.

TS 3.8, Specification m, requires that the normal operating procedures include controls on the use or exclusion of corrosive, flammable, and toxic materials in experiments or in the RCB. It also requires these procedural controls to include a current list of prohibited materials and the specific controls and procedures applicable to the use of corrosive, flammable, or toxic materials that are authorized. The NRC staff finds that TS 3.8, Specification m, helps ensure that prohibited materials are not used in experiments. In its response to RAI No. A.32 (Ref. 21), the licensee states that the materials are controlled by the reactor utilization request (RUR) procedure that invokes MURR Form FM-33, "Containment Building Restricted Materials." The NRC staff finds that this specification helps ensure that the core components are not damaged from experiment malfunction. Based on the information above, the NRC staff concludes that TS 3.8, Specification m, is acceptable.

TS 3.8, Specification n, requires that, if the failure of an experiment could release radioactive gases or aerosols into the containment building atmosphere, the experiment shall be limited to that amount of material such that the airborne concentration of radioactivity when averaged over a year will not exceed the limits of Table 1, "Occupational Values," in Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage," to

10 CFR Part 20. An exception is stated for fueled experiments that produce I-131 through I-135 and nonfueled experiments that are intended to produce I-131 (see TS 3.8, Specifications f and h). The NRC staff finds that this specification provides an effective radiological dose limit to ensure that an experiment failure could not release radioactive gases or aerosols into the containment atmosphere in excess of the limits of Table 1 in Appendix B to 10 CFR Part 20. The exception allows the licensee to conduct experiments that produce Iodine isotopes above the limits in Table 1 in Appendix B to 10 CFR 20, but imposes additional controls required in TS 3.8, Specifications f and h, and which were found acceptable by the NRC staff, as discussed above. Based on its review, the NRC staff concludes TS 3.8, Specification n, is acceptable.

TS 3.8, Specification o, requires that experiments be designed and operated so that identifiable accidents, such as a loss of primary coolant flow or loss of experiment cooling, will not result in a release of fission products or radioactive materials from the experiment. The NRC staff finds that TS 3.8, Specification o, requires experiment design to consider the potential for postulated reactor accidents such as a loss of flow accident or loss-of-coolant accident and helps to ensure that the experiment design considers the potential for an accidents and includes provisions to avoid a release of fission products or radioactive materials. The NRC staff finds that this specification helps ensure that anticipated reactor transients will not result in radiological consequences that are greater than the accident analysis in SAR Chapter 13. Based on its review, the NRC staff concludes TS 3.8, Specification o, is acceptable.

TS 3.8, Specification p, requires that experiments be designed such that a failure of an experiment will not lead to a direct failure of another experiment or a failure of a reactor fuel element nor interfere with the action of the reactor control and safety systems, or other operating components. The NRC staff finds that TS 3.8, Specification p, helps ensure that the experiment design prevents the failure of an experiment from leading to a failure of another experiment, or reactor fuel, or interfering with the operation of the RCS, or other safety components. In addition, the NRC staff finds that this specification helps ensure that failure of individual experiments will not cause cascading malfunctions because the experiment review must consider the failure mode for the experiment and ensure that it cannot fail in a way that causes the failure of another experiment, degrade or damage a fuel plate, or compromise the operability of any RCS component (e.g., control blade motion or valve actuation), or compromise the operation of any other reactor component. In addition, limits on the design of an experiment such that a failure will not have adverse consequences is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes TS 3.8, Specification p, is acceptable.

TS 3.8, Specification q, prohibits experiments from being placed in the reactor pressure vessel or water annulus surrounding the center test hole other than for reactor calibration. The NRC staff finds that TS 3.8, Specification q, helps ensure that the region inside the reactor pressure vessel (where the fuel is located) or the annulus of water surrounding the flux trap will not be used as an experiment location and prevents the possibility of experiments generating gases or causing void in these high reactivity worth regions. The NRC staff also finds that the exception will allow the MURR staff to perform reactor calibration, which is an operational requirement (physics testing), is not an experiment, and helps to support the neutronics calculations performed to characterize core performance. Based on the information above, the NRC staff concludes TS 3.8, Specification q, is acceptable.

TS 3.8, Specification r, requires that cooling be provided to prevent the surface temperature of a submerged irradiated experiment from exceeding the saturation temperature of the cooling medium. The NRC staff finds that TS 3.8, Specification r, helps control the surface temperature of a submerged experiment to prevent localized boiling, reduce the likelihood of an experiment boundary failure, and reduce the potential for vaporization of radioactive material into the atmosphere. Additionally, the NRC staff also finds that this specification helps ensure that void formation and fluid conditions are not created that could conflict with assumptions in the thermal-hydraulic analysis in provided in SER Section 4.6. Based on the information above, the NRC staff concludes TS 3.8, Specification r, is acceptable.

TS 3.8, Specification s, requires that irradiation containers used in the reactor, in which a static pressure will exist or in which a pressure buildup is predicted, be designed and tested for a pressure exceeding the maximum expected pressure by at least a factor of 2. The NRC staff finds TS 3.8, Specification s, helps ensure that irradiation containers to be used in an experiment shall be designed and tested to withstand double the maximum expected pressure to prevent container failure. In its response to RAI No. 13.9.b (Ref. 19), the licensee provides the methods and calculations used to determine the container pressure, which the NRC staff finds acceptable. In addition, the NRC staff finds that limits on the design of the experimental container used in experiments involving a static pressure is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on its review of TS 3.8, Specification s and response to RAI the NRC staff concludes TS 3.8, Specification s, is acceptable.

TS 3.8, Specification t, requires that the maximum temperature of a fueled experiment shall be restricted to at least a factor of 2 below the melting temperature of any material in the experiment. This specification also requires that first-of-a-kind fueled experiments be instrumented to measure temperature. The NRC staff finds that TS 3.8, Specification t, helps ensure that the temperature of a fueled experiment is controlled so that the maximum value is a factor of 2 below the expected melting temperature of any material in the experiment, including the fuel. The NRC staff also finds that this specification also helps to ensure that temperature changes do not lead to unexpected changes in phase, pressures, off-gas rates, or other conditions that could lead to unexpected or detrimental consequences to the reactor or release of radioactive material. Based on the information above, the NRC staff concludes that TS 3.8, Specification s, is acceptable.

TS 3.8, Specification u, requires that only movable experiments (as defined in TS 1.14) in the center test hole be removed or installed with the reactor operating and restricts the removal or installation of all other experiments in the center test hole unless the reactor is shutdown. This specification also requires secured experiments to be rigidly held in place during reactor operation. The NRC staff finds that TS 3.8, Specification u, helps ensure that only movable experiments in the center test hole are removed or installed when the reactor is operating to avoid reactivity changes that have not been analyzed or bounded by the reactivity analysis in SER Chapter 13. Based on the information above, the NRC staff concludes that TS 3.8, Specification u, is acceptable.

TS 3.8, Specification v, requires that experiments that are intended to produce I-131 be processed in hot cells that are vented to the exhaust stack directly through charcoal filters that are continuously monitored for an increased radiation level. The NRC staff finds that TS 3.8, Specification v, helps ensure that any iodine released from the hot cell is mitigated by the charcoal filters and dispersion of the exhaust stack and dilution of the main exhaust flow. The NRC staff finds that these controls are effective to minimize any potential iodine dose to the public. The NRC staff reviewed the effectiveness of the charcoal filters in reducing any potential doses to any members of the public during its review of LA No. 37. The NRC staff found that TS 3.8. Specification v, to be acceptable. The NRC staff also finds that any hypothetical accident doses to the public are bounded by the failed fueled experiment analysis, as provided in the licensee's response to RAI No. 7 (Ref. 33), which the NRC staff evaluated and found acceptable in SER Section 13.1. Based on the information above, the NRC staff concludes that TS 3.8, Specification v, is acceptable.

The NRC staff reviewed TS 3.8, Specifications a through v, and finds that they are supported by the safety analyses in SAR Chapter 13, and are consistent with the guidance in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," issued February 1996 (Ref. 51). Based on the information provided above, the NRC staff concludes that TS 3.8, Specifications a through v, are acceptable.

TS 4.8 Experiments

TS 4.8 states:

Specification:

a. The criteria of Specification 3.8 shall be evaluated and found acceptable prior to inserting an experiment in the reactor or its experimental facilities.

b. The reactivity worth of an experiment shall be estimated or measured, as appropriate, before reactor operation with said experiment.

TS 4.8, Specification a, requires that the criteria of Specification 3.8 be evaluated before inserting an experiment in the reactor or its experimental facilities. The NRC staff finds this specification helps ensure that the applicable requirements in TS 3.8, Specifications a through v, are considered before the performance of an experiment to avoid conditions adverse to safe operation. The NRC staff finds that this surveillance interval is consistent with the guidance in NUREG-1537, and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.8, Specification a, is acceptable.

TS 4.8, Specification b, requires the reactivity worth of an experiment to be estimated or measured, as appropriate, before reactor operation with the experiment. The NRC staff finds that TS 4.8, Specification b, helps ensure that the appropriate experimental worth is determined before the performance of the experiment to ensure compliance with TS 3.8, Specifications a through e. The NRC staff finds that the surveillance interval is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 4.8, Specification b, is acceptable.

The NRC staff reviewed TS 4.8, Specifications a and b, regarding the controls applicable to experiments in the MURR facility and finds that the surveillance requirements support the limiting conditions for operation in TS 3.8, meet the regulations in 10 CFR 50.36, and the surveillance intervals are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes TS 4.8, Specifications a and b, are acceptable.

10.4 Conclusions

The NRC staff reviewed the experimental facilities associated with the MURR facility and finds that the design and use of the experimental facilities, the review process for experiments, and the applicable TSs, provides assurance that appropriate precautions will minimize any associated risk. The NRC staff also finds that the licensee has an independent organization for experiment review (RAC), which has a diverse and independent membership, as well as acceptable experience and expertise. The procedures and methods used at the MURR facility help to ensure that a detailed review of all potential safety and radiological risks that an experiment may pose to the MURR staff and the public have sufficient administrative controls to protect the operations personnel, experimenters, and general public from potential hazards caused by the experiments. The NRC staff finds that the expected radiation doses from experiment do not exceed the limits of 10 CFR Part 20, are consistent with the facility ALARA Program, and that the TSs required by 10 CFR 50.36, ensure acceptable implementation of the review and approval of experiments.

Based on its review of the information above, the NRC staff concludes that experimental facilities and the review process provide reasonable assurance that the use of experiments or experimental facilities will not damage the fuel, will ensure that any potential release of radioactive material is below regulatory limits, and will not endanger the health and safety of the public or facility staff.

11 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

11.1 Radiation Protection

SAR Chapter 11 describes the Radiation Protection Program at the University of Missouri-Columbia Research Reactor (MURR or the reactor). Activities involving radiation at the MURR facility are controlled through the Radiation Protection Program, which must meet the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) 20.1101, "Radiation Protection Programs," and minimize radiation exposure. The regulation, 10 CFR 20.1101 requires, in part, that each licensee develop, document, and implement a radiation protection program commensurate with the scope of extent of licensed activities and to ensure compliance with 10 CFR Part 20, and requires the licensee to 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 as low as is reasonably achievable (ALARA).

SAR Section 9.5 indicates that the licensee holds NRC-issued Broad Scope Materials License No. 24-00513-32, issued under 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," which allows the University of Missouri-Columbia to possess those radioisotopes listed in the license. The facility operating license, R-103, provides the authority and responsibilities associated with the reactor and the radioactive materials produced by the reactor. Broad Scope Materials License No. 24-00513-32 authorizes the facility to use licensed materials in support of research and development, which may not currently be covered under the facility operating license.

SAR Section 9.5.1, states that MURR facility handles or produces a spectrum of by product materials. The MURR facility has been licensed to handle isotopes ranging from tritium through transuranic elements. The MURR facility utilizes radioisotopes it produces by the activation of materials, radioactive sources, and that are used to calibrate and verify the operation of radiation detection instruments. Isotopes that the MURR facility is currently licensed to possess and use under the Broad Scope Material License are summarized in SAR Table 9-2. Because byproduct material continuously changes as part of the normal operation of the reactor and the experimental program, the licensee indicates that the information presented in SAR Table 9-2 should be considered representative rather than an exact listing of radioisotopes (see SER Section 9.5)

In its response to RAI No. 9.3.a (Ref . 17) the licensee states that radioactive materials within the reactor facility, whether such materials are licensed under the broad scope materials license or the facility operating license, are subject to the same radiation protection controls. However, the organizational structure that provides the review and approval process for the use of radiation sources and radioactive materials under the broad scope materials license differs from that of the facility operating license. The reactor and health physics managers review and approve the uses of radioactive materials produced by the reactor. The Isotope Use Subcommittee of the Reactor Advisory Committee has advisory responsibility for the actions of the reactor and health physics managers with regard to the use of radioactive materials and radiation sources under the facility operating license. The Radiation Safety Committee (RSC) reviews and approves the uses of radioactive materials and radiation sources that are covered by the broad scope materials license. The radiation safety officer (RSO) maintains the records of the review and approval process. The basic aspects of the radiation protection program include occupational and general public exposure limits, surveys and monitoring, and personnel dosimetry.

The NRC inspection program routinely reviews radiation protection and radioactive waste management at MURR. A review of NRC inspection reports (IRs) from the years 2010 to 2016 (Refs. 45, 46, 47, 48, 49, 50, and 84) found no violations of radiation protection or radioactive waste management at the MURR facility.

11.1.1 Radiation Sources

SAR Chapter 11 describes the radiation sources, including the inventories of each, their physical form, and location. The MURR Radiation Protection Program and Radioactive Waste Management Program monitor and control the radiation sources; these sources are categorized as airborne, liquid, or solid.

Airborne Radiation Sources

SAR Section 11.1.1.1 indicates that during normal operations, the reactor generates neutrons for a number of research purposes. Beam ports for experiments allow neutrons to pass from the reactor through a side of the reactor pool and through the biological shield to experiment areas. The facility employs a number of other irradiation methods to use the radiation from the reactor core. The reactor also has a pneumatic transfer system for in-core experiments, which can create radioactive materials.

SAR Section 11.1.1.1 also indicates that MURR airborne sources consist mainly of argon-41 (Ar-41), which has a radioactive half-life of 1.8 hours and accounts for greater than 99 percent of the radioactivity released through the facility ventilation exhaust stack. In its response to Request for Additional Information (RAI) No. 11.1 (Ref. 17), the licensee describes the potential for the release of Ar-41. Minimization of Ar-41 production is accomplished by minimizing the circulation of ambient air into areas subjected to neutron bombardment. The principal sources of Ar-41 generation are the pneumatic-tube (p-tube) system, thermal column, and the beam ports. Almost all (98 percent) of the Ar-41 is generated in the p-tube terminus located in the graphite reflector region. SAR Section 10.3.4 describes the operation of the p-tube. The p-tube air contains Ar-41 and is exhausted from the system through a high-efficiency particulate air filter to the facility ventilation exhaust stack. Ar-41 produced in the thermal column and beamports is ducted to the 16-inch (40.64-centimeter) hot exhaust line, which also exhausts to the facility ventilation exhaust stack.

In its response to RAI No. 11.1 (Ref. 17) concerning the potential for a p-tube pipe failure leading to releases of Ar-41 into the containment or laboratory buildings, the licensee references the p-tube system description in SAR Section 10.3.4 and states that there are no credible failures that could lead to releases into the building. The licensee states that the direction of airflow in the p-tube system ensures that the airflow is always into the p-tube system should a leak develop in the sample carrier tubing. Additionally, the solenoid-operated control valves are positioned (de-energized state) such that a continuous flowpath for air exists through the sample carrier tubing even when the p-tube system is secured; therefore, the pneumatic system is always at a negative pressure regardless if the system is in use or not.

Based on its review of the information above, the NRC staff concludes that the description and characterization of the airborne radiation sources at the MURR are reasonable for a non-power reactor of this type and size and that the information provides sufficient details to evaluate consequential doses to the members of the public and the operational personnel.

Occupational Doses from Ar-41

The licensee states that a limited amount of Ar-41 can be found in the reactor containment building (RCB) during reactor operation. SAR Section 11.1.1.1.2 provides measured Ar-41 concentrations in the RCB, which were less than 1 percent of the derived air concentration limit of $3.0x10^{-6}$ microcuries per milliliter (μCi/ml), as listed in Table 1, "Occupational Values," in Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage," to 10 CFR Part 20, "Standards for Protection against Radiation." Normally, there are 12 staff members (4 reactor operators, 4 operations management staff, 1 administrative assistant, and 3 researchers) within the containment building at any one time. At this level of exposure to Ar-41 within the containment, the licensee calculated a 12-staff worker dose of 0.60 person-rem (roentgen equivalent man), or an individual-staff average dose of 50 millirem (mrem), annually.

Based on its review of the information above, the NRC staff concludes that the estimated occupational dose from Ar-41 to be reasonably conservative because not all the licensee staff will stay in the containment 100 percent of the time annually.

Public Dose from Ar-41

As stated above, the radioactive air containing Ar-41 from the p-tube system, thermal column, and beam ports is discharged to the ventilation exhaust stack, which is 70 ft (21 m) above the grade level. In SAR Appendix B, as supplemented by its responses to RAIs (Ref. 22 and Ref. 33), the licensee provides detailed calculations of the public dose estimates from Ar-41 released from the exhaust stack.

TS 3.7, Specification b (see SER Section 11.2.2) requires that the annual average concentration of Ar-41 in airborne effluents released from the exhaust stack be limited to 350 times the 1×10^{-8} μCi/ml air effluent concentration (AEC) value for Ar-41 listed in Appendix B, Table 2 of 10 CFR Part 20 (i.e., 3.5×10^{-6} μCi/ml). Using the maximum value from TS 3.7, Specification b, with the maximum main stack exhaust flow rate of 30,500 ft^3/min, the licensee calculated the maximum possible Ar-41 release rate of 5×10^{-5} Ci/sec. Based on this maximum release rate and using average meteorological data for the Columbia, Missouri area for the period from 1960 through 1969, the licensee calculated the estimated annual dose to a member of the public at two locations directly north of the facility stack. The licensee chose the locations directly north because the wind most frequently blows from south to north, based on the meteorological data. The locations chosen were the EPZ boundary, at 150 m (492 ft) from the stack, and the nearest residence, at approximately 760 m (2,493 ft) from the stack. In response to RAI No. 5.b (Ref. 33), the licensee indicates that the calculated annual doses at the EPZ boundary and the nearest residence would be 0.46 mrem and 2.35 mrem, respectively, for a member of the public who remained at those locations for an entire year. The NRC staff finds that the licensee's calculated doses are below the annual limit of 100 mrem for members of the public stated in 10 CFR 20.1301, and are also within the 10 mrem ALARA constraint on dose to members of the public from air emissions of radioactive material to the environment stated in 10 CFR 20.1101(d).

The licensee's calculations used effective stack heights to account for both plume rise (i.e., the increase in the effective release height due to the upward momentum of the effluent air released from the stack) and the elevation differences between the stack and the receptor locations. In its response to RAI No. 5.a (Ref. 33), the licensee states that it calculated the plume rise from the stack diameter, stack exhaust velocity, and wind speed using the Davidson equation. The licensee uses the Davidson equation in "Briggs, G.A., *Plume Rise,* AEC Critical Review Series, U.S. Atomic Energy Commission, Division of Technical Information, 1969" (Ref. 78). In this report, Briggs indicated that the Davidson equation underestimates the plume rise. The licensee states that the underestimation in the plume rise results in decreasing the effective stack height and leads to an overestimation of the calculated radiological doses.

The NRC staff reviewed the information above and finds that the Davidson equation under-predicts the plume rise (resulting in less dispersion and higher radionuclide concentrations), which would lead to an overestimation of the dose in an elevated release. The NRC staff also finds that the licensee's adjustment to the effective release heights to account for the receptor locations' elevation above the grade level of the base of the stack to be consistent with established practice, and conservative because it decreases the effective release heights, resulting in higher calculated doses.

The NRC staff also evaluated the applicability of the 1960 to 1969 meteorological data (provided in SAR Appendix B) used by the licensee for the Ar-41 dose calculations, to ascertain if changes in weather patterns may have occurred during this time period. In response to RAI No. 7.e (Ref. 33), the licensee provides its review of two sets of meteorological data, one from 1961 through 1990 and another from 1970 through 1990, and states that the averaged wind rose data for these two periods trended similarly to a set of data from 1961 through 1969. The licensee provides the meteorological data sets as Attachments 10 through 12 to its RAI responses (Ref. 33). The NRC staff reviewed the responses to RAI and finds that the licensee provided adequate justification for its use of the 1960 to 1969 meteorological data.

In addition, the NRC staff reviewed the locations chosen by the licensee for the Ar-41 dose calculations. The licensee chose the EPZ boundary, at a distance of 150 m (492 ft) from the exhaust stack, and the nearest permanently inhabited residence, at a distance of 760 m (2,493 ft) from the exhaust stack, to represent the locations of the highest annual public dose. In its response to RAI No. B.1 (Ref. 22), the licensee states that the intent of the calculations for the different locations was not to indicate the maximum dose that an individual at any location might receive from Ar-41, but to indicate the maximum dose at locations that are reasonably expected to be occupied on a permanent or semi-permanent basis. For the nearest residence (760 m), members of the public are assumed to be present for the entire year. For the EPZ boundary (150 m), there is an office building associated with the University of Missouri-Columbia Research Park, which is occupied approximately 40 to 50 hours per week by members of the public (office workers). However, in its dose calculation, the licensee conservatively assumed the office workers (members of the public) would be present in the building for the entire year.

The licensee states that the actual highest calculated annual Ar-41 dose (assuming full-time occupancy) was between the EPZ boundary and the nearest residence, at a location approximately 350 m (1,148 ft) north of the facility. The licensee calculated the dose for full-time occupancy at this location to be 4.4 mrem, which was higher than the doses calculated for the EPZ boundary (0.46 mrem) and nearest residence (2.35 mrem). However, this location, 350 m (1,148 ft) north of the facility, is in an unoccupied area (no building), so full-time occupancy at this location is not expected. Near this location, there are two maintenance sheds affiliated with the University of Missouri Golf Course. The licensee conservatively assumed a 24 percent occupancy factor for this location (based on a member of the public working 40 hours per week), and calculated the annual dose at this location to be 1.1 mrem. This is below the annual dose calculated for the nearest residence (2.35 mrem). Therefore, based on the occupancy assumptions used in the calculations, the licensee concluded that the highest expected annual dose from Ar-41 occurs at the location of the nearest residence.

The NRC staff reviewed the SAR and the responses to RAI and finds the licensee's justification for the use of the location of the nearest resident as the maximum annual dose to any member of the public from Ar-41 to be acceptable.

NRC Staff Confirmatory Calculations of Ar-41 Doses to Members of the Public

The NRC staff performed confirmatory calculations of the doses from Ar-41 to members of the public located at the EPZ boundary and the nearest residence. The NRC staff's calculations used the maximum Ar-41 release rate of 5×10^{-5} Ci/sec (derived from TS 3.7, Specification b), the effective stack heights, and the 1960 to 1969 meteorological data provided by the licensee. For its confirmatory calculations, the NRC staff used the meteorological data provided by the licensee to determine the dispersion parameters that would be used. The NRC staff performed calculations using Ar-41 DCFs of 8.84×10^{9} mrem/year per Ci/m^3 and 7.59×10^{9} mrem/year per Ci/m^3 based on NRC RG 1.109 (Ref. 76) and Federal Guidance Report (FGR) No. 12 (Ref. 77), respectively. The results of the licensee's calculations and the NRC staff's confirmatory calculations are provided in Table 11-1 below.

Table 11-1 Annual Ar-41 Doses to Members of the Public

Location	Calculated Dose (Licensee)	Calculated Dose (NRC Staff, RG 1.109 DCF)	Calculated Dose (NRC Staff, FGR No. 12 DCF)	10 CFR 20 Dose Limit
Nearest Resident (760 meters)	2.35 mrem	4.15 mrem	3.57 mrem	**100 mrem**
EPZ Boundary (150 meters)	0.46 mrem	0.81 mrem	0.69 mrem	**100 mrem**

In SAR Appendix B, the licensee states that it used a dose conversion factor (DCF) for Ar-41 of 8.84×10^{9} mrem/year per Ci/m^3 for the Ar-41 dose calculations provided in SAR Appendix B, in accordance with the guidance in Table B-1 of NRC RG 1.109 (Ref. 75). In response to RAI No. 5.b (Ref. 33), the licensee provides updated dose calculations that corrected minor issues in the dispersion parameters used for the calculations in SAR Appendix B. The NRC staff noted that the updated calculations also appeared to have been revised to use a lower DCF for Ar-41 of 5×10^{9} mrem/year per Ci/m^3. The NRC staff finds that the licensee may have derived this DCF value based on incorrect assumption that the AEC for Ar-41 in Appendix B, Table 2 of 10 CFR Part 20 is based on a dose of 50 mrem per year. Since the AECs for radionuclides in which submersion (external dose) is limiting, such as Ar-41, in Appendix B, Table 2 of 10 CFR Part 20 are based on a dose of 100 mrem per year, a derivation of a DCF for Ar-41 based on 50 mrem would be incorrect. The licensee's use of the lower DCF explains the discrepancy between the updated licensee calculation provided in response to RAI No. 5.b (Ref. 33) and the NRC staff's calculation. However, as the NRC staff's calculations above demonstrate, Ar-41 doses calculated using accepted Ar-41 DCFs from NRC RG 1.109 (Ref. 75) and FGR No. 12 (Ref. 77) are within regulatory limits.

Although the licensee calculated the annual dose at the nearest residence based on the maximum annual release of Ar-41 allowed by TS 3.7, Specification b, based on the NRC staff's review of the licensee's annual reports for 2010 to 2015 (Refs. 39 through 44), historical releases have been below the TS limit (see Figure 11-1 below). During 2010 through 2015, the annual average concentration of Ar-41 in airborne effluents released from the stack ranged from 1.58×10^{-6} μCi/ml (45.1 percent of the TS limit) to 2.73×10^{-6} μCi/ml (78.1 percent of the TS limit). These annual average concentrations would have resulted in doses that are proportionally lower than the dose corresponding to the release of Ar-41 at the TS limit.

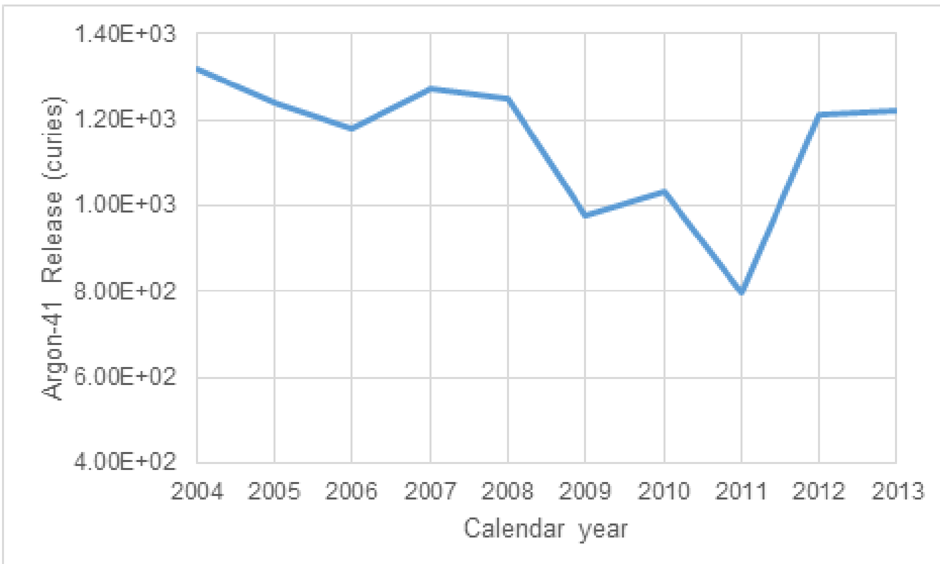

Figure 11-1 Ar-41 production during MURR operation conditions

The NRC staff reviewed the analyses provided in the SAR and responses to RAI related to public doses from Ar-41. The NRC staff reviewed the licensee's calculation methodologies and assumptions, and finds that they are conservative and consistent with accepted industry practices, except for the DCF used. The NRC staff used appropriate DCFs in it confirmatory calculations of the dose from Ar-41 to members of the public at the nearest residence and the EPZ boundary. Based on the confirmatory analyses that demonstrate that public doses from MURR routine Ar-41 releases would be within 10 CFR Part 20 limits, and the NRC staff's review of historical Ar-41 releases at the facility as described in the licensee's annual operating reports, the NRC staff concludes that the licensee's Ar-41 releases are acceptable for the renewal period.

Liquid Radioactive Sources

SAR Section 11.1.1.2 indicates that liquid radiation sources at the MURR consist primarily of activation products in the coolant and reactor components, principally tritium, nitrogen-16, and Ar-41, but they also include magnesium-56 and sodium-24. Nitrogen-16 has a 7-second half-life and is only a radiation hazard during reactor operations or immediately after reactor shutdown. According to the SAR, the reactor pool water is normally kept clean, although occasionally contaminants in the water may create activation products. A demineralizer is used to maintain water purity. Tritium may be produced in the reactor pool in small quantities but is limited by the small natural abundance of deuterium in light water. Liquid radiation sources at the MURR facility also include laboratory wastes and, as states in the SAR, is the most significant source in terms of volume. All potentially radioactive liquid wastes are directed to a liquid waste retention and disposal system located on the below-grade level of the laboratory building. Liquid waste is then retained or chemically treated until an assay indicates activity levels are less than the limits specified in 10 CFR Part 20 for disposal by release into sanitary sewerage. According to the annual operating reports reviewed by the NRC staff, for the years 2010 through 2015 (Refs. 39 through 44), tritium normally accounts for about 81 percent of the total activity released each year (see SAR Table 11-6).

Based on the above discussion, the NRC staff concludes that the description and characterization of the liquid radiation sources at the MURR facility are reasonable for a non-power reactor. The information is sufficient to evaluate the facility's Radiation Protection Program and controls described in the remainder of this section.

Solid Radioactive Sources

MURR operations generate solid radioactive materials. Chief among these are the spent fuel assemblies. After irradiation in the core, the spent fuel assemblies are stored in fuel racks. Chapter 4 of this SER discusses in-core sources. SER Chapter 9 discusses spent fuel movement and storage.

Other solid radioactive sources include reactor resins and filters, reactor components, experiment components from high flux locations, and activated samples. Solid radioactive waste is disposed of in accordance with appropriate NRC regulations and is transferred to organizations authorized to receive the material.

Based on the above discussion, the NRC staff concludes that the description and characterization of the solid radiation sources at the MURR facility is reasonable for a non-power reactor. The information is sufficient to evaluate the radiation protection program and controls described in the remainder of this section.

Radioactive Sources - Conclusions

The NRC staff reviewed the description of potential radiation sources and associated doses, including the inventories, chemical and physical forms, and locations of radioactive materials, and other facility radiation and operational parameters related to radiation safety presented in the SAR. This review included a comparison of the bases for identifying potential radiation safety hazards with the process and facility descriptions to verify that such hazards were accurately and comprehensively identified. This review and evaluation confirm that the SAR identifies the potential radiation safety hazards associated with MURR, and provides an acceptable basis for the development and independent review of the facility's Radiation Protection Program.

11.1.2 Radiation Protection Program

SAR Section 11.1.2 describes the MURR Radiation Protection Program required by 10 CFR 20.1101. This program includes the stated policy to employ the ALARA concept in all operations at MURR. Radiation protection activities at MURR are performed by the Health Physics Branch, which is supervised by the reactor health physics manager. This position reports to the MURR Reactor Facility Director through the Associate Director, Regulatory Affairs Group. However, there is a communications/consultation line from the reactor health physics manager to the Office of the Provost. This line of communications/consultation allows the reactor health physics manager access to upper University management if reactor facility management does not address radiation protection concerns to the satisfaction of the health physics manager. This reporting chain is described SAR Chapter 12 in response to RAI No. 11.7 (Ref. 17) and is detailed in TS 6.1. SER Chapter 12 discusses TS 6.1.

In its response to RAI No. 11.6 (Ref. 17), the licensee stated that normal staffing of the Health Physics Branch is seven full-time individuals. The staff comprises the health physics manager, two staff health physicists, and four health physics technicians. SAR Section 11.1.2.1 also states that the staff includes an assistant reactor health physics manager.

SAR Section 11.1.2.2 states that the licensee also uses an Isotope Use Subcommittee that acts as an advisory group to the Reactor Advisory Committee, as described in SAR Section 10.4.4, regarding matters relating to the custody and use of radiation and radioisotopes within the MURR facility.

SAR Section 11.1.2.3 also describes the MURR RSC, which is responsible for establishing the policies relating to the management of programs using radioactive material and radiation sources that are covered by the broad scope material license. The RSC reports to the Chancellor through the Vice Chancellor for Research on all matters pertaining to the safe use of radiation in these programs. The Vice Chancellor for Research is responsible for appointing members to the RSC and for assigning a Chairman.

In its response to RAI No. 11.7 (Ref. 17), the licensee stated that MURR has an RSO for radiation protection duties under the University's broad scope materials license. The RSO and reactor health physics manager may be the same individual. The RSO is responsible for the implementation of the policies established by the RSC. The RSO is appointed by the Chancellor upon recommendation of the Vice Chancellor for Research. The RSO reports to the Reactor Facility Director.

SAR Section 11.1.2.5 describes radiation safety training which is given to all individuals who work with radioactive materials or receive personnel monitoring. The training covers basic health physics principles, as well as the regulations of 10 CFR Part 19, "Notices, Instructions and Reports to Workers: Inspection and Investigations"; 10 CFR Part 20; and 10 CFR Part 21, "Reporting of Defects and Noncompliance," and rules regarding use of radioactive material at the University of Missouri-Columbia. The licensee indicates that this training is commensurate with the level of activities of the individual and the potential for radiation exposure. In its response to RAI No. 11.2 (Ref. 17), the licensee corrected the statement in SAR Section 11.1.2.5 to provide Class I training for all individuals requesting permission to direct or supervise the work of others in using radioactive materials under the reactor license. In addition, the licensee indicates that all training at MURR applies to and is sufficient to allow work with byproduct material for either the broad scope material license or the reactor license, and this practice has been done in the past and the licensee plans to continue this practice into the future.

SAR Section 11.1.2.6 indicates and TS 6.4 requires that written procedures are used for radiological control activities. The reactor health physics manager approves health physics procedures, and the Reactor Procedures Review Subcommittee reviews changes to these procedures. The reactor health physics manager reviews the procedures annually.

SAR Section 11.1.2.7 discusses and TS 6.2 (SER Section 12.2) requires health physics audits. MURR management or its authorized delegates must perform periodic audits to verify the adequacy and the implementation of the programs and operating procedures designed to ensure that radiation safety and compliance with applicable regulations are maintained. The audits must be conducted annually and include a selective (but comprehensive) examination of logs, operating records, data sheets, and other documents.

Records relating to personnel dosimetry or exposure investigations, as well as effluent records, are retained for the life of the facility as required by 10 CFR 20.2106, “Records of Individual Monitoring Results,” and 10 CFR 20.2107, “Records of Dose to Individual Members of the Public,” and are discussed in SAR Section 11.1.2.8 and TS 6.7. Facility surveys are retained for a minimum of 3 years. This meets the requirements of 10 CFR 20.2103, “Records of Surveys.”

The NRC staff reviewed the structure and methods of the Radiation Protection Program for the MURR facility and finds that it is consistent with the guidance in American National Standards Institute/American Nuclear Society (ANSI/ANS)-15.11-1993, “Radiation Protection at Research Reactor Facilities,” issued 1993 (Ref. 73). The NRC inspection routinely reviews the Radiation Protection Program at MURR facility. Review of the annual operating reports from the years 2010 to 2015 (Refs. 39, 40, 41, 42, 43, and 44) and NRC inspection reports (IRs) from the years 2010 to 2016 (Refs. 45, 46, 47, 48, 49, 50, and 84) demonstrates that adequate measures are in place to minimize radiation exposure to personnel and provide adequate protection against operational releases of radioactivity to the environment.

The NRC staff’s review finds that the MURR Radiation Protection Program acceptably describes:

(1) the roles, responsibilities, authorities, organization, and staffing of the radiation protection organization,

(2) the roles, responsibilities, authorities, staffing, and operation of committees responsible for the review and audit of the Radiation Protection Program,

(3) the effectiveness and comprehensiveness of the radiation protection training program,

(4) radiation protection plans and information that form the bases of procedures and the management systems employed to establish and maintain them,

(5) the effectiveness and comprehensiveness of the Radiation Protection Program for independent oversight reviews and audits of the program,

(6) the effectiveness and comprehensiveness of the process to evaluate the Radiation Protection Program to improve the program and the process to examine problems and incidents at the facility, and

(7) the management of records relating to the Radiation Protection Program.

Based on the above findings, the NRC staff concludes that the Radiation Protection Program at the MURR facility complies with applicable requirements and that the MURR Radiation

Protection Program provides reasonable assurance that facility staff and the public will be protected from the effects of radiation and that radioactive materials are handled safely.

