Safety Evaluation Report

Related to Renewal of the Facility Operating License for the University of Massachusetts Lowell Research Reactor

License No. R-125
Docket No. 50-223

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 University of Massachusetts Lowell (the licensee) for a 20-year renewal of Facility Operating License No. R-125 to continue to operate the University of Massachusetts Lowell Research Reactor (UMLRR). In its safety review, the NRC staff considered information submitted by the licensee, past operating history recorded in the licensee's annual operating reports to the NRC, inspection reports prepared by NRC staff, and the NRC staff's observations during site visits. Based on its review, the NRC staff concludes that the licensee can continue to operate the UMLRR for the 20-year term of the renewed facility license, in accordance with the license, without endangering public health and safety, UMLRR staff, or the environment.

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List of Abbreviations, Acronyms, Symbols, and Units

Abbreviation/Acronym/Symbol	Definition
10 CFR	Title 10 of the *Code of Federal Regulations*
β_{eff}	effective delayed neutron fraction
Λ	neutron generation time
AEA	Atomic Energy Act of 1954, as amended
AEC	U.S. Atomic Energy Commission
ALARA	as low as is reasonably achievable
ALI	annual limit on intake
Am-Be	americium-beryllium
ANSI/ANS	American National Standards Institute/American Nuclear Society
Ar-41	argon-41
ARM CDAS	Area Radiation Monitor Computer Data Acquisition System
ARMS	area radiation monitoring system
B_4C	boron carbide
BOL	beginning-of-life
CAM	continuous air monitor
CDE	committed dose equivalent
CEDE	committed effective dose equivalent
CIC	compensated ion chamber
CM	configuration management
Co-60	cobalt-60
Cs-138	cesium-138
CsI	cesium iodide
CP	construction permit
DAC	derived air concentration
DC	direct-current
DCF	dose conversion factor
DCS	drives control system
DDE	deep dose equivalent
DECON	immediate decontamination
DIB	digital interface board
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DOE	U.S. Department of Energy
DRS	Director of Radiation Safety
ECP	estimated critical position
ESF	engineered safety feature
FGR 11	Federal Guidance Report No. 11
FGR 12	Federal Guidance Report No. 12
GA	General Atomics
GREA	general radiation emergency alarm
GRVS	general reaction in the ventilation system
GUI	graphical user interface
HEPA	high efficiency particulate air

HEU	Highly enriched uranium
HMI	human-machine interface
I&C	instrumentation and control
I/O	input/output
I-131	iodine-131
I-135	iodine-135
I_2	elemental iodine
IFI	inspector follow-up item
ISG	interim staff guidance
k_{eff}	k-effective
Kr-89	krypton-89
Kr-90	krypton-90
LC	license condition
LCC	limiting core configuration
LCD	liquid crystal display
LCO	limiting condition for operation
LED	light-emitting diode
LEU	low-enriched uranium
LOCA	loss of coolant accident
Log	logarithmic
LRA	license renewal application
LREA	local radiation emergency alarm
LSSS	limiting safety system setting
MCNP	Monte Carlo N-Particle Transport
MHA	maximum hypothetical accident
MMC	metal matrix composite
MTR	materials test reactor
MURR	Missouri University Research Reactor
MWD	megawatt-days
N-16	nitrogen-16
NI	nuclear instrumentation
NIM	nuclear instrumentation module
NRC	U.S. Nuclear Regulatory Commission
NVLAP	National Voluntary Laboratory Accreditation Program
OCC	operational core configuration
ONB	onset of nucleate boiling
ONBR	onset of nucleate boiling ratio
ORP	operator requalification program
PARET	Program for the Analysis of Reactor Transients
PCI	process controls and instrumentation
PCS	process controls system
P-GREA	potential-general radiation emergency alarm
PID	proportional gain, integral, derivative
P-LREA	potential-local radiation emergency alarm
PPM	period power module
PSP	physical security plan
Pu	plutonium
Pu-Be	plutonium-beryllium
QA	quality assurance

RAI	request for additional information
RCS	reactor control system
RG	regulatory guide
RINSC	Rhode Island Nuclear Science Center
RO	reactor operator
RPS	reactor protection system
RSC	Radiation Safety Committee
RSO	radiation safety officer
RSP	Radiation Safety Program
RSSC	Reactor Safety Subcommittee
RTD	resistance temperature detector
RTR	research and test reactor
SAR	safety analysis report
Sb-Be	antimony-beryllium
SDM	shutdown margin
SER	safety evaluation report
SGI-M	Safeguards Information – Modified Handling
SL	safety limit
SNM	special nuclear material
SOI	statement of intent
SR	surveillance requirement
SRM	staff requirements memorandum
SRO	senior reactor operator
TAA	trip actuator amplifier
TEDE	total effective dose equivalent
TFS	Thermo-Fisher Scientific
TNT	trinitrotoluene
TRACE	TRAC/RELAP Advanced Computational Engine
TRIGA	Training, Research, Isotopes, General Atomics
TS	technical specification
TSD	touchscreen display
U-235	uranium-235
U_3Si_2-Al	aluminum-clad uranium-silicide
UAl_x-Al	aluminum-clad uranium-aluminide
UML	University of Massachusetts Lowell
UMLRR	University of Massachusetts Lowell Research Reactor
V&V	verification and validation
WDT	watchdog timer
WPI	Worcester Polytechnic Institute
X/Q	chi over q
Xe-137	xenon-137
Xe-139	xenon-139

Unit	Definition
°C	degree Celsius
°F	degree Fahrenheit
% Δk/k	percent delta k over k
µCi	microcurie
µCi/mL	microcurie per milliliter
µCi/s	microcurie per second
µmho/cm	micromho per centimeter
Ci	curie
cfm	cubic feet per minute
cps	counts per second
gpm	gallon per minute
kWt	kilowatt-thermal
MeV	mega electron-volt
msec	millisecond
mrem	millirem
MWt	megawatt-thermal
n/cm^2-s	neutrons per square centimeter per second
VAC	alternating-current volts
Vdc	direct-current volts

1. INTRODUCTION

1.1 Overview

By letter dated October 20, 2015 (Ref. 1), as supplemented by letters dated March 16, 2016 (Ref. 19); November 30, 2016 (Ref. 1); March 31, 2017 (Ref. 23); July 11, 2017 (Ref. 3); August 7, 2017 (Ref. 4); September 13, 2017 (Ref. 5); January 6, 2018 (Ref. 43); February 1, 2018 (Ref. 44); March 5, 2019 (Ref. 71); April 10, 2019 (Ref. 73); October 18, 2019 (Ref. 79); October 24, 2019 (Ref. 80); December 19, 2019 (Ref. 81); December 20, 2019 (Ref. 82); February 24, 2020 (Ref. 85); September 30, 2020 (Ref. 98); January 30, 2021 (Ref. 99); February 16, 2021 (Ref. 101); April 5, 2021 (Ref. 90); and April 20, 2021 (Ref. 92), the University of Massachusetts Lowell (UML, the licensee) submitted to the U.S. Nuclear Regulatory Commission (NRC, the Commission) a license renewal application (LRA) for a 20-year renewal of the Class 104.c Facility Operating License No. R-125 (NRC Docket No. 50-223) for the University of Massachusetts Lowell Research Reactor (UMLRR, the facility), which is a materials test reactor (MTR)-type facility located on the UML campus in Lowell, Massachusetts. A Notice of Opportunity to Request a Hearing was published in the *Federal Register* on March 19, 2018 (Ref. 46, 83 FR 12036). No requests for hearing were received.

Title 10 of the *Code of Federal Regulations* (10 CFR) 50.51(a) states, in part, that "[e]ach license will be issued for a fixed period of time to be specified in the license but in no case to exceed 40 years from date of issuance." On December 24, 1974, the U.S. Atomic Energy Commission (AEC) issued Facility Operating License No. R-125 to the Lowell Technological Institute (later changed to the University of Lowell by License Amendment No. 2, dated October 15, 1975). By License Amendment No. 9, dated November 21, 1985 (Ref. 105), the NRC renewed Facility Operating License No. R-125 for a period of 30 years, expiring on November 21, 2015. License Amendment No. 11, dated February 20, 1992 (Ref. 106), changed the licensee name to UML. Although the current license has a November 21, 2015, expiration date, under 10 CFR 2.109(a), because UML filed its October 20, 2015, LRA in a timely manner, the current operating license for the UMLRR does not expire until the NRC staff makes a final determination on the LRA. The proposed renewal would authorize continued operation of the UMLRR for an additional 20 years from the date of issuance of a renewed license.

The NRC staff conducted its review of the LRA based on information contained in the LRA, as well as information in supporting supplements and licensee responses to requests for additional information (RAIs) issued by the NRC staff on the LRA. The LRA included the safety analysis report (SAR), the technical specifications (TSs), the emergency plan (EP), the physical security plan (PSP), and the operator requalification program (ORP).

The NRC staff issued RAIs by letters dated November 2, 2016 (Ref. 6); February 1, 2017 (Ref. 7); June 21, 2017 (Ref. 8); July 14, 2017 (Ref. 9); November 7, 2017 (Ref. 35); November 5, 2018 (Ref. 40); and July 19, 2019 (Ref. 75). UML responded to the NRC staff's RAIs by letters dated November 30, 2016 (Ref. 1); March 31, 2017 (Ref. 23), July 11, 2017 (Ref. 3); August 7, 2017 (Ref. 4); September 13, 2017 (Ref. 5); January 6, 2018 (Ref. 43); March 5, 2019 (Ref. 71); October 18, 2019 (Ref. 79); October 24, 2019 (Ref. 80); December 19, 2019 (Ref. 81); December 20, 2019 (Ref. 82); and February 24, 2020 (Ref. 85).

As part of its review of the LRA, the NRC staff also reviewed annual reports of the facility operation submitted by the licensee for 2005-2006 through 2019-2020 (Ref. 10) and inspection reports prepared by NRC staff from 2006 through 2019 (Ref. 11). The NRC staff also

conducted site visits at the facility to observe facility conditions and to discuss RAIs on February 14-15, 2017; July 25-27, 2017; and January 9-10, 2019.

Also as part of its review of the LRA, the NRC staff conducted two regulatory audits: an onsite audit conducted on July 26-27, 2017 and an audit conducted from February 10, 2020 – March 2, 2021, by teleconference and using an online portal for document review. The NRC staff provided UML with audit plans by letters dated July 19, 2017 (Ref. 87), for the 2017 audit, and February 4, 2020 (Ref. 88), for the 2020-2021 audit. By letter dated December 17, 2020 (Ref. 86), the NRC staff provided UML with a combined final audit report on the 2017 audit and interim audit report on the 2020-2021 audit. By letter dated March 4, 2021 (Ref. 100), the NRC staff provided UML with a final audit report on the 2020-2021 audit.

With the exception of the PSP and portions of the SAR, RAI responses, and the EP that contain security-related, proprietary, and/or copyrighted information, material pertaining to the NRC staff review of the LRA may be examined or copied, for a fee, at the NRC's Public Document Room, Room P1-B35, at One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852. To make an appointment to visit the Public Document Room, a request must be submitted to the Public Document Room staff using the contact information below. The NRC also maintains an Agencywide Documents Access and Management System (ADAMS), which provides text and image files of the NRC's public documents. Publicly available documents related to the NRC staff review of the LRA may be accessed through the ADAMS Public Documents Library on the Internet at https://www.nrc.gov/reading-rm/adams.html. The NRC Public Document Room reference staff may be contacted regarding problems in accessing the documents located in ADAMS at 1-800-397-4209 or 301-415-4737, or by e-mail to pdr.resource@nrc.gov. The entire PSP is protected from public disclosure under 10 CFR 73.21, "Protection of Safeguards Information: Performance Requirements." Redacted versions of other documents that contain security-related information, and some documents that contain proprietary information, are publicly available in ADAMS.

The dates and associated ADAMS accession numbers for documents related to the NRC staff review of the LRA are listed in Chapter 8, "References" of this safety evaluation report (SER).

In conducting its review, the NRC staff evaluated the facility against the requirements of the NRC's 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 guidance in applicable NRC regulatory guides (RGs) and relevant accepted industry standards, such as the American National Standards Institute/American Nuclear Society (ANSI/ANS)-15 series is also considered. The NRC staff also used to the guidance in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," issued February 1996 (Ref. 14). Because there are no specific accident-related regulations for research reactors, the NRC staff compared calculated dose values for accidents against the requirements in 10 CFR Part 20 (i.e., the standards for protecting employees and the public against radiation).

In SECY-08-0161, “Review of Research and Test Reactor License Renewal Applications,” dated October 24, 2008 (Ref. 16), the NRC staff provided the Commission with information regarding plans to streamline the review of LRAs for research and test reactors (RTRs). The Commission subsequently issued its staff requirements memorandum (SRM)-SECY-08-0161, dated March 26, 2009 (Ref. 15). The SRM directed the NRC staff to streamline the renewal process for RTRs, 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 the alternative safety review approach described in Enclosure 1 of SECY-08-0161. In the alternative safety review approach, the NRC staff considers the results of past NRC staff evaluations when determining the scope of the review. A basic requirement, as contained in the SRM, is that licensees be in compliance with applicable regulatory requirements.

The NRC staff developed interim staff guidance (ISG)-2009-001, “Interim Staff Guidance on the Streamlined Review Process for License Renewal for Research Reactors” (Ref. 17), for the review of LRAs. The streamlined review process is a graded approach based on licensed power level. Under this process, facilities are divided into two tiers. Facilities with a licensed power level of 2 megawatts thermal (MWt) and greater, or 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 LRA and relies on past NRC reviews for certain findings. The NRC staff issued a draft of the ISG for public comment and the NRC staff considered public comments in its development of the final ISG.

The NRC staff conducted its review of the UMLRR LRA using the guidance in the final ISG. Since the licensed power level for the UMLRR is less than 2 MWt, and since UML is not requesting a power level increase for the UMLRR, the NRC staff performed a focused review of the LRA. Specifically, the review focused on reactor design and operation, instrumentation and control systems, radiation protection, accident analysis, TSs, financial requirements, environmental assessment, and changes to the facility made during NRC review of the LRA. The ISG states that as part of a focused review, the NRC staff does not normally need to perform a detailed review of instrumentation and control systems. However, because UML proposes to install new instrumentation and control systems, which it described in its license renewal SAR, as supplemented, and due to the scope of instrumentation and control changes that UML made under 10 CFR 50.59, “Changes, tests, and experiments,” since the issuance of the 1985 renewed UMLRR license (see SER Section 1.8), the NRC staff included a full review of instrumentation and control systems within the scope of its focused review in SER Section 3.0.

TSs, pursuant to 10 CFR 50.36, “Technical specifications,” include items in the following categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. UML has proposed TSs to satisfy the requirements in 10 CFR 50.36. The NRC staff also used the guidance in ANSI/ANS-15.1-2007, “The Development of Technical Specifications for Research Reactors” (Ref. 18), in evaluating the UMLRR TSs. Although the guidance in NUREG-1537, issued in 1996, endorses ANSI/ANS-15.1-1990, the NRC staff used ANSI/ANS-15.1-2007 in its review consistent with the updated guidance in ISG-2009-001.

This SER summarizes the findings of the safety review of the application for a renewed UMLRR operating license, and delineates the technical details considered in evaluating the radiological

safety aspects of continued operation. This SER provides the basis for renewing the license for operation of the UMLRR at thermal power levels up to and including 1 MWt.

This SER was prepared by Edward Helvenston and Eben Allen, Project Managers, and Duane Hardesty, Senior Project Manager, from the NRC's Office of Nuclear Reactor Regulation, Division of Advanced Reactors and Non-power Production and Utilization Facilities, Non-power Production and Utilization Facility Licensing Branch; Joseph Staudenmeier, Senior Reactor Systems Engineer from the NRC's Office of Research, Division of Systems Analysis, Code and Reactor Analysis Branch; and Nicole Newton, Mable Henderson, and Kosmas Lois, Financial Analysts from the Financial Assessment Branch, Division of Rulemaking, Environmental, and Financial Support in the NRC's Office of Nuclear Material Safety and Safeguards. Energy Research Incorporated, and Oak Ridge National Laboratory, NRC contractors, also provided input to this SER.