11.1.3 As Low As Reasonably Achievable Program

SAR Section 11.1.3 states that the University of Missouri-Columbia has a defined ALARA policy for exposure to radiation that meets the requirements of 10 CFR 20.1101(b). The ALARA policy states that MURR is dedicated to the fundamental principle of maintaining individual exposures and radioactive effluents ALARA. The program to implement this policy is based on the guidelines in ANSI/ANS-15.11-1993. The program is applied through written procedures and guidelines. The licensee reviews all proposed experiments and operational procedures for ways to minimize potential exposure to personnel. The MURR Health Physics Branch participates in experiment planning to minimize both personnel exposure and the generation of radioactive waste. Additionally, unanticipated or unusual reactor-related exposures are investigated to develop methods to prevent recurrence.

The review of controls for limiting access and personnel exposure in the MURR facility provides reasonable assurance that radiation doses to the public and facility personnel will be ALARA. SAR Table 11-11 lists Investigation Levels I and II. Investigation levels for occupational radiation exposures and effluent concentrations are established. When these levels are exceeded, the Health Physics Branch initiates a review or investigation that focuses on determining the cause of the exposure so that appropriate ALARA actions, if any, can be applied. In its response to RAI No. 11.5 (Ref. 17), the licensee stated that Section 7, “MURR ALARA Program,” of MURR Policy Manual POL-3, “Radiation Protection Program,” describes the investigation levels.

The NRC staff’s review considered recent NRC IRs from the years 2010 to 2016 (Refs. 45, 46, 47, 48, 49, 50, and 84) and the ALARA Program at MURR. The policies and the bases for procedures give reasonable assurance that doses to occupational workers and the public will be maintained below regulatory limits and ALARA. The controls and procedures for limiting access and personnel exposure (including allowable doses, effluent releases, ALARA goals, and the criteria used for the action levels in radiation alarm systems) is consistent with the guidance in ANSI/ANS-15.11-1993, provide reasonable assurance that radiation doses to the environment, the public, and facility personnel will be ALARA. The ALARA Program is adequately supported at the highest levels of management for the facility. The NRC staff concludes that the MURR ALARA Program complies with 10 CFR 20.1101(b), is acceptable, and provides reasonable assurance that radiation doses will be maintained ALARA for all facility activities.

11.1.4 Radiation Monitoring and Surveying

As described in SAR Section 11.1.4, the radiation protection organization maintains numerous fixed and portable radiation detection instruments throughout the MURR facility. SAR Table 11-14 summarizes the radiation monitoring equipment used at MURR; this listing is not intended to be all-inclusive and should be considered representative rather than an exact listing. Ten fixed area radiation monitors and six continuous air monitors are located throughout the facility to alert staff and operators to changing radiation conditions. All of the area monitors read out in the control room. The area monitors have local alarm lights. The continuous air monitors have local alarms and readouts.

TS 3.7 (see SER Section 11.2.2) requires that the reactor bridge radiation monitor, RCB exhaust plenum radiation monitor, and stack radiation monitor to be operable for reactor operation. TS 4.7 (see SER Section 11.2.2) requires surveillance requirements for the radiation and effluents monitoring systems. The stack instrumentation monitors the level of radioactive effluent, while the bridge instrumentation monitors gamma-radiation levels to alert personnel to changes in conditions. Additional monitoring is performed on an as-needed basis to support non-routine activities. Fixed radiation monitors are used in the facility for detection of personnel contamination. These contamination monitors are located at the main airlock to the reactor building and other locations as needed. Portable instrumentation is available to survey areas in the MURR facility for all types of radiation and radioactive contamination that may be present from facility operations. This includes ion chambers and friskers.

In its response to RAI No. 3.I (Ref. 37), the licensee reiterated its response to RAI No. A.28 (Ref. 20), and stated that "Isolation of the reactor containment building at 10 times the normal previously established radiation levels is necessary to allow for sample handling within the reactor pool or when removing samples from the pool. Normal pool surface radiation levels are around 20 mrem per hour while those at the containment building exhaust plenums are around 0.15 mrem per hour. Operational experience at MURR has demonstrated that the 10 times factor provides sufficient margin to minimize inadvertent reactor scrams without allowing for the potential of unacceptable exposure rates to personnel in containment. Ten times the routine dose rates equate to 200 mrem at the bridge monitor and 1.5 mrem at the exhaust plenum. Dose rates at this level do not constitute an unreasonable risk and could not go unidentified for any significant period of time. Radiation monitor indications are recorded at set intervals in the reactor log book and any increase above normal would be identified by and responded to by Reactor Operations. The functions of the Reactor Bridge and Reactor Containment Building Exhaust Plenum Radiation Monitors are described in SAR Section 7.8. The NRC staff reviewed the licensee's radiation monitor setpoints and finds that they are properly established to alert to operators to a change in the radiation levels, and to investigate or evacuate the area, as appropriate.

According to the licensee, facility surveys (radiation and contamination) are conducted on a nominally weekly basis, with more frequent surveys based on work levels and types. For areas not normally in use, surveys are performed before allowing activities in those areas.
In addition to the fixed and portable radiation detection equipment, additional laboratory monitoring equipment is available to support analyses. This includes a National Institute of Standards and Technology-traceable calibration facility used for calibrations of area, air, and portable survey instruments. Calibration activities are controlled by approved facility procedures.

The NRC staff reviewed the number, placement, and types of radiation detection equipment in use at the MURR facility. The NRC staff finds that the installed and available radiation detection equipment is of the proper type, range, and sensitivity to detect and quantify the types of radiation at MURR. Further, the NRC staff finds that the program to use and maintain the equipment and the frequency of surveys satisfy the requirements of 10 CFR 20.1501(a) and (b).

The NRC staff reviewed NRC IRs from the years 2010 to 2016 (Refs. 45, 46, 47, 48, 49, 50, and 84) and the design of radiation monitoring and sampling provisions at the facility. The fixed and portable equipment used for radiation monitoring and sampling inside the facility is selected, located, calibrated, tested, and maintained in accordance with guidance contained in recognized national standards and the manufacturers' instructions and with applicable regulations. The methods and bases of procedures used to determine the placement of the equipment, the circumstances under which the equipment is used, and the selection of the equipment function and sensitivity are appropriate to the facility and give reasonable assurance that appropriate types of radiation in significant intensities will be detected, monitored, and sampled consistent with 10 CFR Part 20 requirements and the facility ALARA Program. Based on its review, the NRC staff concludes that the licensee's equipment for detecting the types and intensities of radiation likely to be encountered within the facility, the program for calibrating and maintaining that equipment, and the surveillance frequencies provide reasonable assurance that doses to personnel will be kept below the limits required in 10 CFR 20.1201, "Occupational Dose Limits for Adults."

11.1.5 Radiation Exposure Control and Dosimetry

SAR Section 11.1.5 describes the radiation exposure control and dosimetry program in the Radiation Protection Program at the MURR facility. MURR is located in a controlled access building. The MURR building, laboratories, and classrooms meet the definition of a controlled area as defined by 10 CFR 20.1003, "Definitions." Access to the building requires training appropriate to the level of access to radioactive materials. According to SAR Section 11.1.5.5.2 and based on observations during the facility tour, all personnel entering the areas where radiation and radioactive material could be present use individual dosimetry. In its response to RAI No. 11.2 (Ref. 17), the licensee states that it currently uses optically stimulated luminescence (OSL) dosimeters for both area monitoring and personnel dosimetry. The licensee uses a contract dosimetry supplier who is certified by the National Voluntary Laboratory Accreditation Program as required by 10 CFR 20.1501(c). Supplementary dosimetry is available as needed.

As described in SAR Section 11.1.4, the licensee uses portable radiation detection equipment to monitor radiation levels throughout the facility. A review of the MURR annual operating reports the years 2010 through 2015 (Refs. 39 through 44) reveals that the highest annual whole body dose received by a facility employee has been 1,565 mrem (10 CFR 20.1201 limit is 5,000 mrem). The highest annual extremity exposure for the same period was 5,524 mrem (10 CFR 20.1201 limit is 50,000 mrem). Both of these exposures occurred in 2013 and are below the dose limit of 10 CFR 20.1201, which is 5,000 mrem total effective dose equivalent, and 50,000 mrem shallow-dose equivalent to an extremity.

Internal monitoring is not normally required at the MURR facility. According to SAR Section 11.1.5.5.2, urine sampling for tritium uptake is periodically performed for individuals frequenting the RCB. Other bioassay measurements can be made if they are deemed necessary, depending on the particular radiological circumstance. According to SAR Section 11.1.5.5.1, respiratory protection devices are not used at the MURR facility for limiting radiological uptakes because engineering controls have been effective and sufficient to limit airborne radioactive material. The NRC staff finds this practice to be consistent with the guidance in NUREG-1537 and ANSI/ANS-15.11-1993, and the regulations in 10 CFR Part 20.

Entry points with locked gates control high radiation areas. The SAR identifies the mechanical equipment area, the demineralizer cell area, and the beam port area as areas meeting the criteria for high-radiation areas and requiring the locked gate access control. The RSO controls the keys to these areas. Direct surveillance or warning devices may supplement controls for short-term experiments. This meets the requirements of 10 CFR 20.1601, "Control of Access to High Radiation Areas" that entryways are locked, except during periods when access to the areas is required, with positive control over each individual entry, and in place of the controls required for a high radiation area, the licensee may substitute continuous direct or electronic surveillance that is capable of preventing unauthorized entry.

SAR Section 11.1.5.1 discusses the various shielding analyses that have been performed at MURR to maintain personnel doses ALARA. Areas evaluated include the following:

- biological shield
- spent fuel transfer and storage
- experimental facilities
- primary and pool coolant systems

The NRC staff reviewed the licensee's radiation exposure control and dosimetry processes, which considered the engineered radiation exposure controls employed at the MURR facility. The NRC staff finds that the licensee has given sufficient information about the design of the containment, radiological shielding, ventilation, remote handling, decontamination equipment, and entry control devices to allow for an assessment of the design of these radiological protection features. The NRC staff finds that entry control devices employed are adequate to alert the workers to prevent entry into radiological areas, including high or very high radiation areas. The NRC staff finds that containment system design provides reasonable assurance that uncontrolled radiological releases to the unrestricted environment, controlled area, or the restricted work area will not occur during any anticipated normal operations.

The NRC staff finds that the licensee has discussed the procedures for use of personal dosimetry at the facility. Provisions exist in the licensee's Radiation Protection Program for external and internal radiation monitoring of all individuals required to be monitored. The proposed dosimetry program meets the requirements in 10 CFR Part 20. The NRC staff also finds that the Radiation Protection Program incorporated design features such as personal dosimetry, shielding, ventilation, remote handling, and decontamination equipment and provides reasonable assurances that radiation doses are maintained ALARA and within applicable regulations.

Based on its review of MURR Radiation Protection Program, the NRC staff concludes that the exposure control and dosimetry program at MURR is consistent with the guidance in ANSI/ANS-15.11 1993, and is adequate to monitor and control exposures to personnel below the limits in Subpart C, "Occupational Dose Limits," of 10 CFR Part 20 and.

11.1.6 Contamination Control

SAR Section 11.1.4.1 discusses contamination surveys. SAR Section 11.1.6 discusses the Contamination Control Program at MURR.

According to SAR Section 11.1.6, contamination control at the MURR facility is accomplished through staff training and surveys, as needed to detect contamination. Survey equipment is available for personnel to use to monitor for contamination. The NRC staff observed during site visits that the licensee had monitoring equipment at locations with the potential for contamination and that MURR staff used the monitors to ensure that contamination was not present. The NRC IRs for the years 2010 through 2016 (Refs. 45, 46, 47, 48, 49, 50, and 84) includes reviews of the licensee's contamination monitoring and control as part of the Radiation Protection Program (Refs. 45 through 50). Postings of surveys of contamination are available for control of potentially contaminated areas. Decontamination supplies are available for the cleanup of any spilled material. According to the licensee, routine surveys of the facility are performed on a nominally weekly basis and supplemented as needed based on the type of activity taking place. Experiments likely to generate significant contamination are identified in the experiment review process, and appropriate controls are included in the design.

The NRC staff reviewed the contamination control program as described in the SAR and NRC IRs from the years 2010 through 2016 (Refs. 45, 46, 47, 48, 49, 50, and 84) and find that adequate controls are in place to preclude the spread of contamination throughout the facility. The NRC staff concludes that the MURR Contamination Control Program is adequate for minimizing the potential for increased personnel doses from contamination.

The NRC staff examined recordkeeping for contamination and historical information about occurrences of radioactive contamination at the facility, which helps to confirm that the program is effective. The NRC staff concludes that the program for contamination control is adequate to ensure the effective control of radioactive contamination and to provide reasonable assurance that the health and safety of the facility staff, the environment, and the public will be protected.

11.1.7 Environmental Monitoring

Environmental monitoring is performed at MURR as part of the Radiation Monitoring Program and is described in SAR Section 11.1.7. The procedures for carrying out the environmental monitoring program are contained in the MURR Regulatory Assurance Procedures Manual. The MURR operations annual reports include results of environmental sample analyses.

According to the SAR Section 11.1.7, environmental samples of soil and vegetation are taken at eight locations, and water samples are taken at three locations semiannually around the reactor facility, including one at the City of Columbia sewage treatment and wastewater treatment facilities. Forty-two environmental radiation monitors, including three control OSLs, are placed around MURR, including five at the facility.

A review of the MURR annual operating reports the years 2010 through 2015 (Refs. 39 through 44) indicates that the maximum environmental OSL dosimeter dose was reported in the 2013 MURR Annual Operating Report (Ref. 43) which indicated that the OSL dosimeter No. 9 was 83.0 mrem net (above background) at a distance of 27 m from the main exhaust stack, with almost all of the remaining dosimeters recording significantly less (OSL dosimeter No. 15 was 65 mrem). OSL dosimeters No. 9 and No. 15 are located near the loading dock and receive most of their radiation from shipments in transit, not from routine operation of the MURR. The remaining OSL dosimeters read 10 mrem or less, which is below the limits of 10 CFR Part 20, which is 50 mrem in a year.

Based on its review, the NRC staff concludes that the environmental monitoring locations are sufficient to properly characterize the public dose from MURR and to demonstrate compliance with the dose limits of 10 CFR 20.1301, "Dose Limits for Individual Members of the Public."

The NRC staff reviewed the environmental monitoring program presented in the SAR and finds that the environmental monitoring program is appropriate to the facility and commensurate with its projected radiological impact on the environment. The NRC concludes that the environmental monitoring program can be effectively implemented and sustained during the day-to-day operation of the facility, and that any radiological impact on the environment will be accurately assessed.

11.2 <u>Radioactive Waste Management</u>

11.2.1 Radioactive Waste Management Program

According to SAR Section 11.2, all individuals who work with radioactive materials at MURR are required to have training approved by the reactor health physics manager. This training includes instruction dealing with radioactive waste. Implementation of the ALARA principle, as described in SAR Section 11.1.3, also includes the minimization of the generation of radioactive waste. The design of the experiments incorporates ALARA reviews to minimize unnecessary generation of radioactive material. The Health Physics Branch provides oversight of the Radioactive Waste Management Program. As states in the SAR, the Radioactive Waste Management Program is periodically audited as part of the Radiation Protection Program and other radiation safety programs (ALARA Program), in addition to the required audit in TS 6.2. The audit is performed by MURR management or its authorized delegates to verify the adequacy of the program and its compliance with applicable regulations.

Radioactive material is stored for decay as much as practical. Waste minimization practices are used throughout the facility to minimize disposal costs. These practices include the use of materials with low neutron activation potential.

In its response to RAI No. 11.9 (Ref. 17), the licensee clarified that the reactor health physics manager, with the assistance of the Health Physics Branch, is responsible for the safe disposal of radioactive waste generated from materials under the facility operating license. In its response to RAI No. 11.8 (Ref. 17), the licensee discussed the access to solid waste disposal facility sites for the license renewal period. In the response, the licensee stated that the majority of the radioactive wastes generated at MURR is Class A waste, which meets the waste acceptance criteria for disposal at a facility in Utah. The generated Class B and C wastes are now maintained in temporary long-term storage in the reactor building. The licensee added that, given the current generation of these wastes, sufficient safe and secure storage space exists for storing these wastes during the license renewal period.

The NRC staff reviewed the Radioactive Waste Management Program as described in the SAR, as supplemented by RAI responses, the NRC IRs, and the facility annual operating reports and finds that effluents and waste generation by the MURR facility are within Federal requirements and the effluents and wastes produced are maintained ALARA.

11.2.2 Radioactive Waste Controls

According to SAR Section 11.2.2.1, solid waste materials are collected at the point of generation in marked waste containers. The waste materials are consolidated with other laboratory radioactive material for final disposition. Material shipped for disposal is packaged to meet transportation and burial requirements and shipped to a licensed disposal site.

SAR Section 11.2.2.2 states that potentially radioactive liquid wastes are collected in a series of four tanks below grade at MURR. Liquid radioactive waste may be processed using chemical precipitation and filtering to remove as much radioactive material as practicable. Liquid radioactive waste for disposition is sampled and analyzed to confirm that waste released to the sanitary sewer meets the requirements in 10 CFR 20.2003, “Disposal by Release into Sanitary Sewerage,” for concentration, pH and solubility requirements for the local sewage treatment facility. The annual operating reports (Refs. 39 through 44) provide a list of isotopes released to the sanitary sewer annually. The reports indicate that the released concentrations comply with

10 CFR 20.2003, which references the concentration limits in Table 3, “Releases to Sewers,” of Appendix B to 10 CFR Part 20. The annual released quantities are well below the limits cited in 10 CFR 20.2003(a)(4). The release is mainly tritium, with maximum quantities released of less 0.2 Ci per year. This is well below the limit of 5 Ci allowed under 10 CFR 20.2003(a)(4).

Although Ar-41 and other radioactive gases are released from the facility through the ventilation system exhaust stack, this release is not considered to be waste in the same sense as the solid and liquid wastes previously described. Releases through the MURR stack are classified as gaseous effluent, which is a routine part of the normal operation of the reactor. SER Section 11.1.1 discusses Ar-41 production, release and resulting radiological doses.

TS 3.7 applies to radiation monitoring channels that must be available to the reactor operator during reactor operation and specifies the maximum allowed discharge rate from the ventilation system exhaust stack. TS 4.7 contains the associated surveillance requirements that ensure that the reactor bridge, RCB exhaust plenum, and stack radiation monitors are operable, their source or channel is checked on a periodic frequency, and they are properly calibrated.

TS 3.7 Radiation Monitoring Systems and Airborne Effluents

TS 3.7 states:

Specification:

a. The reactor shall not be operated unless the following radiation monitoring channels are operating:

	Channel	**Minimum Numbers Operating** Mode I	Mode II	Mode III
1.	Reactor Bridge Radiation Monitor	1[(1)]	1[(1)]	1[(1)]
2.	Reactor Containment Building Exhaust Plenum Radiation Monitor	1	1	1
3.	Off-Gas (Stack) Radiation Monitor	1[(2)]	1[(2)]	1[(2)]

(1) The trip setting may be temporarily set upscale during periods of maintenance and sample handling. During these periods, the radiation monitor indication shall be closely observed.

(2) The stack radiation monitor may be placed out of service for up to two (2) hours for calibration and maintenance. During this out-of-service time, no experimental or maintenance activities shall be conducted which are likely to result in the release of airborne radioactivity.

b. The maximum discharge rate through the ventilation exhaust stack shall not exceed the following:

Type of Radioactivity	Max. Concentration Averaged Over One Year	Max. Controlled Instantaneous Release Concentration
Particulates and halogens with half-lives greater than 8 days	AEC	AEC
All other radioactive isotopes	350 AEC	3,500 AEC

AEC = Air Effluent Concentration as listed in Appendix B, Table 2, Column I of 10 CFR 20, "Standards for Protection Against Radiation."

c. An environmental monitoring program shall be carried out and shall include, as a minimum:

(1) Analysis of samples from surface waters from the surrounding areas, and vegetation or soil,

AND

(2) Placement of film badges, thermoluminescent dosimeters, or other devices at control points.

TS 3.7, Specification a, requires the minimum number of radiation monitoring channels to be operating for each mode of reactor operation. The NRC staff finds that this specification helps to ensure that the monitors required to facilitate monitoring and safety functions are operating so that the expected facility response to radiological events (e.g., scram or containment isolation), as assumed in the safety analysis, are maintained. The monitors selected and the required number operating in each mode of reactor operation are consistent with the guidance in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," issued February 1996 (Ref. 51), and ANSI/ANS-15.1-2007, "The Development of Technical Specifications for Research Reactors," issued 2007 (Ref. 57). This specification allows various radiation monitors to be out of service temporarily for maintenance. The NRC staff finds the maintenance periods to be reasonable based on the anticipated work and the limitations on other activities during this period. Based on the information provided above, the NRC staff concludes that TS 3.7, Specification a, is acceptable.

TS 3.7, Specification b, limits the discharge rate of radioactive isotopes from the ventilation exhaust stack. For each particulate and halogen radionuclide with half-life greater than eight days, TS 3.7, Specification b, requires that both the maximum release concentration averaged over one year and the maximum controlled instantaneous release concentration not exceed the corresponding AEC in 10 CFR Part 20, Appendix B for that radionuclide. For particulate and halogen radionuclides, these AECs are airborne concentrations that would result in a dose of 50 mrem from inhalation of a given radionuclide, if a member of the public were exposed to that concentration of that radionuclide for an entire year.

In the basis for TS 3.7, Specification b, the licensee states that the TS limits the release concentrations of these particulates and halogens to the AEC, without the inclusion of a dilution factor to account for dilution of the radionuclides between the release point and the receptor. This is done to help ensure that when these radionuclides are released from the stack, any public dose from those radionuclides will remain within 10 CFR Part 20 limits, even when any reconcentration of these radionuclides that may occur in the environment (resulting in doses from pathways other than inhalation, such as ingestion of contaminated food products) is considered. The licensee states the dilution factor between the stack and the nearest residence due north of MURR, for conditions in which the wind blows toward the nearest residence, is approximately 1,900. Additionally, since the wind only blows from south to north a portion of the year, a member of the public located at the nearest residence would only be exposed a portion of the year; therefore, the 1,900-dilution factor is conservative.

The licensee cited J. K. Soldat, "The Relationship between I-131 Concentrations in Various Environmental Samples," (Ref. 103) to support the use of a reconcentration factor of approximately 400, which is applicable to the milk ingestion exposure pathway for I-131. (This particular reconcentration factor may be used to estimate doses from I-131 ingestion based on I-131 air concentrations, for a situation in which I-131 is deposited on grass and consumed by dairy cows, which then produce milk containing I-131 that is consumed by humans.) The licensee noted that this reconcentration factor is well below the 1,900-dilution factor.

The NRC staff noted that I-131 is, in general, a significant radionuclide of concern for environmental reconcentration. The NRC staff reviewed the licensee's cited reference (Ref. 103), and noted that the reference stated that the milk ingestion pathway was the primary exposure pathway related to I-131 reconcentration in the environment, and that the cited reconcentration factor of 400 was applicable to children; for adults, the reconcentration factor was approximately an order of magnitude lower. The NRC staff also noted that, as reported in each of the MURR annual reports the years 2010 through 2015 (Refs. 39 through 44), I-131 was one of the top two radionuclides released from the stack, by percent of TS limit, other than Ar-41 (although I-131 was still released at less than one percent of the TS limit in each of those years). Therefore, the NRC staff finds that a reconcentration factor on the order of 400 is a reasonable bounding approximation for particulates and halogens with half-lives longer than eight days that are likely to be released from MURR. The NRC staff also finds that the licensee's stated dilution factor of 1,900 at the nearest residence when the wind blows toward the nearest residence is reasonable, given weather conditions near the MURR site. The dilution factor also includes a significant degree of conservatism, since the wind only blows from south to north about 13 percent of the average year. The NRC staff finds that there is a large margin of safety between the likely actual reconcentration and dilution factors for particulates with half-lives longer than eight days that may be released from MURR. Therefore, the NRC staff concludes that TS 3.7, Specification b, helps provide assurance that the collective public dose from all of these radionuclides that could potentially be released in one year, plus the dose from other radionuclides such as Ar-41, will be below the 100 mrem limit in 10 CFR 20.1301, because the dilution would be greater than any reconcentration of the particulates and halogens in the environment.

For all other radioactive isotopes, TS 3.7, Specification b, requires that the maximum release concentration averaged over one year not exceed 350 times AEC, and the maximum controlled instantaneous release concentration not exceed 3,500 times AEC. The primary radionuclide released by MURR, which falls into this category of radioactive isotopes, is Ar-41 (accounting for over 99 percent of the radioactivity released, as discussed in SER Section 11.1.1). As the analyzed in SER Section 11.1.1, releases of Ar-41 from the facility stack at an annual average concentration equal to 350 times the AEC for Ar-41 will result in maximum doses to members of the public that are well below the 100 mrem public dose limit in 10 CFR 20.1301. In its response to RAI No. A.33 (Ref. 23), the licensee discusses the instantaneous concentration limit of 3,500 AEC on the consequences of Ar-41 releases. As explained in the licensee's basis for the TS, the normal short burst releases at the facility are 5 to 10 seconds in duration, and occur an average of 10 times per day, 5 days per week. These spikes are almost completely comprised of Ar-41, and mostly occur due to the operation of the pneumatic tube system or from the opening of sample cans, which have small amounts of irradiated air inside of them. Assuming bursts of 10-second duration occurring 50 times per week (or 500 seconds total duration of bursts per week), the release increases the concentration by less than 1 percent over the 350 AEC limit, when averaged over a week. Should the duration of the bursts increase to a total of one hour or 2.5 hours per week, the concentration would increase by 8 percent or 20 percent, respectively, when averaged over a week. The NRC staff reviewed this information, and finds that the short bursts would not significantly increase the Ar-41 release concentrations when they are averaged over one day.

The NRC staff reviewed the information above, and also reviewed the licensee's environmental dose records provided in the annual operating reports for the years 2010 through 2015 (Refs. 39 through 44). The NRC staff finds that these records provide additional support for the analyses, discussed above, showing that TS 3.7, Specification b, helps ensure that airborne radioactive effluents from MURR will not cause the public dose limit in 10 CFR 20.1301 to be exceeded. Based on the information above, the NRC staff concludes that TS 3.7, Specification b, is acceptable.

TS 3.7, Specification c, requires an environmental monitoring program that includes analysis of samples of surface water and vegetation or soil and film badges, thermoluminescent dosimeters, or other devices at controlled locations. The NRC staff finds that this specification helps to ensure that the collection and analysis of water, soil, or vegetation samples will provide information on regulatory compliance with environmental limits. SAR Section 11.1.7 describes how film badges, thermoluminescent dosimeters, or other devices placed at control points provide a measurement of radiation fields at appropriate locations. The NRC staff finds that the environmental monitoring program, as described in SAR Section 11.1.7, helps to verify that operation of the facility presents no significant risk to the general public health and safety. The TS is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 3.7, Specification c, is acceptable.

The NRC staff reviewed TS 3.7, Specifications a, b, and c. The NRC staff finds that TS 3.7, Specifications a, b, and c, requires radiation monitors and environmental monitoring controls that help to ensure that operation of MURR does not endanger the public or the environment. Based on the information provided above, the NRC staff concludes that TS 3.7, Specifications a, b, and c, are acceptable.

<u>TS 4.7 Radiation Monitoring Systems and Airborne Effluents</u>

TS 4.7 states:

<u>Specification</u>:

a. Radiation monitoring instrumentation required by Specification 3.7.a shall be verified operable by monthly radiation source checks or channel tests.

b. Radiation monitoring instrumentation required by Specification 3.7.a shall be channel calibrated on a semiannual basis.

c. Surveillance of the environmental monitoring program shall include:

    (1) A collection of water, and vegetation or soil samples semiannually.

AND

(2) A collection of film badges, thermoluminescent dosimeters, or other devices semiannually.

TS 4.7, Specification a, requires the radiation monitoring instrumentation required by TS 3.7, Specification a, to be verified operable by monthly radiation source checks or channel tests. The NRC staff finds that this specification helps to ensure that the radiation monitors are operable by performing monthly source checks or channel checks. The surveillance method and surveillance interval are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 4.7, Specification a, is acceptable.

TS 4.7, Specification b, requires the radiation monitoring instrumentation, required by TS 3.7, Specification a, to be channel calibrated on a semiannual basis. The NRC staff finds that this specification helps ensure that the radiation monitors are operable to support monitoring and safety functions. The NRC staff finds that this specification also helps to ensure that the radiation monitoring equipment is providing an accurate indication of radiological conditions and the surveillance interval is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 4.7, Specification b, is acceptable.

TS 4.7, Specification c, requires a surveillance of the Environmental Monitoring Program to include a collection of water and vegetation or soil samples and a collection of film badges, thermoluminescent dosimeters, or other devises. The NRC staff finds this specification helps to ensure the Radiation Monitoring Program would be effective to detect adverse trends in the release of radioisotopes from the MURR facility. The NRC staff also finds that this specification helps to ensure that the Radiation Monitoring Program is collecting the required data, which is an essential feature of program effectiveness, and that the surveillance interval is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff finds that TS 4.7, Specification c, is acceptable.

TS 4.7, Specifications a, b, and c, help to ensure that the radiation monitoring equipment required by TS 3.7, Specifications a, b, and c, is verified operable, calibrated, and collecting data on an appropriate frequency to provide a high confidence that the systems and components will perform their expected safety functions and that the monitoring program is provided data on an appropriate frequency. The surveillance interval is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007.

Based on its review, the NRC staff concludes that the radioactive waste controls are acceptable, as the licensee described in the SAR methods by which the waste products from all procedures and processes will be monitored or otherwise assessed for radioactive material contents; the licensee has controls established on the waste streams and products designed to prevent uncontrolled exposures or escape of radioactive waste; and the descriptions of the plans and procedures provide reasonable assurance that radioactive wastes will be controlled at all times in a manner that protects the environment and the health and safety of the facility staff and the public.

11.2.3 Release of Radioactive Waste

SAR Section 11.2.3 describes the releases of liquid and gaseous waste from MURR. The release of gaseous Ar-41 and particulate activity through the facility ventilation exhaust stack has been previously discussed in SER Section 11.1.1.1. The maximum rate of discharge shall not exceed limits as specified in the TS. These limits ensure that exposure the general public resulting from the radioactivity released to the environment will not exceed the limits of 10 CFR Part 20.

Liquid radioactive waste is retained until an assay indicates that the specific activity of all radioactive isotopes is less than the limit specified in 10 CFR Part 20 for disposal by release into sanitary sewerage. In addition to the limit on each isotope, 10 CFR Part 20 also limits the total activity that can be annually released from MURR to the sanitary sewerage. MURR policy is to use 5 percent of the total limit of each isotope as an administrative limit, although a few isotopes have a higher administrative limit. These limits ensure that the liquid waste is retained as long as practical to allow the activity to decay.

The transfer of solid radioactive waste is normally to an authorized solid waste broker or brokerage service. However, the facility may opt to ship solid radioactive waste directly to a waste disposal site without the use of a broker.

The NRC staff reviewed the release of radioactive material as described in SAR Section 11.2.3 and finds that the effluent Radiation Monitoring Program at MURR is adequate to quantify and characterize the gaseous and liquid effluents released from the facility and keeps effluent concentrations below the limits of Appendix B to 10 CFR Part 20, TSs related to the Radiation Monitoring Program meet the regulation in 10 CFR 50.36 requirements for LCOs and SRs. Furthermore, the NRC staff concludes that the program is sufficient to provide reasonable assurance that doses to members of the public from effluents are well below the limits of 10 CFR 20.1301.

11.3 <u>Conclusions</u>

Based on its review of the information presented in the MURR SAR, as supplemented by responses to RAIs, observations of the licensee's operations, review of annual operating reports, and the results of the NRC inspection program, the NRC staff concludes the following:

- The MURR Radiation Protection Program complies with the requirements of 10 CFR 20.1101(a), is acceptably implemented, and provides reasonable assurance that the facility staff, the public, and the environment 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 MURR ALARA Program complies with the requirements of 10 CFR 20.1101(b). A review of controls for radioactive material at MURR provides reasonable assurance that radiation doses to facility personnel and the public and effluent releases to the environment will be ALARA.

- The results of radiation surveys carried out at MURR, doses to the individuals issued dosimetry, and results of the environmental monitoring program help verify that the Radiation Protection Program and ALARA Program are effective.

- The licensee adequately identifies and describes potential radiation sources and sufficiently controls them.

- TSs discussed above are consistent with the SAR and satisfy 10 CFR 50.36 requirements.

- Facility design and procedures limit the production of Ar-41 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 MURR staff and the public will be below applicable 10 CFR Part 20 limits.

- The radioactive waste management program provides reasonable assurance that radioactive waste released from the facility will not exceed the limits in 10 CFR Part 20 or endanger the public or the environment during the renewal period.

12 CONDUCT OF OPERATIONS

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

12.1 Organization

Section 12.1 of the safety analysis report (SAR) and the licensee's response to Request of Addition Information (RAI) No. 12.1 (Ref. 24) describe the University of Missouri-Columbia Research Reactor (MURR or the reactor) organization. The licensee for MURR is the Board of Curators for the University of Missouri System. The University of Missouri System is governed by a nine-member Board of Curators appointed by the Governor and confirmed by the State Senate to serve a six-year term. As the facility licensee, the Board of Curators is responsible for ensuring adherence to all the requirements of the facility operating license and the Technical Specifications (TSs), thus reasonably ensuring that the health and safety of the general public will not be endangered as a result of operating the reactor. The Board of Curators delegates this responsibility to the MURR Director. The Director of MURR reports to the Office of the Provost, who reports to the University President, who ultimately reports to the Board of Curators. The reactor manager reports to the MURR Director. The reactor operations staff, which includes the licensed senior reactor operators (SROs) and reactor operators (ROs), report to the reactor manager.

The organization chart presented in SAR Section 12.1 is consistent with Technical Specification (TS) Figure 6.0 (see Figure 12-1 of this safety evaluation report (SER). In its responses to RAI No. 12.2.a and RAI No. 12.2.b (Ref. 17), the licensee discusses minimum staffing requirements. The radiation protection organization has a reporting chain independent of the reactor operations staff. The reactor health physics manager reports to the MURR Director. SAR Section 12.1.2 describes the responsibilities of the MURR primary staff.

TS 6.1 Organization

TS 6.1 states:

6.1 Organization

a. The organizational structure of the University of Missouri-Columbia (MU) relating to the University of Missouri Research Reactor (MURR) shall be as shown in Figure 6.0.

b. The following positions shall have direct responsibility in implementing the Technical Specifications as designated throughout this document:

(1) Office of the Chancellor (Level 1): Shall be responsible for directing MU's research mission, the quality and effectiveness of all programs and dedicating university resources necessary to ensure that all research, education and service are conducted in accordance with applicable federal, state and local regulations and accreditation requirements.

(2) Reactor Facility Director (Level 2): Shall be responsible for establishing the policies that minimize radiation exposure to the public and to radiation workers, and that ensures that the requirements of the license and Technical Specifications are met.

(3) Reactor Manager (Level 3): To safeguard the public and facility personnel from undue radiation exposure, the Reactor Manager shall be responsible for:

i. Compliance with Technical Specifications and license requirements regarding reactor operation, maintenance and surveillance; and

ii. Oversight of the experiment review process.

(4) Reactor Health Physics Manager (Level 3): To safeguard the public and facility personnel from undue radiation exposure, the Reactor Health Physics Manager shall be responsible for:

i. Compliance with Technical Specifications and license requirements regarding radiation safety, byproduct material handling and the shipment of byproduct material; and

ii. Implementation of the Radiation Protection Program.

(5) Reactor Operations Staff (Level 4): Shall be responsible for the manipulation of reactor controls, monitoring of instrumentation, and operation and maintenance of reactor-related equipment.

(6) Reactor Health Physics Staff (Level 4): Shall be responsible for directing research, training, and monitoring programs in order to protect personnel from radiation hazards and to assure compliance with federal, state, and MU regulations.

c. At a minimum during reactor operation, there shall be two (2) facility staff personnel at the facility. One of these individuals shall be a Reactor Operator or a Senior Reactor Operator licensed pursuant to 10 CFR 55. The other individual shall be knowledgeable of the facility.

d. 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:

(1) Management personnel;

(2) Reactor Health Physics personnel; and

(3) Reactor Operations personnel.

e. A Senior Reactor Operator licensed pursuant to 10 CFR 55 shall be present at the facility or readily available on call at all times during operation. Readily available on call means an individual who:

   (1) Has been specifically designated and the designation known to the operator on duty;

   (2) Can be rapidly contacted by phone, by the operator on duty; and

   (3) 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).

f. Events requiring the presence of a Senior Reactor Operator at the facility are:

   (1) Initial startup and approach to power;

   (2) All fuel or control rod relocations within the reactor core region;

   (3) Relocation of any experiment with a reactivity worth greater than 0.0074 $\Delta k/k$; and

   (4) Recovery from an unplanned or unscheduled shutdown or a power reduction of 2 MWs or greater.

g. The selection, training, and requalification of operations personnel should be in accordance with the requirements of ANSI/ANS-15.4-2007, “Selection and Training of Personnel for Research Reactors.” Qualification and requalification of licensed reactor operators shall be performed in accordance with a U.S. Nuclear Regulatory Commission (NRC) approved program.