1.1.1 Operator Requalification Program Review

As part of its review of the LRA, the NRC staff reviewed the UMLRR ORP. The NRC staff reviewed the ORP, Revision 2, dated April 2008 (Ref. 19), and determined that additional information was needed to complete its review of the ORP. Therefore, by letter dated November 2, 2016 (Ref. 6), the NRC staff issued an RAI regarding the ORP. UML responded to the NRC staff's RAI by letter dated November 30, 2016 (Ref. 1). UML's RAI response included the ORP, Revision 3, dated November 2016. The NRC staff found the ORP, Revision 3, dated November 2016, to be consistent with guidance contained in ANSI/ANS-15.4-2016, "Selection and Training of Personnel for Research Reactors" (Ref. 20), and in accordance with the applicable regulations contained in 10 CFR Part 55, "Operators' Licenses." Therefore, by letter dated February 7, 2018 (Ref. 12), the NRC staff stated that it had concluded that the ORP, Revision 3, dated November 2016, is acceptable for use at the UMLRR, and requested that UML implement its approved program.

1.1.2 Emergency Plan Review

The NRC staff reviewed the UMLRR EP as part of its review of the LRA. The NRC staff reviewed the EP, Revision 7, dated May 23, 2013 (Ref. 27). The NRC staff found that the EP, Revision 7, is consistent with the following applicable guidance:

- NRC RG 2.6, "Emergency Planning for Research and Test Reactors," Revision 1, dated March 1983 (Ref. 28);
- ANSI/ANS-15.16-2015, "Emergency Planning for Research Reactors" (Ref. 29);
- NUREG-0849, "Standard Review Plan for the Review and Evaluation of Emergency Plans for Research and Test Reactors" (Ref. 30);
- NRC Information Notice 97-34, "Deficiencies in Licensee Submittals Regarding Terminology for Radiological Emergency Action Levels in Accordance with the New Part 20" (Ref. 31); and
- NRC Information Notice 92-79, "Non-Power Reactor Emergency Event Response" (Ref. 32).

The NRC staff also found that the EP complies with 10 CFR 50.34(b)(6)(v). Based on these findings, the NRC staff concluded that the UMLRR EP, Revision 7, is acceptable, as documented in a letter to UML dated August 27, 2019 (Ref. 76). The NRC staff notes that UML may make changes to the EP without prior NRC approval, if, as stated in 10 CFR 50.54(q), UML

performs and retains an analysis that demonstrates that the changes do not reduce the effectiveness of the EP and the EP, as changed, continues to meet the NRC's requirements. The NRC staff also reviewed the UMLRR inspection reports from 2006 through 2019 (Ref. 11), and found no significant issues related to emergency preparedness or compliance with the EP.

1.1.3 Physical Security Plan Review

As part of its review of the LRA, the NRC staff reviewed the UMLRR PSP. The NRC staff reviewed the PSP, Revision 7, which was previously submitted to the NRC by letter dated January 13, 2015 (Ref. 21), and determined that additional information was needed to complete its review of the PSP. Therefore, by letter dated July 14, 2017 (Ref. 9), the NRC staff issued an RAI regarding the PSP. UML responded to the NRC staff's RAI by letter dated August 7, 2017 (Ref. 4). UML's RAI response included the PSP, Revision 8. Following a teleconference on September 12, 2017 (summarized in an e-mail from the NRC staff to UML (Ref. 22)), during which the NRC staff and UML discussed minor corrections to the PSP, UML submitted the PSP, Revision 9, to the NRC by letter dated September 13, 2017 (Ref. 5).

Based on its review, the NRC staff found that the PSP, Revision 9, meets the applicable guidance in NRC RG 5.59, "Standard Format and Content for a Licensee Physical Security Plan for the Protection of Special Nuclear Material of Moderate of Low Strategic Significance" (Ref. 33), and the applicable regulations in 10 CFR Part 73, "Physical Protection of Plants and Materials." The NRC staff also found that the PSP, Revision 9, incorporates the site-specific compensatory security measures discussed in Confirmatory Action Letter No. NRR-03-009, dated October 22, 2003 (Ref. 34), which were put in place after the attacks of September 11, 2001, to enhance security at research reactors. Based on these findings, the NRC staff concluded that the PSP, Revision 9, is acceptable, as documented in a letter to UML dated August 27, 2019 (Ref. 77). The NRC staff notes that UML may make changes to the PSP without prior NRC approval, in accordance with 10 CFR 50.54(p), if the changes do not decrease the effectiveness of the PSP. The NRC staff also reviewed the UMLRR security-related inspection reports from 2007, 2011, 2015, and 2018 (Ref. 11.b, Ref. 11.h, Ref. 11.m, and Ref. 11.q, respectively), and found no significant issues related to physical security or compliance with the PSP.

1.1.4 Environmental Review

The NRC staff separately evaluated the environmental impacts of the renewal of the license for the UMLRR 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 August 4, 2021 (Ref. 104, 86 FR 41998), which concluded that the renewal of the UMLRR operating license will not have a significant effect of the quality of the human environment.

1.2 Summary and Conclusions on Principal Safety Considerations

The NRC staff's evaluation of the LRA considered the information submitted by the licensee, including past operating history recorded in the licensee's annual reports to the NRC; inspection reports prepared by NRC staff; and observations made by the NRC staff during site visits conducted for the review of the LRA. On the basis of this evaluation and resolution of the principal issues reviewed for the UMLRR, the NRC staff concludes the following:

- The design and use of reactor structures, systems, and components important to safety during normal operation described in Chapter 4 of the SAR, as supplemented, in

accordance with the TSs, are safe, and safe operation can reasonably be expected to continue.

- The UMLRR will continue to be useful in the conduct of research and development activities. As discussed in SAR Section 1.6.1, UML's Radiation Laboratory, which includes the UMLRR, is a major research focal point of UML. Much research is correlated with safety and efficiency in the nuclear and radiation industries, including research related to nuclear power generation, pharmaceuticals, medical applications, and health effects. Research is also done by workers in other fields who use the Radiation Laboratory's unique facilities as analytical tools. Additionally, the Radiation Laboratory's facilities are used in the course work of various departments of UML, other universities, governmental agencies, and, to a limited extent, industrial organizations.

- In SAR Chapter 13, as supplemented, the licensee considered the expected consequences of a broad spectrum of postulated credible accidents and a maximum hypothetical accident (MHA), emphasizing those that could lead to a loss of integrity of fuel plate cladding and a release of fission products. The licensee performed conservative analyses of the most serious credible accidents and the MHA, and determined that the calculated potential radiation doses for facility personnel and members of the public would not exceed 10 CFR Part 20 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.

- Operation of the facility and the handling of radioactive material under the control of the UMLRR 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 as low as is reasonably achievable (ALARA) principles.

- When operated in accordance with the TSs, the systems provided for the control of radiological effluents are adequate to ensure that releases of radioactive materials from the facility are within the limits of the Commission's regulations and are ALARA.

- The TSs, which state limits controlling the operation of the facility, are such that there is reasonable assurance that the facility will be operated safely and reliably. There has been no significant degradation of the reactor, 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 facility.

- The licensee's program for providing for the physical protection of the facility and its special nuclear material (SNM) complies with the requirements of 10 CFR Part 73, which provides reasonable assurance that the licensee will continue to provide for 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 senior reactor operators and the ORP provide reasonable assurance that the licensee will continue to have qualified staff that can safely operate the facility.

On the basis of these findings, the NRC staff concludes that there is reasonable assurance that the licensee can continue to operate the UMLRR in accordance with the Atomic Energy Act of 1954, as amended (the AEA), NRC regulations, and the renewed facility operating license, without endangering public health and safety, facility staff, 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 Description of the Facility

SAR Chapter 1, as supplemented by responses to RAIs, provides a general description of the UMLRR. The UMLRR is a water moderated and cooled, open pool type research reactor located within a confinement building (changed from a containment to a confinement building as part of this renewal, as discussed in SER Section 1.8 and Chapter 6), which is also referred to as the reactor building, and which is connected to the adjacent Pinanski building on UML's main campus in Lowell, Massachusetts. It is licensed to operate with forced-convection cooling at power levels up to 1 MWt, or up to 0.1 MWt with natural convection cooling.

The UMLRR has MTR fuel elements with uranium enriched to 19.75 percent uranium-235 (U-235). The fueled core region is reflected by water and graphite reflector elements. The current UMLRR fuel is aluminum-clad uranium-silicide (U_3Si_2-Al) fuel. The U_3Si_2-Al is also referred to as silicide fuel in this SER. Following license renewal, the UMLRR core may also include aluminum-clad uranium-aluminide (UAl_x-Al) fuel, obtained from the Worcester Polytechnic Institute (WPI) following the permanent shutdown of the WPI research reactor (the use of this fuel was approved as part of this license renewal, as discussed in SER Sections 1.8 and 1.10 and Chapter 2). The UAl_x-Al fuel is also referred to as aluminide fuel or WPI fuel in this SER. The UMLRR U_3Si_2-Al and WPI UAl_x-Al fuel elements fit interchangeably within the UMLRR core grid support structure (core box), which includes the grid plate and grid box. The grid plate is installed at the bottom of the grid box, whose four sides are enclosed. The top of the grid box is open to the pool, and the bottom connects to an enclosed plenum for coolant flow. Control of the reactor core is by four boron carbide (B_4C) and aluminum control blades and one BORAL servo-controlled regulating rod.

The reactor pool is divided into two interconnected sections: a high-power section (the stall section) and a low-power section (the bulk section). The reactor bridge, core suspension structure, and core are designed to permit the core and its suspension structure to be moved along a rail system mounted on the top of the pool walls. The core grid plate is suspended approximately 8 meters (26.3 feet) below the pool water surface. When the reactor core is positioned in the high-power section and coupled to the coolant system it may be operated in power ranges above 0.1 MWt using forced circulation (there are also connections for coupling the reactor to the coolant system in the low-power section, allowing the reactor to be operated above 0.1 MWt using forced convection in the low-power section, although UML does not typically use this configuration). The coolant system, which is discussed in SAR Chapter 5, is a double loop coolant system that transfers heat from the reactor to the atmosphere via a primary coolant system, heat exchanger, a secondary coolant system, and a cooling tower. For operation in power ranges of 0.1 MWt and below, the reactor may be operated in any location in either pool section, using natural convection cooling.

Surrounding the reactor pool, at the main floor level, is a thick wall of reinforced high-density concrete to provide biological shielding for personnel working in the reactor building. Coolant piping and experimental facilities protrude through the concrete shield structure.

As discussed in SAR Chapter 10, UMLRR experimental facilities include a fast neutron irradiation facility and other irradiation spaces adjacent to the reactor, three beam tubes, a thermal column, gamma ray irradiation facilities (which may utilize nitrogen-16 generated in the reactor coolant, used fuel elements, or a cobalt-60 (Co-60) source stored in the reactor pool), a hot cell for material irradiation and remote handling of samples, and a pneumatic tube system. Figure 1-1, “Reactor Layout,” shows the layout of the UMLRR reactor pool, including experimental facilities.

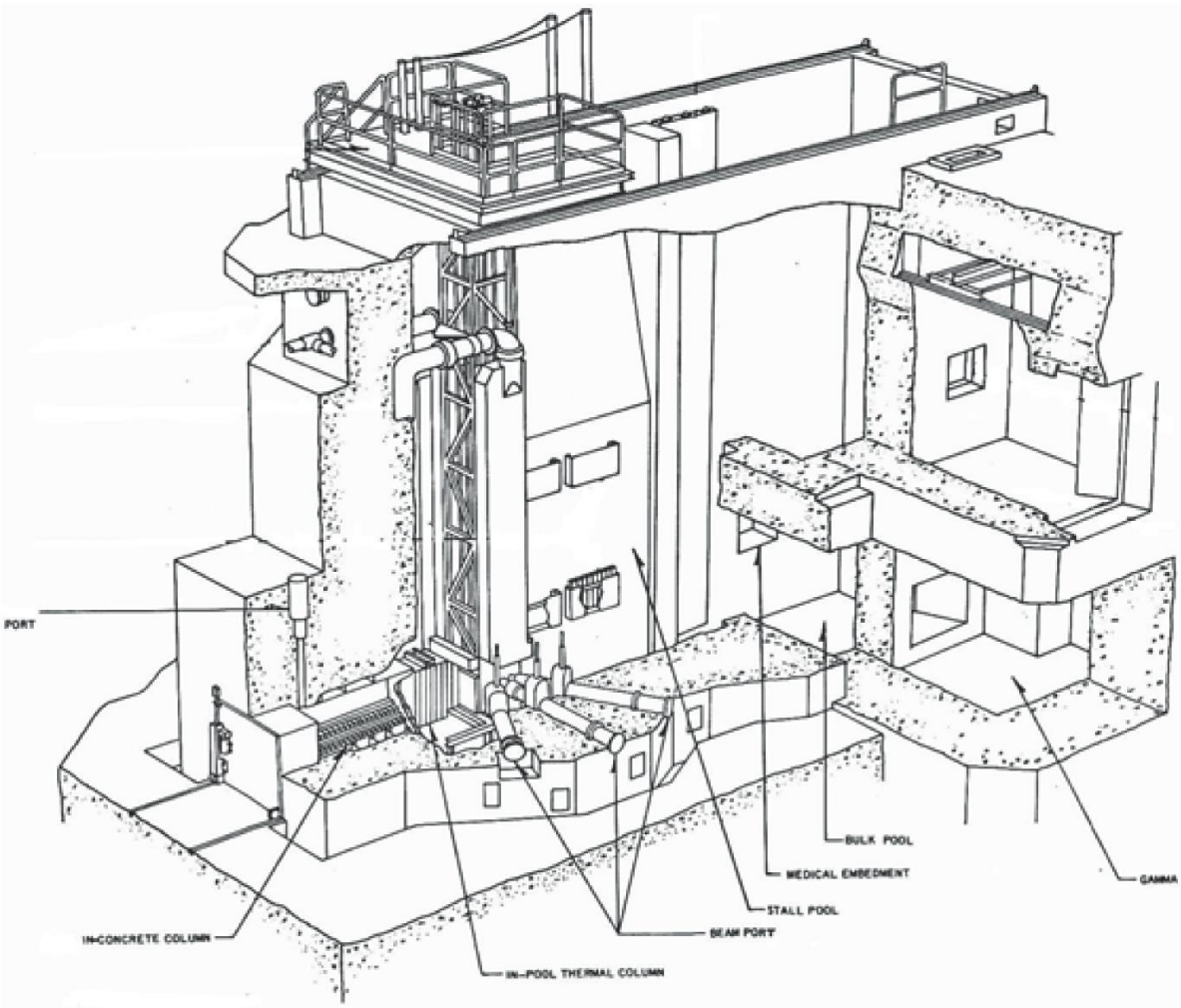

Figure 1-1 Reactor Layout

1.4 Shared Facilities and Equipment

Shared facilities and equipment are described in SAR Section 1.4. Additionally, SAR Chapter 8 provides further information on shared electrical distribution systems.

The UMLRR reactor building has a shared heating and cooling system with the adjacent Pinanski building; the Pinanski building supplies heat and cooling directly into the UMLRR reactor building ventilation system. Hot water for the UMLRR facility is generated by the central gas-fired boiler system for the North Campus of UML.

Electricity is supplied at 13,800 volts from a pole on Riverside Street near the UMLRR, down underground conduits leading to switch and metering gear near the nearby UML power plant, from which point it is distributed to several university buildings. The Pinanski building and UMLRR are fed by a 4160-volt line, through two transformers in the Pinanski building, to two main distribution switchboards in the Pinanski building basement. Both of these switchboards feed areas and equipment both in the Pinanski building and in the reactor building.

The Pinanski building and the UMLRR are also served by a 100-kilowatt natural-gas fired three-phase generator, located in the Pinanski building. This capacity is split between the UMLRR and Pinanski building. The generator can support all operations necessary to shut down and secure the UMLRR during any loss of power events that may occur.

1.5 Comparison with Similar Facilities

SAR Section 1.5 compares the UMLRR with two similar facilities, the Massachusetts Institute of Technology Reactor-II and the Ohio State University Research Reactor. In response to RAI-13.2 (Ref. 23), UML also stated that the UMLRR is similar to the Rhode Island Nuclear Science Center (RINSC) reactor. These four reactors all utilize MTR plate-type fuel and light water moderation and cooling. UMLRR and RINSC are also similar because their open reactor pools, reactor core support structures, and beam ports have identical characteristics, although there are some other structural and core differences between the two facilities, for example, UMLRR's (nominally) 21-element, graphite-reflected core versus RINSC's 14-element, beryllium-reflected core.