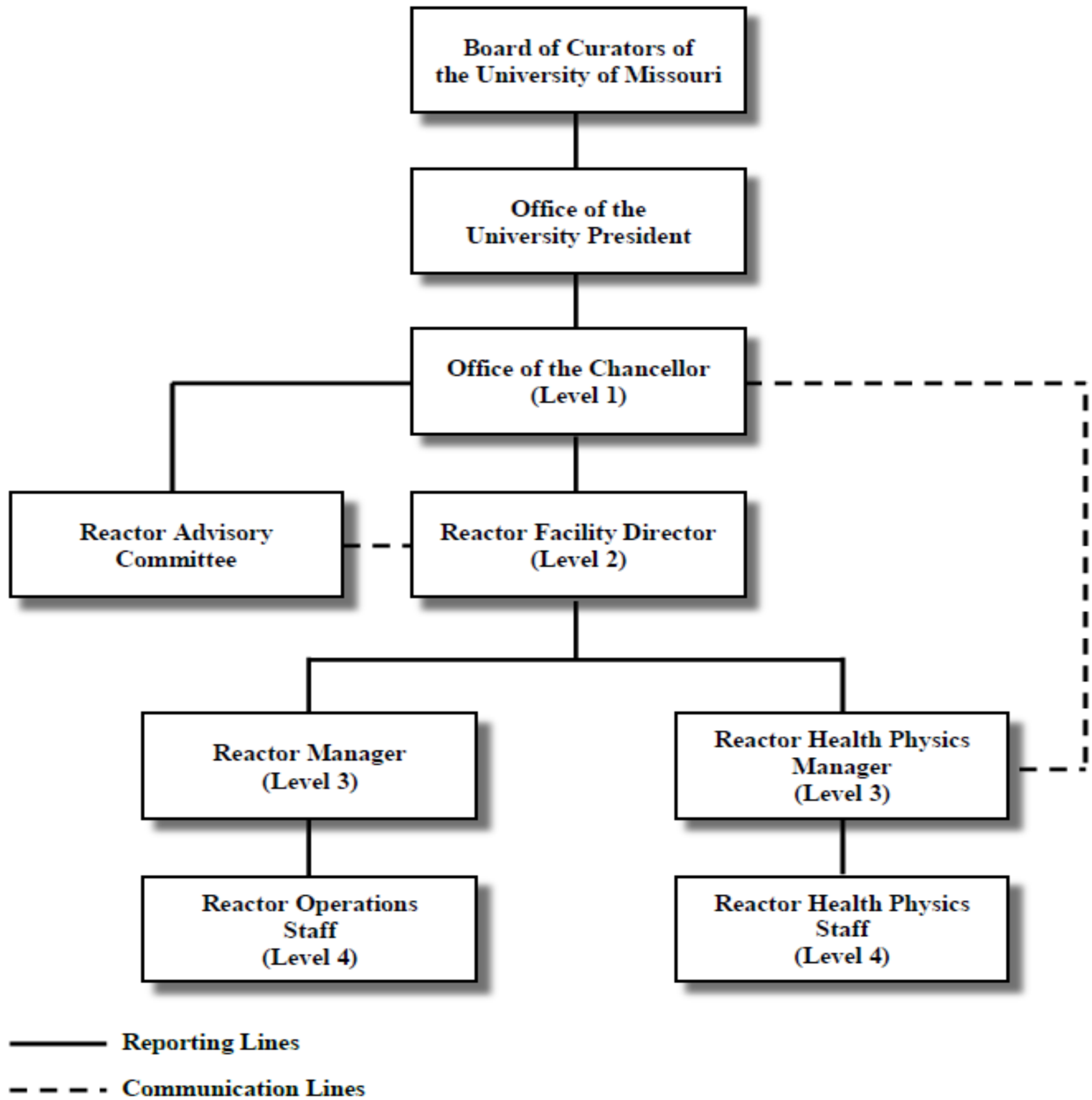

FIGURE 6.0
UNIVERSITY OF MISSOURI RESEARCH REACTOR (MURR)
ORGANIZATION

Figure 12-1 TS Figure 6.0

TS 6.1, Specification a, requires the MURR organization structure to be organized as shown in TS Figure 6.0. The NRC staff finds that this specification helps ensure that the TS properly delineates the MURR organization structure, including the communication and reporting lines. The NRC staff finds that the MURR organizational structure TS Figure 6.0 is consistent with the guidance in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," issued February 1996 (Ref. 51), and American National Standards Institute/American Nuclear Society (ANSI/ANS)-15.1-2007, "The Development of Technical Specifications for Research Reactors," issued 2007 (Ref. 57). Based on the information above, the NRC staff concludes that TS 6.1 and TS Figure 6.0 are acceptable.

TS 6.1, Specification b, specifies the positions at MURR that must be responsible for implementing the TS. The NRC staff finds that this specification helps to ensure that staff members in key positions in the MURR organizational structure fulfill their TS responsibilities. The NRC staff finds that the organizational responsibilities described in TS 6.1, Specification b, are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 6.1, Specification b, is acceptable.

TS 6.1, Specification c, requires that, during reactor operation, there shall be at least two facility staff personnel at the facility. One of these individual must be an RO or an SRO licensed pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 55, "Operators' Licenses." The other individual must have considerable knowledge of the facility. The NRC staff finds that this specification meets the regulation in 10 CFR 50.54(k), which states that "an operator or senior operator licensed pursuant to part 55 of this chapter shall be present at the controls at all times during the operation of the facility." The NRC staff also finds that this specification is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 6.1, Specification c, is acceptable.

TS 6.1, Specification d, requires that a list of reactor facility personnel by name and telephone number be readily available for the operator in the control room. The list must include (1) management personnel, (2) reactor health physics personnel, and (3) reactor operations personnel. The NRC staff finds that this specification describes those key personnel whose name and telephone numbers must be readily available in the control. The NRC staff also it is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 6.1, Specification d, is acceptable.

TS 6.1, Specification e, requires an SRO, licensed pursuant to 10 CFR Part 55, be present at the facility or readily available on call at all times during operation. TS 6.1, Specification e, defines "readily available on call" as an individual who: (1) has been specifically designated and the designation is known to the operator on duty; (2) can be rapidly contacted by phone, by the operator on duty; and (3) 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). The NRC staff finds this specification meets the regulations in 10 CFR 50.54(m)(1), which state that "a senior operator licensed pursuant to part 55 of this chapter shall be present at the facility or readily available on call at all times during its operation, and shall be present at the facility during initial startup and approach to power, recovery from an unplanned or unscheduled shut down or unscheduled reduction in power, and refueling, or as otherwise prescribed in the facility license." The NRC staff also finds that this specification is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 6.1, Specification e, is acceptable.

TS 6.1, Specification f, requires an SRO to be present at the facility, during initial startup and approach to power, during fuel or control rod relocation within the reactor core region, during relocation of any experiment with a reactivity worth greater than 0.0074 $\Delta k/k$, and during a recovery from an unplanned or unscheduled shutdown or significant power reduction. The NRC staff finds that this specification is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 6.1, Specification f, is acceptable.

TS 6.1, Specification g, requires that the selection, training, and requalification of operations personnel be performed in accordance with the requirements of ANSI/ANS-15.4-2007, "Selection and Training of Personnel for Research Reactors," issued 2007 (Ref. 70). The NRC staff finds that this specification helps to ensure the selection, training, and requalification of operators is accomplished using the guidance in ANSI/ANS-15.4-2007. The NRC staff also finds that this specification ensures that qualification and requalification of licensed ROs is performed in accordance with an NRC-approved program, which is also consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 6.1, Specification g, is acceptable.

The NRC staff reviewed TS 6.1, Specifications a through g, and finds that these specifications help ensure that licensee's staff is technically qualified to operate the MURR facility. The NRC staff finds that TS 6.1, Specifications a through g, meet the requirements for technical specifications in 10 CFR 50.36(c)(5), and are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 6.1, Specifications a through g, are acceptable.

TS 6.3 Radiation Safety

TS 6.3 states:

6.3 Radiation Safety

a. The Reactor Health Physics Manager shall be responsible for the implementation of the Radiation Protection Program. The requirements of the Radiation Protection Program are established in 10 CFR 20. The program should use the guidelines of American National Standard "Radiation Protection at Research Reactor Facilities," ANSI/ANS-15.11-1993 (R2004).

TS 6.3, Specification a, requires the reactor health physics manager to be responsible for the implementation of the Radiation Protection Program. The requirements of the Radiation Protection Program are established in 10 CFR Part 20, "Standards for Protection against Radiation." The licensee states that the MURR Radiation Protection Program uses the guidelines of ANSI/ANS-15.11-1993, "Radiation Protection at Research Reactor Facilities," reaffirmed 2004 (R2004). The NRC staff finds that this specification helps identify the responsible person for the implementation of the Radiation Protection Program. The NRC staff also finds this specification helps to ensure that the radiation safety aspects of the MURR organization structure are properly delineated. Furthermore, the requirements of the position and the responsibility for the Radiation Protection Program are stated and appropriate. The NRC staff also finds that this specification is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.11-1993 (R2004). Based on the information above, the NRC staff concludes that TS 6.3, Specification a, is acceptable.

Additionally, SAR Section 12.1.4 references ANSI/ANS-15.4-1988, "Selection and Training of Personnel for Research Reactors," R1999, as guidance for selecting and training personnel (Ref. 74). The NRC staff finds that the implementation of ANSI/ANS-15.4-1988 R1999 is consistent with the guidance in NUREG-1537. The NRC staff also finds that this specification helps ensures that MURR facility staff will be selected in a manner that meets the minimum qualifications for each position. Based on this information, the NRC staff concludes that this TS is acceptable.

Based on its review, the NRC staff concludes the following:

- The licensee presented an organizational structure that reflects the complete facility organization from the official license holder to the reactor operations staff. The organization meets the guidance in ANSI/ANS-15.1-2007 and ANSI/ANS-15.4-1988 (R1999).

- The licensee described the responsibilities of the persons in the organizational structure, and their responsibility for safe operation of the facility and for the protection of the health and safety of the facility staff and the public.

- The licensee described the facility staffing requirements that demonstrates the technical ability of staff members to safely operate the facility and protect the health and safety of the facility staff and the public. The staffing meets the requirements of in 10 CFR Part 55 and 10 CFR 50.54(i).

- ROs will be trained in a program that meets the standards for non-power reactors and the requirements of the regulations. Radiation protection training and specialized training will be conducted at an acceptable level.

- The licensee described a radiation safety organization that is acceptable to the NRC staff. The organization has direct access to upper management and the review and audit committee to express concerns, if necessary.

12.2 Review and Audit Activities

SAR Section 12.2 states that independent review and audit functions are performed by the Reactor Advisory Committee (RAC). The Office of the Chancellor appoints members of the RAC. According to SAR Section 12.2 and based on the licensee's responses to RAI No. 12.4.a (Ref. 24), RAI No. 12.4.b (Ref. 24), and RAI No. 12.4.c (Ref. 24), the RAC members are chosen for their relevant expert knowledge and meet at least once each calendar quarter. SAR Section 12.2 also outlines quorums, frequencies of audits, and lists audit activities. Annual audits are specified for the Quality Assurance (QA) Program, the Radiation Protection Program, the As Low As Is Reasonably Achievable Program, and operating procedures (among other topics).

TS 6.2 Review and Audit

TS 6.2 states:

6.2 Review and Audit

a. A Reactor Advisory Committee (RAC) shall provide independent oversight in matters pertaining to the safe operation of the reactor and with regard to planned research activities and use of the facility building and equipment. The RAC shall be composed of at least five (5) members who have knowledge of experimental activities, reactor operations, University business policy, or related subjects. The Committee members shall be appointed by, and report to, the Office of the Chancellor. The RAC shall review:

(1) Determinations that proposed changes in the facility, and procedures, and the conduct of tests or experiments are allowed without prior authorization by the NRC, pursuant to 10 CFR 50.59;

(2) All new procedures and major revisions thereto having safety significance, proposed changes to reactor facility equipment, or systems having safety significance. Changes to procedures that do not change their original intent may be made without prior RAC review if approved by the TS-designated manager, either the Reactor Health Physics Manager or Reactor Manager, or a designated alternate who is a member of Reactor Health Physics or a Senior Reactor Operator, respectively. All such changes to the procedures shall be documented, reviewed pursuant to 10 CFR 50.59, and subsequently reviewed by the RAC;

(3) Proposed experiments significantly different from any previously reviewed or which involve a question pursuant to 10 CFR 50.59;

(4) Proposed changes in the Technical Specifications or the license;

(5) The circumstances of reportable occurrences and violations of the Technical Specifications or license and the measures taken to prevent a recurrence;

(6) Violations of internal procedures or operating abnormalities having safety significance; and

(7) Reports from audits required by the Technical Specifications.

b. The RAC may appoint subcommittees consisting of knowledgeable members of the public, students, faculty, and staff of MU when it deems it necessary in order to effectively discharge its primary responsibilities. When subcommittees are appointed, these subcommittees shall consist of no less than three (3) members with no more than one (1) student appointed to each subcommittee. The subcommittees may be authorized to act on behalf of the RAC.

The RAC and its subcommittees shall maintain minutes of meetings in which the items considered and the committees' recommendations are recorded. Dissemination of the minutes to the Office of the Chancellor, the RAC and its subcommittees shall be done within three (3) months after the meetings. Independent actions of the subcommittees shall be reviewed by the parent committee at the next regular meeting. A quorum of the committee or the subcommittees consisting of at least fifty percent of the appointed members shall be present at any meeting to conduct the business of the committee or subcommittee. Additionally, reactor facility staff shall not constitute greater than fifty percent of the quorum for a meeting of the RAC. Reactor facility staff shall not constitute a majority of the RAC. The RAC shall meet at least quarterly.

A meeting of a subcommittee shall not be deemed to satisfy the requirement of the parent committee to meet at least once during each calendar quarter.

c. Any additions, modifications or maintenance to the systems described in these Specifications shall be made and tested in accordance with the specifications to which the system was originally designed and fabricated or to specifications approved by the NRC.

d. Following a favorable review by the NRC, the RAC, or the Reactor Facility Management, as appropriate, and prior to conducting any experiment, the Reactor Manager shall sign an authorizing form which contains the basis for the favorable review.

e. Audits:

   (1) Audits of the following functions shall be conducted by an individual or group without immediate responsibility in the area to be audited:

      i. Facility Operations, for conformance to the Technical Specifications and license conditions, at least annually;

      ii. Operator Requalification Program, for compliance with the approved program, at least every two (2) years;

      iii. Corrective Action items associated with reactor safety, at least annually; and

      iv. Emergency Plan, at least every two (2) years.

   (2) Audit findings which affect reactor safety shall be immediately reported to the Reactor Facility Director. A written report of the findings shall be submitted to the Reactor Facility Director, the RAC and its subcommittees within three (3) months after the audit has been completed.

TS 6.2, Specification a, requires the RAC to provide independent oversight in matters pertaining to the safe operation of the reactor and with regard to planned research activities and use of the facility building and equipment. TS 6.2, Specification a also delineates items that must be reviewed by the RAC such as 10 CFR 50.59 changes, new procedures and major revision to procedures having safety significant, proposed experiments significantly different from any previously reviewed, proposed changes to the TS, reportable occurrences and violations of TS and license, violation of internal procedures or operating abnormalities, and reports from audits required by TS. The NRC staff finds that this specification helps to ensure that the RAC requirements are properly delineated to provide independent oversight of the MURR facility. The NRC staff also finds that this specification is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007, noting that it also implements the guidance in Section 6.2.3 of NUREG-1537 by requiring the RAC to review items pursuant to 10 CFR 50.59, “Changes, Tests and Experiments.” Based on the information above, the NRC staff concludes that TS 6.2, Specification a, is acceptable.

TS 6.2, Specification b, provides the requirements for the charter and rules for the RAC. TS 6.2, Specification b, describes provisions for meeting frequency, voting rules, quorums, use of subcommittees, and minutes. This specification also establishes a quorum of not less than half the voting membership, where the operating staff does not constitute a majority. The licensee requested a clarification to indicate that the requirement that the operating staff did not constitute a majority was applicable to the RAC, and not to the subcommittees (Ref. 108). The licensee indicated that the subcommittees report to the RAC and all work performed by the subcommittees must be reviewed and approved by the RAC. The NRC staff review finds this request acceptable as it is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. The NRC staff finds that the requirements in this specification are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 6.2, Specification b, is acceptable.

TS 6.2, Specification c, requires that any additions, modifications, or maintenance to the systems described in these specifications be made and tested in accordance with the specifications to which the system was originally designed and fabricated or to specifications approved by the NRC. The NRC staff finds that this specification helps to ensure that any additions, modifications, or maintenance to the facility are made and tested to the specifications with which the systems were originally designed and fabricated. The NRC staff also finds that this specification helps ensure that the systems are maintained to their original requirements. Based on the information above, the NRC staff concludes that TS 6.2, Specification c, is acceptable.

TS 6.2, Specification d, requires that the Reactor Manager or the RAC to authorize experiments on a form which contains the basis for the favorable review accepted by the NRC, RAC or Reactor Facility Management, prior to conducting any experiment. The NRC staff finds this specification helps ensure that the reactor manager has reviewed the information, which is consistent with the management philosophy and structure expressed in TS 6.1. Based on the information above, the NRC staff concludes that TS 6.2, Specification d, is acceptable.

TS 6.2, Specification e, requires that audits of the following functions be conducted by an individual or group without immediate responsibility in the area to be audited: 1) facility operations, for conformance to the TSs and license conditions, at least annually; 2) Operator Requalification Program, for compliance with the approved program, at least every 2 years; 3) corrective action items associated with reactor safety, at least annually; and 4) the Emergency Plan, at least every 2 years. In addition, audit findings that affect reactor safety must be immediately reported to the reactor facility Director. The NRC staff finds this specification helps ensure that audit functions are fulfilled. The NRC staff also finds that this specification is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 6.2, Specification e, is acceptable.

The NRC staff reviewed TS 6.2, Specifications a through e, and finds that TS 6.2, Specifications a through e, are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on its review, the NRC staff concludes that TS-6.2, Specifications a through e, are acceptable.

SAR Section 10.4 and the licensee’s response to RAI No. 12.6 (Ref. 21) describe significant aspects of experiment review and approval.

<u>TS 6.5 Experiment Review and Approval</u>

TS 6.5 states:

6.5 Experiment Review and Approval

a. Approved experiments shall be carried out in accordance with established and approved procedures. Procedures related to experiment review and approval shall include the following:

   (1) All new experiments or class of experiments shall be reviewed by the RAC and approved in writing by the Reactor Manager.

   (2) Substantive changes to previously approved experiments shall be made only after review by the RAC and approved in writing by the Reactor Manager.

TS 6.5, Specification a.(1), requires the review by the RAC and approval from Reactor Manager for new experiments. The NRC staff finds that this specification involves the criteria provided in TS 3.8, Experiments, which is the NRC staff evaluated and found acceptable in SER Section 10.3. The NRC staff also finds that this specification consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 6.5, Specification a.(1), is acceptable.

TS 6.5, Specification a.(2), requires the review by the RAC and approval by the Reactor Manager of previously approved experiments if substantive changes have been made. The NRC staff finds that this specification is consistent with the guidance in NUREG-1537; ANSI/ANS-15.1-2007; and the provisions of Section C.3 of Regulatory Guide (RG) 2.2, "Development of Technical Specifications for Experiments in Research Reactors," (Ref. 97). Based on the information above, the NRC staff finds that this specification is acceptable.

The NRC staff reviewed TS 6.5, Specifications a.(1) and a.(2), and finds that TS 6.2, Specifications a.(1) and a.(2), are consistent with the guidance in NUREG-1537, RG 2.2, and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 6.2, Specifications a.(1) and a.(2), are acceptable.

The NRC staff reviewed the review and audit activities at MURR and finds they are consistent with the guidance in ANSI/ANS-15.1-2007, as follows:

- The licensee's TSs specify the composition of the review and audit committee. The structure provided is acceptable.

- The licensee's TSs discuss the charter and rules that govern the operation of the committee. The attendance frequency, definition of a quorum, reporting, and publication of minutes are acceptable.

- The TSs requirements for review and audit functions, periodicity, and the scope are appropriate.

Based on its review, the NRC staff concludes that the licensee's TSs provide sufficient criteria and controls to ensure that the review and audit functions are effective to support the safe operation of the facility during the renewal period.

12.3 Procedures

In SAR Section 12.3, as supplemented by its response to RAI No. 12.5 (Ref. 24), the licensee indicated that written, approved procedures govern all aspects of the reactor facility's operation and use. SAR Section 12.3 specifies the scope of required procedures. These procedures encompass, but are not limited to, the following areas:

- startup, operation, and shutdown of the reactor
- core loading, unloading, and fuel handling
- testing of reactor control and instrumentation systems
- emergency procedures
- health physics procedures

TS 6.4 Procedures

TS 6.4 states:

6.4 Procedures

a. Written procedures shall be in effect for operation of the reactor, including the following:

   (1) Startup, operation, and shutdown of the reactor;

   (2) Fuel loading, unloading and movement within the reactor;

   (3) Maintenance of major components of systems that could have an effect on reactor safety;

   (4) Surveillance checks, calibrations and inspections that may affect reactor safety;

   (5) Administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect reactor safety or core reactivity; and

   (6) Implementation of the Emergency and Physical Security Plans.

b. Written procedures shall be in effect for radiological control, and the preparation for shipping and the shipping of byproduct material produced under the facility operating license.

c. The Reactor Manager shall approve and annually review the procedures for normal operations of the reactor and the Emergency Plan implementing procedures. The Reactor Health Physics Manager shall approve and annually review the radiological control procedures and the procedures for the preparation for shipping and the shipping of byproduct material.

d. Deviations from procedures required by this Specification may be enacted by a Senior Reactor Operator or member of Reactor Health Physics, as applicable. Such deviations shall be documented, reviewed pursuant to 10 CFR 50.59, and reported within 24 hours or the next working day to the Reactor Manager or Reactor Health Physics Manager or designated alternate.

TS 6.4, Specifications a through d, require written procedures be in effect and controlled regarding reactor operations and radiological control as well as shipping of byproduct materials at MURR. The NRC staff reviewed TS 6.4, Specifications a through d, and finds that these specifications are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007, and meet the requirements in 10 CFR 50.36 (c)(5). In its response to RAI No. AA102 (Ref. 103), the licensee indicates that it interprets radiological control to include procedures that involve the use or handling of byproduct material at quantities sufficient to present a radiological hazard. Additionally, the licensee requires that byproduct radiological controls procedures necessary to ensure the safety of the worker be reviewed by the Health Physics Manager and a subcommittee of the RAC. The NRC staff finds this TS sufficient to help ensures that MURR radiological control procedures include the handling of byproduct material produced by operations and shipping of byproduct materials. Based on the information above, the NRC staff concludes that TS 6.4, Specifications a through d, are acceptable.

The NRC staff reviewed the use of procedures at MURR and finds that the licensee's TSs adequately describe the review and approval process for procedures, the method for making minor and substantive changes to existing procedures, and the process for the temporary deviation from procedures during operations. Based on its review, the NRC staff concludes that the process and methodology provided in the SAR and TSs ensure proper control and review of procedures during renewal period.

12.4 Reportable Events and Required Actions

In SAR 12.4, the licensee provides an outline of the incidents and conditions relating to the operation of the reactor require that the NRC be informed, including occurrences that are considered reportable events also require certain actions prior to returning the reactor to its normal condition.

In its responses to RAI No. 12.7, RAI No. 12.8.a, and RA No. 12.8.b (Ref. 20), the licensee proposed a revised TS 6.6 to add requirements for reactor shutdown, NRC notification, follow-up reports, and specific NRC authorization for restart of the reactor from a safety limit (SL) violation.

The licensee defined a group of incidents as reportable events (TS 1.1, Abnormal Occurrences) and described the required actions that it will take if a reportable event occurs. The definition of reportable events gives reasonable assurance that the licensee will report safety-significant events. The licensee also included actions to be taken if an SL is violated or a reportable event occurs. The NRC staff finds that these processes will help to ensure that the licensee will take the actions that are necessary to protect public health and safety.

TS 6.6 Reportable Events and Required Actions

TS 6.6, Specifications a, b, and c, states:

6.6 Reportable Events and Required Actions

a. Safety Limit Violation - In the event of a safety limit violation, the following actions shall be taken:

(1) The reactor shall be shut down and reactor operation shall not be resumed until authorized by the NRC pursuant to 10 CFR 50.36(c)(1);

(2) The safety limit violation shall be promptly reported to the Reactor Manager and Reactor Facility Director, or designated alternates.

(3) The safety limit violation shall be promptly reported to the NRC. Prompt reporting of the violation shall be made by MU, by telephone and subsequently confirmed in writing or email, to the NRC Operations Center no later than the following working day;

(4) A detailed follow-up report shall be prepared. The report shall include the following:

i. Applicable circumstances leading to the violation including, when known, the causes and contributing factors;

ii. Date and approximate time of the occurrence;

iii. Effect of the violation upon the reactor and associated systems;

iv. Effect of the violation on the health and safety of the facility staff and general public; and

v. Corrective actions to prevent recurrence.

(5) The follow-up report shall be submitted within fourteen (14) days to the NRC Document Control Desk.

b. Release of Radioactivity - Should a release of radioactivity greater than the allowable limits occur from the reactor facility boundary, the following actions shall be taken:

(1) Reactor conditions shall be returned to normal or the reactor shall be shut down;

(2) The release of radioactivity shall be promptly reported to the Reactor Manager and Reactor Facility Director, or designated alternates;

(3) The release of radioactivity shall be promptly reported to the NRC. Prompt reporting of the violation shall be made by MU, by telephone and

subsequently confirmed in writing or email, to the NRC Operations Center no later than the following working day;

(4) If it is necessary to shut down the reactor to correct the occurrence, operations shall not be resumed until authorized by the Reactor Facility Director, or designated alternate; and

(5) A detailed follow-up report shall be prepared. The follow-up report shall be submitted within fourteen (14) days to the NRC Document Control Desk.

c. Other Reportable Occurrences - In the event of an Abnormal Occurrence, as defined by Specification 1.1, the following actions shall be taken:

(Note: Where components or systems are provided in addition to those required by these Technical Specifications, the failure of the extra components or systems is not considered reportable provided that the minimum numbers of components or systems specified or required perform their intended reactor safety function.)

(1) The Abnormal Occurrence shall be promptly reported to the NRC. Prompt reporting of the Abnormal Occurrence shall be made by MU, by telephone and subsequently confirmed in writing or email, to the NRC Operations Center no later than the following working day;

(2) The Abnormal Occurrence shall be promptly reported to the Reactor Manager and Reactor Facility Director, or designated alternates;

(3) A detailed follow-up report shall be prepared. The follow-up report shall be submitted within fourteen (14) days to the NRC Document Control Desk; and

(4) The reactor shall be shut down or placed in a safe condition and return to normal reactor operations shall not be allowed until authorized by the Reactor Facility Director, or alternate.

TS 6.6, Specification a, requires the reactor to be shut down and the safety limit (SL) violation to be reported to the NRC. It also requires that a detailed follow-up report be made to the NRC, as required by in 10 CFR 50.36(c)(1). The NRC staff finds that this specification helps to ensure that prompt action and reporting are performed should an SL violation occur. The NRC staff also finds that this specification is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 6.6, Specification a, is acceptable.

TS 6.6, Specification b, requires the reactor to be shut down or returned to normal operation if a release of radioactivity greater than allowable limits occurs. The NRC staff finds that this specification helps to ensure that appropriate controls and responses are in place in the event that a release of radioactivity in an amount that is greater than allowable limits occurs. The NRC staff also finds that this specification is comprehensive, employs appropriate steps, requires reasonable reporting of the event, takes appropriate action to correct the event and prevent recurrence, and appropriately involves MURR management before the resumption of operation, if the reactor was shutdown. The NRC staff also finds that this specification is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 6.6, Specification b, is acceptable.

TS 6.6, Specification c, requires the reporting of an abnormal occurrence, as defined in TS 1.1. The NRC staff finds that this specification helps to ensure that abnormal occurrences, as defined in TS 1.1, are reported and that, if necessary, the reactor is shut down until operation is allowed to resume when authorized by the reactor manager. The NRC staff also finds that this specification is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 6.6, Specification c, is acceptable.

The NRC staff reviewed TS 6.6, Specifications a, b, and c, and finds that TS 6.6, Specifications a, b, and c, are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007, and the requirements of 10 CFR 50.36. Based on the information above, the NRC staff concludes that TS 6.6, Specifications a, b, and c, are acceptable.

Based on its review, the NRC staff concludes that the required actions are appropriate and provide reasonable assurance that the facility will respond to unanticipated occurrences in a manner that emphasizes reactor safety and protection of public health and safety.

12.5 Reports

SAR 12.4 describes the reportable events and required action. SAR Sections 12.4.4 and 12.4.5 describe reporting requirements.

TS 6.6 Reportable Events and Required Actions

TS 6.6, Specifications d and e, states:

d. Other Reports - A written report shall be submitted to the NRC Document Control Desk within thirty (30) days of:

   (1) Any significant change(s) in the transient or accident analyses as described in the SAR; and

   (2) Permanent changes in the facility organization involving the Office of the Chancellor or the Reactor Facility Director.

e. Annual Report - An annual operating report shall be submitted to the NRC within sixty (60) days following the end of each calendar year. The report shall include the following information for the preceding year:

   (1) A brief narrative summary of (a) operating experience (including operations designed to measure reactor characteristics), (b) changes in the reactor facility design, performance characteristics, and operating procedures related to reactor safety occurring during the reporting period, and (c) results of surveillance tests and inspections;

   (2) A tabulation showing the energy generated by the reactor (in megawatt-days);

(3) The number of emergency shutdowns and inadvertent scrams, including the reasons therefore and corrective action, if any, taken;

(4) Discussion of the major maintenance operations performed during the period, including the effects, if any, on the safe operation of the reactor;

(5) A summary of the changes to the facility and procedures, and conduct of tests or experiments carried out under the conditions of 10 CFR 50.59;

(6) A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as measured at or prior to the point of such release or discharge;

(7) A description of any environmental surveys performed outside the reactor facility; and

(8) A summary of radiation exposures received by facility staff, experimenters, and visitors, including the dates and time of significant exposure, and a brief summary of the results of radiation and contamination surveys performed within the facility.

TS 6.6, Specification d, requires a written report be submitted to the NRC Document Control Desk within 30 days of (1) any significant change(s) in the transient or accident analyses as described in the SAR and (2) permanent changes in the facility organization involving the Office of the Chancellor the Reactor Facility Director. The NRC staff finds this specification helps to establish controls over the reporting of changes to certain analyses or to the MURR organization. The NRC staff also finds this specification consistent with the guidance in NUREG-1537, and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 6.6, Specification d, is acceptable.

TS 6.6, Specification e, requires the submission of an annual operating report to the NRC within 60 days following the end of each calendar year and that the report included certain information. The NRC staff finds that this specification helps to ensure that important information will be provided to the NRC in a timely manner. The NRC staff also finds this specification consistent with guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 6.6, Specification e, is acceptable.

The NRC staff reviewed TS 6.6, Specifications d and e, and finds that TS 6.6, Specifications d and e, are consistent with the guidance NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 6.6, Specifications d and e, are acceptable.

The NRC staff also reviewed SAR Sections 12.4.4 and 12.4.5 and finds that the licensee described the content and the timing of the submittal and the distribution of the reports to ensure that it will provide important information to the NRC in a timely manner.

Based on its review, the NRC staff concludes there is reasonable assurance, per SAR Sections 12.4, 12.4.4, and 12.4.4 and TS 6.6, that licensee will report appropriate information on routine operation, non-routine occurrences, and changes to the facility and personnel to the NRC in a timely manner.

12.6 <u>Records</u>

SAR Section 12.5 describes records to be maintain and retained.

<u>TS 6.7 Records</u>

TS 6.7 states:

6.7 Records

Records of the following activities shall be maintained and retained for the periods specified below. The records may be in the form of logs, data sheets, or other suitable forms or documents. The required information may be contained in single or multiple records, or a combination thereof.

a. Lifetime Records—The following records shall be retained for the lifetime of the reactor facility: (Note: Applicable annual reports, if they contain all of the required information, may be used as records in this section.)

   (1) Gaseous and liquid radioactive effluents released to the environs;

   (2) Off-site environmental-monitoring surveys required by the Technical Specifications;

   (3) Radiation exposure for all monitored personnel;

   (4) Updated drawings of the reactor facility; and

   (5) Reviews and reports pertaining to a violation of a safety limit, limiting safety system setting, or limiting conditions for operations.

b. Five Year Records—The following records shall be maintained for a period of at least five (5) years or for the life of the component involved, whichever is shorter:

   (1) Normal reactor facility operation (but not including supporting documents such as checklists, log sheets, etc. 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 Technical Specifications;

   (5) Reactor facility radiation and contamination surveys required by applicable regulations;

   (6) Experiments performed with the reactor;

   (7) Fuel inventories, receipts and shipments;

(8) Approved changes to operating procedures; and

(9) Records of meetings and audit reports of the review and audit group.

c. Operator Licensing Records—Record of training and requalification of licensed reactor operators and senior reactor operators shall be retained at all times the individual is employed or until the license is renewed.

TS 6.7, Specifications a, b, and c, require certain records to be maintained for the life of the facility, for 5 years, or for the requalification or employment period of an operator. TS 6.7, Specification a, specifies type records that will need to be retained for the life of the facility. TS 6.7, Specification b, specifies type of records that will need to be retained for 5 years. TS 6.7, Specification c, requires that training and requalification of licensed operators and senior reactor operators be retained at all times the individual is employed or until the license is renewed. The NRC staff finds that TS specification helps to ensure the retention of certain records and are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information above, the NRC staff concludes that TS 6.7 is acceptable.

Based on its review, the NRC staff finds that, as described in SAR Section 12.5, the licensee has adequately described the types of records that will be retained and the period of retention to ensure important records will be retained for an appropriated time. The NRC staff also finds TS 6.7 is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007, and meets the regulations in 10 CFR 50.36. Based on the information above, the NRC staff concludes that the licensee requirements for record retention are acceptable.

12.7 Emergency Planning

The guidance for implementation of Emergency Planning for research and test reactors is provided in NUREG-1537, Regulatory Guide 2.6, "Emergency Planning for Research and Test Reactors," March 1983 (Ref. 106), and ANSI/ANS-15.16-2008, "Emergency Planning for Research Reactors," issued 2008 (Ref. 92).

The MURR Emergency Plan (EP) (Ref. 81) describes the radius of the MURR emergency planning zone (EPZ) (see Figure 12-2 below), which is greater than the value listed in Table 2 of ANSI/ANS-15.16-2008, for research reactors authorized to operate at a power level of 10 megawatts thermal. It is the area bounded by a 150-meter (492-foot) radius from the MURR exhaust stack, which lies completely within the site boundary. The licensee has not identified any credible accidents for the facility that would result in radiological effluents exceeding the Protective Action Guide at the EPZ boundary or exceeding the alert action levels listed in Table I of ANSI/ANS-15.16-2008, at the site boundary. However, the MURR EP describes three standardized classes of emergency situations grouping the accidents according to the severity of offsite radiological consequences: (1) notification of unusual event; (2) alert; and, (3) site area emergency. The latter classification is included to be conservative and to provide for consultation with offsite authorities and handling of information for the public through offsite authorities in the unlikely event of a site area emergency.

The licensee recognizes emergencies of lesser consequences than the notification of unusual events classification. These include physical occurrences within the facility requiring facility emergency organization response. The initial assessment should indicate that it is unlikely that an offsite hazard will be created. Protective evacuations or isolations of certain areas within the facility may be necessary.

The licensee indicates that the maximum fission product dose is generated by the maximum hypothetical accident, which is a failed fueled experiment (see SER Section 13.1). Both the licensee's and the NRC's conservative confirmatory analyses estimate that the resulting radiological dose will be below the alert action levels in Table 1 of ANSI/ANS-15.16-2008. Therefore, the NRC staff finds that the statement in the MURR EP that the licensee has not identified any credible accidents that would result in radiological effluents exceeding the alert action levels at the site boundary to be acceptable. The action levels are defined based on the projected doses in the first 24 hours. The 24-hour radiation shine (no release) at the EPZ for the maximum hypothetical accident is 12.34 millirem (mrem) (licensee calculation) and 22.58 mrem (NRC confirmatory calculation), without radionuclide decay. For assumed radionuclide decay, the 24-hour radiation shine dose would be 4.87 mrem. The alert action level is 75 mrem.

The NRC staff reviewed the MURR emergency preparedness program and documented its findings in NRC IR No. 50-186/2014-202 (Ref. 109). In the inspection, the NRC staff found that MURR conducted its emergency preparedness program in accordance with the MURR Emergency Plan. Training was conducted annually as required. Emergency response equipment was available, maintained and inventoried as required. Emergency drills were conducted annually as required by the Emergency Plan with support organizations participating biennially. The NRC staff met with the City of Columbia Fire Department, reviewed the licensee's memorandum of understanding with the City of Columbia Fire Department and found that the department's training and participation in drills, hazmat certification of personnel, and response times acceptable. Additionally, the NRC staff met with the University of Missouri Hospital personnel and considered the hospital's (1) ambulance response and transfer of a potentially contaminated person to an isolated decontamination room, (2) decontamination prior to patient transfer to the emergency room area, (3) ambulance decontamination, and (4) participation in drills with Callaway Nuclear Power Plant and determined that the hospital is well prepared to handle any medical emergency. Based on the information above, the NRC staff concludes that the City of Columbia Fire Department and the University of Missouri Hospital are capable of handling any fire or medical emergency at MURR during the license renewal period.

In its renewal application (LRA), the licensee indicates no changes were needed to the MURR EP. However, as part of its review of the LRA, the NRC staff reviewed Revision 17 to the MURR EP, dated October 17, 2014 (Ref. 81). The NRC staff completed its review and by letter dated December 9, 2015 (Ref. 82), and acknowledged that the MURR EP, Revision 17, dated October 17, 2014, complies with the regulations and is consistent with the applicable guidance. The NRC staff routinely inspects the licensee's compliance with the requirements of the EP, and no violations have been identified in recent years based on the NRC staff's review of inspection reports for years from 2010 to 2016 (Refs. 45, 46, 47, 48, 49, 50 and 84). The NRC staff concludes that the licensee is required to maintain the EP, in compliance with regulation 10 CFR 50.54(q), "Emergency Planning," which require research reactor EPs to adhere to the requirements in Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," and which provides reasonable assurance that the licensee will be prepared to assess and respond to emergency events.

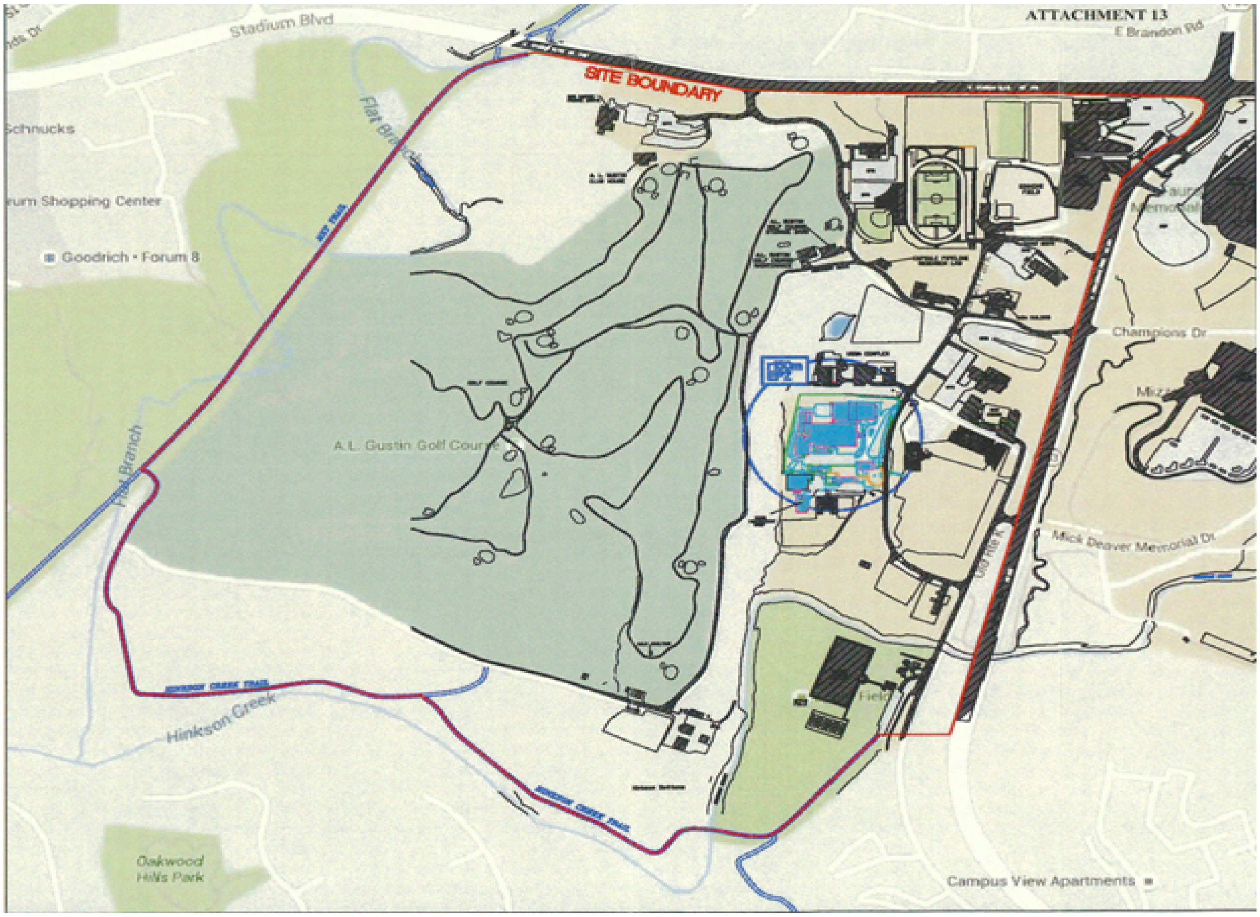

Figure 12-2 MURR emergency planning zone

12.8 Security Planning

In its LRA, the licensee indicates no changes were needed to the MURR PSP. However, as part of its review of the LRA, the NRC staff reviewed the PSP entitled, "Physical Security Plan for the University of Missouri Research Reactor," Revision dated October 8, 2013, and as changed under 10 CFR 50.54(p). Changes to the physical security plan can be made, by the licensee, in accordance with 10 CFR 50.54(p), as long as those changes do not decrease the effectiveness of the plan. The NRC staff issued RAIs to the licensee in a letter dated September 14, 2014 (Ref. 79), and the licensee provided its responses by letters dated December 2, 2015 (Ref. 80), and November 15, 2016 (Ref. 105), which included a revised MURR PSP.

The NRC staff reviewed the revised MURR PSP, and finds that the MURR PSP is in compliance with the applicable regulations contained in 10 CFR Part 73, "Physical Protection of Plants and Materials," in accordance with Regulatory Guide 5.59, "Standard Format And Content For A Licensee Physical Security Plan For The Protection Of Special Nuclear Material of Moderate Or Low Strategic Significance," and the site specific security measures committed to in the Confirmatory Action Letter, dated October 28, 2002 (ADAMS Accession No. ML022810673). The licensee maintains the program to provide the physical protection of the facility and its SNM in accordance with the requirements of 10 CFR Part 73. Changes to the PSP can be made, by the licensee, in accordance with 10 CFR 50.54(p), as long as those changes do not decrease the effectiveness of the plan.

In addition, the NRC staff performs routine inspections of the licensee's compliance with the requirements of the PSP. The NRC staff's review of the NRC inspection reports from the years 2010 through 2016 (Refs. 45, 46, 47, 48, 49, 50, and 84) for the MURR facility identified no violations of the PSP requirements.

Based on its review, the NRC staff finds that the licensee maintains a PSP for the facility and its SNM, in accordance with the requirements of 10 CFR Part 73. Therefore, based on the information above, the NRC staff concludes that there is reasonable assurance that the licensee will continue to provide physical protection of the facility and its SNM, and that continued operation of the MURR will not be inimical to the common defense and security.

12.9 Quality Assurance

SAR Section 12.8 describes the QA Program, which involves the use, testing, maintenance, and repair of shipping containers identified by 10 CFR Part 71, "Packaging and Transportation of Radioactive Material." Any activity that could significantly affect the ability of such a structure, system, or component to perform safely and as specified falls within the scope of the QA Program. Shipping casks covered under 10 CFR Part 71 will be released for shipping only after they have satisfactorily met the requirements of the QA Program. The licensee states that the Associate Director of the MURR Regulatory Assurance Group is responsible for the QA Program, which shall be annually reviewed and revised as necessary.

The NRC staff finds the scope and oversight of the QA program consistent with the guidance in NUREG-1537. Based on the information provided above, the NRC staff concludes that the MURR QA Program is acceptable.

12.10 Operator Training and Requalification

SAR Section 12.9 describes the MURR Operator Requalification Program which is designed to provide assurance that all operators certified at the reactor operator and senior reactor operator levels, pursuant to 10 CFR 55, maintain competence and proficiency in all aspects of licensed activities. The licensee states that the objectives of the program are to review/retrain in areas of infrequent operation, to review facility and procedural changes, to address subject matter not reinforced by direct use, and to improve in areas of performance by direct use and to improve in areas of performance weakness.

In its LRA, the licensee indicates that no changes were needed to the MURR Operator Requalification Program. However, as part of its review of the MURR LRA, the NRC staff reviewed the MURR Operator Requalification Program, dated January 7, 1997, that the licensee provided by letter dated January 4, 2013 (Ref. 83).

The NRC staff completed its review and by letter, dated April 9, 2013 (Ref. 55) approved the University of Missouri – Columbia Operator Requalification Program, dated January 7, 1997. The NRC staff concluded that the MURR Operator Requalification Program, dated January 7, 1997, meets the applicable requirements of 10 CFR Sections 55.41, 55.43, 55.53, and 55.59 and consistent with guidance in ANSI/ANS-15.4-2007, "Selection and Training of Personnel for Research Reactors."

12.11 Startup Plan

NUREG-1537, Section 12.11 states that a startup plan is required for a new facility and for license amendments authorizing modifications that require verification of operability before normal operations are resumed. However, since the licensee has operated MURR for many years, and is not submitting any facility modifications with LRA, the NRC staff finds that a startup plan is not required as part of the license renewal review.

12.12 Conclusions

The NRC staff reviewed SAR Chapter 12, as supplemented by responses to RAIs, and the applicable specification in TS Chapter 6, which discusses 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 NUREG-1537 and the ANSI/ANS-15.1-2007. The NRC staff finds that the licensee's proposed conduct of operations in the areas reviewed is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. The NRC staff also reviewed the applicable proposed MURR TS Chapter 6 against the requirements in 10 CFR 50.36 "Technical Specifications," including 10 CFR 50.36(c)(5) and (7) and finds that the TSs meet the requirements of the regulations.

Based on its reviewed of information above, the NRC staff finds that the licensee has sufficient oversight, management positions and responsibilities structure, and procedures to provide reasonable assurance that the reactor will continue to be managed in a way that will not cause significant risk to public health and safety. The NRC staff also finds that the licensee's procedures for training its reactor operators and the operator requalification plan give reasonable assurance that the licensee will continue to have qualified personnel who can safely operate the reactor.

Based on the above discussion, the NRC staff concludes that MURR has the appropriate organization, experience levels, and adequate controls through the TSs to provide reasonable assurance that MURR is managed and operated in a manner that will not endanger the facility staff or the public during the renewal period.

13 ACCIDENT ANALYSES

Chapter 13 of the safety analysis report (SAR), as supplemented by the licensee's responses to requests for additional information (RAIs), describes a series of accident analyses to demonstrate that the health and safety of the public and workers are protected during analyzed reactor transients and other hypothetical accident scenarios. The accident analyses provide the basis to establish the University of Missouri-Columbia Research Reactor (MURR or the reactor) technical specifications (TS) described in this safety evaluation report (SER). The accident analysis helps ensure that no credible accident could lead to unacceptable radiological consequences to the MURR staff, the public, or the environment. Additionally, the licensee analyzed the consequences of a maximum hypothetical accident (MHA), which is considered the worst-case accident scenario for MURR that would lead to the maximum potential radiation hazard to facility staff and/or members of the public. The results of the MHA are used to evaluate the ability of the licensee to respond and mitigate the consequences of this postulated radioactive release.

NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," issued February 1996 (Ref. 51), recommends that licensees consider the applicability of each of the following accident scenarios:

- fuel failure
- insertion of excess reactivity
- loss of coolant
- loss of coolant flow
- mishandling or malfunction of fuel
- experiment malfunction
- loss of electrical power
- external events
- mishandling or malfunction of equipment

In SAR Section 13.2, the licensee describes and identifies the MHA as the fuel failure during reactor operation. However, in subsequent responses RAI, the licensee determined that the failed fueled experiment could lead to potential occupational and public doses that were greater. As such, the licensee subsequently identifies the failed fueled experiment accident as the MURR MHA. Therefore, the U.S. Nuclear Regulatory Commission (NRC) staff's review will include both accident scenarios. The MHA (failed fueled experiment) and the fuel failure during reactor operation are discussed in SER Sections 13.1 and 13.5.2, respectively.

13.1 Maximum Hypothetical Accident—Failed Fueled Experiment

In its RAI responses, the licensee states that it has determined that the failed fueled experiments this accident is now the MHA for MURR (Ref. 34, page 56 of 67), and provides an analysis of this accident scenario (Ref. 36). The release of the radioisotopes of krypton (Kr), xenon (Xe), and radioiodine from a 5-gram (g) low-enriched uranium experiment serves as the basis for the source term for the dose calculations of this accident. The licensee adds that a complete failure of the fueled experiment is unrealistic; the worst that can be expected is a partial melting and a partial release. However, the accident analysis assumes that 100 percent of the total activity of the experiment is released into the reactor pool.

MHA Scenario

The licensee states that the MHA scenario assumes that the accident occurs after a 5-gram LEU experiment is irradiated for 150 hours (normal weekly operating cycle) at a thermal neutron flux of 1.5×10^{13} n/cm^2-sec, producing about 150 Ci of total Iodine. In order to present a worst-case dose assessment for an individual that remains in the containment building following experiment failure, the scenario assumes that 100 percent of the total activity in the experiment is released into the reactor pool. The scenario also considers that a reactor scram and actuation of the containment building isolation system occurs by action of the pool surface radiation monitor. Actuation of the isolation system will prompt operations personnel to ensure that a total evacuation of the containment building is accomplished promptly. This evacuation is usually completed within 2.5 minutes. A conservative 5-minute evacuation time is considered as the basis for the stay-time in the dose calculations for facility staff that are in containment during experiment failure

In its response to RAI No. 6.c and RAI No. 10.a (Ref. 33), the licensee states that MURR performs an evacuation drill every year and the typical time period for all personnel to evacuate the containment building, including verification by operations personnel, is 2.5 minutes. However, for the purposes of the MHA and fuel handling accident (FHA) calculations discussed in SER Section 13.5.1, an assumption of 5 minutes is used. The NRC staff considers the stay time of 5 minutes to be reasonable and conservative, because in comparison to accidents involving fuel failure during operation, the MHA and FHA accidents do not require reactor staff to secure the pool cooling system, which leads to additional stay time in containment (see SER Section 13.5.2).

The licensee also states that the radioiodine released into the reactor pool is conservatively assumed to be instantly and uniformly mixed into the 20,000 gal (75,708 l) of bulk pool water. The noble gas fission products are assumed to rise through the pool instantaneously (although, as discussed below, they are assumed to have decayed for a short period of time prior to their release in the air, to represent the time for the gas bubbles to rise to the pool surface), and radioiodine enters the containment ambient air through pool evaporation. As discussed below, the licensee assumed the evaporation to be 20 gal (75.7 l) over 5 minutes, because following this amount of evaporation, the containment air is at 100 percent relative humidity.

In addition, the scenario assumes that gaseous fission products form a uniform concentration in the containment air. For calculation of occupational doses to the facility staff inside containment, it was conservatively assumed that no radioactive material leaks from containment during the 5-minute stay time. Because the containment is isolated by the action of the pool surface radiation monitor, any release into the environment would be through containment leakage that enters the laboratory building that surrounds the containment structure, mixes in the laboratory building ventilation system, and exhausts through the main exhaust stack at a rate of approximately 30,500 standard cubic feet per minute (scf/m) (863.67 m^3 per minute (m^3/m)?. The licensee states that the laboratory building ventilation system would remain operating following the accident.

The NRC staff reviewed the assumptions for the release to the environment, and finds that keeping the laboratory building ventilation system on following the accident would help maintain doses to the facility staff and the public ALARA. Keeping the laboratory ventilation system on would help reduce radionuclide concentrations in the laboratory areas of the facility, would help ensure that any effluents are diluted before release, and would help ensure that effluents are released from the stack, providing additional dilution after release. The NRC staff also finds that the assumption that the effluents are released through the main exhaust stack appears valid, given that the laboratory building exhaust system has two redundant 100 percent capacity fans (one operates while the other remains in standby mode), and both fans are connected to the emergency power electrical bus, which provided electrical power by the emergency diesel generator in the case of a loss of normal electrical power (see SAR Sections 8.2.4 and 9.1.2.2). Additionally, and consistent with the assumptions in the safety analysis, TS 3.4, Specification a.(6), requires a negative pressure in the reactor containment building of 0.25 inches, which indicates that the exhaust fans must be operating during operation of the reactor or when irradiated fuel with less than 60 days of decay is being handled as required by TS 3.4, Specification b. These TSs were evaluated and found acceptable in SER Sections 6.2 and 11.2.2, respectively.

The NRC staff finds the above MHA scenario assumptions would lead to conservative conditions for dose estimates to both the occupational workers and the members of the public, since all gaseous fission products in the experiment are released. Based on its review, the NRC staff concludes that the licensee's assumptions are reasonably conservative, and acceptable.

Nuclide Inventory

In Attachment 16 of its RAI responses (Ref. 34), the licensee uses the ORIGEN 2 computer code to calculate the MHA source term radionuclide inventories (Ref. 110). The ORIGEN (Oak Ridge Isotope Generation) 2 code was developed by Oak Ridge National Laboratory to compute isotope generation and depletion and its use is consistent with established nuclear industry practice. The ORIGEN 2 code output assumes that a 19.75 percent U-235 experiment is irradiated for 150 hours with a neutron flux density of 1.5×10^{13} n/cm^2-second at a power level of 7.13×10^{-4} MW. Approximately 0.014 Ci of Sr-90 is also generated in the experiment.

The NRC staff reviewed the input used for the estimation of the inventories and finds the input used to be consistent with the assumptions in the analysis. The use of the ORIGEN 2 code to calculate the MHA source term radionuclides is an industry-accepted code and methodology. The licensee's provided MHA source term radionuclide and the inventories are listed in Table 13-1 below.

The NRC staff performed two confirmatory calculations to generate the inventories presented in Table 13-1, below. The values in the first column (Adjusted MURR Estimate from Analysis of Record) are derived from the fission product inventory for the entire MURR core, which is provided in SER Table 13-13 (see SER Section 13.5.2). Additionally, these values are derived by adjusting the values in SER Table 13-13 by the ratio of 5 grams (g) of target U-235 in a fueled experiment to the total U-235 in the core (5,474 g). In addition, in order to ensure that these values are consistent with the licensee's performance of fueled experiments in compliance with TS 3.8, Specification f (see SER Section 10.3), which limits the total inventory of I-131 through I-135 in fueled experiments to 150 Ci, the inventory values were normalized such that the value for total iodine matched the licensee's ORIGEN 2 code total iodine inventory. The second column (Based on Industry Average Fission Yields and Target Power) inventories are calculated using the fission yields and the target power level of 7.13×10^{-4} MWt. These values are then normalized in a manner similar to the first column (Adjusted MURR Estimate from Analysis of Record). Table 13-1 also provides a comparison of estimates of radionuclide inventories for select halogens and noble gases, as provided by the licensee; and those estimated by the NRC staff in the two confirmatory calculations. Even though there are differences in the estimated inventories in Table 13-1, the overall results are very similar for many of the more radiologically important isotopes. Although the licensee and NRC staff values for total iodine (148.365 Ci) are below the 150 Ci TS limit, the NRC staff finds that this difference is small (about 1 percent), and consequently any difference in calculated doses from using the slightly smaller inventory would also be small (also about 1 percent).

Based on the information above, the NRC staff finds the licensee's estimate of the MHA source term inventories acceptable.

Table 13-1 MHA - Licensee and NRC MHA Inventory Estimates

Nuclides	Licensee Estimate from ORIGEN (Ci)	NRC Confirmatory Estimates	
		Adjusted MURR Estimate from Analysis of Record (Ci)	Based on Industry Average Fission Yields and Target Power (Ci)
I-131	6.755	11.68	15.33
I-132	18.635	22.67	22.85
I-133	39.875	35.03	35.61
I-134	45.405	43.27	41.12
I-135	37.695	35.72	33.44
Total Iodine	148.365	148.365*	148.365*
Kr-85m	7.58	7.56	6.71
Kr-85	0.002	0.03	1.46
Kr-87	15.405	14.42	13.37
Kr-88	21.660	20.61	18.97
Kr-89	27.740	26.10	25.31
Kr-90	27.410	26.10	30.83
Total Krypton	99.797	94.820	96.65
Xe-133	18.925	28.85	35.63
Xe-135m	6.762	6.46	6.45
Xe-135	13.63	6.59	34.72
Xe-137	35.800	33.66	32.49
Xe-138	37.380	35.72	33.88
Xe-139	30.68	28.85	34.02
Total Xenon	143.170	140.12	177.20

* The individual inventories were adjusted to have similar total iodine.

Release Fractions

The licensee states that it assumes that the noble gases and iodine in the experiment are instantaneously released to the reactor pool. The iodine radionuclides uniformly mix in the pool water and enter the containment air through pool water evaporation. For calculation of occupational doses, this evaporation was assumed to occur gradually over a period of five minutes; for calculation of public doses, the evaporation was assumed to occur instantaneously. The licensee stated that containment air with a temperature of 75 degrees F (23.9 degrees C) and 100 percent relative humidity contains water vapor equivalent to 40 gal (151.4 l) of liquid water. Therefore, assuming an initial relative humidity of air in the containment of 50 percent, the analysis considers 20 gal (75.7 l) of pool water evaporation to result in 100 percent relative humidity. Once 100 percent relative humidity is reached, no additional pool water can evaporate. Based on these considerations, the calculated equivalent iodine release fraction is then 0.1 percent, or 20 out of the 20,000 gal (75,700 l). For the noble gases, the licensee assumes no retention in the pool water. For the occupational dose calculations, the licensee assumed that the noble gases and radioiodines entering the containment air had undergone 17 seconds of radioactive decay before their release to the air, reflecting the gas bubble rise time from the reflector region to the pool surface (for the radioiodines, this 17-second decay time is in addition to the decay prior to evaporation). The licensee states that the 17-second bubble rise time is a measured value unique to the conditions in the MURR pool. This licensee used this time delay to decay all released gaseous fission products (both the noble gases and iodines) before they enter the containment air, either by instantaneous release (for the noble gases) or by gradual pool water evaporation (for iodines). The licensee did not apply the 17-second time delay for the public dose calculations.

The analysis assumes that the release fractions to the environment to be 50 percent of the values for halogens due to retention and plate-out and 100 percent of the values (no retention) for noble gases in the containment air.

The NRC staff reviewed the licensee's assumptions on the releases of the iodines and noble gases to the containment and the environment. The NRC staff finds the 50 percent release fraction to the environment for halogens due to retention and plate-out, and the 100 percent release fraction to the environment for noble gases, to be consistent with established nuclear industry practice. These release fractions to the environment are conservative because they assume that all of the noble gases, and half of the halogens, would be released to the environment, which in actuality more than half of the halogens would likely remain in containment due to adsorption and adherence to surfaces within containment and/or settling. Therefore, the NRC staff finds these release fractions to the environment to be acceptable. The NRC staff also finds the 0.1 percent release fraction for radioiodines in pool water evaporating to containment to be reasonable and conservative, and therefore acceptable, over a short period of time such as the 5- to 10-minute stay time assumed for the occupational dose calculations for the MHA and the fuel failures discussed in SER Section 13.5. However, the NRC staff finds that over a longer period of time, such as the 16.5 hour release/exposure period assumed for public dose analyses, additional pool water could evaporate if the relative humidity is allowed to decrease below 100 percent. Therefore, in its confirmatory analysis of MHA doses to members of the public, which is described below, the NRC staff conservatively assumes that 100 percent of the radioiodines that are released to the pool are also instantaneously released to the containment building at the beginning of the release/exposure period.

As stated above, for the occupational dose calculations, the licensee assumes a delay time of 17 seconds for iodine release to the containment. The NRC staff considers the assumption of the additional time delay in decay calculations for iodine to be inconsistent with the assumption that the released iodine is instantaneously mixed in the pool water and then enters the containment through gradual pool water evaporation. Therefore, the NRC staff's confirmatory analyses, which are described below, do not consider this delay time in any calculation of the occupational or public doses. As explained below, this additional decay time will not have any significant effect on the calculated occupational dose from iodine.

Atmospheric Dispersion

The licensee used a constant dilution factor of 292 to determine the concentration of the released radionuclides at the nearest residence, 760 m (2,493 ft) meters north of MURR. In its response to RAI No. 7.g.vi (Ref. 34), the licensee states that the 292 dilution factor is the ratio of the TS 3.7, Specification b maximum Ar-41 release concentration limit of 3.5×10^{-6} microcuries per cubic centimeters (μCi/cc) and the downwind concentration at the point where the highest calculated Ar-41 concentration (1.2×10^{-8} μCi/cc) occurs. Therefore, this factor represents the worst-case scenario dilution factor for the nearest residence, based on the Pasquill-Guifford model and any stability class. The licensee states that although this dilution factor is determined based on Ar-41, the dilution factor will remain constant regardless of the isotope or isotope concentration that is emitted at the end of the exhaust stack. The licensee further states that for all other locations greater than 100 m (328 ft) north of MURR, the dilution factors are greater than 292. The licensee states that the 292 dilution factor, corresponding to the nearest residence 760 m (2,493 ft) north of MURR, is most conservative.

The NRC staff reviewed the information above and the licensee's Ar-41 dose calculations described in SAR Appendix B and in response to RAI No. 5.b (Ref. 33). The NRC staff finds that the cited dilution factor (292) is smaller (more conservative) than any dilution factor calculated for Ar-41 at the nearest residence. However, the NRC staff also reviewed LA No. 37 relating to an experiment to produce I-131 solution at MURR (Ref. 90), and noted that for the accident analysis provided by MURR (Ref. 99) in conjunction with that amendment request, which was also based on the Pasquill-Guifford dispersion model, the highest potential dose to a member of the public would occur at another location 400 m (1,312 ft) north of MURR. Although this location is not continually occupied by members of the public, it is a location where members of the public could potentially be present. Since the accident analysis for LA No. 37, like the MHA, involves a release of radioactive material from the stack, the atmospheric dispersion, dispersion factors, and location where the maximum dose is received should be similar (although the potential doses will differ, because the type and quantity of released material differs).

The NRC staff reviewed information provided by the licensee in response to RAI No. 4.i associated with LA No. 37 (Ref. 99), regarding the determination of the worst-case offsite public dose for the I-131 experiment accident evaluation. The NRC staff previously reviewed and accepted this information (Ref. 90). Using this information, the NRC staff independently calculated a worst-case dilution factor of 112 for analysis of accident doses to a member of the public located 400 m (1,312 ft) north of MURR. (This location is slightly different than the highest-dose location for Ar-41, 350 m north of MURR, discussed in the Ar-41 evaluation in SER Section 11.1.1, because the Ar-41 analysis assumes varying meteorological conditions over the course of one year, while the MHA analysis uses worst-case meteorological conditions. Both the location, and the value, of the highest dose due to an airborne release of radioactive material vary depending on meteorological conditions.) This dilution factor is lower (more conservative) than the dilution factor of 292 cited by the licensee. Therefore, the NRC staff's confirmatory analysis of MHA doses to members of the public, which is described below, includes an evaluation of doses that is based on the more conservative worst-case dilution factor of 112.

Dose Calculations

In its RAI response (Ref. 36), the licensee provided an updated Failed Fueled Experiment analysis. The licensee calculated the potential MHA total effective dose equivalent (TEDE) for an occupational worker in the containment, and the public TEDE at the nearest location outside the reactor building. Boundary conditions for these calculations included assuming the following: (a) the failure of the fueled experiment; (b) containment minimum volume of 225,000 cu ft (6,371.29 m^3) (used for the occupational dose calculations; the licensee's public dose calculations used a larger containment volume, as discussed below); (c) reactor pool water volume of 20,000 gal (75,708 l); and (d) the evaporation of 20 gal (75.7 l) of pool water.

The licensee also considered the following assumptions:

- Restricted area inside the containment structure:
  - the occupational workers will be exposed to the airborne gaseous fission products with credit for radionuclide decay,
  - the containment ventilation system is off and isolated,
  - an evacuation time of 5 minutes for the occupational workers is needed to secure the reactor to a safe condition.
- Unrestricted area outside the MURR facility for members of the public:
  - the laboratory ventilation system is operating to carry the gaseous fission products that leak from the containment into the building ventilation stack with a ventilation flow rate of 30,500 scfm (863.666 m^3 per minute),
  - credit is taken for the radionuclide decay inside the containment,
  - containment leakage duration of 16.5 hours is needed for the leakage rate to become zero (see explanation with public dose estimates below).

For the occupational and public dose calculations, the licensee followed the DAC and AEC approach of 10 CFR Part 20, Appendix B. Appendix B of 10 CFR Part 20 provides generic submersion and inhalation DAC values for radionuclides that have decay modes other than alpha emissions and half-lives less than 2 hours, and that are not otherwise listed in 10 CFR Part 20, Appendix B. Because the list of the radionuclides in the MHA included isotopes with very short half-lives that are not listed in 10 CFR Part 20 Appendix B, the licensee developed nuclide specific DAC values for the missing radionuclides, rather than using the generic values. The MHA identifies four radionuclides (Kr-89, Kr-90, Xe-137, and Xe-139) that are not listed in Appendix B. The licensee used the approach and data in Table II.4 and Figure II.25 in Federal Guidance Report (FGR) No. 12 (Ref. 77) to determine the submersion dose values from gamma and beta emissions, respectively. The licensee provided these calculations in the RAI response Attachment 4 (Ref. 36). Table 13-2 lists the calculated DACs for the aforementioned four nuclides.

Table 13-2 Calculated Occupational DAC Values

Nuclide	DAC (µCi/cc)
Kr-89	$1.9x10^{-6}$
Kr-90	$2.8x10^{-6}$
Xe-137	$2.0x10^{-5}$
Xe-139	$3.7x10^{-6}$

The NRC staff reviewed the process, input data, and the calculation approach and finds the results to be consistent with the intent of the regulations in Appendix B to 10 CFR Part 20 and FGR No. 12. Therefore, the NRC staff concludes that the calculated values can be used as the nuclide-specific DAC values for the occupational dose calculations.

Occupational Dose Estimates

As discussed above, the licensee assumed that 100 percent of the noble gas radioisotopes released to the pool are also released to containment, and all noble gases are released simultaneously. One tenth of a percent (0.1 percent) of the radioiodines released to the pool are released to containment, and these radioiodines are released gradually over a five-minute evaporation period. Also as discussed above, the licensee took credit for 17 seconds of decay to account for bubble rise for both radioiodines and noble gas isotopes. The licensee also took credit for the decay of the radioiodines in the pool prior to their release into the containment structure, as well as the decay of both radioiodines and noble gases after their release into containment. For the occupational dose calculations, the licensee used five 1-minute time steps first to determine evaporation volumes and the related containment concentration values for each radionuclide in conjunction with its decay constant, and then to calculate the average concentrations over the 5-minute evacuation time. Using the assumptions and boundary conditions described above, the licensee calculated the potential occupational dose for an individual inside the containment. As discussed above, in these calculations, the licensee extended the decay time by the 17-second time delay for the bubble rise for both iodine and noble gas nuclides.

The NRC staff reviewed the licensee's analysis approach and the related calculations. The NRC staff finds the licensee’s use of the additional 17-second time delay in decay calculations for iodine to be inconsistent with the assumption that the released iodine is instantaneously mixed in the pool water and then enters the containment through gradual pool water evaporation. The 17-second bubble rise time is not applicable for iodine, since unlike the noble gases, the iodines are not released in bubbles that rise to the surface. The NRC staff addresses this in its confirmatory calculations.

In addition, the NRC staff notes that the licensee’s decay calculations did not appear to consider the buildup of daughter products of the decayed radionuclides, which could be present in the same environment. For example, both iodine (I)-135 and xenon (Xe)-135 are present. As I-135 decays, Xe-135 builds up because it has a longer half-life than its parent nuclide (I-135). Therefore, when radioactive decay is credited, the radiological dose calculation needs to consider both nuclides during the decay process when both the parent and progeny nuclides are present. Alternatively, radioactive decay can, conservatively, be neglected altogether. The

NRC staff's confirmatory analysis, which is discussed below, does not consider radioactive decay in its calculations.

The NRC staff adjusted the licensee's calculations to show how the results of the licensee's calculated doses vary with and without the use of the 17-second time delay decay assumption for radioiodines and noble gases. The NRC staff also performed a separate confirmatory analysis assuming no decay of the radioiodines or noble gases at any point in time, either when they are still in the pool, or once they have been released to containment. The NRC staff's analysis additionally assumed that the 20 gal (75.7 l) of pool water containing radioiodines evaporates instantaneously, in contrast with the licensee's analysis, which assumed that the pool water evaporates over 5 minutes. For the confirmatory analysis, the NRC staff used the licensee's DAC analysis approach as discussed above, and the licensee's MHA inventory. Table 13-3 provides the comparison of the licensee-calculated and NRC-calculated occupational doses within the 5-minute evacuation time. This table shows that the use of the 17-second time delay does not have a measurable effect on the committed effective dose equivalent (CEDE) from the iodine, and the change in the deep dose equivalent (DDE) from the noble gases is small. Additionally, the results of the NRC staff's confirmatory calculation show that the expected occupational TEDE is well below the 5,000 mrem regulatory occupation dose limit of 10 CFR 20.1201, without taking credit for the radionuclides' decay.

The licensee's dose calculations do not account for contributions from semi-volatiles such as Cesium and Strontium. The licensee identifies the potential inventories of the Sr-90 and the ORIGEN 2 code output lists the inventory for the Cs-137. The licensee adds that the addition of Sr-90 will increase the above stated TEDE (whole body) by less than one percent. The NRC staff performed an additional analysis to determine the impact of these nuclides, conservatively assuming that they behave like iodine. The NRC staff calculated the contributions from Cs-137 and Sr-90 and found that they were very small (less than 0.01 percent of the total TEDE), and concluded that the licensee's assumption that the dose contribution was negligible, was acceptable.

Public Dose Estimates

For the public dose calculations, the licensee calculated the potential containment leakage rate based on the assumption of air leakage that could occur as a result of normal changes in atmospheric pressure and pressure equilibrium between the inside of the containment structure and the outside atmosphere. The licensee assumes that, due to changing weather conditions, a barometric pressure drop had occurred outside containment in conjunction with the experiment failure, and considers a pressure differential of one inch of mercury (0.333 psig) between the inside of the isolated containment building and the inside of the adjacent laboratory building. Using this pressure differential and the following equation for the air leakage rate from containment, the licensee calculates average leakage rates for multiple time steps until an equilibrium pressure is established between the buildings.

Table 13-3 MHA - 5-minute Occupational Dose Estimates in the Restricted Area

Dose Parameters	MURR-Calculated Dose (mrem)	NRC-Adjusted MURR-Calculated Dose, without 17-second delay (mrem)	NRC Confirmatory Analysis, without 17-second delay (mrem)	10 CFR 20.1201 Occupational Dose Limit (mrem)
Committed effective dose (thyroid)	130.36	130.37	261.11	50,000
Committed effective dose equivalent (CEDE)	3.91	3.91	7.85	5,000
Deep dose equivalent (DDE)	1176.42	1227.31	1975.26	5,000
Total effective dose equivalent (TEDE)	1180.33	1231.22	1983.11	5,000

$$\text{Leak Rate} = 17.68 \times (\text{Containment Pressure} - 14.7)^{1/2}$$

To determine the average leak rate over each time period, the licensee uses a recursive expression to determine pressure drop after each time step until the leakage rate decreases to zero. The licensee determines that it would take about 16.5 hours for the leakage rate to become zero. Any radionuclides contained in the portion of the containment air that does not leak out during this 16.5-hour period remain in the containment and are not released to the environment.

For determining the containment air concentrations used to determine environmental releases, the licensee considered the 20 gal (75.7 l) pool evaporation to occur instantaneously, and then decayed each radionuclide in nine time periods with durations ranging from 1 to 4 hours. In this calculation, the licensee uses a containment volume of 229,800 standard cubic feet (scf) (6,507 m^3). This volume is larger than that used in the occupational dose calculation (225,000 scf) (6,371 m^3), which is the TS minimum volume to be used. Using these concentrations and calculated average leakage rate over each time period, the licensee calculated the average radionuclide concentrations in the air released from the building exhaust stack, and determined the concentration at the nearest residence at a distance of 760 m (2,493 ft) by using a dilution factor of 292.

The licensee used the 10 CFR Part 20, Appendix B AEC approach to calculate the public dose. Similar to the discussions for the missing radionuclide DAC values in 10 CFR Part 20, Appendix B for the occupational dose calculations, the licensee generated nuclide specific AEC values for the four missing nuclides. To determine the AEC values, the licensee followed the guidance in Appendix B, Table 2, of 10 CFR Part 20 by dividing the calculated occupational DAC values by 219.

The NRC staff reviewed the licensee's analysis approach and the related calculations. As indicated above, the NRC staff considered the use of the 229,600 scf for the containment volume to be inconsistent with, and less conservative than, the accepted approach of using the minimum TS volume.

In reviewing the licensee's AEC approach for calculating public MHA dose, the NRC staff noted that the licensee's approach appears to be based on an incorrect assumption that the AECs for noble gases in Appendix B, Table 2, of 10 CFR Part 20 are based on a TEDE of 50 mrem per year. Because the AECs for noble gases in Appendix B, Table 2, of 10 CFR Part 20 are based on a TEDE of 100 mrem per year, dose calculations based on a TEDE of 50 mrem per year would have been less conservative than the interpretation based on a TEDE of 100 mrem per year. Therefore, the NRC staff's confirmatory analysis, which is discussed below, uses a DAC approach that considers noble gas AECs to correspond to a TEDE of 100 mrem per year.