As discussed in SER Section 2.2.1, the aluminide MTR fuel that UML has requested NRC approval to use in its core is fuel that was previously used at the now-shutdown (and decommissioned) WPI research reactor, and it is also similar to fuel that was previously used at the now-shutdown (and decommissioned) University of Michigan research reactor.

1.6 Summary of Operations

As discussed in SAR Section 1.6.2, UML has been operating the UMLRR for more than 40 years. As one of the higher flux university research reactors, the UMLRR has significant potential to carry out a wide range of research and educational programs. The UMLRR is used to provide tours, briefings, and training to high school students, college classes, and members of the public from a wide geographical area. Reactor utilization by the local research and industrial community outside UML is fostered and encouraged. The reactor operating schedule makes the reactor available for full power operation during most of the year on a daily (one shift) basis.

Total annual usage of the UMLRR has varied over time. As discussed in SAR Section 1.1, initial criticality was achieved at the UMLRR in January 1975. The UMLRR core was converted from highly enriched uranium (HEU) to low-enriched uranium (LEU) in 2000 (Ref. 36); prior to the conversion, the UMLRR was operated 288.23 megawatt-days (MWD) (Ref. 24). Based on the information in the UMLRR Annual Reports for 2005-2006 through 2019-2020 (Ref. 10), the

annual operation of the UMLRR ranged from a low of 2.28 MWD for the 2005-2006 reporting period, to a high of 7.10 MWD for the 2014-2015 reporting period. Total operation since the HEU to LEU conversion in 2000, through the 2019-2020 reporting period, has been 85.84 MWD.

As part of its review of the LRA, the NRC staff reviewed annual reports of the facility operation submitted by the licensee for 2005-2006 through 2019-2020 (Ref. 10), and inspection reports prepared by the NRC staff from 2006 through 2019 (Ref. 11). The inspection reports reviewed included two violations: (1) a cited Severity Level IV violation issued in 2007 due to the failure of a senior reactor operator to take required requalification examinations or annual operating tests and (2) a non-cited violation in 2009 due to UML's failure to perform a required equipment surveillance within the specified timeframe. These violations were closed by NRC inspections in 2008 (Ref. 11.d) and 2009 (Ref. 11.e), respectively. The annual reports for 2014-2015 through 2016-2017 indicated that the reactor control blades were in the process of being replaced because of minor blistering on the blades. Also, the annual report for 2015-2016 noted that the secondary coolant system piping had been replaced due to corrosion, and the annual report for 2017-2018 and inspection report issued in 2019 noted that some leaks in the reactor pool had developed and been repaired. However, neither the annual reports nor the inspection reports reviewed indicated any significant degradation of fuel element integrity, control blade operability issues, or radiological exposure concerns.

1.7 Compliance with the Nuclear Waste Policy Act of 1982

Section 302(b)(1)(B) of the Nuclear Waste Policy Act of 1982, as amended, 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 entered into an agreement with the Department of Energy (DOE) for the disposal of high-level radioactive waste and spent nuclear fuel that may result from the use of such license. By letter dated May 3, 1983 (Ref. 25), R.L. Morgan (DOE) informed H. Denton (NRC) that DOE determined that universities and other governmental agencies operating non-power reactors entered into contracts with DOE stating that DOE retains title to the fuel and is obligated to take the spent nuclear fuel and high-level radioactive waste for storage or reprocessing. An e-mail, dated December 11, 2020 (Ref. 26), sent from Douglas Morrell (DOE) to Duane Hardesty (NRC), reconfirmed this contractual obligation with respect to the fuel at the UMLRR (DOE Contract No. 78293), valid from March 1, 2009, to December 31, 2025. Additionally, DOE renews these contracts prior to their expiration to ensure that they remain valid. Therefore, by entering into such an agreement with DOE, the licensee satisfied the requirements of the Nuclear Waste Policy Act of 1982, as amended.

1.8 Facility Modifications and History

On April 20, 1965, the AEC issued Construction Permit No. CPRR-87 to the Lowell Technological Institute for the construction of a 1 MWt pool-type reactor on its campus. On December 24, 1974, the AEC issued Facility Operating License No. R-125 to the Lowell Technological Institute. The reactor achieved initial criticality in January 1975.

On October 15, 1975, the NRC issued License Amendment No. 2, which changed the name of the licensee from the Lowell Technological Institute to the University of Lowell. The licensee requested the change to reflect the June 9, 1975, merger of the Lowell Technological Institute with the State College at Lowell to form the combined University of Lowell.

The license was last renewed, for a period of 30 years, by License Amendment No. 9, which was issued to the University of Lowell on November 21, 1985. The NRC staff documented its previous renewal review in NUREG-1139, "Safety Evaluation Report Related to the Renewal of the Operating License for the Training and Research Reactor at the University of Lowell" (Ref. 13). The NRC amended the license five times, by License Amendments No. 10 through 14, since the 1985 renewal.

License Amendment No. 10, issued January 15, 1992 (Ref. 83), consisted of changes to the TSs to have the control blade removal inhibited rather than the regulating rod when the reactor period is less than 15 seconds, and to remove the pool water level channel from being classified as a measuring channel.

License Amendment No. 11, issued February 20, 1992 (Ref. 84), changed the TSs to more clearly define the minimum complement of people required to be present at the UMLRR while the reactor is operating, and also changed the name of the licensee from the University of Lowell to UML to reflect a change in the university's name resulting from a reorganization by the Commonwealth of Massachusetts to put all state universities under one Board of Trustees.

License Amendment No. 12, approved by an Order issued July 31, 1997 (Ref. 37), which authorized the conversion of the UMLRR from HEU fuel to LEU fuel, modified the UMLRR license and TSs to reflect the conversion. The Order was issued in accordance with 10 CFR 50.64, "Limitations on the use of highly enriched uranium (HEU) in domestic non-power reactors," which requires that non-power reactors, such as the UMLRR, convert to LEU fuel under certain conditions. The fuel conversion was completed in 2000 (Ref. 36).

License Amendment No. 13, issued June 2, 2010 (Ref. 38), revised the UMLRR license conditions (LCs) and TSs by increasing the SNM possession limit to allow the receipt and possession (but not the use) of fuel from the WPI research reactor, removing a LC for the possession of HEU that was no longer needed, and correcting wording in the possession limits and TSs.

License Amendment No. 14, issued June 22, 2011 (Ref. 39), revised the UMLRR LCs to allow the receipt and possession of byproduct material and SNM contained in WPI fuel and produced by operation of the WPI fuel in the WPI research reactor (while License Amendment No. 13 had previously authorized the receipt and possession of SNM consisting of up to 5.0 kilograms of contained U-235 enriched to less than 20 percent in the form of WPI MTR-type reactor fuel, it did not specifically authorize receipt and possession of byproduct material and additional SNM produced by prior use of the WPI fuel), and also corrected a typographical error in the LCs.

SAR Table 1-5 (in SAR Section 1.8), as supplemented and updated by UML's response to RAI-7.3 (Ref. 79) and UML's letter dated September 30, 2020 (Ref. 98), lists facility changes since the last renewal for which UML stated that it either conducted a full review in accordance with 10 CFR 50.59 to ensure that no NRC license amendment was required, or performed a screen for 10 CFR 50.59 applicability, but determined that the change did not require a full 10 CFR 50.59 evaluation. UMLRR annual reports (Ref. 10) also discuss facility changes that were evaluated under 10 CFR 50.59 and/or screened for 10 CFR 50.59 applicability. The list of changes in SAR Table 1-5, as supplemented and updated by information in UML's response to RAI-7.3 and UML's letter dated September 30, 2020, and UMLRR annual reports, is reproduced in SER Table 1-1. SER Table 1-1 includes both full 10 CFR 50.59 reviews and screens for 10 CFR 50.59 applicability, as indicated in the table.

SER Table 1-1 includes two changes for which UML performed a 10 CFR 50.59 screen, but has not implemented the change. As noted in UML's response to RAI-7.3 and as identified by the NRC staff during site visits to the UMLRR, although UML evaluated its linear channel replacement in December 2014 and determined that it screened out of 10 CFR 50.59 applicability, UML decided to wait until the NRC reviewed and approved its license renewal SAR, including descriptions of its upgraded instrumentation and control systems which include the new linear channels, before implementing the change. Additionally, as discussed in UML's response to RAI-7.3 and its April 10, 2019, SAR supplement (Ref. 73), UML no longer plans to implement the log N channel replacement (with a General Atomics-manufactured channel) that was the subject of the 10 CFR 50.59 screen in December 2013. Instead, UML proposes to replace its existing log N channel (model NLI-1000) with a wide range logarithmic power/period instrument (model TR-10) manufactured by Thermo Fisher Scientific, and it requested NRC review and approval of this change in conjunction with license renewal. The NRC staff's review and approval of UML's instrumentation and control system upgrades (linear and log N channel replacements) proposed as part of the LRA is documented in Chapter 3 of this SER.

Table 1-1 Facility Changes for Which UML Did Not Request an NRC License Amendment

Year	Description	Type	Implemented
2019-2020	Reactor Building Air Handler Unit Replacement	Screen	Yes
2019	Procedure FP-4 for Movement of Objects by Crane	Screen	Yes
2016	Control Room Annunciator Panel Replacement	Screen	Yes
2015	Center Flux Trap Replacement	Screen	Yes
2015	Control Blade Replacement	Screen	Partial[a]
2014	Linear Channel Replacement	Screen	No
2014	Addition of Panel Indicators	Screen	Yes
2013	Log-N Channel Replacement	Screen	No
2013	Beamport Irradiation Facility Enclosure	Screen	Yes
2012	Stack Monitor Replacement	Screen	Yes
2012	Cooling Tower Replacement	Screen	Yes
2012	Chart Recorder Replacement	Full	Yes
2011	Pneumatic Tube Control System Upgrade	Screen	Yes
2010	Upgrades to Drives Control System	Screen	Yes
2010	Reactor Test Using Downcomer Flow Mode	Screen	Yes
2008	Secondary Cooling System Remote Control	Full	Yes
2003	Drives Control System	Full	Yes
2002	Clean-up and Make-up System Upgrade	Full	Yes
2001	Upgrade of UMLRR Process Control Cabinet	Full	Yes
2001	Installation of Ex-Core Fast Neutron Irradiation Facility	Full	Yes
2000	Radiation Monitoring System Upgrade	Full	Yes
1998	Power Detector Mechanical Height Adjusters	Full	Yes
1997	Nuclear Instrumentation and Radiation Monitoring System Upgrades	Full	Partial[b]

[a] See SER Section 2.2.2.

[b] The nuclear instrumentation upgrade associated with a 10 CFR 50.59 review conducted by UML in 1997 was implemented, but the radiation monitoring system upgrade associated with this same review was deferred until after completion of a new 10 CFR 50.59 review in 2000 (see UML's response to RAI-7.3 and SER Sections 3.6.1 and 3.7.1).

The NRC staff reviewed the list of changes in Table 1-1, as well as information related to the changes in the SAR, as supplemented, and the discussions of changes in annual reports of UMLRR operation submitted by the licensee for 2005-2006 through 2019-2020 (Ref. 10). The modifications to the UMLRR mostly involved technological upgrades to instrumentation, and minor changes to the existing design that either enhanced its capability or improved reactor operations, for which UML did not submit a license amendment request. All of these modifications were subjected to screening and/or evaluation under 10 CFR 50.59 to ensure that there was no adverse impact on the safety of the UMLRR. The NRC staff also reviewed the UMLRR inspection reports from 2006 through 2019 (Ref. 11), and noted that there were no significant issues related to UML's compliance with 10 CFR 50.59. Although there was an inspector follow-up item (IFI) in 2015 related to clarifying 10 CFR 50.59 applicability in UML's procedure dealing with procedural changes, the IFI was resolved and closed during a subsequent inspection in 2016 (in its response to RAI-7.3.a, UML stated that although it did not conduct formal 10 CFR 50.59 evaluations of procedure changes prior to 2016, UMLRR procedure changes were reviewed by the Reactor Safety Subcommittee as required by TSs). Therefore, based on the above, the NRC staff concludes, in general, that the changes that UML has implemented without a license amendment appear to be reasonable and that the licensing actions taken over the years appear to be appropriate.

In its LRA, as supplemented, UML requested NRC approval to make certain changes to the facility or facility operations, LCs, and/or TSs, coincident with the renewal of the license. Four particularly significant changes are summarized below:

- UML requested that the license's SNM and byproduct material possession limits be amended to allow possession of plutonium-beryllium (Pu-Be) sources and byproduct material with atomic numbers 3 through 83 (for checks, calibrations, and characterizations of radiation monitoring instruments), HEU in the form of fission chamber linings, and SNM for use in experiments, under the reactor license (as opposed to under a different UML license, as it is currently possessed). The NRC staff review of these proposed changes is discussed in Section 1.10 of this SER.

- UML requested that the license's LCs be amended to allow the use of the WPI MTR-type reactor fuel that is currently possessed under the license. The NRC staff review of these proposed changes is discussed in Section 1.10 and Chapter 2 of this SER.

- UML requested approval to make modifications to its instrumentation and control systems, including power monitoring channel replacements, and radiation monitoring system changes (addition of ratemeters and audible alarms). The NRC staff review of these equipment modifications is discussed in Chapter 3 and Section 4.1.4 of this SER.

- UML requested approval to change the designation of the UMLRR reactor building from a containment to a confinement and to make associated changes to the operation of its ventilation system, and also requested associated TS changes. The NRC staff review of these changes is primarily discussed in Section 6.3.4 of this SER (the NRC staff review of some associated TS changes is also discussed in SER Sections 6.3.5 and 6.4).

In addition to the proposed changes listed above, UML requested to replace its current (pre-renewal) TSs in their entirety, including the TS specification, applicability, and objective sections. (UML also provided replacement TS bases for the proposed TSs. Although the TS bases are included on the replacement TS pages as a reference, pursuant to 10 CFR

50.36(a)(1), the TS bases are not part of the TSs.) The proposed changes to the TSs include updates in format, content, and structure to make the TSs more consistent with the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007, and with TSs for other recently-renewed research reactor licenses. The proposed TSs for the renewed UMLRR license are discussed and found acceptable in Chapters 2, 4, and 6 of this SER. Notable changes between the current TSs (i.e., the pre-renewal TSs) and the renewed license TSs are also discussed in these SER chapters.

1.9 Financial Considerations

By letter dated October 20, 2015 (Ref. **Error! Reference source not found.**), UML submitted the LRA to the NRC requesting renewal of the facility operating license for the UMLRR, which is located at the Radiation Laboratory on the UML campus in Lowell, Massachusetts, for an additional twenty years. Chapter 15 of the SAR submitted as part of the LRA, as supplemented by UML's response to RAI-FR-1 (Ref. 3), UML's response to RAI-15.1 (Ref. 79), UML's letter dated February 1, 2018 (Ref. 44), and UML's letter dated January 30, 2021 (Ref. 99), includes financial information supporting the LRA.

1.9.1 Financial Ability to Operate the Reactor

The financial requirements for non-electric utility nuclear reactor licensees are in 10 CFR 50.33(f):

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

UML does not qualify as an "electric utility," as defined in 10 CFR 50.2, "Definitions"; therefore, the LRA must include the financial information that is required in an application for an initial license. Accordingly, UML must demonstrate that it meets the financial qualifications requirements of 10 CFR 50.33(f) and is subject to a full financial qualifications review by the NRC. As required by 10 CFR 50.33(f)(2), UML must provide information to demonstrate that it possesses or has reasonable assurance of obtaining the funds necessary to cover estimated operation costs for the period of the license. UML must submit estimates for total annual operating costs for each of the first five years of facility operations from the expected license renewal date and indicate the source(s) of funds to cover those costs. The guidance described in NUREG-1537, as it pertains to the NRC staff's review of UML's financial qualifications, is consistent with the 10 CFR 50.33(f)(2) requirement.

In its letter dated January 30, 2021 (Ref. 99), UML submitted updated projected operating costs for the UMLRR, for each of the fiscal years (FY) 2020 through FY 2025. The operating costs for the UMLRR are projected to be $1,284,000 in FY 2020, $1,323,000 in FY 2021, $1,362,000 in FY 2022, $1,403,000 in FY 2023, $1,445,000 in FY 2024, and $1,489,000 in FY 2025. According to the LRA, as supplemented, UMLRR expenses are broken down by category of spending and include salaries, wages, travel, supplies, and equipment. Based on the applicant's submittals, there are two categories of salaries, each dependent on a different source of funds: University-funded salaries are part of the UML operating budget and contract

salaries and student wages are funded by facility users, including commercial users. The university-funded salaries figure includes fringe (direct) and overhead (indirect) costs; the contract salaries and student wages include only direct costs.

UML states that revenues are derived from two sources. The university operating budget provides funding for the direct costs of three full-time staff. Likewise, "user costs," for use of the reactor and gamma facilities provide the remainder of the revenues. While the annual revenue derived from users can vary, the university operating budget provides a fixed source of revenue for the staff and services necessary to operate the UMLRR. As part of its review, the NRC staff considered guidance in NUREG-1537, as well as the projected operating costs and associated funding for similar research reactor facilities. The NRC staff found UML's estimates and sources of funds to be reasonable.

The UMLRR is currently licensed under Section 104.c of the AEA, 42 U.S.C. § 2134(c), as a facility that is useful in the conduct of research and development activities. Pursuant to 10 CFR 50.21(c) and 50.22, if a facility is to be licensed under Section 104.c as a non-commercial, non-power reactor facility that is useful in the conduct of research and development activities, then the facility is to be used so that not more than 50 percent of the annual cost of owning and operating the facility is devoted to the production of materials, products, or energy for sale or commercial distribution, or to the sale of services, other than research and development or education or training. Additionally, pursuant to Section 104.c of the AEA, if a facility is to be licensed under Section 104.c, then the licensee shall recover not more than 75 percent of the annual costs to the licensee of owning and operating the facility through sales of nonenergy services, energy, or both, other than research and development or education and training, of which not more than 50 percent may be through sales of energy. The UMLRR facility was originally licensed by the AEC in 1974, and re-licensed by the NRC in 1985, as a non-commercial facility, and continues as an academic, non-commercial facility. According to the LRA, as supplemented, the UMLRR is used for research and education programs, and providing tours, briefings, and training to high school students, college classes, and the public from a large geographical area. In its response to RAI-FR-1 (Ref. 3), UML stated that the annual cost of conducting the commercial activities at the UMLRR is less than 50 percent of the annual cost of owning and operating the UMLRR facility. In its response to RAI-15.1 (Ref. 79), UML also stated that 75 percent or less of the annual costs of owning and operating the UMLRR are recovered through sales of nonenergy services, energy, or both, other than research and development or education and training, and 50 percent or less of the annual costs of owning and operating the UMLRR are recovered from sales of energy. Because UML confirmed in the LRA, as supplemented, that the UMLRR is used so that it meets the statutory requirements in Section 104.c of the AEA and the regulatory requirements in 10 CFR 50.21(c), the NRC staff concludes that the renewed license can be issued pursuant to Section 104.c of the AEA.

Based on the above discussion, the NRC staff finds that UML provided the appropriate information for operating costs and has also demonstrated reasonable assurance for obtaining the necessary funds to cover these costs for the period of the renewed facility operating license. Accordingly, the NRC staff finds that UML meets the acceptance criteria in NUREG-1537 and the financial qualifications requirements in 10 CFR 50.33(f).

1.9.2 Financial Ability to Decommission the Facility

Pursuant to 10 CFR 50.33(k), the NRC requires that an applicant for an operating license for a utilization facility submit 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). Pursuant to 10 CFR 50.75(d)(2), the report must contain a cost estimate for decommissioning the facility, an indication of the funding method(s) to be used to provide funding assurance for decommissioning, and a description of the means of adjusting the cost estimate and associated funding level periodically over the life of the facility. The acceptable methods for providing financial assurance for decommissioning are described in 10 CFR 50.75(e)(1). The NRC staff applied guidance in NUREG-1537 to complete its review of the LRA as it pertains to financial assurance for decommissioning. The original UMLRR decommissioning cost estimate was developed using the methodology of NUREG/CR-1756, "Technology, Safety and Costs of Decommissioning Reference Research and Test Reactors" (Ref. 54), for a reference test reactor using the immediate decontamination (DECON) option. In its letter dated January 30, 2021 (Ref. 99), UML provided an updated decommissioning cost estimate of $5.2 million in 2020 dollars. The cost estimate included itemized costs for labor, equipment service, indirect costs, radioactive waste volume reduction, packaging, shipping, burial activities, and a 25-percent contingency factor. The estimate did not include the cost associated with nuclear fuel removal and transport to a DOE facility. The estimate also did not include the cost of Co-60 source removal and disposal, but the NRC staff notes that the decommissioning cost associated with 100,000 Curies (Ci) of Co-60 sources is small relative to the overall UMLRR decommissioning cost estimate, including the contingency factor. UML stated that, for estimating the future cost of decommissioning the facility, the decommissioning cost estimate shall be adjusted once every 5 years using calculations based on historical inflation data. Based on the NRC staff's review of the LRA using guidance from NUREG-1537 and NUREG/CR-1756, the NRC staff concludes that the decommissioning approach and decommissioning cost estimate submitted for the UMLRR are reasonable.

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

In its letter dated January 30, 2021 (Ref. 99), UML provided an updated SOI, dated January 15, 2021, stating that the Vice Chancellor for Finance and Operations will "request that funds be made available when necessary to decommission the University [of] Massachusetts Lowell Research Reactor, located in Lowell, Massachusetts." Further, the signator states that he will "request and obtain these funds sufficiently in advance of decommissioning to prevent delay of required activities." The updated SOI is signed by UML's Vice Chancellor for Finance and Operations.

To support the SOI and UML's qualifications to use an SOI, the LRA stated that UML is a non-profit educational institution and a part of the government of the Commonwealth of Massachusetts, and included documentation that corroborates this statement. The LRA also provided information supporting UML's representations that the decommissioning funding obligations of UML are backed by the full faith and credit of the Commonwealth of Massachusetts.

The NRC staff reviewed UML's information on decommissioning funding assurance as described above and finds that UML is a State government licensee under 10 CFR 50.75(e)(1)(iv), that the SOI is acceptable, that the decommissioning cost estimate is

reasonable, and that UML's means of adjusting the cost estimate and associated funding level periodically over the life of the facility is reasonable.

Based on the above, the NRC staff concludes that funds will be made available to decommission the facility and that the financial status of the licensee regarding decommissioning costs conforms to the guidance in NUREG-1537 and meets the requirements of 10 CFR 50.33(k) and 10 CFR 50.75. Therefore, the NRC staff concludes that there is reasonable assurance that funds will be provided for decommissioning the facility.

1.9.3 Foreign Ownership, Control, or Domination

Section 104.d of the AEA, as amended, prohibits the NRC from issuing a license under Section 104 of the AEA to "any corporation or other entity if the Commission knows or has reason to believe it is owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government." The NRC regulations at 10 CFR 50.33(d) and 10 CFR 50.38 contain language to implement this prohibition. The UMLRR is owned and operated by UML, an entity of the Commonwealth of Massachusetts. According to the LRA, UML is a Commonwealth of Massachusetts government entity and is not owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government. Based on the above, the NRC staff concludes that the UMLRR is not owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government.

1.9.4 Nuclear Indemnity and Insurance

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 NRC staff notes that UML has a current indemnity agreement for the UMLRR with the Commission that will not expire until the license terminates. Therefore, UML will continue to be a party to the indemnity agreement following issuance of the renewed operating license. As required by Subpart D, "Provisions Applicable Only to Nonprofit Education Institutions," of 10 CFR Part 140, the Commission will indemnify UML for any claims arising under its indemnity agreement pursuant to Appendix E, "Form of indemnity agreement with nonprofit educational institutions," of 10 CFR 140.95 above $250,000 and up to $500 million. In accordance with Subpart B, "Provisions Applicable Only to Applicants and Licensees Other Than Federal Agencies and Nonprofit Educational Institutions," of 10 CFR Part 140, UML, as a nonprofit educational institution, is not required to provide nuclear liability insurance. Finally, as a non-power reactor licensee, UML is not required to maintain property insurance pursuant to 10 CFR 50.54(w).

In conjunction with this renewal, the NRC is issuing Amendment No. 11 to Indemnity Agreement No. E-54 between the UML and the NRC to update the agreement and to reflect changes since Amendment No. 10 to Indemnity Agreement No. E-54, which was effective on December 18, 1989 (Ref. 78), including changes associated with this renewal. Specifically, Amendment No. 11 to Indemnity Agreement No. E-54 updates the licensee's name from "University of Lowell" to "University of Massachusetts Lowell" for consistency with the current licensee name as changed by License Amendment No. 11 to Facility Operating License No. R-125 for the UMLRR on February 20, 1992. Amendment No. 11 to Indemnity Agreement No. E-54 also updates the licensee address to reflect that "1 Textile Avenue" is now "1 University Avenue," and removes SNM license No. SNM-1220 which has not been applicable for Indemnity Agreement No. E-54 since 1974. Additionally, Amendment No. 11 to Indemnity Agreement No. E-54 updates the "Location" subject to the agreement to be consistent with the licensed boundary for Facility

Operating License No. R-125, which is reflected in UMLRR TS 5.1 for the renewed license (see SER Section 6.5).

1.9.5 Financial Considerations Conclusions

Based on its review as discussed above, the NRC staff concludes that there is reasonable assurance that the necessary funds will be available to support the continued safe operation of the UMLRR and, when necessary, to shut down the facility and carryout the decommissioning activities per guidance in NUREG-1537 and NUREG/CR-1756. In addition, the NRC staff concludes that there are no foreign ownership, control, or domination issues or insurance issues that would preclude the issuance of the renewed license.

1.10 Facility Operating License Possession Limits and License Changes

The renewal of Facility Operating License No. R-125 for the UMLRR authorizes the receipt, possession, and use of SNM and byproduct materials. The SNM that will be possessed under the renewed license is:

- LEU in the form of MTR-type reactor fuel (silicide fuel and WPI aluminide fuel);
- Plutonium (Pu) produced in the reactor fuel, neutron detectors (fission chamber linings), and fueled experiments;
- Uranium of any enrichment in fission plates, foils, and solutions;
- HEU in the form of neutron detectors (fission chamber linings); and
- Pu in the form of sealed Pu-Be neutron sources.

The byproduct material that will be possessed under the renewed license is:

- Activation and fission products in the reactor fuel;
- Activation and fission products in fission chamber linings;
- Activation and fission products in experiments;
- Activation products in reactor components;
- Americium in a sealed americium-beryllium (Am-Be) neutron source;
- Activation and fission products in sealed Pu-Be and Am-Be neutron sources;
- Activation products in a sealed antimony-beryllium (Sb-Be) neutron source;
- Sealed Co-60 irradiation sources; and
- Sealed and/or plated sources used for checks, calibrations, and characterizations of radiation monitoring instruments.

TS 5.1, which is discussed and found acceptable in SER Section 6.5, defines the area controlled under Facility Operating License No. R-125 as the reactor building and the attached three-story Pinanski building. UML may possess and use radioactive materials within this area under Facility Operating License No. R-125. In addition, TS 5.1 requires that the reactor building be the minimum restricted area (as defined in 10 CFR Part 20) for the reactor license. The NRC staff reviewed the UMLRR inspection reports from 2006 through 2019 (Ref. 11), and finds that UML has shown that it has procedures and equipment to safely and securely handle licensed material within the licensed and restricted areas.

License Changes Requested by UML

In its letter dated October 20, 2015 (Ref. **Error! Reference source not found.**) and in SAR Section 9.5, as supplemented by its response to RAI-1.1 (Ref. 43) and its letters dated September 30, 2020 (Ref. 98), and April 5, 2021 (Ref. 90), UML requested changes to the LCs, including possession limits, for the UMLRR as part of license renewal.

Current LC 2.B.(2) allows UML to receive, possess, and use at any one time up to 6.0 kilograms of contained U-235 in LEU MTR-type reactor fuel in conjunction with the operation of the reactor. Current LCs 2.B.(4) and 2.B.(5) allow UML to receive and possess, but not use, 5.0 kilograms of contained U-235 in WPI LEU MTR-type reactor fuel, and SNM and byproduct material contained in the fuel and generated during operation of the WPI reactor. UML requested deletion of current LCs 2.B.(4) and 2.B.(5), and revision of current LC 2.B.(2) to allow receipt, possession, and use at any one time of up to 11.0 kilograms (i.e., the 6.0 kilograms in silicide fuel, plus 5.0 kilograms in WPI aluminide fuel) of contained U-235 in LEU MTR-type reactor fuel in conjunction with the operation of the reactor. UML requested these changes to allow it to operate the UMLRR using its WPI fuel, which was not previously authorized by its LCs. SAR Chapters 4 and 13, as supplemented, provide analyses demonstrating that the WPI fuel, on its own or in conjunction with silicide fuel, can be safely used in the UMLRR. These analyses are reviewed and found acceptable in Chapters 2 and 5 of this SER. Therefore, the NRC staff finds that the requested changes in the LCs to allow reactor operation with WPI fuel are reasonable, and are supported by analyses showing that the UMLRR can be safety operated with the WPI fuel.

Current LC 2.B.(3) allows UML "to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the reactor, and to receive, possess, and use up to 5 Ci Am-Be and 10 Ci Sb-Be neutron sources in connection with operation of the reactor, and to receive, possess, use and transfer byproduct materials activated in reactors other than the University of Massachusetts Lowell reactor, in the form of Cobalt-60, in quantities not to exceed 1,500,000 curies at any time." UML requested that current LC 2.B.(3) be revised to allow it to additionally receive, possess, and use, in connection with operation of the UMLRR facility, up to 50 grams of Pu contained in Pu-Be neutron sources; up to 100 grams of contained U-235 of any enrichment in the form of neutron detectors (fission chamber linings), fission plates, foils, and solutions; and up to 20 millicuries (mCi) per radionuclide, and 50 mCi total, of radionuclides with atomic numbers 3 through 83. UML also requested revision of current LC 2.B.(3) to reduce the Co-60 possession limit from 1,500,000 Ci to 100,000 Ci. Additionally, although current LC 2.B.(3) does not permit UML to separate any byproduct materials or SNM produced by the operation of the reactor, UML requested that its LCs for a renewed license allow it to separate byproduct material produced in non-fueled experiments.

In SAR Section 9.5, UML stated that the Pu-Be sources, fission chamber linings, and radionuclides with atomic numbers 3 through 83 that it requests be added to the reactor license are currently possessed under UML's separate Commonwealth of Massachusetts byproduct material or SNM licenses. UML stated that the Pu-Be sources and atomic number 3 through 83 material are for the purposes of checks, calibrations, and characterizations of radiation monitoring instruments, and the fission chamber linings are used in reactor power level monitoring instrumentation. In response to RAI-1.1(a), UML additionally stated that the atomic number 3 through 83 material is in the form of sealed and/or plated sources, and that these sources will only be used under the reactor operating license and for reactor-related purposes. SAR Sections 10.2.3.1 and 11.1.1 state that UML's Co-60 and neutron (Am-Be, Sb-Be, and Pu-Be) sources are sealed sources. In its letter dated April 5, 2021, UML stated that the

uranium-235 contained in uranium of any enrichment that it proposes to add to the reactor license includes not only the uranium in fission chamber linings, but also additional uranium in the form of fission plates, foils, and solutions, and the additional uranium is for use in education and research activities at the facility.

In its letter dated October 20, 2015, UML stated that it requests a decrease in its Co-60 possession limit in current LC 2.B.(3) because the current limit exceeds the amount needed for its ongoing research and development programs.