The NRC staff also considered the licensee's method for taking credit for radioactive decay to be inconsistent with the need to consider daughter radionuclides that may build up when parent radionuclides decay. As discussed above, the NRC staff noted that over the 16.5-hour released exposure time used for the public dose calculations, greater than 20 gal (75.7 l) of pool water could potentially evaporate, leading to a release factor of greater than 0.1 percent for release of radioiodines from the pool to containment. In addition, the NRC staff determined that a dilution factor of 292, corresponding to the nearest residence, may not be appropriate for determining the worst-case dose to any offsite member of the public. As discussed above, a dilution factor of 112, corresponding to a location 400 m (1,312 ft) north of MURR, is more conservative for calculation of the worst-case public dose.

The NRC staff performed an independent confirmatory analysis of public doses from the MHA. This confirmatory analysis used several conservative assumptions that were not used in the licensee's analysis. First, the containment volume was assumed to be equal to the TS minimum containment volume. Second, the confirmatory analysis assumes no decay of the radionuclides within the pool or the containment. Third, this analysis also assumes that all of the radioiodines and noble gases released to the pool enter the containment air (i.e., there is no holdup in the pool), and that all containment air (containing all noble gases, and the 50 percent of radioiodines that do not plate out in the containment) enters the environment through the laboratory ventilation system over the 16.5-hour release period. Last, the confirmatory analysis used a dilution factor of 112, instead of the licensee's dilution factor of 292. For the confirmatory analysis, the NRC staff used the AEC analysis approach as discussed above, and the licensee's MHA inventory. Table 13-4 provides a comparison of the licensee- and NRC-calculated public doses for the 16.5-hour release and exposure period. The results of the NRC staff's confirmatory calculation show that the expected public TEDE is below the regulatory public dose limit of 100 mrem in 10 CFR 20.1301, even without taking credit for radioactive decay, using the TS minimum containment volume, using a conservative dilution factor of 112, assuming that all fission products in the reactor pool will enter the containment air, and assuming that all fission products in the reactor air (except the 50 percent of radioiodines that plate out in containment) enter the environment. These assumptions are very conservative because the radionuclides with short half-lives (in the range of seconds or minutes) will be decayed out well before the end of the 16.5-hour release and exposure period, and most of the radioiodine will remain in the reactor pool, where it will not impact workers or the environment.

Table 13-4 MHA - Public Dose Estimates in the Unrestricted Area

Dose Parameters	MURR-Calculated Dose (mrem)	NRC Confirmatory Analysis (mrem)	10 CFR 20.1301 Public Dose Limit (mrem)
Committed effective dose equivalent (CEDE)	2.95E-04	40.20	100
Deep dose equivalent (DDE)	1.09E-02	25.66	100
Total effective dose equivalent (TEDE)	1.12E-02	65.86	100

The licensee also performed a calculation of the radiation shine through the containment structure. This calculation represents a condition where the containment is isolated with no leakage. The calculation of exposure rate from the experiment failure, which was provided as Attachment 12 (Ref. 36) to the RAI response, was performed using the computer program MicroShield 8.02, which is a comprehensive photon/gamma ray shielding and dose assessment program that is widely used by industry for designing radiation shields. The source inventory for this calculation is the all of the gaseous fission products released to the reactor pool, which (unlike in the licensee's other MHA analyses discussed above) are all assumed to enter the containment air, with no decay. The licensee provided exposure rate values for the radiation fields at 1 ft (30.5 cm) from a 12 in (30.5 cm) thick ordinary concrete containment wall and at the EPZ boundary of 150 m (492.1 ft).

The NRC staff performed confirmatory dose calculations to members of the public at EPZ 150 m (492 ft) and at the nearest resident location 760 m (2,493 ft) assuming a confinement model and considering direct radiation from all radioactivity that was released into the containment (no leakage). The calculations conservatively assume that the gaseous fission products released into the containment will not decay, but take credit for the shielding of the released contents by the 1 ft (30.5 cm) concrete around the building. These assumptions are consistent with those used by the licensee using the MicroShield computer program. Furthermore, the calculations assume the building can be represented by a point source with the radiation emitted uniformly in all directions and each disintegration is accompanied by one 1-MeV photon (gamma rays). The NRC staff's confirmatory calculation uses a simple equation and is only an approximate method with no credit for decay energy distribution and air attenuation; therefore, the dose result is conservative. Table 13-5 summarizes the calculated direct radiation dose rates at the EPZ boundary and the nearest resident (the NRC staff did not perform a calculation for the location 1 foot from the building wall, because this is an area that is not publicly accessible).

Emergency Planning action levels are based on a projected exposure of 24 hours or less. Assuming no radioisotope decay, the calculated dose rate will remain constant for the duration of the stay time. Therefore, the 24-hour direct radiation dose (no release) at the EPZ for the MHA ranges between 12.34 mrem (MURR dose rate) and 22.58 mrem (NRC confirmatory), with no decay. This assumption is very conservative, because in few hours, about 42 percent of the source inventory will be decayed out. The NRC staff performed additional direct radiation dose calculations to an individual at the EPZ location for a 24-hour, 1-month, and 1-year stay time and took credit for the nuclides decays. The total DDE dose for the 24-hour, 1-month, and

1-year stay times were 4.87, 17.2, and 17.7 mrem, respectively. Although they account for decay, these results are conservative because they do not account for leakage from confinement, which would be significant over the extended periods considered. These results indicate that the dose to each individual member of public at the EPZ location will be less than the public dose limit in 10 CFR 20.1301 (100 mrem).

Table 13-5 MHA - Maximum Radiation Shine through the Containment Building

Parameters	MURR	NRC Analysis
Exposure rate 1 foot from the building wall (mrem/hr)	74.69	-
Exposure rate at emergency planning zone (mrem/hr)	0.514	0.941
Exposure rate at the nearest resident location (mrem/hr)	-	0.037

Based on its review of the licensee's dose calculations and the results of its confirmatory calculations, the NRC staff concludes that the MHA dose results (provided above) demonstrate that the maximum TEDEs are below the occupational limit in 10 CFR 20.1201 and the public dose limit in 10 CFR 20.1301.

13.2 Insertion of Excess Reactivity

SAR Section 13.2.2, describes insertion of excess reactivity accidents which may present two distinct challenges to MURR safety. Accidents that rapidly change system reactivity tend to present a limiting challenge to the fuel temperature safety limit (SL). Accidents that are gradual, such as a ramp insertion of reactivity, tend to challenge the ability of the reactor control and safety systems to respond with appropriate trip functions. This latter category usually results in lesser challenges to the SL but is just as important to the justification of the reactor control and safety systems.

13.2.1 Rapid Reactivity Insertion Accident

SAR Section 13.2.2 describes the evaluation of a rapid reactivity insertion accident. In its response to RAI No. 6.a (Ref. 34), the licensee presents its analysis. The initiating power is conservatively assumed to be 11.5 MWt. The fuel temperature coefficient used is -6.0×10^{-5} Δk/k (absolute reactivity) per degree Fahrenheit (°F); it is the least negative value allowed by TS 5.3, Specification b, and is a conservative assumption for this accident scenario. TS 3.8, Specification t, establishes the maximum allowed reactivity insertion of 0.006 Δk/k ($0.81). The licensee states that the objective of the analysis is to ensure that the MURR response to a reactivity insertion event does not result in operation that exceeds the SL (TS 2.1) for any core configuration.

The rapid reactivity insertion accident analysis for MURR is analyzed using the PARET code. To perform this analysis, the licensee assumes a conservative insertion of reactivity of 0.006 Δk/k ($0.81) from an initial reactor power level of 11.5 MWt. This is greater than the licensed power (10 MWt) and equal to the power that would cause a rod run-in (11.5 MWt); the run run-in function, which is conservative and is equivalent to the assumption of a single failure, is not credited.

The licensee assumes the coolant flow rate of 3,200 gallons per minute (gpm) (12,113 liters per minute (lpm)), pressurizer pressure of 75 pounds per square inch, absolute (psia), and a reactor core coolant inlet temperature of 155 °F (68 degrees Celsius (°C)) are used because they are the limiting safety system setting (LSSS) setpoints. This provides additional conservatism to the analysis. According to the licensee's PARET analysis, the measured peak reactor power would increase from 10 MW to approximately 37.4 MW. A 150-millisecond hysteresis effect delay is assumed before allowing the blades to inert. This is appropriately conservative. The high-power scram initiates at 12.5 MW (TS 2.2, Specification a). The event terminates with the highest fuel temperature of 227.4 °C (441 °F), which is well below the SL. In this analysis, the control blade insertion times are based on the TS 3.2, Specification c, requirement for blade insertion to the 20 percent withdrawn positon in less than 0.7 seconds. In its response to RAI No. 13.2.b (Ref. 27), the licensee states that the prompt power rise experienced by the reactor results in a temperature increase that is substantially less than the TS 2.1 SL because of the short-pulse duration and the lag in temperature rise caused by the slower heat transfer mechanisms of the fuel plates.

In its response to RAI No. 13.2.d (Ref. 25), the licensee investigates the effect of oxide layer buildup on reactivity transients. Using the PARET code, a 2.0-mil-thick (0.002-in-thick) oxide layer is added to a fresh fuel plate to construct a bounding conservative model of fuel temperature analysis. The licensee states that, with a 0.006-Δk/k reactivity step insertion, the peak fuel temperature increased from 214.7°C (418.5 °F) for the no-oxide layer case to 305.4 °C (581.7 °F) with the 2.0 mil oxide layer, which is consistent since the heat transfer from the fuel to the coolant is inhibited by the oxide layer thus raising the peak centerline temperature attained during the transient. The maximum temperature reached is still well below the melting point or the blister temperature of the fuel. The NRC staff reviewed the analysis and finds that the effect of a bounding layer of oxide formation on the hottest channel does not significantly challenge the SL.

The NRC staff reviewed the results of the step reactivity insertion analyses presented by the licensee and finds that the methodology used was consistent with the guidance in NUREG-1537, and consequences of this accident do not challenge the fuel temperature SL, even with a single failure, and concludes that no failure of the fuel element cladding or fission product release would be expected from such an accident under any mode of operation allowed by the TS.

13.2.2 Ramp Reactivity Insertion Accident

In SAR Section 13.2.2.1.2, the licensee provides analyses of different ramp reactivity insertion accidents.

Reactivity Insertion at the TS 3.2 Limit with Initiation at Subcritical Power

The license provides an analysis for a 0.0003 Δk/k per second (Δk/k/s) ($0.04 per second ($0.04/s)) reactivity insertion accident, corresponding to the limit in TS 3.2, Specification e, for all four shim control blades continuously withdrawing simultaneously. The reactivity worth of the regulating blade was not included as its simultaneous withdraw is prohibited by administrative control (see below). The accident is initiated from a subcritical condition (1.0 watt (W)) and is terminated by a short reactor period scram of 8 seconds. The entire transient is dominated by the effect of control blade withdrawal because power levels during the entire transient are low. No major feedback mechanisms exist to effect reactor behavior. Although the control blades have reduced reactivity worth at the top and bottom of the withdrawal, the maximum TS reactivity rate is used in the analysis. The final power attained, assuming the failure of the blade run-in, occurs after reaching a power level of 64 watts thermal (Wt).

Reactivity Insertion at TS 3.2 Limit with Initiation at Full Power (10 MWt)

The license also provided an analysis of the same ramp reactivity insertion accident; however, starting with the reactor at full power operation (10 MWt), rather than subcritical, and also using the TS insertion rate of 0.0003 Δk/k/s ($0.04/s). The results of this analysis show that the transient is terminated, approximately 4.5 seconds following initiation, when reactor power reaches the high-power level scram LSSS setpoint of 12.5 MWt.

Reactivity Insertion with all Four Shim Blades and the Regulating Blade

In its response to RAI No. 13.3 (Ref. 25), the licensee provide an analysis for the withdrawal of the regulating blade in addition to the four shim blades in the accident analyzed above starting at an initial subcritical power level. For this analysis, the maximum positive reactivity insertion rate allowed by the TSs from the withdrawal of the regulating blade is superimposed on top of the maximum-allowed reactivity insertion resulting from the simultaneous withdrawal of the four shim blades. The total reactivity worth of the regulating blade is limited to 0.006 Δk/k (TS 3.1, Specification c) and at a maximum reactivity addition rate of 0.00025 Δk/k/s (TS 3.1, Specification c). Because adding the effect of the regulating blade withdrawal increases the total reactivity addition made, the time for reaching the scram setpoint is reduced, and, consequently, the total power attained is lower. The reactivity addition from withdrawal of the regulating blade occurs only during the first 24 seconds of the continuous control blade withdrawal accident scenario. Beyond that, the only reactivity addition will be a result of shim blade motion. The resulting final power for this accident is 22.8 Wt.

In its response to RAI No. 1.b (Ref. 33), the licensee clarifies the use of scram times and reactivity in MURR accident analysis, which states that the

> *Reactivity Insertion Accidents are the only accidents analyzed by the licensee that explicitly use the control blade drop times (scram times) and their corresponding negative reactivity insertion rates. Because approximately 91 percent of the total control blade worth is inserted when the blades are 80 percent inserted from their fully withdrawn position (see Table 2 in response to Question 3.c) [provided in response to RAI No. 3.c, Ref. 34], this negative reactivity worth value (corresponding to the 80-percent-inserted position) is conservatively used for analyzing the step reactivity insertion accidents. As shown in the response to Question [RAI No.] 6.a, the most severe step reactivity insertion accident analyzed is terminated by the inherent negative reactivity effects combined with 91 percent of the total control blade worth.*

The NRC staff reviewed the limitations on the rate of reactivity insertion in conjunction with the ramp reactivity addition accidents analysis in SAR Chapter 13, as described in the three cases analyzed above. In the first two cases, the analyzed accidents used a conservative addition rate of 0.0003 Δk/k/s. In the first case, the reactor safety system (RSS) terminates the ramp after a minimum reactor period of 8 seconds is reached. The NRC staff finds that the total reactivity inserted is significantly below the allowable step insertion limit of 0.006 Δk/k (TS 3.1, Specification k, and TS 3.1, Specification c). In the second case, the transient is terminated by the high power LSSS setpoint (12.5 MWt) after 4.5 seconds. Although the licensee did not report fuel temperatures for these events, the NRC staff concludes that the SL is not challenged by these accident scenarios because of the low power level achieved prior to the short reactor period scram (8 seconds) and the response time for the LSSS scram (4.5 seconds).

In the third case, the NRC staff finds that the licensee's analysis indicates that reactor power rises continuously, reaching a value of 11.0 kilowatts at 140 seconds. No credit is taken for the RSS initiation of a rod run-in or scram at 120 seconds upon exceeding the short-period setpoint of 11 seconds. The NRC staff finds that initiation of the RSS at 120 seconds would reduce the maximum power reached to 22.8 Wt. This power level is much less than the LSSS power, which has previously been shown to result in fuel temperatures significantly below the SL (see SER Section 4.6).

The NRC staff reviewed the ramp reactivity insertion analyses and finds that the methodology used was consistent with the guidance in NUREG-1537, and the results of this accident do not challenge the fuel temperature SL because the transients are terminated at very low power levels or shortly after initiation of the LSSS high power setpoint. Based on its review, the NRC staff concludes that no failure of the fuel element cladding or fission product release would be expected from such an accident under any mode of operation allowed by the TS.

13.3 Loss-of-Coolant Accident

SAR Section 13.2.3, SAR Appendix C, and responses to RAI No. 13.4 (Ref. 19; Ref. 26; Ref. 27), RAI No. 13.5 (Ref. 17), RAI No. C.3 (Ref. 27), RAI No. 6.b (Ref. 33), and RAI No. 8 (Ref. 33) describe the loss-of-coolant accident (LOCA) analysis for MURR. The licensee considers the consequences of the double-ended rupture of the largest diameter primary coolant piping. For all pipe break conditions except for those that occur between the reactor pool and either isolation valve (hot leg or cold leg), the core is expected to remain covered because, the primary coolant isolation valves adjacent to the reactor pool penetrations would close to prevent the core from being uncovered. A rupture in a section of in-pool primary coolant piping would cause the reactor to scram and pool water to be admitted into the ruptured system until the isolation valves stop the flow, but, the core would remain covered, and no significant reactor safety issues would exist. In both cases, the decay heat would transfer to the reactor pool through the inner and outer reactor pressure vessels and in-pool heat exchanger (HX).

Therefore, the loss of primary coolant accident scenario considered a break in the hot leg or cold leg just outside of the reactor pool (biological shield) before the isolation valves. Once such a break occurs, upon loss of pressure, the following actions would occur:

- A reactor scram would occur.
- The primary coolant circulation pumps would stop.
- The primary coolant isolation valves would close.
- The anti-siphon system isolation valves would open.
- The in-pool HX isolation valves would open.
- The pressurizer surge line isolation valve would close.

When anti-siphon system isolation valves V543A and V543B open, the anti-siphon system pressurized air is applied to the top of the in-pool PCS inverted loop. If the break is in the cold leg, the air pressure and the back core backflow causes the inlet check valve on the cold leg to close within the first second of the transient, thereby isolating the core. If the break is in the hot leg, the air pressure that is admitted to the top of the inverted loop by the anti-siphon system causes the section of primary piping from the top of the inverted loop to the break location to quickly drain.

The licensee uses the RELAP5/MOD3.3 code to determine the peak fuel plate temperature after the accident. In this RELAP model, the licensee used the following modeling assumptions in comparison to the normal reactor operating parameters, as listed in Table 13-6, below.

Table 13-6 Normal and the Modeled Reactor Operating Conditions

Parameters	Normal Conditions	Modeled Conditions
Reactor power (MW)	10	11
Coolant inlet temperature (˚F (˚C))	120 (49)	155 (68)
Core inlet flow rate (gpm (lpm))	3,800 (14,385)	3,800 (14,385)
Pool Temperature ˚F (˚C)	100 (38)	120 (49)
Pressurizer pressure (psig)	62–66	60
Anti-siphon pressure (psig)	36	26

Using the modeled condition parameters in Table 13-6 above in the RELAP5 model, the licensee calculates the peak steady-state fuel temperature of 272.1 °F (133.4 °C) at the centerline of fuel plate No. 1. After the piping rupture, the peak fuel plate centerline temperatures take place within the first second of the transient. Because of the rapid decrease in primary coolant pressure, a reactor scram signal is automatically initiated, causing the control rods to drop. During the cold-leg break, the highest peak centerline temperature of 311.7 °F (155.4 °C) occurs in fuel plate No. 3 at 0.5 second after the rupture occurs. During the hot-leg break, the highest peak centerline temperature of 281.2 °F (138.4 °C) occurs in fuel plate No. 1 at 0.2 second. Because peak fuel plate temperatures remain more than 500 °F (260 °C) below the "no fuel plate blister verification temperature" of 900 °F (482 °C), the NRC staff finds that the methodology used (RELAP5) to be acceptable and the results of both LOCA scenarios are not expected to result in any fuel damage.

The NRC staff reviewed the information in the SAR, as supplemented by RAI responses, and finds that some parameters (i.e., reactor power and core flow) used in the RELAP model differed from those listed in the TSs and requested a reference for the 900 °F (482 °C) blister test. In its responses to RAI No. 13.4.a (Ref. 26) and RAI No. 6.b (Ref. 33), the licensee provided additional clarifications on the use of operating parameters that differ from those in the TSs. The licensee states that the facility license authorizes it to operate MURR up to a maximum steady-state power level of 10 MWt. Therefore, for the LOCA analysis, a maximum steady-state power level of 11 MWt or a 10-percent increase in the decay heat was used. The NRC staff finds that the steady-state operation is limited to a maximum of 10 MWt and the use of 11 MWt for the decay heat buildup before the accident is conservative. This is because, the LSSS power is not the power at which the reactor would be normally operate, and therefore should not be used for the determination of decay heat or inventory for this accident.

In its response to RAI No. 8 (Ref. 33), the licensee states that during a hot-leg break LOCA, the anti-siphon system actuates and injects air into the PCS vertical 12-in (30-cm) diameter piping above the inverted loop to the level of the in-pool HX outlet. The expanding air quickly voids the upper section of the potential PCS natural convention flowpath. The natural circulation flowpath through the in-pool HX is eliminated. Therefore, it precludes any further loss of coolant. The NRC staff reviewed the RAI responses and finds that the licensee's RAI response is acceptable, and the coolant loss assumptions are reasonable for the reasons stated.

In its response to RAI No. 6.b (Ref. 33), the licensee states that the loss of flow accident (LOFA) and LOCA analyses were performed with appropriate LSSS variables—core coolant inlet temperature, core coolant flow rate, and pressurizer pressure—at their respective setpoints of 155 °F (68.3 °C), 3,200 gpm (12,113 lpm), and 75 psia. The licensee adds that the peaking factors (PFs) used in the updated analyses are those used in support of License Amendment (LA) No. 36 (Ref. 65). The licensee states that the new peak fuel plate temperature for the cold-leg break LOCA is 413.9 °F (212.2 °C) and the peak fuel plate temperature for the LOFA is 292.3 °F (144.6 °C). The NRC staff finds that these temperatures are well below the SL peak fuel temperature of 986 °F (530 °C).

The NRC staff reviewed the response to RAI No. 6.b and finds that the revised fuel plate peak temperatures are well below the SL peak fuel temperature of 986 °F (530 °C). The NRC staff finds that the cause of the temperature differences between the new analyses and the previous analyses summarized above can be attributed to the changes in the fuel PFs, the primary coolant flow rate, and other assumptions in control and actuation timings related to isolation valves on the reactor coolant, anti-siphon, and in-pool HX. For example, the lower primary coolant flow rate will lead to a higher initial steady-state fuel plate temperature. The revised PF after LA No. 36 (Ref. 66), which is about 5 percent lower, leads to a lower generated decay heat that is effective after the trip. In addition, the assumption of the valve closure timing only affects the thermal-hydraulic response of the reactor core with the reactor pool.

In its response to RAI No. 13.4.b (Ref. 27), the licensee states that, because of the lack of the actual LOCA data, the RELAP5 code and model cannot be benchmarked for accuracy. However, the licensee has benchmarked the MURR RELAP5 code and model using LOFA data, as described in SER Section 13.4.

In its response to RAI No. C.3 (Ref. 27), the licensee provides discussions similar to those included in its response to RAI No. 13.4.a (Ref. 26) on the use of operational parameter values that differed from those in the TSs and added discussions on the effect of the oxide layer accumulation on the fuel plate. The NRC staff finds that the discussions on the operational parameters were evaluated above and found acceptable. On the effect of oxide layer accumulation on the fuel clad, the licensee states that the effect is offset by the reduction in heat flux because of the power history of the element. In its responses to RAI No. 4.15 (Ref. 26), the licensee indicates that the maximum fuel and clad temperatures occur in the hot spot of low burnup fuel element. The NRC staff evaluated the responses to RAI No. 4.15 and finds them acceptable in SER Section 13.2.1. This effect of the oxide layer is clearly presented in the licensee's response to RAI No. 4.15 (Ref. 26), which shows that the maximum fuel and clad temperatures occur in the hot spot of low burnup fuel elements. The NRC staff finds this conclusion acceptable based on the same justifications presented in SER Section 13.2.1.

The NRC staff reviewed the analysis and conclusions in SAR Section 13.2.3 and responses to RAI discussed above, and concludes that a LOCA would not result in the failure of the fuel element cladding or the release of fission products under any mode of operation allowed by the TSs.

13.4 Loss of Flow Accident

In SAR Section 13.2.4, SAR Appendix C, and licensee's responses to RAI No. 13.4 (Ref 19; Ref. 26; Ref. 27), RAI No. 13.5 (Ref. 17), RAI No. C.3 (Ref. 27), RAI No. 6.b (Ref. 33), and RAI No. 8 (Ref. 33), the licensee describe the LOFA analysis for MURR. The licensee considers that a LOFA could occur by anyone or a combination of the following: (1) loss of facility electrical power (or coolant circulation pump power), (2) inadvertent closure of coolant loop isolation valve(s), (3) inadvertent loss of pressurizer pressure, (4) locked rotor in a coolant circulation pump, and (5) failure of a coolant circulation pump coupling. The licensee provided the results of a LOFA from the SAR Section 13.4.2.1, event (c), "Inadvertent Loss of Pressurizer Pressure," which it considers to be the worst-case scenario. The loss of the facility electrical power or an inadvertent closure of the coolant isolation valves will have similar effect but with a lower consequential effects. A locked rotor or failure of a coolant circulation pump will result in only one pump failure, which then leads to a reactor scram on low flow rate.

The licensee uses the MURR RELAP5 model with the same operational assumptions used in the LOCA analysis. The sequences of events from a LOFA caused by a loss of pressure in the PCS pressurizer is similar to those in a LOCA. Upon loss of pressure, the following actions would occur:

- A reactor scram would occur.
- The primary coolant circulation pumps would stop.
- The primary coolant isolation valves would close.
- The anti-siphon system isolation valves would open.
- The in-pool HX isolation valves would open.
- The pressurizer surge line isolation valve would close.

In this scenario, the flow rate through the core transitions from forced to natural circulation with the primary coolant isolation valves closed; therefore, the timing of the isolation valve closure is an important input to the RELAP5 model. The licensee indicates that, upon the initiation of a trip, the hot-leg isolation valve would close in 8.4 seconds, and the cold-leg isolation valve would close in 9.1 seconds. The licensee assumes a closure time of 9.5 seconds for both valves.

Starting from the same steady-state condition as that discussed in the LOCA analysis above, the licensee calculated peak centerline fuel plate temperatures following a LOFA transient. The calculations indicate that the highest fuel plate centerline temperature of 280.3 °F (137.9 °C) occurs in fuel plate No. 1 at 0.3 seconds into the transient. Fuel plate centerline temperatures then decrease as reactor power decreases from the insertion of the control blades. After the first second of the transient, the highest centerline temperature of 277.9 °F (136.6 °C) occurs in fuel plate No. 22 during a flow reversal event, 17 seconds after the loss of pressure transient starts. Because peak fuel plate temperatures remain more than 500 °F (260 °C) below the "no fuel plate blister verification temperature" of 900 °F (482 °C), a LOFA scenario is not expected to result in any fuel damage.

The calculated coolant channel temperatures indicate a peak coolant temperature of 237 °F (114.2 °C). This temperature occurs just below the core horizontal centerline (between the centerline and 5 in (12.7 cm) below centerline). However, this temperature is about 40 °F (22 °C) subcooled, considering the fluid saturation temperature of 277 °F (136 °C), at the top of the core. In addition, subcooled nucleate boiling occurs during the first second in the peak heat flux region of fuel plate No. 1 and fuel plate No. 22.

The NRC staff reviewed the information in the SAR and finds that some parameters (e.g., reactor power and core flow) used in the RELAP5 model differed from those listed in the TSs and requested additional information on the benchmarking of the MURR RELAP5 model and the effect of the closure timing of the isolation valves on the calculated fuel temperature. The NRC staff reviewed the licensee's response in RAI No. 6.b (Ref. 33) on the use of operational parameters and finds them acceptable as discussed in SER Section 13.3.

In its response to RAI No. 13.4.b (Ref. 27), the licensee states that the RELAP5 code and model is benchmarked against known LOFA data from previous LOFA events that have occurred at MURR as recently as April 2008. The benchmarked results focused on the in-pool HX temperature calculation response to the modeled LOFA because, in 2007, a high-speed digital recorder was connected to the two (hot-leg and cold-leg) in-pool HX resistance temperature detector to record potential transients. On April 12, 2008, MURR experienced a LOFA caused by the rupturing of a diaphragm upstream where the pressurizer connects to the PCS. This accident is similar to that analyzed in MURR SAR Section 13.2.4.

Table 13-7 below provides a comparison of the actual (measured—the initial conditions of the April 12, 2008 event) versus the modeled operational parameters by RELAP5.

Table 13-7 Comparison of the Modeled and Actual Operating Conditions

Parameter	Modeled Conservative Assumption	April 12, 2008—Actual Operating Values
Reactor power level (MW)	11	9.95
Reactor inlet water temperature (°F (°C))	155 (68)	120.1(48.9)
Core flow rate (gpm (lpm))	3,800 (14,385)	3,793 (14,359)
Pool coolant temperature (°F (°C))	120 (49)	100.2 (37.9)
Pressurizer pressure (psig)	60	69.8
Anti-siphon system pressure (psig)	26	36

In its response to RAI No. 13.4.b (Ref. 27), the licensee provides two figures showing the benchmarked results using the conservative and the actual operational parameters in the MURR RELAP model. Table 13-7, which is presented graphically in Figure 13-1, show very similar temperature response times between the loss of flow event recorded on April 12, 2008, and the RELAP5-modeled LOFA using the actual operating values listed. The temperature peaks for the RELAP5-modeled LOFA are very close to the actual event. The event starting point in the figure corresponds to 500-second. However, there is a difference in starting temperatures before the event begins. The RELAP5-modeled transient starts with the same in-pool HX coolant inlet and outlet temperatures as the reactor pool water temperature of 100.2 °F (37.9 °C). The assumption in RELAP is that the in-pool HX isolation valves are closed. This condition is true, but it does not consider a ½-in-diameter (1.27-cm-diameter) bypass line around each isolation valve. These bypass lines allow very limited amounts of the primary coolant with a temperature of about 120 °F (49 °F) (inlet temperature) to flow through the in-pool

HX during normal operation. The exit temperature is at 121 °F (49.4 °C) because of the effect of the core exit flow, which directly connects with the HX outlet-piping header.

In addition, the higher temperature difference between the in-pool HX coolant inlet and outlet temperatures in the RELAP5 model indicate that transfer of heat from the PCS coolant through the inner and outer pressure vessels and into the reactor pool is greater than that provided by the RELAP5 modeling because of the limitations of the one-dimensional flow code. This also supports how conservative the RELAP5-modeled LOCA is. In the LOCA analysis, no coolant flows through the in-pool HX. All of the heat transfer from the PCS around the reactor core region is through the inner and outer pressure vessels.

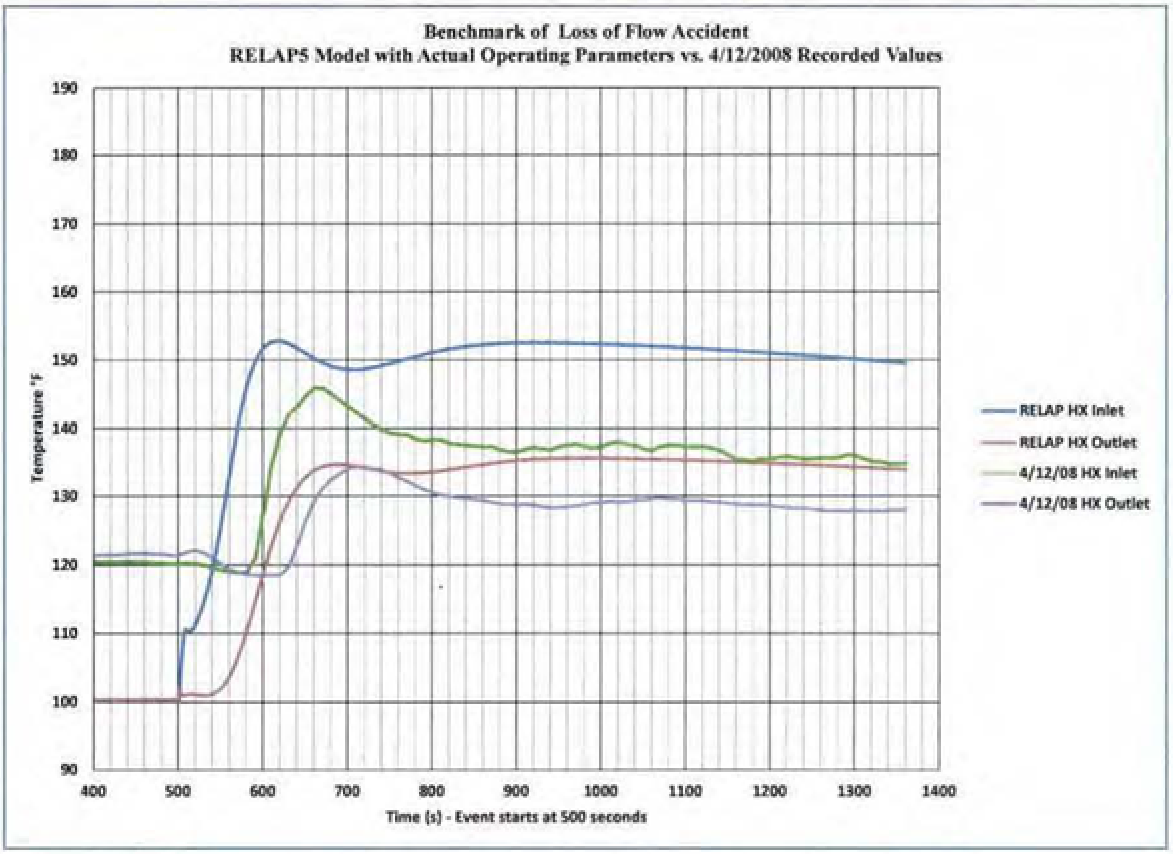

Figure 13-1 Benchmark of loss of flow accident

In its response to RAI No. 13.7 (Ref. 27), the licensee states that the assumption of longer rather than shorter closure times for PCS isolation valves V507A and V507B is conservative for the LOCA but not necessarily for the LOFA. The valve closure times provided in SAR Section 13.2.4, are based on the best time measurements that were obtained before 2006. In 2007, response time measurements were performed for primary coolant isolation valves V507A and V507B, anti-siphon isolation valves V543A and VS43B, and in-pool HX isolation valves V546A and V546B using a high-speed, portable digital data recorder. The licensee reran the LOFA analysis with the revised isolation valve response times to compare them to those used in the SAR, as listed in Table 13-8, below.

Table 13-8 Valve Closure Times (seconds)

Component	SAR	Revised Benchmark
Valves V507A and V507B (PCS isolation)		
Start to Close	4.5	4.2
Fully Closed	9.5	9.0
Valves V543A and V543B (anti-siphon isolation)		
Start to Open	0.085	0.40
Fully Open	0.185	0.60
Valves V546A and V546B (in-pool HX isolation)		
Start to Open	0.135	0.40
Fully Open	0.385	0.80

The NRC staff finds the licensee's revised benchmarked valve response time results illustrate that there is a minimal difference between the highest fuel plate centerline temperatures and the timing of their peaks. The highest fuel plate centerline temperature of 278 °F (136.7 °C) occurs in plate No. 22, 17 seconds after the event starts. In comparison to the values in the SAR as presented above, the highest fuel plate centerline temperature of 280.3 °F (137.9 °C) occurs in plate No. 1, 0.3 seconds into the transient, with the peak centerline temperature of 277.9 °F (136.6 °C) in plate No. 22 after 17 seconds into the transient. Because the differences in the calculated centerline temperatures are small, the NRC staff finds the valve response times used in the original calculations for LOFA acceptable.

In its response to RAI No. C.3 (Ref. 27), the licensee states that a revised RELAP5 analysis of LOFA was performed using a core inlet flow rate of 3,260 gpm (12,340 lpm) and keeping other modeled operational parameters (e.g., reactor power, pool temperature, and other such parameters) unchanged. This analysis is an attempt to model the LOFA using the LSSS core flow of 3,200 gpm (12,113 lpm). However, because the core exit temperature with the reactor core power of 11 MWt exceeded the reactor outlet water temperature scram setpoint, the core flow was set at 3,260 gpm (12,340 lpm). In this analysis, the revised benchmarked valve responses given in Table 13-8 were used.

The NRC staff reviewed the response to RAI No. C.3 and finds that the fuel plate centerline temperature, as shown in Figure 5 of the licensee's response to indicate a peak temperature of about 292 °F (1,44.4 °C) shortly (less than a second) after the accident, followed by a peak temperature of 284 °F (140 °C), 8 seconds after the LOFA started. These temperatures are well below the SL of 986 °F (530 °C). Therefore, the NRC staff finds the methodology and results of the LOFA analysis acceptable.

The NRC staff reviewed the LOFA analysis and responses to RAI and finds that the SAR valve closure time assumptions are acceptable based on its review of the benchmarking completed by the licensee and finds that the fuel centerline temperatures remain well below the SL peak fuel temperature of 986 °F (530 °C). The NRC staff also finds that the subcooled nucleate boiling that occurs during the first second in the hot channels along fuel plate No. 1 and fuel plate No. 22 is not expected to result in any fuel plate damage because the peak centerline temperature of fuel plate No. 1 at this location is at 280.3 °F (137.9 °C), which is well below the blister temperature of 900 °F (480 °C). Based on its review, the NRC staff concludes that the resulting temperatures are below the SL and consequently, no fuel damage is expected.

13.5 <u>Mishandling or Malfunction of Fuel</u>

In SAR Section 13.2.5, the licensee identifies the following three scenarios that could cause an accident related to the mishandling or malfunction of fuel:

(1) simple failure of the fuel cladding caused by a manufacturing defect or corrosion

(2) overheating of fuel element with subsequent potential cladding failure caused by loss of primary coolant or coolant flow

(3) a fuel-handling event in which a fuel element is dropped in a cask or underwater and damaged severely to breach the cladding

In its response to RAI No. 4.1 (Ref. 18), the licensee states that corrosion can result in aluminum cladding failure due to pitting or oxide film formation. Historically, there has been only one fuel element retired early due to increased I-131 levels in the PCS caused by a suspected manufacturing defect in the element. Pitting corrosion is not catastrophic in nature and can be detected by the TS-required monitoring in place at MURR. Therefore, the consequences of the fuel cladding failure due to a manufacturing defect in scenarios ((1) above) are bounded by scenarios (2) and (3) above. With regard to scenario (2), fuel overheating due to loss of primary coolant or coolant flow, the NRC staff evaluates this scenario and finds that no fuel damage would occur in SER Sections 13.3 and 13.4. Although fuel damage from this type of event (which could result from a flow blockage) is not credible, the licensee provides an analysis of a malfunction of fuel during operation (fuel failure accident (FFA)) assuming that fuel damage does occur. The NRC staff evaluates and finds this FFA analysis acceptable in SER Section 13.5.2 below.