The NRC staff reviewed UML's requests for possession limit changes. The NRC staff finds that the materials which UML has requested to add to the reactor license are materials that will be used solely in connection with activities conducted under NRC Facility Operating License No. R-125 for the UMLRR (including operation of the gamma irradiation facilities as well as reactor operation), and that it is therefore appropriate for these materials to be on the reactor license. The NRC staff also finds that the additional quantities of material are reasonable given the purposes for which they will be used, that the additional materials do not change the security requirements for the facility (in accordance with 10 CFR Parts 37 and 73), and that the additional materials do not represent an increased risk to public, worker, or facility safety (as stated in SAR Section 9.5, some of the materials being added to the reactor license have already normally been stored within the reactor building). The NRC staff finds that UML's possession of uranium in the form of fission plates, foils, and solutions under the UMLRR license provides material that may be used in fueled experiments, which UML is permitted to conduct in accordance with TS 3.7.2(5), which is discussed and found acceptable in SER Section 6.3.7. The NRC staff additionally finds that the lower Co-60 possession limit is appropriate because it is a decrease in the possession limit and better aligns the possession limit with the quantity of material needed for facility operations.

Given the location of the Co-60 sources in the reactor pool, and the potential for the sources to affect reactor operation, the NRC staff finds that it is appropriate for the Co-60 sources to continue to be controlled under NRC Facility Operating License No. R-125 for the UMLRR. As documented in a letter and NRC safety evaluation dated February 5, 1998 (Ref. 70), the NRC previously determined that, because UML's Co-60 sources are located in a research reactor pool, and the sources and gamma cave are part of a research reactor facility, the regulations in 10 CFR Part 36, "Licenses and Radiation Safety Requirements for Irradiators," do not apply for the UMLRR.

Regarding UML's request that its renewed license allow it to separate byproduct material produced in non-fueled reactor experiments, the NRC staff finds that such an allowance is reasonable and appropriate given the types of activities typically conducted at research reactors, and is consistent with LCs for similar research reactor facilities. Experiments at the UMLRR, including non-fueled reactor experiments involving separation of byproduct material, are subject to the review and approval (by the UMLRR Reactor Safety Subcommittee and Reactor Supervisor, respectively) requirements of TS 6.5, which is discussed and found acceptable in SER Section 6.6.5. Furthermore, all UMLRR experiments, including non-fueled reactor experiments involving separation of byproduct material, are required to be designed such that a credible failure of the experiment shall not result in releases or exposures in excess of 10 CFR Part 20 limits, in accordance with TS 3.7.2(1), which is discussed and found acceptable in SER Section 6.3.7.

As discussed above, the NRC staff finds that UML's requested LC changes are reasonable, appropriate, and supported by SAR analyses, as applicable, and that UML has historically

demonstrated that it has procedures and equipment to safely and securely handle licensed material within the restricted area. Therefore, the NRC staff concludes that the requested authorization to use WPI fuel, the requested possession limit changes, and the requested allowance to separate byproduct material produced in non-fueled experiments, for the renewed license are acceptable. (The NRC staff notes that, in the renewed license, the NRC staff made editorial and format changes, as discussed below; these changes included renumbering and reorganization of LCs discussed above.)

UML Request for Delayed Implementation of Renewed License and TSs

In its letter dated September 30, 2020 (Ref. 98), UML requested that its renewed license be effective upon issuance, but implemented within 60 days of issuance. UML requested the delayed implementation to allow it time to make facility and procedure changes, and to conduct operator training, consistent with the renewed license and TSs. The NRC staff finds that UML's request is reasonable given the scope of the changes associated with the renewal (see SER Section 1.8), and given the reissuance of the entire license, including TSs. The NRC staff also finds that delayed implementation of the renewed license and TSs does not pose an undue risk to public health and safety because the licensee will be required to meet all of the conditions in the renewed license, including TSs, within a reasonable period following issuance, and must meet its current LCs and TSs in the interim. The NRC staff further finds that the delayed implementation of the renewed license gives UML appropriate scheduling flexibility to implement changes associated with the renewal, and helps minimize impacts on reactor operation. Therefore, based on the above, the NRC staff concludes that UML's request to implement its renewed license within 60 days of issuance is acceptable.

License Changes Made by the NRC Staff

Although current LC 2.B.(5) allows UML to receive and possess, but not use or separate, byproduct material and SNM produced during operation of the WPI reactor, the NRC staff noted that UML's requested revised possession limits for the renewed license did not include an allowance to possess (or use) this byproduct material and SNM. Given that the WPI fuel still contains byproduct material and SNM produced during operation of the WPI reactor, and given that the WPI fuel is now permitted to be used in the UMLRR, the NRC staff added language to the LCs of the renewed license to allow UML to receive, possess, and use, but not separate, byproduct material and SNM produced during operation of the WPI reactor, in addition to byproduct material and SNM produced during operation of the UMLRR.

For the byproduct material and SNM produced during UMLRR operation, the NRC staff worded the renewed license to allow receipt, possession, and use (but not separation, except for byproduct material produced in non-fueled experiment) of byproduct material and SNM produced during operation of the "facility." The language used is consistent with renewed LC 2.A, which indicates that the "facility" in Facility Operating License No. R-125 refers to the UMLRR.

The NRC staff also worded the renewed LCs to specify that the Co-60 and neutron (Am-Be, Sb-Be, and Pu-Be) sources are sealed (encapsulated), and that the atomic number 3 through 83 material is in the form of sealed and/or plated sources, consistent with the descriptions in the SAR, as supplemented. Also consistent with descriptions in the SAR, as supplemented, stating that UML possesses one of each type of source, the NRC staff worded the renewed LCs to clarify that one each of the Am-Be and Sb-Be sources may be possessed.

Current LC 2.B.(3) includes an allowance for UML to “transfer” (in addition to receive, possess, and use) byproduct materials activated in reactors other than the UMLRR, in the form of Co-60. UML’s requested possession limits for the renewed license also include the “transfer” language. However, the NRC staff notes that any NRC licensee may transfer licensed material it possesses to another authorized user (i.e., another NRC or agreement state licensee authorized to receive and possess the material), in accordance with NRC regulations. Therefore, because it is not necessary for the license to explicitly state that licensed material may be transferred, and consistent with other recently issued research reactor licenses, the NRC staff deleted the unnecessary “transfer” language from the Co-60 LC (LC 2.B.3.c in the renewed license).

Consistent with other recently issued research reactor licenses, and to enhance the clarity of the LCs, the NRC staff worded all of the renewed possession limit LCs to state that radioactive material under the facility license may be received, possessed, and used “in connection with the operation of the facility” (UML had also requested that some of the renewal LCs include this language). The NRC staff notes that it considers activities that may be conducted at the UMLRR under the reactor license, but which may not necessarily involve reactor operation (e.g., Co-60 irradiations), to still be operation of the “facility,” as defined in renewal LC 2.A, because equipment such as the Co-60 sources and gamma cave are part of the UMLRR and controlled under the reactor license.

Current LC 2.C.(3) requires UML to fully implement and maintain its PSP, and states that the PSP contains information withheld from public disclosure under 10 CFR 2.790. In January 2004 (69 FR 2236), the NRC renumbered 10 CFR 2.790 as 10 CFR 2.390. However, because UML is licensed to operate a research reactor, and to possess “special nuclear material of moderate strategic significance” (i.e., a Category II quantity of SNM), as defined in 10 CFR 73.2, 10 CFR 73.21 requires UML to protect the types of information contained in the UMLRR PSP as Safeguards Information – Modified Handling (SGI-M). The UMLRR PSP, Revision 9, submitted by letter dated September 13, 2017 (Ref. 5), which, as discussed in SER Section 1.1.3, the NRC staff reviewed and found acceptable as part of the review of the LRA, is designated as SGI-M. Therefore, for the renewal LCs, the NRC staff worded LC 2.C.(3) to state that the UMLRR PSP consists of documents withheld from public disclosure pursuant to 10 CFR 73.21.

Current LC 2.C.(4) states that UML shall “submit a startup test report within six months of the initial criticality with low-enriched uranium reactor fuel in accordance with Amendment No. 12….” The NRC staff notes that the UMLRR conversion to LEU was completed in 2000, and UML sent the required report to the NRC by letter dated April 9, 2001 (Ref. 36). Therefore, the NRC staff finds that current LC 2.C.(4) is no longer applicable, and this LC is deleted.

In addition to the above, the NRC staff made editorial and format changes to the LCs (including renumbering and reorganization of LCs) to make them easier to read and understand and consistent with other recently issued research reactor licenses. By e-mail dated April 6, 2021 (Ref. 91), the NRC staff provided UML with a draft of the LCs for the renewed license to allow UML to review the draft LCs and by letter dated April 20, 2021 (Ref. 92), UML stated that the renewed license LCs are acceptable. By e-mail dated January 27, 2022, the NRC staff provided UML with an updated draft of the LCs with additional minor editorial changes, and UML acknowledged this by e-mail also dated January 27, 2022 (Ref. 109).

2. REACTOR DESCRIPTION

2.1 Summary Description

Safety analysis report (SAR) Chapter 4, as supplemented by responses to requests for additional information (RAIs), provides information describing the principal features, operating characteristics, and parameters of the University of Massachusetts Lowell (UML) Research Reactor (UMLRR). UML stated that the analysis in SAR Chapter 4 shows that the reactor is conservatively designed for safe operation and shutdown under all credible operating conditions. The information in SAR Chapter 4 also provides the design bases for many systems, subsystems, and functions discussed elsewhere in the SAR, and for many of the technical specifications (TSs).

As discussed in SAR Sections 1.3, 4.1, 4.2, 4.3, and 4.5, the UMLRR is a water moderated and cooled, open-pool type reactor that uses a flat-plate (materials test reactor (MTR)-type) fuel element design. The UMLRR is licensed to operate at power levels up to 1 megawatt-thermal (MWt). The fueled core region is reflected primarily by a combination of graphite and water reflector elements. The water reflector elements are also referred to in the SAR and this safety evaluation report (SER) as water-filled radiation baskets, radiation baskets, or irradiation baskets, and in addition to serving as neutron reflectors, they can also contain experiments.

The UMLRR core was converted from highly enriched uranium (HEU) to low-enriched uranium (LEU) in 2000 (Ref. 36). The current UMLRR fuel is aluminum-clad uranium-silicide (U_3Si_2-Al), which is also referred to as silicide fuel in this SER. As discussed in SER Section 1.10, UML requested U.S. Nuclear Regulatory Commission (NRC, the Commission) approval to operate the UMLRR using aluminum-clad uranium-aluminide (UAl_x-Al) fuel, also referred to in this SER as aluminide fuel, which UML previously obtained from the Worcester Polytechnic Institute (WPI) following the permanent shutdown of the WPI research reactor. Facility Operating License No. R-125 currently allows UML to possess, but not to use, the aluminide fuel. The aluminide fuel was previously slightly irradiated in the WPI reactor. The current silicide and WPI aluminide fuel elements are similar in overall size and shape, so both types of element fit interchangeably within the UMLRR core grid support structure (core box), which includes the grid plate and grid box. Differences between the fuel types, other than the material composition of the fuel matrix (i.e., silicide versus aluminide), are differences in uranium-235 (U-235) loading, differences in number of fuel plates per element, differences in fuel matrix and plate thickness, and other small differences (see SER Section 2.2.1 for details). UML performed safety analyses including both types of fuel design.

The core grid plate is installed at the bottom of the grid box, whose four sides are enclosed. The grid plate is a 9 by 7 grid, with the four corner positions occupied by the core suspension frame corner posts. The top of the grid box is open to the pool, and the bottom connects to an enclosed plenum for coolant flow. The complete core, including the core box and its contained fuel elements, reflector elements, irradiation baskets, etc., the control blades, and the regulating rod, is referred to in the SAR and in this SER as the core assembly or grid box assembly. The core box, containing some other parts of the complete core assembly, is illustrated in SAR Figure 4-1, which is reproduced below as Figure 2-1.

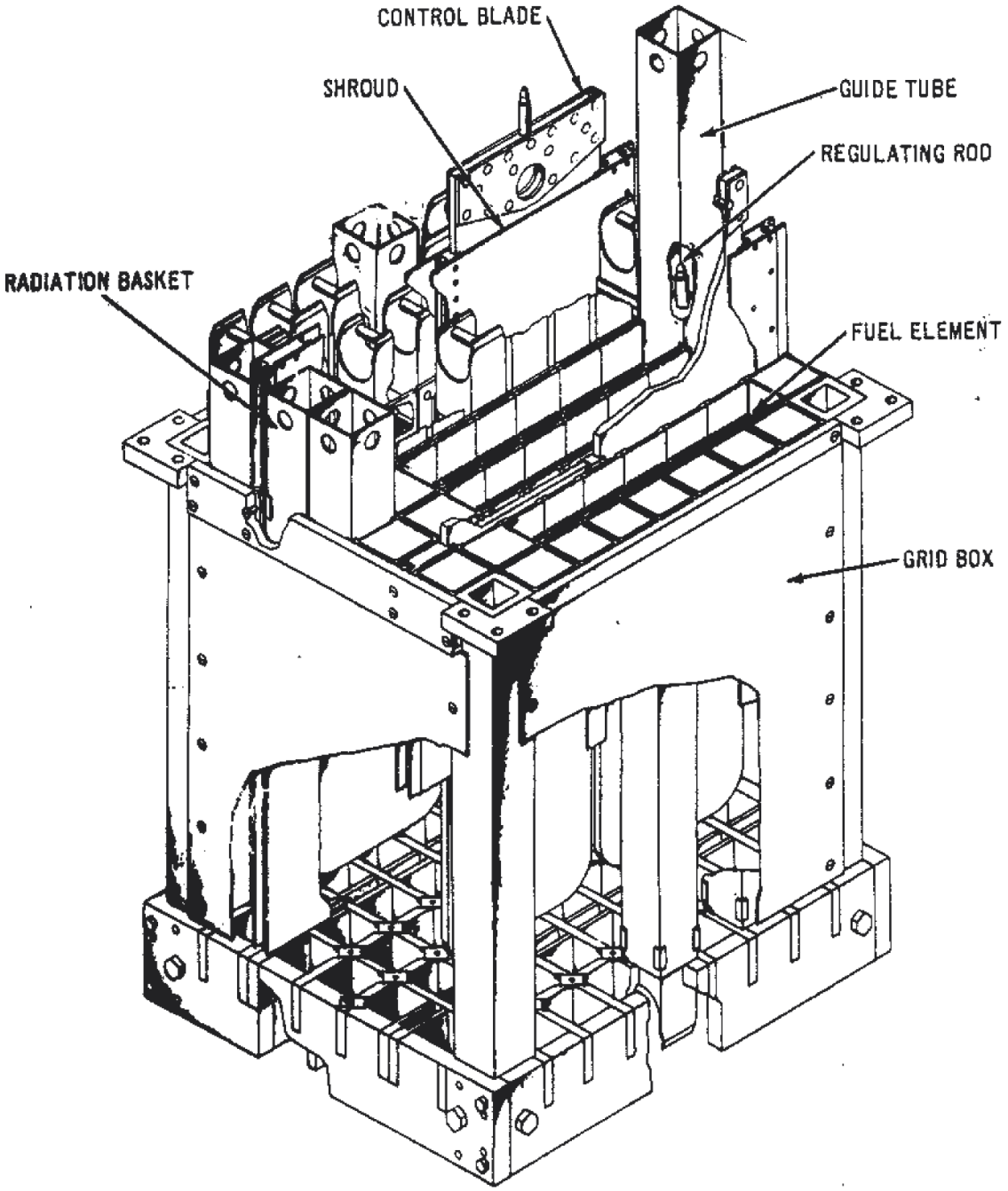

Figure 2-1 Reactor Core Box

The core assembly is located near the bottom of an approximately 31 foot (9.4 meter) deep aluminum-lined pool. The reactor pool is divided into two interconnected sections: a high-power section (the stall section) and a low-power section (the bulk section). The reactor bridge, core suspension frame structure including the four suspension frame corner posts, and core assembly are designed to permit the core assembly and its suspension frame structure to be moved along a mechanical rail system mounted on the top of the pool walls. The core grid

plate is suspended (by the core suspension frame) approximately 8 meters (26.3 feet) below the pool water surface (Ref. 44). The core suspension frame structure is illustrated in SAR Figure 4-11, reproduced below as Figure 2-2. When the reactor core is positioned in the high-power section and coupled to the coolant system, it may be operated in power ranges above 0.1 MWt using forced circulation (there are also connections for coupling the reactor to the coolant system in the low-power section, allowing the reactor to be operated above 0.1 MWt using forced convection in the low-power section, although UML does not typically use this configuration). In the forced circulation mode, a single primary coolant pump circulates approximately 1,700 gallons per minute (gpm) of coolant through the core assembly. For operation in power ranges of 0.1 MWt and below, the reactor may be operated in any location, using natural convection cooling.

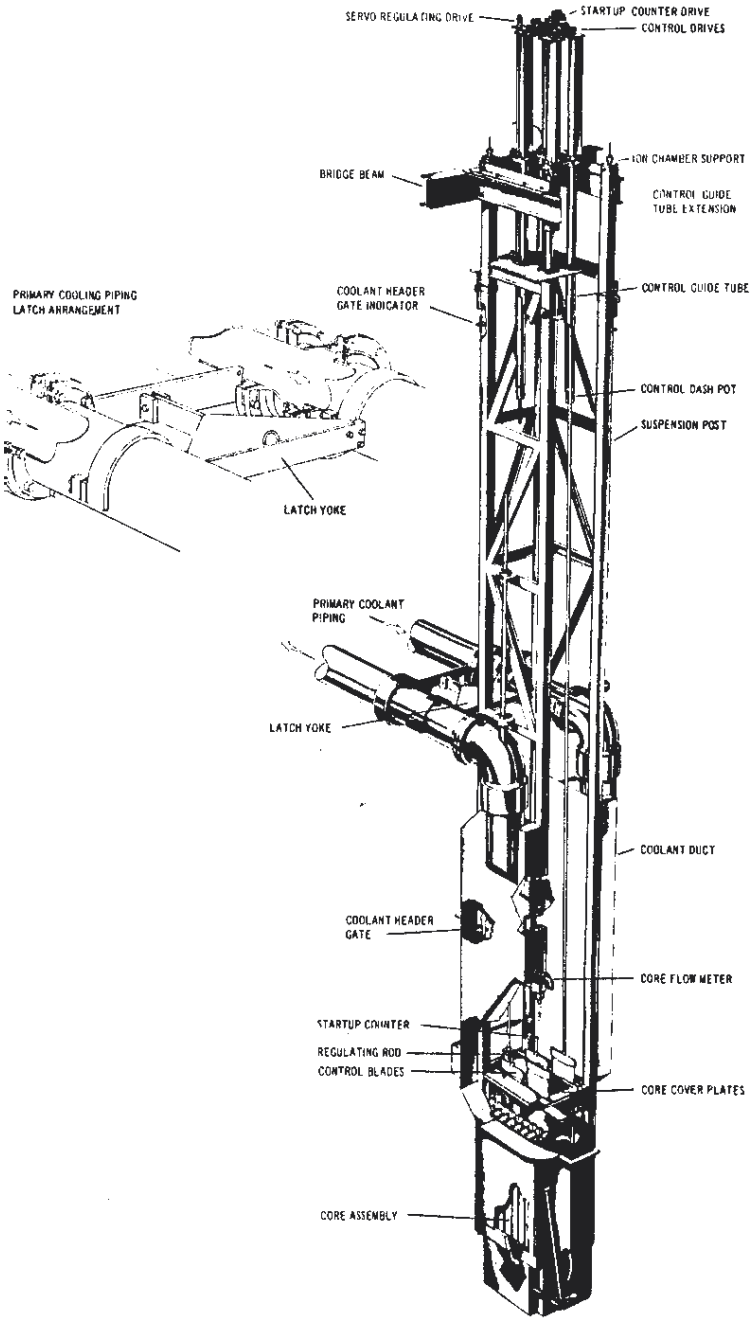

Figure 2-2 Reactor Core Suspension Structure

The fuel elements in the reactor core are arranged in a roughly cylindrical-shaped array within the 9 by 7 core grid. The fuel elements are surrounded on all four sides by reflector elements. The four control blades and a servo-actuated regulating rod control core reactivity. The four control blades each move vertically within separate shrouds (one shroud per control blade). The shrouds divide the core into three sections, and each pair of shrouds extends the length of the core (see Figure 2-1 and Figure 2-3). The confinement of the fuel and reflector elements within the enclosed grid box confines the flow of water between and through the elements.

As discussed in SAR Section 4.1 and SAR Chapter 10, the reactor core accommodates a flux trap (irradiation position which allows experiments to be exposed to the peak neutron flux) at the center of the core. Experiments can also be placed in irradiation baskets along the edge of the core. The reference core configuration illustrated in Figure 2-3 includes the flux trap and irradiation baskets. Other experimental facilities at the UMLRR include a pneumatic tube system, the fast neutron irradiation facility, horizontal beam ports, a thermal column, and dry irradiation rooms.

As discussed in SAR Sections 4.1 and 9.2, new and used reactor fuel that is not in the reactor core may be stored in racks located along the walls of the reactor pool. The used fuel may be used as a source of gamma radiation for experiments.

2.2 Reactor Core

The reactor core is described in SAR Section 4.2. SAR Figure 4-2 provides a diagram of the reference core layout, which is reproduced below as Figure 2-3. Although the exact configuration of the core, including the number of fuel elements, may vary, the figure illustrates a typical core layout. This figure identifies the row and column grid notation used; for example, the D5 location refers to row D and column 5 in the core grid, which is directly in the center of the core. Position D5 contains the flux trap and is commonly referred to as the flux trap location. Additionally, the regulating rod is located in position D9. Partial fuel elements, which contain half the uranium loading of a full silicide UMLRR fuel element, are shown in positions C3 and E3. (In addition to the silicide partial fuel elements, UML also has a WPI (aluminide) variable load element with removable plates available for use.) Lead-void elements (see SER Section 2.2.2) are located in positions A3 through A7. The four control blades are numbered 1 through 4 in the clockwise direction, starting with the lower left quadrant. Figure 2-3 also identifies the location of the beam tubes (note the three tubes illustrated below row G), the fast neutron irradiator, and the large graphite thermal column relative to the reactor core.

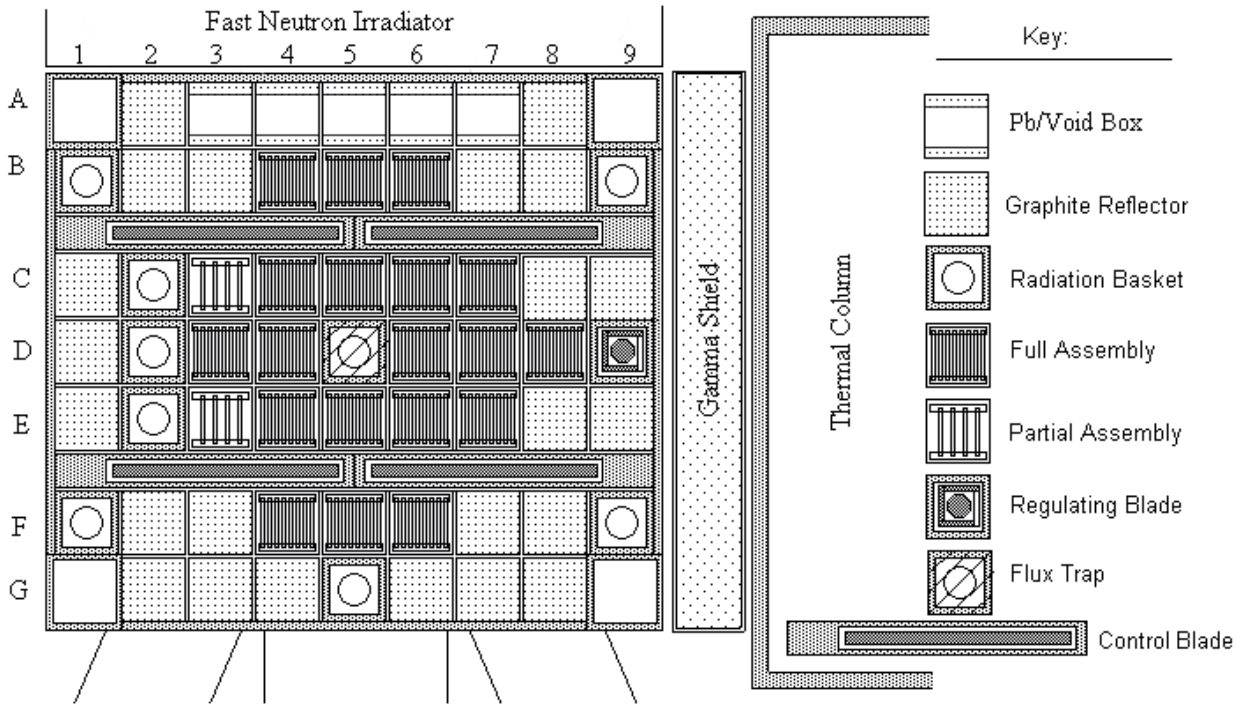

Figure 2-3 Reference Core Layout

Renewed TS 5.3, "Reactor Core and Fuel," would state, in part:

Applicability:

These specifications apply to reactor core and fuel.

Objective:

To specify design features of the reactor core and fuel and allowable fuel configurations.

Specifications:

(1) The reactor core shall consist of a 9 x 7 array of 3-inch square modules with the four corners occupied by posts.

(2) Cores shall contain 21 elements to 26 elements, consisting of any combination of fuel elements as described in specifications 5.3.3, 5.3.4, and 5.3.5.

(3) A standard fuel element shall be either:

a. A flat plate MTR-type element having plates fueled with low enrichment (<20% U-235) U_3Si_2, clad with aluminum. There shall be 18 plates per element with 16 plates containing fuel and two outside plates of aluminum. There shall be 200 ± 2 grams of Uranium-235 per element when new, or

b. A flat plate MTR-type element having plates fueled with low enrichment (<20% U-235) UAl_x, clad with aluminum. There shall be 18 plates per element. There shall be 167 ± 2 grams of Uranium-235 per element when new.

(4) A partial fuel element shall be the same as Specification 5.3(3-a) except each plate shall have approximately half the uranium loading. No more than two (2) partial fuel elements shall be allowed in the core.

(5) A removable plate fuel element shall be the same as Specification 5.3(3-b), except the fuel plates are removable. No more than one (1) removable plate element shall be allowed in the core.

(6) Prior to operating the reactor with a removable plate element, a safety analysis shall be performed for each core configuration and configuration of the element to assure there are no changes to the safety margins presented in the SAR. The analysis shall be reviewed by the reactor safety subcommittee.

….

TS 5.3(1) would require that the reactor core layout consist of a 9 by 7 array of 3-inch square modules, with the four corner modules occupied by the core suspension frame corner posts. TS 3.1.1(3), which is discussed and found acceptable later in this SER section, separately would require that the 59 core grid positions corresponding to the square modules that do not contain posts shall be filled with fuel elements, irradiation baskets, source holders, regulating rod, graphite reflector elements, lead void boxes, or grid plugs. The NRC staff finds that the core layout description in TS 5.3(1) is consistent with information in the SAR, as supplemented, and that TS 5.3(1) helps ensure that the core layout and configurations are consistent with the assumptions of UML's safety analyses of steady-state reactor operation and possible accidents. The NRC staff also finds that TS 5.3(1) appropriately implements the guidance in NUREG-1537, Appendix 14.1, and American National Standards Institute/American Nuclear Society (ANSI/ANS)-15.1-2007 by describing important design parameters of the core. Therefore, based on the above, and also based on its 10 CFR 50.36(c)(4) findings for TSs on design features in SER Section 6.7, the NRC staff concludes that TS 5.3(1) is acceptable.

TS 5.3(2) would require that UMLRR core configurations contain between 21 and 26 fuel elements, and that these fuel elements be a combination of the types of fuel elements described in TSs 5.3(3), 5.3(4), and 5.3(5) (i.e., silicide standard fuel elements, aluminide standard fuel elements, silicide partial fuel elements, and the aluminide removable plate element). In its bases for TS 5.3, UML states that the UMLRR core design analyses (provided in SAR Chapter 4) considered cores with combinations of silicide and aluminide fuel, including up to two silicide partial elements (TS 5.3(4) separately limits UMLRR core configurations to two partial elements), for core loadings from 21 to 26 elements. As discussed in SER Section 2.5.1, for its core design analyses, UML considered various operational cores, including the "operational core configuration" (OCC) which contains 21 silicide elements (19 standard and two partial). UML also analyzed many other feasible core configurations (containing all one fuel type or mixed fuel types) to determine the configuration with the maximum power peaking, which it designated the limiting core configuration (LCC). UML's LCC is a 21-element core containing eight silicide standard fuel elements surrounded by 13 aluminide standard fuel elements. UML used its LCC as the basis for its thermal-hydraulic and accident analyses in the SAR, as supplemented. As discussed in SER Section 2.6 and SAR Section 4.5.7, UML considered

cores between 20 and 26 elements for determination of coolant flow through the elements for use in its steady-state thermal-hydraulic and transient calculations. (UML determined that, of the cores it considered that would be allowed within the UMLRR TSs, cores with 21 aluminide elements are bounding for determination of flow through the fuel and flow bypass around the fuel, and it used the corresponding flow conditions for its steady-state thermal-hydraulic and transient calculations. UML did not consider additional flow bypass that could result from use of the aluminide removable plate element in these calculations, but as discussed later in this SER section, TS 5.3(6) would require UML to perform appropriate analyses to verify that reactor operation would remain within appropriate SAR safety margins prior to any use of the removable plate element in the core.)

The NRC staff notes that cores with fewer elements are typically bounding from both a power peaking (because power is concentrated in fewer elements) and flow bypass (because the fueled region of the core is smaller and more flow can bypass the fueled region) standpoint, although larger cores can increase axial peaking if the critical blade height is lower because of the additional reactivity. The NRC staff finds that TS 5.3(2) helps ensure that the UMLRR is operated with core configurations that are less limiting than the LCC and within the bounds of the analyses in the SAR, as supplemented. The NRC staff notes that notwithstanding TS 5.3(2) and other TSs related to fuel and core configurations, UML is required (in accordance with the regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.59, and UMLRR TS 6.2) to evaluate any core configurations it uses to verify that the configurations are appropriately bounded by analyses in the SAR, as supplemented, in addition to evaluating the configurations for compliance with the UMLRR TSs. The NRC staff also finds that TS 5.3(2) appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 by describing allowable core configurations. Therefore, based on the above, and also based on its 10 CFR 50.36(c)(4) findings for TSs on design features in SER Section 6.7, the NRC staff concludes that TS 5.3(2) is acceptable.

TS 5.3(3), items a. and b., would impose requirements on the specifications of the silicide standard fuel elements and aluminide standard fuel elements, respectively. SAR Section 4.2.1 describes the UMLRR fuel. The NRC staff finds that the fuel specifications in TS 5.3(3), items a. and b., are consistent with information in the SAR, as supplemented, and that TS 5.3(3) helps ensure that the UMLRR fuel design and parameters are consistent with the assumptions of UML's analyses of steady-state reactor operation and possible accidents. The NRC staff also finds that TS 5.3(3) appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 by describing important parameters of the fuel. Therefore, based on the above, and also based on its 10 CFR 50.36(c)(4) findings for TSs on design features in SER Section 6.7, the NRC staff concludes that TS 5.3(3), items a. and b., are acceptable.

TS 5.3(4) would impose requirements on the specifications of the silicide partial fuel elements, and also would require that UMLRR core configurations not include more than two partial fuel elements. SAR Section 4.2.1 describes the UMLRR fuel, including the silicide partial fuel elements. In its bases for TS 5.3, UML states that the UMLRR core design analyses (provided in SAR Chapter 4) considered cores with combinations of silicide and aluminide fuel, including up to two silicide partial elements, for core loadings from 21 to 26 elements. As discussed in SER Section 2.5.1, for its core design analyses, UML considered various operational cores, including the OCC which contains 21 silicide elements (19 standard and two partial). UML also analyzed many other feasible core configurations (containing all one fuel type or mixed fuel types) to determine the configuration with the maximum power peaking, which it designated the LCC. The NRC staff finds that the silicide partial element fuel specifications in TS 5.3(4) are consistent with information in the SAR, as supplemented, and that TS 5.3(4) helps ensure that

the UMLRR fuel design and parameters are consistent with the assumptions of UML's analyses of steady-state reactor operation and possible accidents. The NRC staff also finds that TS 5.3(4) helps ensure that the UMLRR is operated with core configurations that are less limiting than the LCC and within the bounds of the analyses in the SAR, as supplemented. As discussed in SER Section 2.5.1, using a greater number of partial fuel elements in the core could potentially increase radial power peaking. The NRC staff notes that notwithstanding TS 5.3(4) and other TSs related to fuel and core configurations, UML is required (in accordance with the regulations in 10 CFR 50.59, and UMLRR TS 6.2) to evaluate any core configurations it uses to verify that the configurations are appropriately bounded by analyses in the SAR, as supplemented, in addition to evaluating the configurations for compliance with the UMLRR TSs. The NRC staff also finds that TS 5.3(4) appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 by describing important fuel parameters and allowable core configurations. Therefore, based on the above, and also based on its 10 CFR 50.36(c)(4) findings for TSs on design features in SER Section 6.7, the NRC staff concludes that TS 5.3(4) is acceptable.