Scenario (3) above is a fuel-handling event in which a fuel element is dropped in a cask or underwater and damaged severely to breach the cladding. As stated in SAR Section 13.2.5, a significant hazard that could damage the reactor or fuel is associated with placing or removing the shipping cask from the reactor pool. The licensee provides an analysis of a possible accident related to mishandling of fuel (FHA), and the evaluation of this analysis is discussed in SER Section 13.5.1 below. In its response to RAI No. 13.8 (Ref. 17), the licensee states that this hazard is minimized by only moving the spent fuel shipping cask into or out of the reactor pool in an area of the pool called the weir floor, which is well away from the reactor structure and above the elevation of all fuel storage areas. Also during this evolution, all potentially adrift or interfering items such as bins, trays, fixtures and tools are removed to further ensure that inadvertent damage cannot occur to the reactor structure and fuel storage areas. The licensee states that movement of the cask is performed by an experienced crane operator with multiple spotters. Additionally, as described in SAR Section 9.2.2, all cask lifting equipment, including the 15-ton capacity crane, is rigorously maintained, including preventive maintenance and magnetic particle testing of crane components as appropriate.

SAR Section 9.2 describes handling and storage of reactor fuel. In its response to RAI No. 9 (Ref. 33), the licensee states that all fuel handling is performed in accordance with Special Nuclear Material Control and Accounting Procedures and as outlined in the Operations Procedures. Irradiated fuel is handled with a specially designed remote tool. The fuel handling tool is designed to provide a positive indication of latching prior to movement of a fuel element. This feature is tested prior to any fuel handling sequence. The licensee states that it always handle one fuel element at a time so that they are maintained in a criticality safe configuration. New or irradiated fuel may be stored in any one of the 88 in-pool fuel storage locations (not including the core). These storage locations are designed to: (1) ensure a geometry such that the calculated k_{eff} is less than 0.9 under all conditions of moderation, (2) allow sufficient convection cooling, and (3) provide sufficient radiation shielding. Based on the above, the licensee states in the SAR that the fuel handling system provides a safe, effective and reliable means of transporting and handling reactor fuel from the time it enters the facility until it leaves. Nevertheless, the potential for dropping a fuel element while underwater and damaging it severely enough to breach the fuel cladding was considered. The licensee's evaluation of a conservative potential radionuclide release from an FHA, and calculation of the occupational and public exposures from the FHA, are discussed in SER Section 13.5.1 below. .

13.5.1 Mishandling of Fuel

In its response to RAI No. 9 (Ref. 33), the licensee states that in spite of the operating experience, a FHA scenario is postulated that results in the loss of 0.125 grams of U-235 due to 1-square inch (6.4 cm^2) damage area to a fuel plate leading to the loss of fuel meat with a thickness of 0.02 in (0.0508 cm). The licensee provided an updated FHA analysis in an RAI response (Ref. 36). The scenario assumes all iodine and noble gases within the 0.125 g U-235 is released into the reactor pool. The scenario assumes that the damage has occurred to the fuel plate No. 1, located at the highest peak power density with a PF of 4.116. The scenario assumes that the event occurs within 30 minutes after shutdown.

The NRC staff reviewed the 30-minute time period and finds the time period to be conservative because refueling typically occurs no sooner than 1 hour after shutdown, and this accounts for the time required to shut down the reactor, to secure the PCS (required by MURR procedures to stay in operation a minimum of 15 minutes after the control blades are fully inserted), and to remove the reactor pressure vessel head.

The licensee assumes that in the FHA scenario, fission products are released into the reactor pool and will be detected by the pool surface radiation monitor, leading to the actuation of the containment building isolation system. Actuation of the isolation system will prompt operations personnel to ensure that a total evacuation of the containment building is accomplished promptly. This usually occurs within 2.5 minutes. However, a conservative 5-minute evacuation time is used as the basis for the stay-time in the dose calculations for personnel that are in containment during the FHA. The scenario then closely follows that described for the MHA (except that the 17-second delay that was assumed to decay the radioiodines and noble gases in the MHA is not used for the FHA). The radioiodine released into the reactor pool is conservatively assumed to be instantly and uniformly mixed into the of bulk pool water of 20,000 gal (75,708 l). The noble gaseous fission products rise through the pool immediately with no retention, and iodine radionuclides enter the containment ambient air through pool evaporation; the evaporation was assumed to be 20 gal (75.7 l) over 5 minutes. With this evaporation, the containment air is considered to be at 100 percent relative humidity.

In addition, the scenario assumes that gaseous fission products form a uniform concentration in the containment air. For calculation of occupational doses to MURR facility staff inside containment, it was conservatively assumed that no radioactive material leaks from containment during the 5-minute stay time. Because the containment is isolated by the action of the pool surface radiation monitor, any release into the environment would be through containment leakage that enters the laboratory building that surrounds the containment structure, mixes in the laboratory building ventilation system, and exhausts through the stack at a rate of approximately 30,500 scfm (863.6 m^3 per minute). The licensee states that the laboratory building ventilation system would remain operating following the accident.

The NRC staff reviewed the assumptions for the release to the environment, and finds that keeping the laboratory ventilation system on following the accident would help maintain doses to MURR personnel and the public ALARA. Keeping the laboratory ventilation system on would help reduce radionuclide concentrations in the laboratory areas of the facility, would help ensure that any effluents are diluted before release, and would help ensure that effluents are released from the stack, providing additional dilution after release. The NRC staff also finds that the assumption that the effluents are released through the main exhaust stack is reasonable, given that the laboratory building exhaust system has two redundant 100 percent capacity fans (one operates while the other remains in standby mode), and both fans are connected to the emergency power electrical bus, which is provided electrical power by the emergency diesel generator in the case of a loss of normal electrical power (see SAR Sections 8.2.4 and 9.1.2.2). Additionally, TS 3.7, Specification 6, requires a negative pressure in the reactor containment building of 0.25 inches, which indicates that the exhaust fans must be operating during operation of the reactor or when irradiated fuel with less than 60 days of decay is being handled (see TS 3.4, Specification b).

The NRC staff reviewed the FHA scenario assumptions and finds that the assumptions provide conservative conditions for dose estimates to both the occupational workers and the members of the public since all gaseous fission products in the damaged part of the fuel are released. Based on its review, the NRC concludes that the assumptions are reasonable and are acceptable.

Nuclide Inventory

The licensee determined the nuclide inventory in the MURR core for the FHA using the computer program MONTEBURNS. In Attachment 1 to its response to responses to RAIs (Ref. 36), the licensee provides a MONTEBURNS output that lists select fission products (noble gases and iodine nuclides) inventories for the eight fuel elements within the MURR.

Table 13-9 below provides the inventory of the noble gas and radioiodine fission products in the total MURR core 30 minutes after shutdown, assuming 1,200 MWd of burnup. Table 13-9 also provides the fission product inventory for the damaged fuel plate considered in the analysis. The licensee calculates the fission product inventory for the damaged fuel portion by considering the fraction of the core U-235 inventory in the damaged fuel part (0.125 g out of 5,474 g total U-235, or 0.00228 percent) and adjusting for the power PF of 4.116. Because the MONTEBURNS analysis uses the ORIGEN 2 code for the fuel depletion, the NRC staff notes the same variations in the generated inventories in comparison with the saturated inventories using the reactor power and the fission yields as presented in the MHA analysis. Based on this information, the NRC staff finds that the MONTEBURNS-ORIGEN 2 code provides very precise fission yields, and that the licensee's use of this code is acceptable. The NRC staff also finds that the licensee's calculation of the inventory of the damaged fuel plate from the total core inventory is acceptable because it is based on assumptions and parameters that are appropriate for the MURR core.

The NRC staff reviewed the licensee's methodology and assumptions for determining the gaseous fission product inventories in the damaged fuel plate, as discussed above. The NRC staff finds that the licensee methodology and assumptions are reasonable and consistent with established nuclear industry practice, and, therefore, finds them to be acceptable.

Table 13-9 FHA - Licensee Radioiodine and Noble Gas Inventories

Nuclides	Total Core Inventory after a 30-minute decay (Ci)	Inventory for damaged portion of failed plate after a 30-minute decay (Ci)
I-131	2.20E+05	2.07E+01
I-132	3.07E+05	2.89E+01
I-133	5.39E+05	5.07E+01
I-134	5.49E+05	5.16E+01
I-135	4.80E+05	4.51E+01
Kr-85	4.63E+02	4.35E-02
Kr-85m	1.23E+05	1.16E+01
Kr-87	1.58E+05	1.49E+01
Kr-88	2.58E+05	2.42E+01
Kr-89	5.28E+02	4.96E-02
Kr-90	6.31E-12	5.93E-16
Xe-133	3.85E+05	3.62E+01
Xe-135	9.11E+04	8.56E+00
Xe-135m	3.62E+04	3.40E+00
Xe-137	2.24E+03	2.11E-01
Xe-138	1.16E+05	1.09E+01
Xe-139	7.89E-09	7.42E-13

Release Fractions

The licensee assumes that the noble gases and radioiodines in the damaged portion of the fuel plate are instantaneously released into the reactor pool. The iodine radionuclides uniformly mix in the pool water and enter the containment air through pool water evaporation. For calculation of occupational doses, this evaporation was assumed to occur gradually over a period of five minutes; for calculation of public doses, the evaporation was assumed to occur instantaneously.

Based on similar considerations as those used for the MHA and discussed in SER Section 13.1, the licensee assumed the calculated equivalent iodine release fraction due to evaporation to be 0.1 percent (20 gal (75.7 l) out of the initial 20,000 gal (75,708 l) of pool water evaporates). For the noble gases, the licensee assumes no retention in the pool water, and assumes that the entire inventory of noble gases released to the pool is also instantaneously released to containment.

The analysis assumes that the release fractions to the environment to be 50 percent of the values for halogens due to retention and plate-out and 100 percent of the values (no retention) for noble gases in the containment air.

The NRC staff reviewed the licensee's assumptions on the releases of the iodines and noble gases to the containment and the environment. The NRC staff finds the 50 percent release fraction to the environment for halogens due to retention and plate-out, and the 100 percent release fraction to the environment for noble gases, to be conservative and consistent with established nuclear industry practice, and therefore acceptable. The NRC staff also finds the 0.1 percent release fraction for radioiodines in the pool water, which evaporates to containment, to be a reasonable and conservative assumption, and therefore acceptable, for the short period of time, such as the 5- to 10-minute stay time assumed for fuel failures. However, the NRC staff notes that over a longer period of time, such as the 16.5-hour release/exposure period assumed for public dose analyses, additional pool water could evaporate if the relative humidity is allowed to decrease below 100 percent. Therefore, in its confirmatory analysis of FHA doses to members of the public, which is described below, the NRC staff conservatively assumed that 100 percent of the radioiodines that are released to the pool are also instantaneously released to the containment building at the beginning of the release/exposure period.

Atmospheric Dispersion

Consistent with the approach in the MHA analysis, the licensee uses a constant dilution factor of 292 to determine the concentration of the released radionuclides at the nearest residence (760 m (2,493 ft) north of MURR). As discussed in SER Section 13.1, the NRC staff reviewed LA No. 37 (Ref. 90), including information provided by the licensee (Ref. 99) in support of its request for LA No. 37, which the NRC staff previously reviewed and found to be acceptable. Based on this information, the NRC staff noted the potential for accident doses to members of the public at another location, 400 m (1,312 ft) north of MURR, which are higher than those at the nearest residence, 760 m (2,493 ft) north of MURR. Using the information associated with LA No. 37, the NRC staff independently calculated a worst-case dilution factor of 112 for analysis of accident doses to a member of the public located 400 m (1,312 ft) north of MURR. This dilution factor is lower (more conservative) than the dilution factor of 292 used by the licensee. Therefore, the NRC staff's confirmatory analysis of FHA doses to members of the public, which is described below, includes an evaluation of doses that is based on the more conservative worst-case dilution factor of 112.

Dose Calculations

The licensee calculates the potential TEDE for an occupational worker in the containment and members of the public at the nearest unrestricted location outside the reactor building from exposure to releases from the FHA. The boundary conditions for these calculations are similar to those listed under the MHA, without the time delay for the noble gases to pass through the reactor pool. The licensee takes credit for the iodine and noble gas nuclide's decay while in the containment air in both the occupational and public dose calculations.

Similar to the MHA analysis, for the occupational and public dose calculations, the licensee followed the DAC and AEC approach provided in the 10 CFR Part 20, Appendix B. As explained above in SER Section 13.3 MHA, the licensee used the calculated nuclide-specific DAC and AEC values for those nuclides that are not listed in Appendix B because of their short half-lives (less than 2 hours).

For both occupational and public dose calculations, the licensee takes credit for the decay of the radioiodines in the pool prior to their release into the containment structure, as well as the decay of both radioiodines and noble gases after their release into containment.

For the occupational dose calculations, the licensee uses five 1-minute time steps first to determine the release in the evaporation volumes from the reactor pool to calculate the related containment concentration values for each radionuclide in conjunction with its decay constant, and then to calculate the average concentrations over the 5-minute evacuation time. Using the assumptions and boundary conditions described above, the licensee calculates the potential occupational dose to an individual inside the containment.

The NRC staff reviewed the licensee's analysis approach and the related calculations. Similar to the MHA, the NRC staff finds the licensee’s method for taking credit for radioactive decay to be inconsistent with the need to consider daughter radionuclides that may build up when parent radionuclides decay.

The NRC staff performed an independent confirmatory analysis of occupational doses conservatively assuming no decay of the radionuclides within the pool or the containment. The NRC staff’s confirmatory analysis additionally assumed that the 20 gal (75.7 l) of pool water containing radioiodines evaporates instantaneously, in contrast with the licensee’s analysis which assumed that the pool water evaporates over 5 minutes. The NRC staff uses the licensee’s DAC analysis approach as discussed in SER Section 13.1, and the licensee's nuclide inventory in the damaged part of the fuel plate. Table 13-10 below provides the comparison of the licensee-calculated occupational doses and the doses calculated for the NRC staff’s confirmatory analysis within the 5-minute evacuation time. This table shows the results of the NRC staff's confirmatory calculation for the expected occupational TEDE to be well below the regulatory dose limits of 10 CFR 20.1201, even without taking credit for the decay.

The NRC staff finds that the licensee's dose calculations do not account for contributions from semi-volatiles such as Cesium (Cs) and Strontium (Sr). As indicated under the fuel failure analysis, and also based on the analysis performed under the MHA and the considerations of the release fractions for the cesium and strontium in the reactor pool after the fuel failure, the NRC staff considers the contributions from Cs-137 and Sr-90 to be very small.

Table 13-10 FHA - 5-minute Occupational Dose Estimates in the Restricted Area

Dose Parameters	MURR-Calculated Dose (mrem)	NRC Confirmatory Analysis (mrem)	10 CFR 20.1201 Occupational Dose Limit (mrem)
Committed effective dose (CDE) to the thyroid	264.00	528.24	50,000
Committed effective dose equivalent (CEDE)	7.92	15.85	5,000
Deep dose equivalent (DDE)	636.49	654.49	5,000
Total effective dose equivalent (TEDE)	644.41	670.34	5,000

For the public dose calculations, the licensee uses a similar approach as that explained under the MHA analysis. Therefore, the discussions here focus on the NRC staff's review, given the analysis details above. The NRC staff reviewed the licensee's analysis approach and the related calculations. Similar to the MHA, the review of the public dose calculations found the licensee used a containment volume 229,800 scf (6,507 m^3). The NRC staff considered the use of 229,800 scf for the containment volume to be inconsistent with the minimum volume of 225,000 scf (6,371 m^3) allowed by TS 5.5, Specification a (which the NRC staff evaluated and found acceptable in SER Section 6.2). Also similar to the MHA, the NRC staff finds that the licensee's analysis appeared to use a non-conservative interpretation of the AECs for noble gases. Also similarly to the MHA and similar to the occupational dose calculations for the FHA above, the NRC staff considered the licensee's method for taking credit for radioactive decay to be inconsistent with the need to consider daughter radionuclides that may build up when parent radionuclides decay. Additionally, and also similar to the MHA, the NRC staff noted that over the 16.5-hour release exposure time used for the public dose calculations, greater than 20 gal (75.7 l) of pool water could potentially evaporate, leading to a release factor of greater than 0.1 percent for release of radioiodines from the pool to containment. Last, also similar to the MHA, the NRC staff determined that a dilution factor of 292, corresponding to the nearest residence, may not be appropriate for determining the worst-case dose to any offsite member of the public. As discussed above, a dilution factor of 112, corresponding to a location 400 m (1,312 ft) north of MURR, is more conservative for calculation of the worst-case public dose.

The NRC staff performed an independent confirmatory analysis of public doses from the FHA. This confirmatory analysis used several conservative assumptions that were not used in the licensee's analysis. First, the containment volume was assumed to be equal to the TS minimum containment volume. Second, the confirmatory analysis assumes no decay of the radionuclides within the pool or the containment. Third, this analysis also assumes that all of the radioiodines and noble gases released to the pool enter the containment air (i.e., there is no holdup in the pool), and that all containment air (containing all noble gases, and the 50 percent of radioiodines that do not plate out in the containment) enters the environment through the laboratory ventilation system over the 16.5-hour release period. Last, the confirmatory analysis additionally used a dilution factor of 112, instead of the licensee's dilution factor of 292. For the confirmatory analysis, the NRC staff used the conservative AEC analysis approach as discussed in SER Section 13.1, and the licensee's FHA inventory. Table 13-11 below provides a comparison of the licensee- and NRC-calculated public doses for the 16.5-hour release and exposure period. The results of the NRC staff's confirmatory calculation show that the expected public TEDE is below the regulatory public dose limit of 100 mrem in 10 CFR 20.1301, even without taking credit for radioactive decay, using the TS minimum containment volume, using a conservative dilution factor of 112, assuming that all fission products in the reactor pool will enter the containment air, and assuming that all fission products in the reactor air (except the 50 percent of radioiodines that plate out in containment) enter the environment. These assumptions are very conservative because the radionuclides with short half-lives (in the range of seconds or minutes) will be decayed out well before the end of the 16.5-hour release and exposure period, and most of the radioiodine will remain in the reactor pool, where it will not impact workers or the environment.

The NRC staff notes that the FHA public TEDEs in Table 13-11 (both MURR-calculated and NRC confirmatory values) are greater than the public TEDEs calculated for the MHA and shown in Table 13-4 in SER Section 13.1. However, the MHA occupational TEDEs (shown in Table 13-3 in SER Section 13.1) are significantly above the FHA occupational TEDEs listed in Table 13-10 above, and therefore the MHA (fuel experiment failure) is still considered the worst-case accident scenario for MURR. As stated previously, all occupational and public doses for both the MHA and FHA are below applicable limits in 10 CFR 20.1201 and 10 CFR 20.1301.

Table 13-11 FHA - Public Dose Estimates in the Unrestricted Area

Dose Parameters	MURR-Calculated Dose (mrem)	NRC Confirmatory Analysis (mrem)	10 CFR 20.1301 Public Dose Limit (mrem)
Committed effective dose equivalent (CEDE)	6.23E-04	80.48	100
Deep dose equivalent (DDE)	1.14E-02	8.50	100
Total effective dose equivalent (TEDE)	1.20E-02	88.98	100

Similar to the analysis for the MHA, the licensee also performed a calculation of the radiation shine through the containment structure using the computer program MicroShield 8.02. This calculation represents a condition where the containment is isolated with no leakage. The source inventory for this calculation is the all of the gaseous fission products released to the reactor pool, which (unlike in the licensee's other FHA analyses discussed above) are all assumed to enter the containment air, with no decay. The licensee provides exposure rate values for the radiation fields at 1 ft (30.5 cm) from a 12 in (30.5 cm) thick ordinary concrete containment wall and at the EPZ boundary of 150 m (492.1 ft).

The NRC staff reviewed the licensee's FHA shine calculation, and finds that the licensee's inventory used for the calculation is inconsistent with the inventory released to the pool during the FHA and shown in Table 13-9 of this SER section. The inventories of most of the individual isotopes considered in the FHA shine calculations are below the values in Table 13-9; therefore, the licensee's calculated dose rates in Table 13-12 below are underestimated. However, the NRC staff notes that given that most of the inventory differences are small (approximately 20 percent or less), and that some of the inventories used for the FHA shine calculation are also larger than the Table 13-9 values, the overall underestimation of the dose rate is small. Additionally, the licensee's calculation used other conservative assumptions, particularly the assumption that all of the radioiodines released to the pool would also be released to the containment air, where they would contribute to the shine dose; as discussed previously in this section of the SER, this would not occur. The NRC staff's confirmatory calculation, which is discussed below, is based on radionuclide inventories that are consistent with the values in Table 13-9.

The NRC staff performed dose calculations to members of the public at the boundary of the EPZ (150 m) and at the nearest resident location 760 m (2,493 ft), assuming a confinement model and considering the direct radiation from all radioactivity that was released into the containment (no leakage). The calculations assume that the entire inventory of radionuclides that is released to the reactor pool (see Table 13-9) is also released to the air of the containment building. The calculations conservatively assume that the gaseous fission products released into the containment will not decay, but take credit for the shielding of the released contents by the 1 ft (30.5 cm) concrete structural material. These assumptions are consistent with those used by the licensee. Furthermore, the calculations assume the building can be represented by a point source with the radiation emitted uniformly in all directions, and each disintegration is accompanied by one 1-MeV photon (gamma rays). The NRC staff's calculation uses a simple equation and is only an approximate method with no credit for the decay energy distribution and air attenuation; therefore, the dose result is conservative. Table 13-12 below summarizes the calculated direct radiation dose rates at the EPZ boundary and the nearest resident (the NRC staff did not perform a calculation for the location 1 foot from the building exterior wall, because this is an area that is not publicly accessible, and any occupational dose would be monitored and controlled).

The MURR Emergency Plan action levels are based on a projected exposure of 24 hours or less. Assuming no radioisotope decay, the calculated dose rate will remain constant for the duration of the stay time. The 24-hour direct radiation dose (no release) at the EPZ for the FHA ranges between 9.84 mrem (MURR dose rate) and 17.8 mrem (NRC analysis), with no decay. This assumption is conservative because in few hours about 42 percent of the source inventory will be decayed out. The NRC staff performed additional direct radiation dose calculations for an individual at the EPZ location for a 24-hour, 1-month, and 1-year stay time and took credit for decay. The total DDEs for the 24-hour, 1-month, and 1-year stay times were 7.16, 34.20, and 36.3 mrem, respectively. These results (which are very conservative because although they account for decay, they do not account for leakage from containment, which would be significant over the extended periods considered) indicate that the dose due to shine from radioactive material in containment to each individual member of public at the EPZ boundary will be less than the dose limit of 10 CFR 20.1301 (100 mrem).

The NRC staff finds that the sum of the NRC-calculated maximum TEDE from radioactive material released from the stack during the FHA (88.98 mrem received 400 m (1,312 ft) north of MURR), and the NRC-calculated 1-year direct radiation dose at the EPZ boundary (36.3 mrem received 150 m (492 ft) from MURR) could be greater than 100 mrem. However, these worst-case doses for release from the stack and direct radiation are calculated at two different locations, and it is not realistic that an individual member of the public would receive the sum of these doses. The airborne release dose at the EPZ boundary would be less than 88.98 mrem, because at that location much closer to the stack, more material would pass overhead above a person on the ground (causing that person to only receive external dose from the plume passing overhead), rather than reaching ground level where it could result in greater exposure. The 1-year direct radiation dose at 400 m (1,312 ft) would be much less than 36.3 mrem, since direct radiation dose drops off rapidly with increasing distance, as illustrated by Table 13-12. Additionally, neither the EPZ boundary nor the location 400 m (1,312 ft) north of MURR are locations that would be continually occupied over 1 year (the nearest residence is located further away, 760 m (2,493 ft) north of MURR). Conservatively applying a 25 percent occupancy factor to the maximum direct radiation dose received at the EPZ boundary, and adding this dose to the maximum airborne release dose received 400 m (1,312 ft) north of MURR, the total dose would be 98.1 mrem, which is less than 100 mrem. This is a conservative estimate of the total combined dose to a member of the public from airborne release and direct radiation, not only because it is a sum of worst-case doses for two different locations, but also because the estimates of the individual contributions from airborne release and direct radiation are also both conservative, as discussed above.

Table 13-12 FHA - Maximum Radiation Shine through the Containment Building

Parameters	MURR	NRC Analysis
Exposure rate 1 foot from the building wall (mrem/hr)	59.77	-
Exposure rate at emergency planning zone (mrem/hr)	0.41	0.74
Exposure rate at nearest resident (mrem/hr)	-	0.029

Based on its review of the licensee's dose calculations and on the results of its confirmatory calculations, the NRC staff concludes that the FHA dose results (provided above) demonstrate that the maximum TEDEs are well below the occupational dose limit in 10 CFR 20.1201 and the public dose limit in 10 CFR 20.1301.

13.5.2 Malfunction of Fuel during Operation

In SAR Section 13.2.1, malfunction of fuel during operation accident was considered as the MHA event for MURR. In its response to RAI No. 7 (Ref. 33), the licensee states that this accident is no longer the MHA. Nevertheless, the licensee provides a revised fuel failure analysis in its responses to RAI No. 7 (Ref. 33) and RAI No. 7.g (Ref. 36).

Accident Scenario

In its revised fuel failure analysis, the scenario for the fuel failure accident (FFA) during reactor operation is the partial fuel melting of the fuel plate no. 1 into four separate fuel elements, resulting in the release of volatile fission products into the PCS. The four fuel plates are in the peak power region with a power history of continuous 10 MWt operation. The scenario assumes that all gaseous fission products (iodine and the noble gases) in these four plates enter the PCS. The fuel failure within the PCS will result in a quick dispersal of the fission products throughout the system, and particulates will remain in the coolant. Fission product gases that come out of solution are collected in the reactor loop vent system and retained. The primary release pathways to the environment could include the PCS pressure relief valves and pressurizer.

The licensee hypothesizes that the partial melting of the fuel could occur from metal (aluminum cladding)-water (coolant) reaction. However, the licensee's analysis shows the amount of energy released to the PCS from such a scenario to be negligible; therefore, the existence of a pressure surge, which would lift the relief valves, is not anticipated. The gaseous radioactivity trapped in the reactor loop vent tank will cause a reactor scram and actuation of the containment building isolation system by action of the pool surface radiation monitor. Additionally, following actuation of the Anti-Siphon System when the PCS is secured, gases could also collect in the anti-siphon pressure tank. However, because of the locations of these tanks, the shielding provided by the pool water and the biological shield is sufficient to reduce the radiation exposure to facility staff, visitors, and researchers.

The licensee adds that the PCS normally leaks some coolant into the reactor pool through the pressure vessel head packing and flange gasket. The leakage rate is approximately 4×10^{-3} gpm (9.1×10^{-4} m^3 per hour), or 40 gal (151 l) per week. However, for purposes of calculation, the scenario conservatively assumes a leakage rate of 80 gal (303 l) per week. For the FFA, this leakage is the source of fission products that enter the reactor pool and the environment. Upon the activation of the containment building isolation alarm, the scenario assumes the reactor staff will secure the PCS, and verify that the containment building has been evacuated, within 10 minutes. This is longer than the 5-minute evacuation time assumed for the MHA and FHA because it is assumed that the additional 5 minutes will be needed for facility staff to secure the PCS before they can evacuate. After the 10-minute period ends, it is assumed that the PCS is secured and fission products stop leaking into the reactor pool because the driving force for leakage, which is operation of the reactor coolant pumps, no longer exists when the PCS reactor coolant pumps are secured.

The licensee states that all radioiodine and noble gases that leak from the PCS into the reactor pool are conservatively assumed to enter the pool instantaneously at the beginning of the 10-minute leakage period. The radioiodine in coolant that leaks from the PCS into the reactor pool is conservatively assumed to be instantly and uniformly mixed into the 20,000 gal (75,708 l) of bulk pool water. The noble gas fission products that leak from the PCS to the pool rise instantaneously through the pool without retention, and are released to containment. The 17-second delay that was assumed to decay the radioiodines and noble gases in the MHA is not used for the FFA. Radioiodine enters the containment ambient air through gradual pool evaporation; the evaporation was assumed to be 40 gal (151 l) over 10 minutes. The assumed addition of 40 gal (151 l) of water vapor to the containment in 10 minutes overestimates by a factor of two the amount of water that would make the containment air saturated, and far exceeds the normal evaporation rate from the reactor pool, which is approximately 80 gal (303 l) of water per day when the containment ventilation system is in operation.

The scenario also assumes that gaseous fission products form a uniform concentration in the containment air. For calculation of occupational doses to MURR staff inside containment, it was conservatively assumed that no radioactive material leaks from containment during the 10-minute stay time. Because the containment is isolated by the action of the pool surface radiation monitor, any release into the environment would be through containment leakage that enters the laboratory building that surrounds the containment structure, mixes in the laboratory building ventilation system, and exhausts through the stack at a rate of approximately 30,500 scfm (863.6 m^3 per minute). The licensee states that the laboratory building ventilation system would remain operating following the accident.

The NRC staff reviewed the assumptions for the release to the environment, and finds that keeping the laboratory ventilation system on following the accident would help maintain doses to MURR personnel and the public ALARA. Keeping the laboratory ventilation system on would help reduce radionuclide concentrations in the laboratory areas of the facility, would help ensure that any effluents are diluted before release, and would help ensure that effluents are released from the stack, providing additional dilution after release. The NRC staff also finds that the assumption that the effluents are released through the main exhaust stack appears valid because the laboratory building exhaust system has two redundant 100 percent capacity fans (one operates while the other remains in standby mode) and both fans are connected to the emergency power electrical bus that is powered by the emergency diesel generator if there is a loss of normal electrical power (see SAR Sections 8.2.4 and 9.1.2.2). Additionally, TS 3.7, Specification 6, requires a negative pressure in the reactor containment building of 0.25 inches, which indicates that the exhaust fans must be operating during operation of the reactor or when irradiated fuel with less than 60 days of decay is being handled (see TS 3.4, Specification b).

The NRC staff reviewed the licensee's FHA scenario assumptions and finds that the assumptions provide conservative conditions for dose estimates to both the occupational workers and the members of the public, since all gaseous fission products in the four fuel plates involved in the FFA are released. Therefore, based on its review, the NRC staff concludes that the assumptions reasonable, conservative, and acceptable.

Nuclide Inventory

The licensee states that the source terms used in the analysis for the FFA are determined using the computer program MONTEBURNS. Within the MONTEBURNS program, MCNP calculations are used to obtain accurate one-group fluxes and one-group cross sections that are then utilized by the ORIGEN 2 code for fuel depletion and fission product activity calculations (see SER Figure 4-3). In its response to RAI No. 9.b.ii (Ref. 36), the licensee states that MONTEBURNS is used to simulate the burnup of all eight fuel elements, for a core configuration that is based on an all-fresh core that was irradiated for 12 10-day cycles over a 300-day period. In Attachment 1 to its responses to RAIs (Ref. 36), the licensee provides a MONTEBURNS code output that lists select fission product (noble gases and radioiodines) inventories for the eight fuel elements within the MURR.

Table 13-13 below provides the inventories of the noble gases and radioiodines in the total MURR core for a 1,200 MWd burnup. Table 13-13 also provides the fission product inventory for the four failed fuel plates considered in the FFA. The licensee calculates the fission gas inventories in the four fuel plates by considering the fraction of the core U-235 inventory in the failed fuel plates (0.0141) and adjusting for the overall PF (2.423). Because the MONTEBURNS uses the ORIGEN 2 code for the fuel depletion, the NRC staff observed the same variations in the generated inventories in comparison with the saturated inventories using the reactor power and the fission yields as presented in the MHA analysis. Based on this information, the NRC staff finds that the MONTEBURNS-ORIGEN 2 code provides very precise fission yields.

The NRC staff reviewed the assumptions, approach, and the calculations for determining the gaseous fission product inventories in the failed fuel plates, and finds them acceptable.

Table 13-13 FFA - Licensee Radioiodine and Noble Gas Inventories

Nuclides	MURR Core (Ci)	Four No. 1 Fuel Plates (Ci)
I-131	2.20E+05	7.52E+03
I-132	3.08E+05	1.05E+04
I-133	5.42E+05	1.85E+04
I-134	6.11E+05	2.09E+04
I-135	5.06E+05	1.73E+04
Kr-85	4.63E+02	1.58E+01
Kr-85m	1.31E+05	4.48E+03
Kr-87	2.05E+05	7.01E+03
Kr-88	2.91E+05	9.95E+03
Kr-89	3.69E+05	1.26E+04
Kr-90	3.68E+05	1.26E+04
Xe-133	3.85E+05	1.32E+04
Xe-135	7.56E+04	2.59E+03
Xe-135m	3.62E+04	1.24E+03
Xe-137	4.81E+05	1.65E+04
Xe-138	5.01E+05	1.71E+04
Xe-139	4.07E+05	1.39E+04

Release Fractions

The licensee assumes that the noble gases and halogens in the four failed fuel plates are instantaneously released to the PCS, which has a volume of 2,000 gal (7,570 l). The released fission products uniformly mix in the PCS water and enter into the reactor pool through the pressure vessel head packing and flange gasket leakage. The licensee assumes a PCS leak rate of about 0.0079 gpm (0.030 lpm) (based on the conservative 80 gal (303 l) per week PCS leakage rate discussed above), meaning that a total PCS volume of 0.079 gal (0.3 l) enters the reactor pool over the 10-minute period before the PCS is secured. As discussed above, all fission products that leak from the PCS to the pool are conservatively assumed to enter the pool instantaneously at the beginning of the 10-minute leakage period. Based on these considerations, the calculated equivalent release fraction for the iodine and noble gases into the reactor pool is then 3.95×10^{-3} percent (0.079 gal (0.30 l) out of the initial 2,000 gal (7,570 l) of PCS water leaks into the pool). The radionuclides that enter the reactor pool instantly and uniformly mix with the pool water. For calculation of occupational doses, the radioiodine will enter the containment air through gradual pool water evaporation over 10 minutes; for calculation of public doses, the evaporation was assumed to occur instantaneously. It is assumed that all of the radioiodine activity in the 40 gal (151 l) of pool water that evaporates will form a uniform concentration in the containment building air. For radioiodines, this assumption adds another release fraction of 0.2 percent (40 gal (151.4 l) out of the initial 20,000 gal (75,708 l) of pool water evaporates). For the noble gases, the licensee assumes that they immediately pass through the pool water and instantaneously form a uniform concentration in the isolated containment structure, with no retention in the pool water.

Similarly to the MHA and FHA, the analysis assumes that the release fractions to the environment to be 50 percent of the values for iodine due to retention and plate-out and 100 percent of the values (no retention) for noble gases in the containment air.

The NRC staff reviewed the licensee's assumptions on the releases of the iodine and noble gases to the containment and the environment. The NRC staff finds the 50 percent release fraction to the environment for halogens (radioiodines) due to retention and plate-out, and the 100 percent release fraction to the environment for noble gases, to be conservative and consistent with established nuclear industry practice, and therefore acceptable. The NRC staff also finds the 3.95×10^{-3} percent release fraction for fission products leaking from the PCS to the pool to be reasonable, because it is calculated based on assumptions that are conservative and consistent with the design and operating parameters of the reactor. The NRC staff finds the 0.2 percent release fraction for radioiodines in pool water evaporating to containment to be reasonable and conservative, and therefore acceptable, over a short period of time such as the 5- to 10-minute stay time assumed for fuel failures (see discussion of release fractions from pool evaporation in SER Section 13.1). However, the NRC staff noted that the PCS has a higher pressure than the pool water, and the gas contents in the leaked fluid will most likely force some of the iodine to enter the containment without mixing with the 20,000 gal (75,708 l) of the pool water. Because of this, the licensee's assumption (used for calculation of occupational doses) that the radioiodines are released gradually over a 10-minute period, as the pool water evaporates, may not be realistic. Therefore the NRC staff's confirmatory analysis of FFA occupational doses, which is described below, assumes that the radioiodines are released instantaneously at the beginning of the 10-minute leakage period. Additionally, the NRC staff notes that over a longer period of time, such as the 16.5 hour release/exposure period assumed for public dose analyses, additional pool water could evaporate if the relative humidity is allowed to decrease below 100 percent. Therefore, in its confirmatory analysis of FFA doses to members of the public, which is described below, the NRC staff conservatively assumed that 100 percent of the radioiodines that are released to the pool are also instantaneously released to the containment building at the beginning of the release/exposure period.