TS 5.3(5) would impose requirements on the specifications of the aluminide removable plate fuel element, and also would require that UMLRR core configurations not include more than one removable plate fuel element. TS 5.3(6) would require that prior to any use of the removable plate element, UML perform safety analyses to ensure that the use of the element does not challenge the safety margins in the SAR and that these analyses shall be reviewed by the UMLRR Reactor Safety Subcommittee. SAR Section 4.2.1 describes the UMLRR fuel, including the single aluminide removable plate fuel element UML has available for use. In the SAR, as supplemented, UML does not specifically state that it evaluated possible cores containing the removable plate element for its operating cores and/or its LCC. The NRC staff noted that because the removable plate element can have a lower uranium loading than the aluminide standard fuel elements, the use of the removable plate element could potentially increase power peaking in the remaining standard fuel elements. Additionally, the NRC staff noted that if the removable plate element were used in place of a standard fuel element, additional coolant flow through the removable plate element could result in reduced flow through the remaining standard elements, increasing temperatures in the standard elements. In response to RAI-14.5.6 (Ref. 71), UML stated that given these considerations, it added TS 5.3(6) to specifically require a safety analysis and reactor safety committee review prior to any use of the removable plate element in the UMLRR. The NRC staff finds that the aluminide removable plate fuel element specifications in TS 5.3(5) are consistent with information in the SAR, as supplemented. The NRC staff also finds that although the removable plate element is not specifically considered for UML's operating cores and/or LCC, by specifically requiring appropriate evaluation and review of core configurations containing the removable plate element, TS 5.3(6) helps ensure that the UMLRR is operated with core configurations that are less limiting that the LCC and within the bounds of the analyses in the SAR, as supplemented. The NRC staff notes that notwithstanding TS 5.3(5) and other TSs related to fuel and core configurations, and TS 5.3(6), UML is required (in accordance with the regulations in 10 CFR 50.59, and UMLRR TS 6.2) to evaluate any core configurations it uses to verify that the configurations are appropriately bounded by analyses in the SAR, as supplemented, in addition to evaluating the configurations for compliance with the UMLRR TSs. The NRC staff also finds that TSs 5.3(5) and 5.3(6) appropriately implement the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 by describing important fuel parameters and constraints on allowable core configurations. Therefore, based on the above, and also based on its 10 CFR 50.36(c)(4) findings for TSs on design features in SER Section 6.7, the NRC staff concludes that TSs 5.3(5) and 5.3(6) are acceptable.

Design features TSs 5.3(2), 5.3(4), and 5.3(5) would impose restrictions on fuel configurations that may be used in the UMLRR. Although the UMLRR TSs include surveillances (e.g., TSs 4.1(3) and 4.1(4)) for limiting conditions for operation (LCOs) related to core configuration, the NRC staff notes that UMLRR TSs do not include specific surveillance requirements (SRs) to verify that TSs 5.3(2), 5.3(4), and 5.3(5) are met. However, given that standard industry practice for fuel configuration changes includes verifying fuel element positions prior to any subsequent reactor operation, and given that TSs 5.3(2), 5.3(4), and 5.3(5) are design features TSs, the NRC staff finds that no specific surveillance TSs are necessary for TSs 5.3(2), 5.3(4), and 5.3(5). As discussed in SER Section 6.7, the NRC staff finds that UML has surveillance TSs that satisfy the provisions of 10 CFR 50.36(c)(3).

Renewed TS 3.1.1, "Reactivity and Core Configurations," would state, in part:

> Applicability:
>
> These specifications apply to the reactivity condition of the reactor, core configuration, and experiments.
>
> Objective:
>
> To ensure that the reactor can be safety operated and shutdown and maintained in a safe shutdown condition at all times such that the Safety Limit will not be exceeded.
>
> Specifications:
>
> When the reactor is operating, the following conditions shall exist:
>
> …
>
> (3) All core grid positions shall be filled with fuel elements, irradiation baskets, source holders, regulating rod, graphite reflector elements, lead void boxes, or grid plugs.
>
> (4) No more than five (5) of the radiation baskets shall be without flow restricting devices. This specification shall not apply for low power operation ≤100 kW without forced flow.
>
> ....

TSs 3.1.1(3) and 3.1.1(4) would require that all core grid positions be filled and for positions filled with radiation baskets (e.g., peripheral radiation baskets, or the center flux trap, which can allow flow bypass), no more than five be without flow restricting devices. TS 3.1.1(4) does not apply when the reactor is operated in natural convection mode at or below 100 kilowatts-thermal (kWt). SAR Section 4.5.7 describes UML's flow distribution and fuel element flow rate calculations. In its calculations, UML assumes that forced coolant flow can bypass the fuel element flow channels through up to five open radiation baskets (a portion of the flow that passes around the four control blades and the regulating rod also bypasses the fuel; other than the open radiation baskets, control blades, and regulating rod, flow bypass around the fuel channels is assumed to be negligible as long as all core grid positions are filled, based on the design of the grid plate). The results of UML's fuel element flow rate calculations are used in its steady-state thermal hydraulic (SER Section 2.6) and transient (SER Chapter 5) analyses. In its

response to RAI-14.3.4 (Ref. 71), UML stated that it revised the condition for which TS 3.1.1(4) does not apply to natural convection mode at or below 100 kWt (in the previous TSs prior to this license renewal, it did not apply for operation below 10 kWt) because forced convection flow is only required for operation above 100 kW(t), and applying this requirement for natural convection flow operation is not necessary for consistency with its safety analyses. The NRC staff finds that TS 3.1.1(3) and 3.1.1(4) will help ensure that flow bypass around the UMLRR fuel element flow channels is minimized and that the reactor is operated consistent with the assumptions in the thermal-hydraulic and transient analyses. The NRC staff also finds that TS 3.1.1(4) only needs to be applicable for forced-flow operation because it ensures the validity of an assumption that is only used for forced-flow operation analyses. Therefore, based on the above, and based on its 10 CFR 50.36(c)(2) findings for LCO TSs in SER Section 6.7, the NRC staff concludes that TSs 3.1.1(3) and 3.1.1(4) are acceptable.

Renewed TS 4.1, "Reactor Core Parameters," would state, in part:

> Applicability:
>
> This specification applies to surveillance requirements for the various reactor core parameters.
>
> Objective:
>
> To ensure the reactor core parameters meet the specified limiting conditions for operation.
>
> Specifications:
>
> …
>
> (3) Prior to the first reactor start-up of the day, visual verification shall be made that each core grid position is filled with either a fuel element, a radiation basket, a source holder, the regulating rod, a graphite reflector element, a lead void box, or a grid plug.
>
> (4) Prior to the first reactor start-up of the day, visual verification shall be made that no more than five of the radiation baskets are without flow restricting devices. This specification shall be optional for low power operation less than or equal to 100 kW without forced flow.
>
> ….

TSs 4.1(3) and 4.1(4) would require that prior to the first reactor start-up of the day, UML visually verify that, respectively, all core grid positions are filled as required by LCO TS 3.1.1(3) and for core grid positions that contain radiation baskets, no more than five of the baskets are without flow restricting devices, as required by LCO TS 3.1.1(4). The TS 4.1(4) surveillance is optional if only natural convection operation will occur during that day. The NRC staff finds that TSs 4.1(3) and 4.1(4) help ensure that UML verifies, at an appropriate frequency, that UMLRR core configurations meet the requirements of TSs 3.1.1(3) and 3.1.1(4), such that the reactor is operated consistent with the assumptions in UML's thermal-hydraulic and transient analyses. The NRC staff also finds that it is reasonable that TS 4.1(4) is optional when only natural convection operation is planned because TS 3.1.1(4) is only applicable for forced-flow

operation. Therefore, based on the above, and based on its 10 CFR 50.36(c)(3) findings for TSs on SRs in SER Section 6.7, the NRC concludes that TSs 4.1(3) and 4.1(4) are acceptable.

2.2.1 Reactor Fuel

SAR Section 4.2.1 describes the MTR fuel at the UMLRR. For the standard fuel used since the HEU to LEU conversion of the UMLRR in 2000, the fuel plates contain a uranium-silicon intermetallic dispersion matrix, U_3Si_2, which contains uranium enriched to 19.75 percent in U-235. The matrix is clad in aluminum. In the SAR and in this SER, this fuel is referred to as standard fuel or silicide fuel.

As discussed in SER Sections 1.10 and 2.1, UML also requested, coincident with the NRC's approval of the renewal of the UMLRR license, NRC approval to operate the UMLRR using fuel that UML previously obtained from WPI following the permanent shutdown of the WPI research reactor (Facility Operating License No. R-125 currently allows UML to possess, but not to use, this fuel). The fuel from WPI contains a uranium-aluminum dispersion matrix (UAl_x) and, like the silicide fuel, contains uranium enriched to 19.75 percent in U-235. The fuel matrix is clad in aluminum. This UAl_x fuel is referred to in the SAR and this SER as WPI fuel or aluminide fuel. SAR Section 4.1 states that the aluminide fuel was previously slightly irradiated in the WPI reactor.

NUREG-1313, "Safety Evaluation Report Related to the Evaluation of Low-Enriched Uranium Silicide-Aluminum Dispersion Fuel for Use in Non-Power Reactors" (Ref. 52), which UML references in the SAR and in its basis for the UMLRR safety limit (SL) (TS 2.1), provides a general basis for the NRC's acceptance of the use of silicide fuels in non-power reactors. Aluminide fuels were in use at research reactors prior to the development of silicide fuels, which were originally developed to facilitate the conversion of certain non-power reactors from HEU to LEU fuel. NUREG-1313 compares the performance of aluminide and silicide fuels, and states that the materials properties including heat capacity, thermal conductivity, and corrosion behavior are similar for silicide and aluminide fuels. NUREG-1313 states that the irradiation behavior, including thresholds for and propensity to swelling, blistering, and fission product release, is also similar for the two fuel types. Additionally, NUREG-1313 includes information regarding the fabrication of silicide fuels.

As discussed in the decommissioning plan that WPI submitted to NRC for the WPI reactor on March 31, 2009 (Ref. 41), WPI permanently shut down the WPI reactor in 2007. The 10 kWt WPI reactor was converted from HEU fuel to the LEU aluminide fuel in 1989 (although the UMLRR was converted in 2000 to use LEU silicide fuel, the WPI reactor could be converted to use LEU aluminide fuel). The NRC staff directed the HEU to LEU conversion of the WPI reactor by an Order (supported by a safety evaluation) dated September 12, 1988 (Ref. 42), which modified and added license conditions (LCs) and TSs. Although the WPI reactor only operated at 10 kWt, the safety evaluation supporting the WPI conversion Order noted that in 1981, the NRC licensed fuel elements with plates essentially identical to the WPI LEU aluminide plates for use in the University of Michigan research reactor (which the NRC staff notes was licensed to operate at 2MWt), and these fuel elements operated successfully at this higher-power reactor with no safety-significant unpredicted events. The NRC staff notes that LEU and HEU aluminide fuels have been used safely and extensively at other non-power reactors of varying power levels as well.

A UMLRR fuel element is illustrated in SAR Figure 4-3 in SAR Section 4.2.1. This figure is reproduced below as SER Figure 2-4. As described in the SAR, the fuel plates contained within

the elements are arranged such that water gaps (or channels) between the plates allow coolant to pass between the plates as it flows from top to bottom through the elements (or from bottom to top when the reactor is operating in natural convection mode). Two grooved side plates hold the fuel plates in place and support the structure of the fuel element. Identical end boxes on each end of the fuel element position the element in the grid and allow the element to be manipulated using a fuel handling tool. The elements may be inverted and rotated to achieve more efficient utilization of fuel. Physically, the silicide and aluminide elements are nearly identical, and they fit interchangeably within the UMLRR core. Differences between the fuel types, other than the material composition of the fuel matrix (i.e., silicide versus aluminide), are different U-235 loading (200 grams of U-235 per silicide element versus 167 grams of U-235 per aluminide element); different number of fuel plates per element (16 per silicide element versus 18 per aluminide element); different fuel matrix and plate thickness (aluminide plates are thicker); and other small differences in parameters including clad thickness, water gap thickness, plate height and width, and side plate thickness. The silicide fuel elements have two outer plates made of aluminum (occupying the two outer fuel plate "positions" in the grooved side plates) in addition to the 16 fuel plates, while all 18 plates in the aluminide fuel elements are fuel plates. Table 2-1 summarizes some of the similarities and differences between the silicide and aluminide fuel elements, as discussed in the SAR.

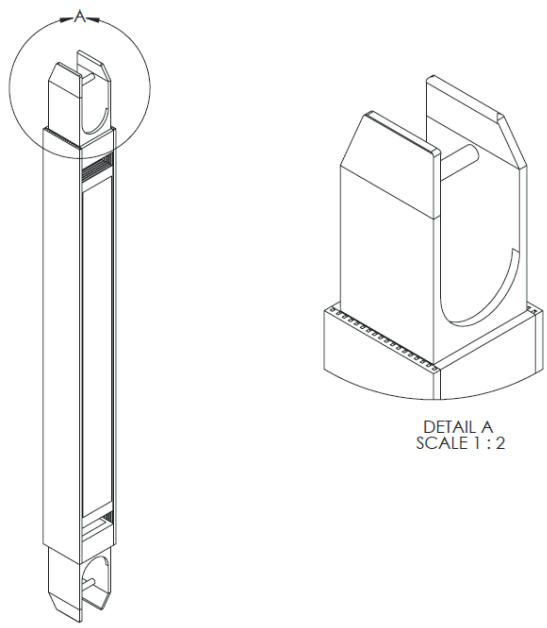

Figure 2-4 UMLRR Fuel Element, Including End Box Detail

Table 2-1 Physical Data for UMLRR Silicide and Aluminide Fuel Elements

Parameter	Silicide Fuel Element	Aluminide Fuel Element
Dispersion Matrix	U_3Si_2	UAl_x
Cladding	Aluminum	Aluminum
Enrichment (percent)	19.75	19.75
Plate thickness (cm)	0.1270	0.1524
Fuel matrix thickness (cm)	0.0510	0.0762

Clad thickness (cm)	0.0380	0.0381
Fuel plates/element	16	18
Aluminum plates/element	2	0
Side plate thickness (cm)	0.5080	0.4572
Channel thickness (cm)	0.2963	0.2709

The UMLRR core may also include silicide or aluminide fuel elements that have less than the full U-235 loading of 200 or 167 grams, respectively. Some silicide fuel elements, which are referred to as partial fuel elements, are identical to the standard silicide fuel elements except that they have half the uranium loading of standard silicide fuel elements (i.e., 100 grams of U-235 instead of the full 200 grams of U-235). An aluminide fuel element, which is referred to as a variable load element, has removable plates.

Because UML proposes to operate the UMLRR using either or both types of MTR fuel interchangeably, UML performed nuclear design, thermal-hydraulic, and accident analyses including both silicide and aluminide fuel. UML's analyses are discussed and found acceptable in Sections 2.5 and 2.6 and Chapter 5 of this SER. In response to RAI-4.1(e) (Ref. 23), UML provided analyses demonstrating that the silicide fuel currently in use is more limiting from a safety analysis standpoint because it has a higher reactivity per element, a higher power per plate, and a lower margin to onset of nucleate boiling (ONB). However, as discussed in SER Section 2.5.1, UML determined that a mixed core consisting of silicide fuel near the center of the core and aluminide fuel at the edges would maximize the power peaking in the core as a whole, and, therefore, UML defined a mixed core as the LCC.