Atmospheric Dispersion

Consistent with the approach in the MHA and FHA analyses, the licensee used a constant dilution factor of 292 to determine the concentration of the released radionuclides at the nearest residence (760 m (2,493 ft) north of MURR). As discussed in SER Section 13.1, the NRC staff reviewed LA No. 37 (Ref. 90), including information provided by the licensee (Ref. 99) in support of its request for LA No. 37, which the NRC staff previously reviewed and found to be acceptable. Based on this information, the NRC staff finds the potential for accident doses to members of the public at another location, 400 m (1,312 ft) north of MURR, which are higher than those at the nearest residence, 760 m (2,493 ft) north of MURR. Using the information associated with LA No. 37, the NRC staff independently calculated a worst-case dilution factor of 112 for analysis of accident doses to a member of the public located 400 m (1,312 ft) north of MURR. This dilution factor is lower (more conservative) than the dilution factor of 292 cited by the licensee. Therefore, the NRC staff's confirmatory analysis of FFA doses to members of the public, which is described below, includes an evaluation of doses that is based on the more conservative worst-case dilution factor of 112.

Dose Calculations

The licensee calculates the potential TEDE for an occupational worker in the containment and members of the public at the nearest residence outside the reactor building from exposure to releases from the FFA. The boundary conditions for these calculations are similar to those listed under the MHA, except for the doubling of the evaporation of the pool water to 40 gal (151 l), and not using the 17-second delay that was assumed to decay the radioiodines and noble gases in the MHA.

The licensee considered the following additional assumptions:

- Restricted area inside the containment structure:
  - the occupational workers will be exposed to the airborne gaseous fission products with credit for the time-dependent leak and radionuclide decay,
  - the containment ventilation system is off and isolated,
  - an evacuation time of 10 minutes for the occupational workers is needed to secure the reactor to a safe condition and verify the containment building has been evacuated.
- Unrestricted area outside the MURR facility for members of the public:
  - the laboratory ventilation system is operating to carry the gaseous fission products that leak from the containment into the building ventilation stack with a ventilation flow rate of 30,500 scfm (863.66 m^3 per minute),
  - credit is taken for the time dependent radionuclide decay inside the containment,

– containment leakage duration of 16.5 hours is needed for the leakage rate to become zero.

Similar to the MHA analysis, for the occupational and public dose calculations, the licensee follows the DAC and AEC approach provided in 10 CFR Part 20, Appendix B. As explained above under the MHA, the licensee used the calculated nuclide-specific DAC and AEC values for those nuclides that are not listed in Appendix B because of their short half-lives (less than 2 hours).

As stated above, for both occupational and public dose calculations, the licensee took credit for the decay of the radionuclides in the pool prior to their release into the containment structure, as well as the decay of both radioiodines and noble gases after their release into containment.

For the occupational dose calculations, the licensee uses two 1-minute and four 2-minute time steps first to determine the release from the PCS to the reactor pool and at the same time determine the evaporation volumes from the reactor pool to calculate the related containment concentration values for each radionuclide in conjunction with its decay constant, and then calculate the average concentrations over the 10-minute evacuation time. Using the assumptions and boundary conditions described above, the licensee calculates the potential occupational dose to an individual inside the containment.

The NRC staff reviewed the licensee's occupational dose analysis approach and the related calculations. Similar to the MHA and the FHA, the NRC staff considered the licensee's method for taking credit for radioactive decay to be inconsistent with the need to consider daughter radionuclides that may build up when parent radionuclides decay. Additionally, as discussed above, the licensee's assumption that the radioiodines are released gradually over a 10-minute period, as the pool water evaporates, may not be realistic.

The NRC staff performed an independent confirmatory analysis of the occupational doses, assuming no decay of the radionuclides within the pool or the containment. The NRC staff's confirmatory analysis also assumed that the 40 gal (75.7 l) of pool water containing radioiodines evaporates instantaneously, in contrast with the licensee's analysis which assumed that the pool water evaporates over 10 minutes. The NRC staff used the licensee's DAC analysis approach as discussed in SER Section 13.1, and the licensee's FFA inventory. Table 13-14 below provides the comparison of the licensee-calculated occupational doses and the doses calculated for the NRC staff's confirmatory analysis within the 10-minute evacuation time. This table shows the results of the NRC staff's confirmatory calculation for the occupational TEDE to be well below the regulatory dose limits of 10 CFR 20.1201, even without taking credit for radioactive decay.

In addition, the NRC staff finds that the licensee's dose calculations do not account for contributions from semi-volatiles such as Cesium and Strontium. In its response to RAI No. 13.1.b (Ref. 18), the licensee states that dose contributions from Cesium and Strontium are less than 1 percent of the total dose from Iodine, Kr, and Xe. Based on the analysis performed under the MHA and the considerations of the release fractions of the Cesium and Strontium in the PCS and the reactor pool after the fuel failure, the NRC staff finds that the contributions from Cs-137 and Sr-90 are very small and may be considered to be negligible.

Table 13-14 FFA - 10-minute Occupational Dose Estimates in the Restricted Area

Dose Parameters	MURR-Calculated (mrem)	NRC Confirmatory Analysis (mrem)	10 CFR 20.1201 Occupational Dose Limit (mrem)
Committed effective dose (CDE) to the thyroid	10.07	30.61	50,000
Committed effective dose equivalent (CEDE)	0.302	0.92	5,000
Deep dose equivalent (DDE)	16.38	70.34	5,000
Total effective dose equivalent (TEDE)	16.68	71.26	5,000

For the public dose calculations, the licensee a uses similar approach as that explained under the MHA analysis. Therefore, the discussions here focus on the NRC staff's review, given the analysis details above. The NRC staff reviewed the licensee's analysis approach and the related calculations, and similar to the MHA and FHA public doses analyses, the NRC staff noted the following. Based on its review of the FFA public dose calculations, the NRC staff finds that the licensee uses a containment volume 229,800 scf (6,507 m^3). The NRC staff considered the use of 229,800 scf for the containment volume to be inconsistent with the minimum volume of 225,000 scf (6,371 m^3) allowed by TS 5.5, Specification a (which is discussed and found acceptable in SER Section 6.2). The NRC staff also finds that the licensee’s analysis appeared to use a non-conservative interpretation of the AECs for noble gases. Similar to the occupational dose calculations for the FFA above, the NRC staff considered the licensee’s method for taking credit for radioactive decay to be inconsistent with the need to consider daughter radionuclides that may build up when parent radionuclides decay. Additionally, the NRC staff finds that over the 16.5-hour release/exposure time used for the public dose calculations, greater than 40 gal (75.7 l) of pool water could potentially evaporate, leading to a release factor of greater than 0.2 percent for release of radioiodines from the pool to containment. Also, the NRC staff determined that a dilution factor of 292, corresponding to the nearest residence, may not be appropriate for determining the worst-case dose to any offsite member of the public. As discussed above, a dilution factor of 112, corresponding to a location 400 m (1,312 ft) north of MURR, is more conservative for calculation of the worst-case public dose.

The NRC staff performed an independent confirmatory analysis of public doses from the FFA. The confirmatory analysis used several conservative assumptions that were not used in the licensee’s analysis. First, the containment volume was assumed to be equal to the TS minimum containment volume. Second, the confirmatory analysis assumes no decay of the radionuclides within the pool or the containment. Third, this analysis also assumes that all of the radioiodines and noble gases released to the pool enter the containment air (i.e., there is no holdup in the pool), and that all containment air (containing all noble gases, and the 50 percent of radioiodines that do not plate out in the containment) enters the environment through the laboratory ventilation system over the 16.5-hour release period. Last, the confirmatory analysis additionally used a dilution factor of 112, instead of the licensee’s dilution factor of 292. For the confirmatory analysis, the NRC staff used a conservative AEC analysis approach as discussed in SER Section 13.1, and the licensee's FFA inventory. Table 13-15 below provides a comparison of the licensee- and NRC-calculated public doses for the 16.5-hour release and exposure period. The results of the NRC staff's confirmatory calculation show that the expected public TEDE is well below the regulatory public dose limit of 100 mrem in 10 CFR 20.1301, even without taking credit for radioactive decay, using the TS minimum containment volume, using a conservative dilution factor of 112, assuming that all fission products in the reactor pool will enter the containment air, and assuming that all fission products in the reactor air (except radioiodines that plate out in containment) enter the environment. These assumptions are very conservative because the radionuclides with short half-lives (in the range of seconds or minutes) will be decayed out well before the end of the 16.5-hour release and exposure period, and most of the radioiodine will remain in the reactor pool, where it will not impact facility staff or the environment.

Table 13-15 FFA - Public Dose Estimates in the Unrestricted Area

Dose Parameters	MURR-Calculated Dose (mrem)	NRC Confirmatory Analysis (mrem)	10 CFR 20.1301 Public Dose Limit (mrem)
Committed effective dose equivalent (CEDE)	1.80E-05	1.17	100
Deep dose equivalent (DDE)	5.52E-02	0.458	100
Total effective dose equivalent (TEDE)	5.53E-02	1.63	100

Similar to the analysis for the MHA and FHA, the licensee also performed a calculation of the radiation shine through the containment structure using the computer program MicroShield 8.02. This calculation represents a condition where the containment is isolated with no leakage. The source inventory for this calculation is the all of the gaseous fission products released to the reactor pool, which (unlike in the licensee's other FFA analyses discussed above) are all assumed to enter the containment air, with no decay. The licensee provides exposure rate values for the radiation fields at 1 ft (30.5 cm) from a 12 in (30.5 cm) thick ordinary concrete containment wall and at the EPZ boundary of 150 m (492.1 ft).

The NRC staff performed dose calculations to members of the public at the EPZ boundary of 150 m (492 ft) and at the nearest resident location of 760 m (2,492 ft), assuming a confinement model and considering the direct radiation from all radioactivity that was released into the containment (no leakage). The calculations conservatively assume that the gaseous fission products released into the containment will not decay, but take credit for the shielding of the released contents by the 1-foot concrete around the building. These assumptions are consistent with those used by the licensee. Furthermore, the calculations assume the building can be represented by a point source with the radiation emitted uniformly in all directions and each disintegration is accompanied by one 1-MeV photon (gamma rays). The NRC staff's calculation uses a simple equation and is only an approximate method with no credit for the decay energy distribution and air attenuation; therefore, the dose result is conservative. Table 13-16 below summarizes the calculated direct radiation dose rates at the EPZ boundary and the nearest resident (the NRC staff did not perform a calculation for the location 1 foot from the building wall, because this is an area that is not publicly accessible).

The MURR Emergency Plan action levels are based on a projected exposure of 24 hours or less. Assuming no radioisotope decay, the calculated dose rate will remain constant for the duration of the stay time. Therefore, the 24-hour direct radiation dose (no release) at the EPZ boundary for the FFA ranges between 0.23 mrem (MURR dose rate) and 0.42 mrem (NRC confirmatory analysis), with no decay. This assumption is very conservative because in few hours about 42 percent of the source inventory will be decayed out. The NRC staff performed additional direct radiation dose calculations to an individual at the EPZ boundary for a 24-hour, 1-month, and 1-year stay time and took credit for decay. The total DDE for the 24-hour, 1-month, and 1-year stay times were 0.11, 4.9, and 5.3 mrem, respectively. These results (which are very conservative because although they account for decay, they do not account for leakage from containment, which would be significant over the extended periods considered) indicate that the dose to each individual member of public at the EPZ location will be less than the public dose limit in 10 CFR 20.1301 (100 mrem).

Table 13-16 FFA - Maximum Radiation Shine through the Containment Building

Parameters	MURR	NRC Analysis
Exposure rate 1 foot from the building wall (mrem/hr)	1.374	-
Exposure rate at emergency planning zone (mrem/hr)	0.0094	0.0177
Exposure rate at nearest resident (mrem/hr)	-	0.00069

Based on its review of the licensee's dose calculations, as well as the results of the NRC staff's confirmatory calculations, the NRC staff concludes that the calculations clearly demonstrate that the maximum occupational and public TEDEs for the FFA are well below the occupational dose limit in 10 CFR 20.1201 and the public dose limit in 10 CFR 20.1301.

13.6 Experiment Malfunction

SAR Section 13.2.6 states that all experiments are subject to strict procedural and regulatory requirements listed in TS 3.8. These requirements are designed to reduce the likelihood of damage to the reactor and the possibility of the radioactivity releases or radiation doses that exceed the limits in 10 CFR Part 20. The restrictions in TS 3.8 ensure that experiments consider failure mechanisms including corrosion, overheating, impact from projectiles, and chemical or mechanical explosions. SAR Section 13.2.6 specifically evaluates fueled experiments, the amount of the explosive materials that may be irradiated, and the limits on experimental reactivity worth.

The licensee states that the MURR utilization request establishes safety reviews for the proposed experiments. These reviews require the performance of specific safety analyses to assess such considerations as criticality or reactivity, or both; heat generation; off-gassing or chemical reactions, or both; and shielding. This review process is an important step in ensuring the safety of reactor experiments and has been successfully used for many years at other research reactors and for nearly 40 years at MURR. Therefore, the continuation of this approach is expected to be an effective measure for ensuring experiment safety at MURR.

In its response to RAI No. 13.9.a (Ref. 22) and in its revised response (Ref. 33), the licensee provides an analysis for the failure of a fueled experiment, which identifies the fueled experiment failure as the MHA. The NRC staff evaluate and finds the analysis acceptable in SER Section 13.1.

In its response to RAI No. 10.b (Ref. 36), the licensee states that LA No. 34 (Ref. 67) revised TS 3.8, Specification o, to state that fueled experiments containing inventories of I-131 through

I-135 greater than 1.5 Ci or Sr-90 greater than 5 millicuries shall be in irradiation containers that satisfy the requirements of TS 3.8, Specification I or shall be vented to the facility ventilation exhaust stack through high-efficiency particulate air and charcoal filters that are continuously monitored for an increase in radiation levels. The licensee adds that SAR Section 13.6.2 is now outdated. The NRC staff considers the resolution of this response (RAI No. 10.b) through LA No. 34 (Ref. 67) to resolve the concerns in RAI No. 10.c regarding the venting of the iodine and its consequential doses.

The licensee limits the amount of the explosive that can be irradiated or that is allowed to be generated in any experiment to 25 milligrams of trinitrotoluene (TNT)-equivalent explosives. This limitation is set to reduce the likelihood of the damage to the reactor or the pool should a detonation occurs. In its response to RAI No. 13.9.b (Ref. 19), the licensee demonstrated that an explosive failure of an experiment within an irradiation container would remain encapsulated within the container and have a safety margin of a factor greater than 2 as required by TS 3.8, Specification i.

NUREG-1537 states that reactivity limits are placed on experiments to ensure that (1) the rate of change of any movable experiment is such that, when the experiment is intentionally set in motion, the capacity of the reactivity control system to provide compensation is not exceeded and (2) the magnitude of the potential reactivity worth of each unsecured experiment is less than the value of reactivity, which would cause a violation of an SL. The reactivity worth of all secured (center test hole) or unsecured experiments is set at 0.00600 $\Delta k/k$. The NRC staff evaluated the reactivity limit for all MURR experiments and finds them acceptable in SER Section 13.2.

The NRC staff reviewed the experimental facilities and provisions for experiment review for MURR. Based on its review, the NRC staff concludes that performance of experiments within the restrictions of the TSs provides reasonable assurance that the potential consequences of experiment malfunctions would be within the dose limits of 10 CFR Part 20 and be bounded by the insertion of excess reactivity accidents.

13.7 <u>Loss of Normal Electric Power</u>

SAR Section 13.2.7 describes the loss of normal power, which is an anticipated event for MURR and would not be expected to cause an accident scenario. Reactor shutdown is a passive action and considered fail safe in that, if normal power is lost, the control rods automatically fall into the core because of gravity, thereby shutting down the reactor. Through an automatic transfer switch, the 275-kilowatt emergency diesel generator provides power to essential reactor monitoring systems to ensure personnel safety following the reactor scram.

In the SAR, the licensee analyzes the worst-case scenario of a complete loss of power (failure of the normal and emergency power system) with and without the reactor operating. When reactor is operating, loss of normal power results in reactor scram followed by a reduction in flow (see SER Section 13.4). The MURR 15-kilovolt-ampere uninterruptable power supply (UPS) would provide power to critical reactor monitoring instrument channels in the control room for 2 hours. After the shutdown, physical observations of the reactor are made by the operators to determine that the reactor is in a secure and safe shutdown condition. Health physics personnel would be able to monitor radiation levels with portable instruments.

The licensee states that the existing compressed air supply has a reservoir (a pressurized tank) that provides air to sealing gaskets. If all electric power is lost, containment function will eventually be lost when the reservoir of compressed air is exhausted because pneumatic sealing gaskets will no longer be operable. The containment ventilation system isolates by closing the backup isolation doors with solid rubber gaskets through gravity. The truck door is closed and sealed and will remained sealed as long as there is sufficient air pressure. The personnel air locks are closed and sealed at the time of the event but can be opened manually. This action would lead to loss of seal gasket (on loss of air) after manual re-closure. However, the reactor is shut down, and no release scenario would be credible; therefore, containment integrity would not be a primary consideration.

The licensee also states that battery-operated emergency lights strategically positioned throughout the facility would provide sufficient lighting in all critical locations, particularly along emergency escape routes. The emergency method (emergency pool system) of adding water to the pool in case of a leak is not dependent on reactor building power. No TSs require building power, UPS power, or diesel generator power when the reactor is secured. Therefore, because the reactor is automatically shut down when all power is lost, no requirements exist for providing emergency electrical power to maintain the reactor in a safe condition.

The NRC staff reviewed SAR Chapter 8 and SAR Section 13.2.7, and finds that a loss of normal electrical power would not pose an undue risk to public health and safety. The NRC staff considers that the lack of the seal gasket on the personnel airlocks after their use during a complete loss of electrical power would result in loss of containment integrity. However, the reactor is shut down, and physical observations of important indicators (e.g., shim blades and valve operator position) by the operators before their exit through the airlock would preclude any potential for a release. Based on its review, the NRC staff concludes that the loss of containment integrity for a short duration of total loss of power does not pose an undue risk to public health and safety.

13.8 External Events

SAR Section 13.2.8 describes various external events. The licensee considers meteorological disturbances, such as hurricanes, tornadoes, extreme winds, or floods (see SAR Chapter 2), as potential external accident-initiating events at MURR but dismissed all of these events based on the geographic location and robustness of the facility. In SAR Section 2.5.2, the licensee provides a review and analysis of the historical seismicity of the region surrounding MURR determined that the maximum earthquake potential for the MURR site is well below the level that would cause damage to the facility. In addition, no other industrial, military, or transportation facilities exist nearby that could cause a credible accident. Therefore, the licensee concludes no other external events could be identified that would prevent safe shutdown of the reactor or damage the reactor.

The NRC staff reviewed SAR Chapter 2 and SAR Section 13.2.8 and concludes that there is reasonable assurance that no external event would pose an unacceptable risk to public health and safety.

13.9 Mishandling or Malfunction of Equipment

SAR Section 13.2.9 discusses various equipment malfunctions. The licensee considers equipment failures leading to a leak in the pool coolant system (PoolCS), shearing of beamports, failure of the in-pool HX isolation valves to open, PCS high-pressure transient, and failure of the neutron startup source and examined the impact of each failure on the reactor fuel integrity and potential release. The licensee concludes that none of the failures would lead to a condition that could exceed the MHA releases.

The NRC staff reviewed and finds the licensee's evaluations of potential malfunctions to be reasonable but needed additional clarifications on the failure of the neutron startup source.

In SAR Section 13.2.9.5, the licensee states that the neutron source (antimony-beryllium) is presently used to perform subcritical multiplication measurements for spent fuel storage racks and shipping casks and to response check installed nuclear instrumentation detectors (see SER Section 4.2.4). A small leak or a sudden rupture of the source capsule could cause a failure of the neutron source. The failure could lead to a gradual release or a sudden large release of activity in the pool. The latter failure was considered not credible because of the robustness of the container and the location of the source inside the pool. The gradual leak would lead to activity buildup over time. The licensee states that a weekly surveillance of the PoolCS coolant is maintained to detect activity sources in the pool system. In its response to RAI No. 13.10 (Ref. 18), the licensee states that, in addition to the weekly surveillance, the pool surface is actively monitored for radiation with readouts in the control room. A breach of the antimony-beryllium neutron source which would release less than 1 percent of the source material into the pool system would increase the dose rate of the pool water by 33 percent (or an activity concentration 1.32 μCi/ml, which is well above the detection limits of the gamma-ray spectral analysis conducted on the weekly sampling of the pool. This large increase in dose rate would also elevate the doses above the pool surface and be detected readily by the control room staff. The licensee states that a pool water sample would then be taken and measured.

The NRC staff reviewed SAR Section 13.2.9 and the licensee's responses to RAIs, and finds that the scope and consequences of the analyzed accident scenarios involving mishandling or malfunction of equipment are bounded by the previously analyzed accidents. Based on the information provided above, the NRC staff concludes that the analyzed accidents do not present an undue risk to public health and safety.

13.10 <u>Conclusions</u>

The NRC staff has reviewed the accident analyses presented in the SAR and in RAI responses, and finds the licensee has considered a sufficient range of accident categories and analyzed limiting scenarios for each category to bound all credible accidents for MURR. Based on its review, the NRC staff concludes the following:

- The licensee considered the expected consequences of a sufficiently broad spectrum of postulated credible accidents and an MHA, emphasizing those that could lead to a fission product release of a fueled experiment, or a loss of integrity of fuel element clad and a release of fission products.

- The licensee analyzed the most significant credible accidents and the MHA and determined that, under conservative assumptions, the most significant credible accidents and the MHA will not result in occupational radiation exposure of the MURR staff or radiation exposure to a member of the public in excess of the applicable 10 CFR Part 20 limits.

- The licensee generally employed appropriate methods in performing the accident and consequence analysis.

- The review of the calculations, including assumptions, demonstrated that a LOCA would not result in unacceptable fuel element temperatures.

- External events that would lead to fuel failure are unlikely.

- The licensee's accident analysis confirms the acceptability of the licensed power of 10 MWt, including the response to anticipated transients and accidents.

- The confirmatory analyses performed by the NRC staff confirmed the acceptability of the assumptions and methods stated in the individual accident analyses provided in the SAR, as supplemented.

The NRC staff reviewed the radiation source term and MHA calculations for MURR. The NRC staff finds the calculations, including the assumptions, demonstrated that the source term assumed and other boundary conditions used in the analysis are acceptable. The radiological consequences to the public and occupational workers at the MURR are in conformance with the requirements in 10 CFR Part 20. The NRC staff also finds that the licensee's review of the postulated accident scenarios provided in NUREG-1537 did not identify any other accidents with fission product release consequences not bounded by the MHA. The MURR design features and administrative restrictions found in the TSs help to prevent the initiation of accidents and mitigate associated consequences. Therefore, based on its review, the NRC staff concludes that there is reasonable assurance that no credible accident would pose significant radiological release and the continued operation of the MURR would not endanger the facility staff, the public during the renewal period.

14 TECHNICAL SPECIFICATIONS

In this section of the safety evaluation report (SER), the U.S. Nuclear Regulatory Commission (NRC) staff provides its evaluation of the licensee's proposed technical specifications (TSs). The TSs for the University of Missouri-Columbia Research Reactor (MURR or the reactor) define specific features, characteristics, and conditions required for the safe operation of the MURR facility. The TSs are explicitly included in the renewal license as Appendix A. The NRC staff reviewed the format and content of the TSs for consistency with the guidance in Chapter 14, "Technical Specifications," of NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content," issued February 1996 (Ref. 51); Appendix 14.1, "Format and Content of Technical Specifications," to NUREG-1537; and American National Standards Institute/American Nuclear Society (ANSI/ANS)-15.1-2007, "The Development of Technical Specifications for Research Reactors," issued 2007 (Ref. 57).

The NRC staff specifically evaluated the content of the proposed TSs to determine whether they meet the requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36(c)(1) through (5) to include Safety Limits (SLs), Limiting Safety System Settings (LSSS), Limiting Conditions for Operation (LCO), Surveillance Requirements (SRs), Design Features, and Administrative Controls. The NRC staff also relied on NUREG-1537 to perform its review. The SER Sections where the TS was evaluated are only referenced in this Chapter if the TS was evaluated previously in the SER.

14.1 Technical Specification Definitions

The licensee proposed the following definitions to be general consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. The licensee's proposed TSs include minor modification to, and some additional facility-specific, definitions.

TS 1.0 states:

1.0 **DEFINITIONS**

1.1 **Abnormal Occurrences -** An abnormal occurrence is any of the following which occurs during reactor operation:

a. Operation with actual safety system settings for required systems less conservative than specified in Section 2.2, Limiting Safety System Settings;

b. Operation in violation of Limiting Conditions for Operations established in Section 3.0;

c. A reactor safety system component malfunction which renders or could render the reactor safety system incapable of performing its intended safety function. If the malfunction or condition is caused by maintenance, then no report is required;

d. An unanticipated or uncontrolled change in reactivity in excess of 0.006 $\Delta k/k$. Reactor trips resulting from a known cause are excluded;

e. Abnormal and significant degradation in reactor fuel or cladding, or both, primary coolant boundary, or containment boundary (excluding minor leaks) where applicable; or

f. 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 involving operation of the reactor.

1.2 **Center Test Hole -** The center test hole is that volume in the flux trap occupied by the removable experiment sample canister.

1.3 **Channel -** A channel is the combination of sensor, line, amplifier, and output devices that are connected for the purpose of measuring the value of a parameter.

1.4 **Channel Calibration -** A channel calibration is an adjustment of the channel such that its output corresponds with acceptable accuracy to known values of the parameter that the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip, and shall be deemed to include a channel test.

1.5 **Channel Check -** A channel check is a qualitative verification of acceptable performance by observation of channel behavior. This verification, where possible, shall include comparison of the channel with other independent channels or systems measuring the same variable.

1.6 **Channel Test -** A channel test is the introduction of a simulated input signal into channel and the observation of proper channel response. When applicable, the test shall include verification of proper safety trip operation.

1.7 **Control Blade (Rod) -** A control blade (rod) is either a shim blade (rod) or the regulating blade (rod). The words blade and rod can be used interchangeably.

1.8 **Core Configuration -** The core configuration includes the number, type, or arrangement of fuel elements, reflector elements, and control blades occupying the core region.

1.9 **Excess Reactivity -** Excess reactivity is that amount of positive reactivity that would exist if all of the control blades were moved to the fully withdrawn position from the point where the reactor is exactly critical (K_{eff} = 1) at reference core conditions.

1.10 **Experiment -** An experiment is any operation, hardware, or target (excluding devices such as detectors or foils) which is designed to investigate non-routine reactor characteristics or which is intended for irradiation within an irradiation facility. Hardware rigidly secured to a core or shield structure so as to be part of their design to carry out experiments is not normally considered an experiment.

1.11 **Flux Trap -** The flux trap is that portion of the reactor through the center of the core bounded by the 4.5-inch inside diameter tube and 15 inches above and below the reactor core horizontal center line.

1.12 **Irradiated Fuel -** Irradiated fuel is any fuel element which has been irradiated and used to an integrated power of:

a. Greater than 0.10 megawatt-day;

OR

b. Less than or equal to 0.10 megawatt-day but greater than 1.0 kilowatt-day and with a decay time of less than 7 days since last irradiation;

OR

c. Less than or equal to 1.0 kilowatt-day and with a decay time of less than 24 hours since last irradiation.

1.13 **Limiting Safety System Settings -** Limiting Safety System Settings (LSSS) are settings for automatic protection devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting shall be so chosen that automatic protective action will correct the most severe abnormal situation anticipated before a safety limit is exceeded.

1.14 **Movable Experiment -** A movable experiment is one which is designed with the intent that it may be moved into, out of, or in the near proximity of the reactor while the reactor is operating.

1.15 **Operable -** Operable means a component or system is capable of performing its intended function.

1.16 **Operating -** Operating means a component or system is performing its intended function.

1.17 **Operational Modes -** The reactor may be operated in any of three (3) operating modes, depending upon the configuration of the reactor coolant systems and the protective system set points.

a. Operational Mode I - Reactor can be operated at a thermal power level of ten megawatts or less.

b. Operational Mode II - Reactor can be operated at a thermal power level of five megawatts or less.

c. Operational Mode III - Reactor can be operated at a thermal power level of fifty kilowatts or less.

1.18 **Protective Action** – Protective action is the initiation of a signal or the operation of equipment within the reactor safety system in response to a parameter or condition of the reactor facility having reached a specified limit.

1.19 **Reactivity Worth of an Experiment -** The reactivity worth of an experiment is the value of the reactivity change that results from the experiment, being inserted into or removed from its intended position.

1.20 **Reactor Containment Building -** The reactor containment building is a reinforced concrete structure within the facility site which houses the reactor core, pool, and irradiated fuel storage facilities that is designed to (1) be at a negative internal pressure to ensure in-leakage, (2) control the release of effluents to the environment, and (3) mitigate the consequences of certain analyzed accidents or events.

1.21 **Reactor Core -** The reactor core shall be considered to be that volume inside the reactor pressure vessels occupied by eight or less fuel elements.

1.22 **Reactor Operator -** A reactor operator is an individual who is licensed to manipulate the controls of a reactor.

1.23 **Reactor in Operation -** The reactor shall be considered in operation unless it is either shutdown or secured.

1.24 **Reactor Safety System -** The reactor safety system is that combination of sensing devices, electronic circuits and equipment, signal conditioning equipment, and electro-mechanical devices that serves to either effect a reactor scram, or activates the engineered safety features.

1.25 **Reactor Scram -** A reactor scram is the insertion of all four (4) shim blades (rods) by gravitational force as a result of removing the holding current from the shim rod drive mechanism electromagnets.

1.26 **Reactor Secured -** The reactor shall be considered secured when:

a. There is insufficient fuel in the reactor core to attain criticality with optimum available conditions of moderation and reflection with all four (4) shim blades (rods) removed,

OR

b. Whenever all of the following conditions are met:

(1) All four shim blades (rods) are fully inserted;

(2) One of the two following conditions exits:

i. The Master Control Switch is in the “OFF” position with the key locked in the key box or in custody of a licensed operator,

OR

ii. The dummy load test connectors are installed on the shim rod drive mechanisms and a licensed operator is present in the reactor control room;

(3) No work is in progress involving the transfer of fuel in or out of the reactor core;

(4) No work is in progress involving the shim blades (rods) or shim rod drive mechanisms with the exception of installing or removing the dummy load test connectors; and

(5) The reactor pressure vessel cover is secured in position and no work is in progress on the reactor core assembly support structure.

1.27 **Reactor Shutdown -** The reactor is shutdown when:

a. It is subcritical by at least 0.0074 $\Delta k/k$ in the reference core condition with the reactivity worth of all installed experiments included,

AND

b. All four (4) of the shim blades (rods) are fully inserted and power is unavailable to the shim rod drive mechanism electromagnets.

1.28 **Reference Core Condition -** Reference core condition is the condition of the core when it is at ambient temperature (cold) and the reactivity worth of xenon is negligible ($< 0.002\ \Delta k/k$).

1.29 **Regulating Blade (Rod) -** The regulating blade (rod) is a low worth control blade (rod) used for very fine adjustments in the neutron density in order to maintain the reactor at the desired power level. The regulating blade (rod) may be controlled by the operator with a manual switch or push button, or by an automatic controller. The regulating blade (rod) does not have scram capability nor will it insert on a rod run-in signal.

1.30 **Removable Experiment -** A removable experiment is any experiment which can reasonably be anticipated to be moved during the life of the reactor.

1.31 **Research Reactor -** A research reactor is defined as a device designed to support a self-sustaining neutron chain reaction for research, development, educational, training, or experimental purposes and that may have provisions for the production of radioisotopes.

1.32 **Research Reactor Facility -** A research reactor facility includes all areas within which the owner or operator directs authorized activities associated with the reactor.

1.33 **Rod Run-In System -** The rod run-in system is that combination of sensing devices, electronic circuits and equipment, signal conditioning equipment, and electro-mechanical devices that serves to effect a rod run-in. A rod run-in is the automatic insertion of the shim blades at a controlled rate should a monitored parameter exceed a predetermined value. This system is not part of the reactor safety system, as defined by Specification 1.24; however, it does provide a protective function by introducing shim blade insertion to terminate a transient prior to actuating the reactor safety system.

1.34 **Safety Limits -** Safety Limits (SL) are limits placed upon important process variables which are found to be necessary to reasonably protect the integrity of the principal physical barriers which guard against the uncontrolled release of radioactivity.

1.35 **Scram Time -** Scram time is the elapsed time between the initiation of a scram signal and insertion of the shim blades to the 20% withdrawn position.

1.36 **Secured Experiment -** A secured experiment is any experiment, experimental apparatus, 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 might be subjected by hydraulic, pneumatic, buoyant, or other forces that are normal to the operating environment of the experiment, or by forces that can arise as a result of credible malfunctions.

1.37 **Senior Reactor Operator -** A senior reactor operator is an individual who is licensed to direct the activities of reactor operators and manipulate the controls of a reactor.

1.38 **Shim Blade (Rod) -** A shim blade (rod) is a high worth control blade (rod) used for coarse adjustments in the neutron density and to compensate for routine reactivity losses. The shim blade (rod) is magnetically coupled to its drive mechanism allowing it to scram when the electromagnet is de-energized. The shim blade (rod) also provides rod run-in functions.

1.39 **Shall, Should, and May -** The word “shall” is used to denote a requirement; the word “should” is used to denote a recommendation; and the word “may” is used to denote permission, neither a requirement nor a recommendation.

1.40 **Shutdown Margin -** Shutdown margin is the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and reactor safety systems starting from any permissible operating condition and with the most reactive shim blade and the regulating blade in the fully withdrawn positions, and that the reactor will remain subcritical without further operator action.

1.41 **Surveillance Intervals -** Surveillance intervals are the maximum allowable intervals established to provide operational flexibility and not reduce frequency. Established frequencies shall be maintained over the long term. The surveillance interval is the time between a check, test or calibration, whichever is appropriate to the item being subjected to the surveillance, and is measured from the date of the last surveillance. Allowable surveillance intervals shall not exceed the following:

a. Biennial – interval not to exceed 2.5 years.

b. Annual - interval not to exceed 15 months.

c. Semiannual - interval not to exceed 7.5 months.

d. Quarterly - interval not to exceed 4 months.

e. Monthly - interval not to exceed 6 weeks.

f. Weekly - interval not to exceed 10 days.

g. Daily – interval not to exceed 1 calendar day.

h. Within a shift – interval not to exceed the reactor shift.

1.42 **True Value -** The true value is the actual value of a parameter.

1.43 **Unscheduled Shutdown -** An unscheduled shutdown is defined as any unplanned shutdown, that occurs after all “Blade Full-In Lights” have cleared, caused by actuation of the reactor safety system, rod run-in system, operator error, equipment malfunction, or a manual shutdown in response to conditions that could adversely affect safe operation, not including shutdowns that occur during testing or checkout operations.

1.44 **Unsecured Experiment -** An unsecured experiment is any experiment which is not secured as defined by Specification 1.36, or the moving parts of secured experiments when they are in motion.

The NRC staff reviewed the TS definitions and finds that they are facility specific and consistent with the SAR (or are standard definitions used in research reactor TSs), enhance the clarity of the TSs, and are consistent with NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that the licensee’s TS definitions are acceptable.

14.2 <u>Safety Limits and Limiting Safety System Settings</u>

14.2.1 TS 2.1 Safety Limits

TS 2.1, Safety Limits, is evaluated and found acceptable in SER Section 4.5.3.

14.2.2 TS 2.2 Limiting Safety System Settings

TS 2.2, Limiting Safety System Settings, is evaluated and found acceptable in SER Section 4.5.3.

Based on the information provided above, the NRC staff concludes that the TSs in Section 2.0 meet the requirements in 10 CFR 50.36 (c)(1), and are acceptable.

14.3 Limiting Conditions for Operation

14.3.1 TS 3.1 Reactor Core Parameters

TS 3.1, Reactor Core Parameters, is evaluated and found acceptable in SER Sections 4.2.1 and 4.5.3.

14.3.2 TS 3.2 Reactor Control and Reactor Safety Systems

TS 3.2, Reactor Control and Reactor Safety Systems, is evaluated and found acceptable in SER Sections 4.2.2 and 7.2.

14.3.3 TS 3.3 Reactor Coolant Systems

TS 3.3, Reactor Coolant Systems, is evaluated and found acceptable in SER Section 5.2.7.

14.3.4 TS 3.4 Reactor Containment Building

TS 3.4, Reactor Containment Building, is evaluated and found acceptable in SER Section 6.2.

14.3.5 TS 3.5 Reactor Instrumentation

TS 3.5, Reactor Instrumentation, is evaluated and found acceptable in SER Section 7.2.

14.3.6 TS 3.6 Emergency Electrical Power System

TS 3.6, Emergency Electrical Power System, is evaluated and found acceptable in SER Section 8.2.

14.3.7 TS 3.7 Radiation Monitoring Systems and Airborne Effluents

TS 3.7, Radiation Monitoring Systems and Airborne Effluents, is evaluated and found acceptable SER Section 11.2.2.

14.3.8 TS 3.8 Experiments

TS 3.8, Experiments, is evaluated and found acceptable in SER Section 10.3.

14.3.9 TS 3.9 Auxiliary Systems

TS 3.9, Auxiliary Systems, is evaluated and found acceptable in SER Sections 5.2.7 and 9.7.

14.3.10 TS 3.10 Iodine Processing Hot Cells

TS 3.10 Iodine-131 Processing Hot Cells

TS 3.10 states:

Specification:

a. The facility ventilation exhaust system shall be operable when processing iodine-131 in the iodine-131 processing hot cells.

b. The facility ventilation exhaust system shall maintain the iodine-131 processing hot cells at a negative pressure with respect to the surrounding areas when processing iodine-131.

c. Processing of iodine-131 shall not be performed in the iodine-131 processing hot cells unless the following minimum number of radiation monitoring channels are operable.

	Radiation Monitoring Channel	**Number**
1.	Off-Gas (Stack) Radiation Monitor	1
2.	Iodine-131 Processing Hot Cells Radiation Monitor	1(1)

(1) Exception: When the required radiation monitoring channel becomes inoperable, then portable instruments may be substituted for the normally installed monitor in Specification 3.10.c.2 within one (1) hour of discovery for a period not to exceed one (1) week.

d. At least three (3) charcoal filter banks each having an efficiency of 99% or greater shall be operable when processing iodine-131 in the iodine-131 processing hot cells.

TS 3.10 requires the conditions of the ventilation, radiation monitoring, and carbon filtration systems needed to process I-131 in the I-131 processing hot cells. The NRC staff previously reviewed TS 3.10 during its review of License Amendment (LA) No. 37 (Ref. 90), dated March 11, 2016. LA No. 37 allowed the licensee to conduct isotope production activity, including producing Iodine-131 for medical use. In its safety evaluation for LA No. 37, the NRC staff concluded that licensee demonstrated that either the routine operation, or any potential failure, of I-131 production experiments conducted in accordance with the TS 3.10 would result in doses to MURR staff and to the public that are within the limits of 10 CFR Part 20. On this basis, the NRC staff concludes TS 3.10 acceptable.

Based on the information provided above, the NRC staff concludes that the TSs in Section 3.0 meet the requirements in 10 CFR 50.36 (c)(2) and are acceptable.

14.4 <u>Surveillance Requirements</u>

14.4.0 General

TS 4.0 states:

Specification:

a. Surveillance frequencies denoted herein are based on continuing operation of the reactor. Surveillance activities scheduled to occur during an operating cycle which cannot be performed with the reactor operating may be deferred to the end of that current reactor operating cycle. A reactor system or measuring channel shall not be considered operable until it is successfully tested. Any time a reactor system or component is modified or repaired, the surveillance for that system shall be performed as part of the operability check of the system or component. This shall be done regardless of when the surveillance was last performed or when it is next due. Surveillance intervals shall not exceed those defined by Specification 1.41. Discovery of noncompliance with any of the surveillance specifications listed in this Section shall limit reactor operations to that required to perform the surveillance.

TS 4.0 helps ensure that the quality of systems and components will be maintained to their original design and fabrication specifications, or, if to new specifications, that those specifications have been reviewed. TS 4.0 also specifies the conduct of surveillance requirements required to allow operational flexibility that does not impact safety. TS 4.0 follows the guidance in NUREG-1537, Appendix 14.1, Section 4.0.

NUREG-1537 and ANSI/ANS-15.1-2007 provide guidance that SRs define the frequency and scope of the surveillance activities required to ensure that the LCO are acceptably maintained. The NRC staff finds that TS 4.0 provides appropriate surveillance practices and is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on its review, the NRC staff concludes that TS 4.0 is acceptable.

14.4.1 TS 4.1 Reactor Core Parameters

TS 4.1, Reactor Core Parameters, is evaluated and found acceptable in SER Sections 4.5.3 and 9.2.

14.4.2 TS 4.2 Reactor Control and Reactor Safety Systems

TS 4.2, Reactor Control and Reactor Safety Systems, is evaluated and found acceptable in SER Sections 4.2.2 and 7.2.

14.4.3 TS 4.3 Reactor Coolant Systems

TS 4.3, Reactor Coolant Systems, is evaluated and found acceptable in SER Sections 6.3 and 5.2.7.

14.4.4 TS 4.4 Reactor Containment Building

TS 4.4, Reactor Containment Building, is evaluated and found acceptable in SER Section 6.2.

14.4.5 TS 4.5 Reactor Instrumentation

TS 4.5, Reactor Instrumentation, is evaluated and found acceptable in SER Section 7.2.

14.4.6 TS 4.6 Emergency Electrical Power System

TS 4.6, Emergency Electrical Power System, is evaluated and found acceptable in SER Section 8.2.

14.4.7 TS 4.7 Radiation Monitoring Systems and Airborne Effluents

TS 4.7, Radiation Monitoring Systems and Airborne Effluents, is evaluated and found acceptable in SER Section 11.2.2.

14.4.8 TS 4.8 Experiments

TS 4.8, Experiments, is evaluated and found acceptable in SER Section 10.3.

14.4.9 TS 4.9 Auxiliary Systems

TS 4.9, Auxiliary Systems, is evaluated and found acceptable in SER Sections 5.2.7 and 9.7.

14.4.10 TS 4.10 Iodine-131 Processing Hot Cells

TS 4.10 Iodine-131 Processing Hot Cells

TS 4.10 states:

> Specification:
>
> a. An operability test of the facility ventilation exhaust system shall be performed monthly.
>
> b. A channel check of the facility ventilation exhaust system to maintain the iodine-131 processing hot cells at a negative pressure with respect to the surrounding areas shall be verified daily prior to any process.
>
> c. The radiation monitors as required by Specification 3.10.c shall be calibrated on a semiannual basis.
>
> d. The radiation monitors as required by Specification 3.10.c shall be checked for operability with a radiation source at monthly intervals.
>
> e. The efficiency of the iodine-131 processing hot cells charcoal filter banks shall be verified biennially or following major maintenance. It shall be verified that the charcoal filter banks have a removal efficiency of 99% or greater for iodine.

TS 4.10 requires the surveillances requirements of the ventilation, radiation monitoring, and carbon filtration systems needed to process I-131 in the I-131 processing hot cells. The NRC staff previously reviewed TS 4.10 during its review of License Amendment (LA) No. 37 (Ref. 90), dated March 11, 2016. LA No. 37 allowed the licensee to conduct isotope production activity, including producing I-131 for medical use. In its safety evaluation for LA No. 37, the NRC staff concluded that licensee demonstrated that the frequency and scope of surveillance, as described in proposed TS 4.10, for equipment required by proposed TS 3.10, are adequate to demonstrate that minimum performance levels of the equipment are maintained. On this basis, the NRC staff concludes that TS 4.10 is acceptable.

Based on the information provided above, the NRC staff concludes that the TSs in Section 4.0 meet the requirements in 10 CFR 50.36 (c)(3), and are acceptable.

14.5 Design Features

14.5.1 TS 5.1 Site Description

TS 5.1, Site Description, is evaluated and found acceptable in SER Section 2.1.1.

14.5.2 TS 5.2 Reactor Coolant Systems

TS 5.2, Reactor Coolant Systems, is evaluated and found acceptable in SER Sections 5.2.7, 6.3, and 5.4.

14.5.3 TS 5.3 Reactor Core and Fuel

TS 5.3, Reactor Core and Fuel, is evaluated and found acceptable in SER Sections 4.2.1, 4.2.2, 4.2.3, 9.2, 10.2.3, 4.5.2, and 7.2.

14.5.4 TS 5.4 Fuel Storage

TS 5.4, Fuel Storage, is evaluated and found acceptable in SER Section 9.2.

14.5.5 TS 5.5 Reactor Containment Building

TS 5.5, Reactor Containment Building, is evaluated and found acceptable in SER Sections 6.2 and 9.2.

14.5.6 TS 5.6 Emergency Electrical Power System

TS 5.6, Emergency Electrical Power System, is evaluated and found acceptable in SER Section 8.2.

Based on the information provided above, the NRC staff concludes that the TSs in Section 5.0 meet the requirements in 10 CFR 50.36 (c)(4), and are acceptable.

14.6 Administrative Controls

14.6.1 TS 6.1 Organization

TS 6.1, Organization, is evaluated and found acceptable in See SER Section 12.1.

14.6.2 TS 6.2 Review and Audit

TS 6.2, Review and Audit, is evaluated and found acceptable in SER Section 12.2.

14.6.3 TS 6.3 Radiation Safety

TS 6.3, Radiation Safety, is evaluated and found acceptable in SER Section 12.1.

14.6.4 TS 6.4 Procedures

TS 6.4, Procedures, is evaluated and found acceptable in SER Section 12.3.

14.6.5 TS 6.5 Experiment Review and Approval

TS 6.5, Experiment Review and Approval, is evaluated and found acceptable in SER Section 12.2.

14.6.6 TS 6.6 Reportable Events and Required Actions

TS 6.6, Reportable Events and Required Actions, is evaluated and found acceptable in SER Sections 12.4 and 12.5.

14.6.7 TS 6.7 Records

TS 6.7, Records, is evaluated and found acceptable in SER Section 12.6.

Based on the information provided above, the NRC staff concludes that the TSs in Section 6.0 meet the requirements in 10 CFR 50.36(c)(1), 10 CFR 50.36(c)(2), 10 CFR 50.36(c)(5), 10 CFR 50.36(c)(7), and deemed necessary under 10 CFR 50.36(c)(8), and are acceptable.

14.7 <u>Technical Specification Conclusions</u>

The NRC staff reviewed and evaluated the MURR TSs as part of its review of the license renewal application. Specifically, the NRC staff evaluated the content of the TSs to determine whether the TSs meet the requirements in 10 CFR 50.36. Based on its review, the NRC staff concludes that the MURR TSs are acceptable for the following reasons:

- To satisfy the requirements of 10 CFR 50.36(a), the licensee provided proposed TSs with the license renewal application. The regulation requires that the proposed TSs include appropriate summary statement of the bases or reasons for submitting the TSs, but shall not be part of the TSs as required by 10 CFR 50.36(a)(1).

- MURR is a facility of the type described in 10 CFR 50.21(c); therefore, 10 CFR 50.36(b) requires that the facility operating license include the TSs. To satisfy the requirements of 10 CFR 50.36(b), the licensee provided proposed TSs derived from analyses in the MURR safety analysis report, as supplemented by responses to RAIs.

- The proposed TSs specifying a safety limit (SL) on the fuel temperature and a limiting safety system setting for the reactor protection system to prevent reaching the SL and satisfy 10 CFR 50.36(c)(1), requirements.

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

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

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

- The proposed TSs contain administrative controls that satisfy the requirements of 10 CFR 50.36(c)(5). The proposed TSs contain requirements for initial notification, written reports, and records that satisfy 10 CFR 50.36(c)(1), (2), and (7); and that the NRC staff deemed necessary in accordance with 10 CFR 50.36(c)(8).

- The proposed TSs acceptably implement the recommendations of Part 1 of NUREG-1537, and ANSI/ANS-15.1-2007, by using definitions that are acceptable.

The NRC staff finds the MURR proposed TSs acceptable and concludes that normal operation of MURR within the limits of the TSs will not result in radiation exposures in excess of the limits specified in 10 CFR Part 20, "Standards for Protection Against Radiation," for members of the general public or occupational exposures for facility staff. The NRC staff also finds that the proposed TSs provide reasonable assurance that the facility will be operated as analyzed in the MURR safety analysis report, and in accordance with the applicable regulations. The NRC staff concludes that adherence to the TSs during the license renewal period will limit the likelihood of malfunctions and the potential accident scenarios discussed in SER Chapter 13, and the conduct of activities by the licensee will not endanger the facility staff or members of the public.

15 FINANCIAL QUALIFICATIONS

15.1 Financial Ability To Operate the Reactor

Title 10 of the *Code of Federal Regulations* (10 CFR) 50.33(f) includes the financial requirements for nonelectric utility licensees. The regulation, 10 CFR 50.33(f), states:

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

The regulation, 10 CFR 50.33(f)(2), states:

> [A]pplicants to renew or extend the term of an operating license for a nonpower reactor shall include the financial information that is required in an application for an initial license.

The University of Missouri-Columbia Research Reactor (MURR or the reactor) is a Class 104c (per Section 104 of the Atomic Energy Act of 1954, as amended (AEA)) research and development facility that does not qualify as an "electric utility," as defined in 10 CFR 50.2, "Definitions," since it does not generate or distribute electricity and recover the cost of this electricity, either directly or indirectly, through rates established by the entity itself or by a separate regulatory authority. Also, 10 CFR 50.33(f)(2), requires that an application to renew or extend the term of any operating license for a non-power reactor include the financial information that is required in an application for an initial license. Accordingly, the licensee for MURR must meet the financial qualifications requirements in 10 CFR 50.33(f) and is subject to a full financial qualifications review. The licensee must provide information to demonstrate that it possesses or has reasonable assurance of obtaining the funds necessary to cover estimated operating costs for the period of the license. Specifically, the licensee 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.

By letter dated August 31, 2006, the licensee provides its initial license renewal application (LRA) to the NRC to renew Amended Facility Operating License No. R-103 for its research reactor (Ref. 1). By letter dated April 8, 2016 (Ref. 35), the licensee provides update to the projected operating costs for MURR for each of the fiscal years (FY) 2016 through 2020, which are projected to be $25,650 (in thousands of dollars) in FY 2016, $26,408 in FY 2017, $27,189 in FY 2018, $27,993 in FY 2019, and $28,882 in FY 2020. According to the licensee, campus allocated funding, service operations, partnerships, grants, and other funding are its primary sources of funding to cover its operating costs. The licensee also stated that campus allocation funding represents an annual allocation of the State of Missouri funds for the University, and service operations funding is primarily based on the sale of irradiation, processing, and analytical services. In addition, grants are a revenue source received from non-University sources. The licensee expects that these funding sources will continue for the aforementioned FYs. The licensee also states that MURR reserves will be used to cover any year-end deficits as needed. Using the guidance in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," issued February 1996 (Ref. 51), the NRC staff reviewed the MURR estimated operating costs and projected sources of funds to determine whether they are acceptable.

Based on its review, NRC staff finds that the licensee demonstrated reasonable assurance of obtaining the necessary funds to cover the estimated facility operation costs for MURR for the period of the renewed license and met the acceptance criteria on financial assurance for operations under NUREG-1537. Accordingly, the NRC staff determined that the MURR has met the financial qualification requirements pursuant to 10 CFR 50.33(f), consistent with the guidance provided in NUREG-1537, and therefore is financially qualified to engage in the proposed activities regarding the MURR facility.

MURR is currently licensed as a facility that is useful in research and development under Section 104.c of the AEA, 42 U.S.C. § 2234(c). The regulation, 10 CFR 50.21(c), provides for issuance of a license to a facility which is useful in the conduct of research development activities if no more than 50 percent of the annual cost of owning and operating the facility is devoted to production of materials, products, or the sale of services, other than research and development or education or training. SAR Section 1.1.2 states that the MURR is a multi-disciplinary research and education facility providing a broad range of analytical, radiographic, and irradiation services to the research community and the commercial sector.

By letter dated August 24, 2016 (Ref. 93), the NRC staff requested additional information to determine whether less than 50 percent of the cost of operating the MURR facility was devoted to commercial activities. In its response to RAI No. 1 (Ref. 94), the licensee states, in part, that less than 50 percent of MURR operating costs were devoted to commercial activities. The licensee also provides financial supporting documentation.

The NRC staff reviewed the proposed conduct of commercial activities at MURR. Because 10 CFR 50.21(c) requires that the majority of MURR operating costs be funded by non-commercial uses and the cost of conducting commercial activities at the MURR is less than 50 percent of the total cost of operating the facility, the NRC staff concludes that the MURR license can be renewed as a Section 104.c license.

15.2 Financial Ability To Decommission the Facility

Under 10 CFR 50.33(k), the NRC 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.

Under 10 CFR 50.75(d)(1), each non-power reactor applicant for or holder of an operating license for a production or utilization facility shall submit a decommissioning report as required by 10 CFR 50.33(k). Under 10 CFR 50.75(d)(2), the report must contain a cost estimate for decommissioning the facility, indicate the funding method(s) to be used to provide funding assurance for decommissioning, and describe the means of adjusting the cost estimate and associated funding level periodically over the life of the facility. The regulation, 10 CFR 50.75(e)(1), describes the acceptable methods for providing financial assurance for decommissioning.

The licensee states that the original MURR decommissioning cost estimate was developed using the methodology of NUREG/CR-1756, "Technology, Safety, and Costs of Decommissioning Reference Research and Test Reactors," issued March 1982, for a reference test reactor using the SAFSTOR (safe storage) option for 30 years. According to the licensee, the reference test reactor approach was used because this was thought to more closely represent the decommissioning efforts that the MURR facility would need. The decommissioning cost estimate summarizes costs by labor, equipment, radioactive shipments, termination survey, and annual storage costs for SAFSTOR for 30 years and a 25-percent contingency factor. The annual costs for SAFSTOR includes security; minor maintenance and repair; major repair; offsite laboratory work and equipment repair; reactor facility services; and laboratory samples, U.S. Environmental Protection Agency reports, and surveillance. In its supplement letter dated September 14, 2009, the licensee states, in part, that the decommissioning cost estimate was $47.3 million in 2009 dollars (Ref. 14). The licensee states that it will update its decommissioning cost estimate at 5-year intervals using the methodology in 10 CFR 50.75(c)(2), based on factors in the most recent version of NUREG-1307, "Report on Waste Burial Charges: Changes in Decommissioning Waste Disposal Costs at Low-Level Waste Burial Facilities," (Ref. 98) and the U.S. Bureau of Labor Statistics for labor, energy, and waste burial. In its supplement letter dated April 8, 2016 (Ref. 35), the licensee states that its decommissioning cost estimate is $58.41 million in 2016 dollars. The NRC staff reviewed the information described above and concludes that the MURR decommissioning approach and decommissioning cost estimate are reasonable.

A licensee may elect to use a statement of intent (SOI) to provide financial assurance, as allowed by 10 CFR 50.75(e)(1)(iv) for a non-power reactor license that is a Federal, State, or local government licensee. The SOI must contain or reference a cost estimate for decommissioning and indicate that funds for decommissioning will be obtained when necessary.

To support the SOI and the licensee qualifications to use an SOI, the licensee in its application states that the University of Missouri-Columbia is a nonprofit educational institution and a part of the State government of the State of Missouri and includes documentation that corroborates this statement. The licensee also provides information supporting its representation that the decommissioning funding obligations of the licensee are backed by the full faith and credit of the State of Missouri. In its letter dated March 12, 2013, the licensee provides documentation signed by Kelly Mescher, Office of the General Counsel, University of Missouri, which states that the University of Missouri is a State university that was created by the Missouri Constitution in Article IX Section 9(a). The licensee also provides documentation verifying that Jacquelyn K. Jones, Vice Chancellor for Administrative Services of the University of Missouri, the signator of the SOI, is authorized to execute contracts on behalf of the University of Missouri.

In its letter dated March 29, 2016 (Ref. 35), the licensee provides an updated SOI, stating that the signator will "request that funds be made available as necessary for the [SAFSTOR] decommissioning of the properties owned by the University of Missouri." Further, the signator states that she will "request and obtain these funds over this period sufficiently in advance of required activities to assure timely funding of required activities." The updated SOI is signed by Rhonda K. Gibler, Vice Chancellor for Finance and Chief Financial Officer, University of Missouri-Columbia.

The NRC staff reviewed the licensee's information on decommissioning funding assurance as described above and finds that the University of Missouri is a State government licensee under 10 CFR 50.75(e)(1)(iv), the SOI is acceptable, the decommissioning cost estimate and the annual costs for the SAFSTOR option are reasonable, and the University of Missouri's means of adjusting the cost estimate and associated funding level periodically over the life of the facility is reasonable.

Based on its review, the NRC staff finds that funds will be available to decommission the MURR facility and that the financial status of the applicant regarding decommissioning costs is in accordance with the requirements in 10 CFR 50.33(k) and 10 CFR 50.75, "Reporting and

Recordkeeping for Decommissioning Planning." The NRC staff also finds the licensee's decommissioning cost estimate is consistent with the methodology in NUREG/CR-1756. Therefore, the NRC staff concludes that the financial qualifications of the applicant for decommissioning of the facility are acceptable.

15.3 Foreign Ownership, Control, or Domination

Section 104.d of the AEA prohibits the NRC from issuing a license under Section 104 of the AEA to "any corporation or other entity if the Commission knows or has reason to believe it is owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government." The regulations, 10 CFR 50.33(d) and 10 CFR 50.38, "Ineligibility of Certain Applicants," contain language that implement this prohibition. MURR is owned and operated by the Curators of the University of Missouri, an entity (component unit) of the State of Missouri. In its license renewal application, the licensee states that the University of Missouri is a State of Missouri government licensee and is not owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government.

The NRC staff reviewed the information above and finds that because the University of Missouri is an entity of the State of Missouri government, the NRC has no reason to believe it is foreign owned, controlled, or dominated.

15.4 Nuclear Indemnity

Pursuant to the requirements of the Price-Anderson Act (Section 170 of the AEA) and the NRC's implementing regulations at 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," the licensee currently has an indemnity agreement with the Commission that will terminate when Facility Operating License No. R-103 expires, provided all radioactive material has been removed from the location and transportation of radioactive material from the location has ended. Therefore, the licensee will continue to be a party to the indemnity agreement following issuance of the renewed license. Pursuant to 10 CFR 170.3, "Definitions," and "Subpart D—Provisions Applicable Only to Nonprofit Educational Institutions," to 10 CFR Part 140, the licensee, is a nonprofit educational institution and is not required to provide nuclear liability insurance. The Commission will indemnify the licensee for any claims arising out of a nuclear incident under the Price-Anderson Act, Section 170 of the AEA, as amended, and in accordance with the provisions under its indemnity agreement pursuant to 10 CFR 140.95, "Appendix E - Form of Indemnity Agreement with Nonprofit Educational Institutions," above $250,000, and up to $500 million. In addition, because MURR is a research reactor, the licensee is not required to purchase property insurance otherwise required by 10 CFR 50.54(w).

15.5 Conclusions

The NRC staff reviewed the financial status of the licensee and finds that there is reasonable assurance that the necessary funds will be available to support the safe operation of the MURR facility during the renewal period and, when necessary, to shut down the facility and carry out the decommissioning activities. The NRC staff also finds that the MURR license can be renewed as a Section 104.c license because it is a 10 CFR 50.21(c) facility that is useful in the conduct of research and development activities. The NRC staff also finds that there are no foreign ownership, control or dominating issues, or insurance issues that would preclude the issuance of a renewed license. In addition, the NRC staff finds that the applicable provisions of 10 CFR Part 140 have been satisfied. The NRC staff finds that the licensee is financially qualified to engage in the activities authorized by the renewed facility operating license in accordance with the rules and regulations of the NRC. Based on its review, the NRC staff concludes that licensee is financially qualified to engage in licensed activities during the renewal period.

16 OTHER LICENSE CONSIDERATIONS

16.1 Prior Use of Reactor Components

Safety Analysis Report (SAR) Section 16.1 describes, in general, the history of the University of Missouri-Columbia Research Reactor (MURR) operation and commitments regarding future operation. MURR first attained criticality in October 1966. The facility was originally licensed to operate at a power level of up to 5 megawatts thermal (MWt) and was subsequently licensed to operate at a power level up to10 MWt. All systems, structures, and components that comprise the facility will continue to be used in the same manner as originally designed.

16.1.1 Fuel and Fuel Cladding

SAR Section 16.1.1 describes the prior use of Fuel and Fuel Cladding and provides a justification for the continued use through the term of the renewed license. MURR inspection processes include visually inspecting fuel elements prior to use, during each refueling, and a representative sample of fuel elements at the end-of-life. The licensee states that this provides adequate confidence in the continued performance of the fuel elements and allows detection of any cladding failure or defects. MURR has used over 700 fuel elements since 1971 with no failures. In addition, regular and comprehensive water chemistry control and monitoring provides additional assurance of acceptable cladding integrity. The licensee also states that MURR fuel cycle is managed such that the fuel elements are fully utilized, and returned to the Department of Energy when burnup limits are reached or the fuel will no longer support the operational needs of the MURR facility.

16.1.2 Primary Coolant System Pressure Boundary

SAR Section 16.1.2 describes the prior use of Primary Coolant System (PCS) Pressure Boundary and provides a justification for its continued use through the term of the renewed license. The licensee states that the design of the PCS and the maintenance and surveillance systems in place provide adequate confidence in the continued performance of the PCS and allow detection of any condition that may require corrective actions. A water clean-up loop provides adequate control of corrosion and no significant deterioration has been reported in annual or IRs. In its response to RAI No. 16.2 (Ref. 18), the licensee provides calculations of the peak thermal neutron fluence, on the limiting location of the pressure vessel, and indicated that the limit for the pressure vessel material, Aluminum 6061-T6, would not be reached until year 2044, well beyond the 20-year limit of this license renewal period (year 2036). Continued analysis and measurements of the thermal neutron fluence on the MURR pressure vessel would dictate future prior use decisions.

16.1.3 In-Pool Components Receiving High Neutron Fluence

SAR Section 16.1.3 describes the prior use of in-pool components receiving high neutron fluence and provides a justification for their continued use through the term of the renewed license. The licensee states that five components/regions that receive a high neutron fluence can be replaced—(1) the inner and outer reactor pressure vessels, (2) the flux trap, (3) the control blades, (4) the beryllium reflector, and (5) the graphite reflector elements. SAR Section 16.1.2 describes, in detail, the material condition of the inner and outer pressure vessels. The remaining four components are replaced periodically because of material condition or before they reach their predicted performance limitations.

16.1.4 Reactor Pool Liner

SAR Section 16.1.4 describes prior use of the reactor pool liner and provides a justification for its continued use through the term of the renewed license. The licensee states that a detailed assessment of the reactor pool liner was performed in June 2000. The inspection focused on the welds and the aluminum plate and components around the welds. No evidence of corrosion, distress, cracks, deformations, bulges, buckling, or tears were found on the inspected locations. The assessment concludes that, based on the condition of the reactor pool liner after 34 years of reactor operations, an additional 34 years of acceptable performance is expected.

16.1.5 Reactor Containment Structure and Isolation System

SAR Section 16.1.5 describes prior use of the reactor containment structure and isolation system and provides a justification for its continued use through the term of the renewed license. The licensee states that a detailed assessment of the Reactor Containment Building (RCB) was performed in June 2000 (Ref. 91). The assessment concludes that the containment building was structurally sound and could acceptably respond to the anticipated earthquake potential (see SAR Section 2.5.2.5). The report further states that, following recommended repairs, the structure would continue to perform its function through the proposed license renewed period. SAR Section 16.1.5 indicates that the repairs were completed. The annual containment building compliance test, which measures the leakage rate of the structure, has shown no indication of degradation or a notable trend toward such degradation.

16.1.6 Reactor Safety and Engineered Safety Features Actuation Systems

SAR Section 16.1.6 describes prior use of the reactor safety and engineered safety features actuation systems and provides a justification for their continued use through the term of the renewed license. The licensee states that all safety system and engineered safety system components have inspection, maintenance, and surveillances performed regularly. The detectors, channels, and circuit components have been thoroughly reviewed and upgraded, where applicable, to ensure that they are suitable for use. The mechanical components associated with the reactor safety and engineered safety features actuation systems, such as the control blades, offset mechanisms, isolation doors, valves, gaskets, isolation valves, and actuators, are also adequately monitored through inspection, maintenance, and surveillance to ensure that the systems are operable. The current operating experience and existing maintenance practices justify the continued use of these components and their performance provides sufficient confidence in their continued performance through the proposed license renewed period.

16.1.7 Area Radiation Monitoring System

SAR Section 6.1.7 describes prior use of the area radiation monitoring system and provides a justification for its continued use through the term of the renewed license. The licensee states that all such components are regularly inspected and maintained. Surveillance is performed on those portions that initiate the isolation system. The licensee states that rare electronic failures have occurred in these modules over time, and adequate spares are on hand or available to ensure that the system as a whole will perform as designed through the proposed license period.

16.1.8 Conclusions on Prior Use of Reactor Components

The U.S. Nuclear Regulatory Commission (NRC) staff concludes that there is reasonable assurance that the continued operation of MURR can be conducted without endangering the health and safety of the public, and will be conducted in compliance with the NRC's regulations. The bases for these conclusions include the assumption that the facility systems and components are in good working condition. 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 at MURR include the fuel cladding and the reactor safety system. SER Section 4.2.1 describes the NRC staff review of the reactor fuel. Technical Specification (TS) 3.8, Specification b, requires that fuel elements with identified anomalies not be used in the reactor. TS 4.5, Specification a, requires that one out of each eight fuel assemblies be inspected at end of life. Because of the high exchange rate of fuel assemblies, the fuel receipt inspections that all new fuel assemblies receive will generally occur a relatively short time before their insertion into the reactor.

Additional considerations supporting the continued use of the fuel include the following:

- The design of the in-pool structures and components minimizes the chance for mechanical impact.
- Reactor components are contained in a pressure vessel.
- Fuel handling requires specially designed tools that do not come into contact with the cladding.
- The pressure vessel shields the fuel elements from tools and small objects in the event that they fall into the reactor pool.

As discussed in SER Section 4.2, the licensee has a Preventive Maintenance Program that monitors PCS water chemistry to detect changes that may indicate component degradation.

The electrical design of the reactor safety system (i.e., safety channel circuitry and control rod magnets) helps to preclude accidents as a result of failure of system components. Failure or removal for maintenance of safety-related instrumentation and control components causes a safe reactor shutdown. TS 4.4 specifies surveillance requirements of the reactor control and safety system. These requirements, which the NRC staff evaluated in SER Chapter 7, are consistent with the guidance in American National Standards Institute/American Nuclear Society-15.1-2007, "The Development of Technical Specifications for Research Reactors," issued 2007 (Ref. 57), and ensure that gradual degradation of system components will be detected. Additionally, the MURR facility staff performs regular preventive and corrective maintenance and replaces system components as necessary.

The NRC staff finds that there is no indication of significant degradation of the instrumentation and components and there is strong evidence that the MURR facility staff will remedy any future degradation with prompt corrective action.

The NRC staff reviewed the prior use of reactor components. Based on its review, the NRC staff concludes that there has been no significant degradation of reactor components to date. Further, the surveillance requirements in the TSs provide reasonable assurance that the reactor components will continue to be adequately monitored for degradation of systems and components during the renewed license period.

16.2 Medical Use of MURR

The licensee does not use MURR for medical irradiations involving the use of special nuclear material for medical therapy.

17 CONCLUSIONS

On the basis of its evaluation of the application as discussed in the previous chapters of this safety evaluation report, the U.S. Nuclear Regulatory Commission (NRC) staff concludes the following:

- The application for license renewal dated August 31, 2006, as supplemented on January 29, July 16, August 31, September 3, September 30, October 29 (two letters), and November 30, 2010; March 11, 2011; September 8, 2011; January 6, 2012; June 28, 2012; January 4, 2013; January 28, July 31, September 15, and October 1, 2015; and February 8, April 8, April 15, May 31, July 25, August 31, November 7, and November 15, 2016 (two letters), complies with the standards and requirements of the Atomic Energy Act (AEA), and the Commission's rules and regulations set forth in Title 10 of the Code of Federal Regulations (10 CFR).

- The facility will operate in conformity with the application, the provisions of the AEA of 1954, as amended, and the rules and regulations of the NRC.

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

- The facility will continue to be useful in the conduct of research and development activities.

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

- The applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," have been satisfied.

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

- The issuance of this license is in accordance with 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," of the NRC's regulations, and all applicable requirements have been satisfied, as documented by the Environmental Assessment and Finding of No Significant Impact published in the *Federal Register* on November 29, 2016 (81 FR 86024), which concluded that renewal of the MURR license will not have a significant effect on the quality of the human environment.

- The receipt, possession, and use of byproduct and special nuclear materials, as authorized by this renewed facility operating license, will be in accordance with the NRC's regulations in 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material."

18 REFERENCES

1. University of Missouri-Columbia, “Application for License Renewal of the University of Missouri-Columbia Research Reactor as per 10 CFR 2.109,” August 31, 2006, Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML062540114 (cover letter), ML092110573 (Chapters 1–9 of the safety analysis report), ML092110597 (Chapters 10–18 of the safety analysis report (redacted version)), and ML062540121 (environmental report (redacted version)).

2. University of Missouri-Columbia, “Preliminary Hazards Summary Report,” March 31, 1961, ADAMS Accession No. ML16215A374 (redacted version).

3. University of Missouri-Columbia, “Hazards Summary Report,” July 1, 1965, ADAMS Accession No. ML16215A386 (redacted version).

4. University of Missouri-Columbia, “Hazards Summary Report,” Addendums 1–5, July 1, 1965, ADAMS Accession No. ML16215A415 (redacted version).

5. U.S. Nuclear Regulatory Commission, “University of Missouri-Columbia Amendment, Re: Extension of License Expiration Date,” License Amendment (LA) No. 32, October 19, 2001, ADAMS Accession No. ML011170201.

6. U.S. Nuclear Regulatory Commission, “University of Missouri-Columbia, Request for Additional Information, Re: License Renewal, Safety Analysis Report, Complex Questions,” May 6, 2010, ADAMS Accession No. ML101160244.

7. U.S. Nuclear Regulatory Commission, “University of Missouri-Columbia, Request for Additional Information, Re: License Renewal, Safety Analysis Report, 45-Day Response Questions,” June 1, 2010, ADAMS Accession No. ML101160266.

8. U.S. Nuclear Regulatory Commission, “University of Missouri-Columbia, Request for Additional Information Regarding the Renewal of Facility Operating License No. R-103 for the University of Missouri-Columbia Research Reactor,” December 3, 2014, ADAMS Accession No. ML14324A707.

9. U.S. Nuclear Regulatory Commission, “University of Missouri-Columbia, Request for Additional Information Regarding the Renewal of Facility Operating License No. R-103 for the University of Missouri-Columbia Research Reactor,” April 17, 2015, ADAMS Accession No. ML15098A648.

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11. U.S. Nuclear Regulatory Commission, “University of Missouri-Columbia, Request for Additional Information Regarding the Proposed Technical Specifications for the Renewal of Facility Operating License No. R-103 for the University of Missouri-Columbia Research Reactor,” October 28, 2015, ADAMS Accession No. ML15280A314.

12. U.S. Nuclear Regulatory Commission, "University of Missouri-Columbia, Clarifications Needed to the U.S. Nuclear Regulatory Commission Staff Request for Additional Information Regarding the Renewal of Facility Operating License No. R-103 for the University of Missouri-Columbia Research Reactor (TAC No. ME1580)," December 17, 2015, ADAMS Accession No. ML15337A175.

13. U.S. Nuclear Regulatory Commission, "University of Missouri-Columbia, Clarifications Needed to U.S. Nuclear Regulatory Commission Staff Request for Additional Information Regarding the Renewal of Facility Operating License No. R-103 for the University of Missouri-Columbia Research Reactor (TAC No. ME1580)," March 23, 2016, ADAMS Accession No. ML16083A025.

14. University of Missouri-Columbia, "Written Communications as Specified by 10 CFR 50.4(b)(1) Regarding the Response to the 'University of Missouri at Columbia Request for Additional Information, Re: License Renewal (TAC No. ME1580),' dated July 10, 2009," September 14, 2009, ADAMS Accession No. ML092590298.

15. University of Missouri-Columbia, "Written Communications as Specified by 10 CFR 50.4(b)(1) Regarding the Response to the 'University of Missouri at Columbia—Request for Additional Information, Re: License Renewal References (TAC No. ME3034),' dated December 30, 2009," January 15, 2010, ADAMS Accession Nos. ML100220371 and ML100221046.

16. University of Missouri-Columbia, "Written Communications as Specified by 10 CFR 50.4(b)(1) Regarding the Response to the 'University of Missouri at Columbia—Request for Additional Information, Re: License Renewal Environmental Report (TAC No. MD3034),' dated December 30, 2009," January 29, 2010, ADAMS Accession No. ML100330073.

17. University of Missouri-Columbia, "Written Communication as Specified by 10 CFR 50.4(b)(1) Regarding Responses to 'University of Missouri at Columbia—Request for Additional Information, Re: License Renewal, Safety Analysis Report, 45-Day Response Questions (TAC No. MD3034),' dated June 1, 2010," July 16, 2010, ADAMS Accession No. ML12354A237 (redacted version).

18. University of Missouri-Columbia, "Written Communication as Specified by 10 CFR 50.4(b)(1) regarding responses to 'University of Missouri at Columbia—Request for Additional Information, Re: License Renewal, Safety Analysis Report, 45-Day Response Questions (TAC No. MD3034),' dated June 1, 2010," August 31, 2010, ADAMS Accession No. ML120050315 (redacted version).

19. University of Missouri-Columbia, "Written Communication as Specified by 10 CFR 50.4(b)(1) Regarding Responses to 'University of Missouri at Columbia—Request for Additional Information, Re: License Renewal, Safety Analysis Report, Complex Questions (TAC No. MD3034),' dated May 6, 2010," September 3, 2010, ADAMS Accession No. ML102500533.

20. University of Missouri-Columbia, "Written Communication as Specified by 10 CFR 50.4(b)(1) Regarding Responses to 'University of Missouri at Columbia—Request for Additional Information, Re: License Renewal, Safety Analysis Report,

45-Day Response Questions (TAC No. MD3034),' dated June 1, 2010," September 30, 2010, ADAMS Accession No. ML12355A019.

21. University of Missouri-Columbia, "Written Communication as Specified by 10 CFR 50.4(b)(1) Regarding Responses to 'University of Missouri at Columbia—Request for Additional Information, Re: License Renewal, Safety Analysis Report, 45-Day Response Questions (TAC No. MD3034),' dated June 1, 2010," October 29, 2010, ADAMS Accession No. ML12355A023 (redacted version).

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