Renewed TS 2.1, "Safety Limit," would state:

> Applicability:
>
> This specification applies to the reactor fuel.
>
> Objective:
>
> The objective is to ensure that the integrity of the fuel cladding is maintained.
>
> Specification:
>
> The reactor fuel clad temperature shall be less than 530°C (986°F).

TS 2.1 would require that the reactor fuel cladding temperature not exceed 530 degrees Celsius (°C) (986 degrees Fahrenheit (°F)) in order to help prevent blistering of the fuel and to ensure that the integrity of the fuel cladding is maintained. In its response to RAI-14.1.2 (Ref. 71), UML stated that the UMLRR SL in TS 2.1 is based on information in NUREG-1313 (Ref. 52). In its basis for TS 2.1, UML stated that based on NUREG-1313, the blister threshold temperature for both uranium silicide and uranium aluminide fuel is above 530 °C (986 °F).

The regulation at 10 CFR 50.36(c)(1)(i)(A) states that SLs for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain physical barriers (such as fuel cladding) that guard against the uncontrolled release of radioactivity. NUREG-1537 (Ref. 14), Part 1, Appendix 14.1, Section 2.1, states that SLs should be developed to avoid failure of the fuel and should be a limit that is not to be exceeded

under any conditions of operation. NUREG-1313 states that the blister threshold on aluminum-clad silicide fuel plates can range from approximately 515 to 575 °C (959 to 1,067 °F) and that this is similar to the blister threshold for aluminum-clad aluminide fuels. NUREG-1537, Part 1, Appendix 14.1, Section 2.1, states that based on information in NUREG 1313, the NRC staff has found that 530 °C (986 °F) is an acceptable SL for aluminum-clad MTR fuels. UML's thermal-hydraulic analyses for the UMLRR, which are reviewed and found acceptable in Section 2.6 of this SER, demonstrate that for any condition of routine reactor operation, cladding (and fuel matrix) temperatures will remain several hundred degrees Celsius below a 530 °C (986 °F) SL. UML's accident analyses, which are reviewed and found acceptable in Chapter 5 of this SER, further show that even during accident conditions, the cladding will remain below 530 °C (986 °F), equivalent to the UMLRR SL, for any credible scenario. Because TS 2.1 would help avoid fuel failure; meets 10 CFR 50.36(c)(1)(i)(A) because it provides a process variable (clad temperature) limit that is necessary to protect cladding integrity and guard against an uncontrolled release of radioactivity; is consistent with information in NUREG-1313 and guidance in NUREG-1537 and ANSI/ANS-15.1-2007; and would not be exceeded during routine operation or any credible accident condition, the NRC staff concludes that TS 2.1 is acceptable.

Renewed TS 3.1.1, "Reactivity and Core Configurations," would state, in part:

> Applicability:
>
> These specifications apply to the reactivity condition of the reactor, core configuration, and experiments.
>
> Objective:
>
> To ensure that the reactor can be safety operated and shutdown and maintained in a safe shutdown condition at all times such that the Safety Limit will not be exceeded.
>
> Specifications:
>
> When the reactor is operating, the following conditions shall exist:
>
> …
>
> (5) The reactor shall not be knowingly operated with damaged fuel except as may be necessary to identify the location of the damaged fuel.
>
> ….

TS 3.1.1(5) would require that the reactor not be knowingly operated with damaged fuel except as needed to locate such fuel. In its response to RAI-14.3.5 (Ref. 71), UML stated that "damaged fuel" in this TS refers to either fuel with physical damage that could affect the mechanical integrity of the fuel element (e.g., a loose or broken end box or side plate) or fuel with actual leakage of fission products. The NRC staff finds that TS 3.1.1(5) helps ensure that the UMLRR is not operated with fuel that is releasing fission products to the primary coolant, unless it is necessary to locate damaged fuel (fuel defects that release fission products may only be detectable during reactor operation), or that has other unacceptable degradation or damage. The NRC staff also finds that TS 3.1.1(5) is consistent with guidance in NUREG-1537, Appendix 14.1, that TSs should prohibit operation with damaged fuel except to locate such fuel.

Therefore, based on the above, and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.1.1(5) is acceptable.

Renewed TS 4.1, "Reactor Core Parameters," would state, in part:

> Applicability:
>
> This specification applies to surveillance requirements for the various reactor core parameters.
>
> Objective:
>
> To ensure the reactor core parameters meet the specified limiting conditions for operation.
>
> Specifications:
>
> …
>
> (8) Visual inspection of one fifth of the in-core reactor fuel elements shall be performed every two years, such that all fuel elements in the core are inspected over a 10 year period.

TS 4.1(8) would require UML to perform a visual inspection of one-fifth of the fuel elements in the reactor core every 2 years, such that all elements are inspected at least once every 10 years (i.e., UML may meet the requirement by inspecting a different one-fifth of the elements during each of five consecutive required biennial inspections). The NRC staff finds that TS 4.1(8) helps ensure that UML inspects all in-core fuel elements at intervals the are appropriate to help ensure that TS 3.1.1(5) is met and that the reactor is not operated with damaged fuel, but also minimizes the fuel handling activity that is necessary (and the risk of possible fuel handling accidents). The NRC staff notes that research reactor fuel inspections typically include looking for any deterioration, corrosion, or other physical changes that could lead to loss of cladding integrity (even if actual fission product leakage may not yet have occurred). The NRC staff also finds that the TS 4.1(8) requirement for periodic visual fuel inspections appropriately implements the guidance in ANSI/ANS-15.1-2007 and NUREG-1537, Appendix 14.1, and exceeds the specific guidance in NUREG-1537, Appendix 14.1, Section 4.1(6), which states that TSs do not typically require routine fuel inspections for reactors with plate-type fuel. Therefore, based on the above, and based on its 10 CFR 50.36(c)(3) findings for TSs on SRs in SER Section 6.7, the NRC staff concludes that TS 4.1(8) is acceptable.

Based on its review of the information provided by UML regarding the UMLRR fuel, the NRC staff finds that UML described the fuel elements to be used in the reactor in sufficient detail. UML provided the SL for the fuel, as well as the basis for the SL, and the NRC staff finds that this SL is appropriate. UML adequately described the constituents, materials, components, and fabrication specifications for the fuel, and compliance with these specifications for all fuel acquisitions will help ensure uniform characteristics and compliance with design bases and safety-related requirements. UML referred to NUREG-1313, which documents the investigation of fuel characteristics and parameters important to safe reactor operation, and discusses fuel design limits that support limits stated in the TSs. Information on the design and performance of the UMLRR silicide and aluminide fuel provides assurance that the fuel can function safety in the reactor without adversely affecting the health and safety of reactor staff or members of the

public. Therefore, based on the above, the NRC staff concludes that the descriptions of the UMLRR fuel in the SAR, as supplemented, and the associated TSs, described above, are acceptable.

2.2.2 Control Blades and Regulating Rod

The control blades and regulating rod are primarily described in SAR Section 4.2.2 and UML's letter dated September 30, 2020 (Ref. 98). SAR Section 3.5.3 and Chapter 7, as supplemented by UML's response to RAI-13.4 (Ref. 23), provide information regarding the interface between the control blades and the reactor control and protection systems.

SAR Section 4.2.2 states that the UMLRR has five independent control devices, including four scrammable control blades and one non-scrammable regulating rod. The SAR and RAI responses use the terms "control blade(s)" and "control rod(s)" to refer to the scrammable control blades, and "regulating blade" and "regulating rod" to refer to the non-scrammable regulating rod, interchangeably throughout. This SER uses "control blade(s)" to refer to the scrammable control blades and "regulating rod" to refer to the non-scrammable regulating rod. "Control device(s)" is used in the SAR and in this SER to refer to the control blades and regulating rod collectively.

The four control blades each move vertically within separate shrouds (one shroud per control blade). The shrouds effectively divide the core into three sections, and each pair of shrouds extends the length of the core (see SER Figure 2-1 and Figure 2-3). The shrouds act as guides for the control blades throughout their travel. When a control blade is fully withdrawn from the core, at least three vertical inches of the control blade remain engaged within the shroud.

A UMLRR control blade is depicted in SER Figure 2-5, which is adapted from SAR Figure 4-6. The active region of the control blade (i.e., the portion of the control blade which contains neutron poison material and can be used to control reactivity) is the entire control blade, except for the hardware at the top of the control blade which is used to secure the control blade to its lift-rod assembly. As described in SAR Section 4.2.2, the control blade active region consists of a single homogeneous metal matrix composite (MMC) material. The MMC material, which is referred to as BORTEC, is a homogeneous mixture of boron carbide (B_4C) and aluminum. When the license renewal SAR was submitted in 2015, UML was in the process of replacing different (older) reactor control blades, which are B_4C in an aluminum matrix clad with aluminum, with the BORTEC blades described in the renewal SAR because UML noted minor blistering of the cladding of one of the old blades (see the UMLRR annual reports for 2014-2015 through 2016-2017 (Refs. 10.j, 10.k, and 10.l)). As discussed in the 2014-2015 annual report (Ref. 10.j), in UML's response to RAI-7.3 (Ref. 79), and in SER Section 1.8, UML screened this change and determined that it did not require a complete evaluation under 10 CFR 50.59. In the 2016-2017 annual report (Ref. 10.l), UML stated that it observed no measurable difference in control blade drop times and reactivity measurements for old blades versus blade replacements done up through the period of time covered in that report. In its letter dated September 30, 2020 (Ref. 98), UML clarified that its control blade status did not match the 2015 license renewal SAR description, because the blade replacements were still in progress as of that date (only three of the four control blades were replaced). UML further clarified that both BORTEC blades (described in the 2015 license renewal SAR) and older blades consisting of B_4C in an aluminum matrix clad with aluminum (i.e., BORAL blades) could be in the reactor core when a renewed license in issued.

TS 4.2.1(2), which is discussed and found acceptable later in this SER section, would require annual visual inspection of control blades; the NRC staff notes that these inspections will help ensure that UML identifies any issues with or degradation of control blades, including any possible blistering of the remaining older blade (consisting of B_4C in an aluminum matrix clad with aluminum).

Figure 2-5 Control Blade

According to SAR Sections 4.2.2.1 and 7.3.2, the four control blades are actuated by electromechanical control blade drive systems that permit each control blade to be moved, held in place, or scrammed. A control blade drive mechanism is illustrated in SER Figure 2-6, which is reproduced from SAR Figure 4-7. The control blade drive mechanisms are located at the top of the core suspension frame structure, above the surface of the reactor pool (see SER Figure 2-2). The control blade drive mechanisms employ direct-current motors. The drive motors are reversible motors with an integral reducing-gear assembly to reduce speed and an integral brake assembly to prevent control blade drift. An optical encoder detects and transmits each revolution of the mechanism drive shaft and the signal pulses are counted and a continuous control blade position indication is provided in units of inches. A ball-bearing screw and nut assembly is used to raise and lower a control blade by converting the rotation of the drive motor to the linear up or down motion of the control blade. As the screw rotates, the nut is driven up or down. The nut is coupled directly to a drive tube, such that the drive tube moves up and down with the nut. The lower end of the drive tube is attached to an electromagnet assembly. When the electromagnet is energized, it engages a cadmium-plated carbon steel anvil that is attached (at a point above the reactor pool surface) to the top of a lift-rod assembly. The lift-rod assembly connects to the control blade through a support and guiding mechanism mounted on a bracket attached to the core suspension structure.

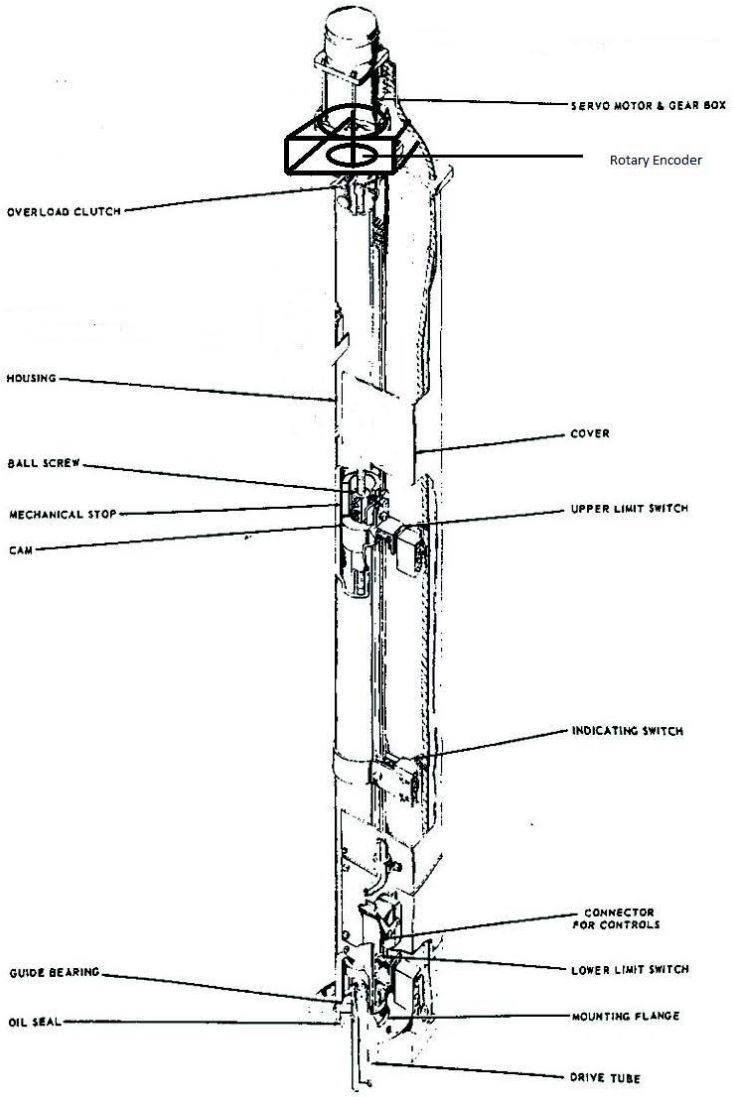

Figure 2-6 Control Blade Drive Mechanism

As discussed in SAR Section 4.2.2.1, the control blade drive mechanism can operate through a stroke of 26 inches at a maximum speed of 3.6 inches per minute in either direction. Coasting of the mechanism is limited to less than 0.1 inch of blade travel. Limit switches at the upper and lower limits of the stroke de-energize the drive motor if the blade reaches its full-in or full-out limits. The limit switches also provide control room indications. In addition, a limit switch within the scram magnet gives an indication in the control room when the electromagnet is energized and engages the anvil at the top of the lift-rod assembly. When the reactor is scrammed, the electromagnet is de-energized, releasing the anvil, and allowing the control blade and lift-rod assembly to drop into the core under the force of gravity. A dash pot assembly cushions the fall of the shim blade during the final 20 percent of travel. To recover the control blades after a scram, the electromagnets are re-energized, the drive mechanisms are rundown, and the electromagnets re-attach to the anvils.

As discussed in SAR Sections 3.5.3 and 7.4.3, and in the response to RAI-13.4, the reactor protection system (RPS) governs power to the electromagnets and initiates a reactor scram

when trip settings (see SER Section 2.5.3) are exceeded by cutting off power to the electromagnets. The RPS is designed for redundancy and diversity and is designed to be fail-safe in that failures result in the initiation of a reactor scram.

As discussed in SAR Section 7.3.2, the control blade drives are hard wired with mechanical relays whose logic prevents more than one control blade from being withdrawn at any time. TS 3.2.2(3), which is discussed and found acceptable later in this SER section, prohibits more than one control blade from being able to be withdrawn at a time. However, all four control blades may be simultaneously inserted. The control blade drive rundown circuit, which is a relay logic system that actuates the simultaneous insertion (rundown) of all four control blade drives following a scram or when initiated by the reactor operator, is discussed in SAR Section 7.3.4.

SAR Section 7.3.3 discusses the control blade withdrawal inhibit circuit, which is a relay logic system that prevents the actuation of the control blade drives and the subsequent withdrawal of the control blades unless the master control switch (i.e., master key switch) controlling power to the reactor control system is in the “on” (or “test”) position, all RPS limit switch contacts in the scram chain are in the normal safe position (i.e., not in the position that would cause a scram), all RPS relay contacts in the scram chain are reset and energized in the normal closed condition, and the linear power channels indicate five percent or more of full scale and the startup channel indicates two counts per second (cps) or greater. In addition, as discussed in SAR Section 7.4.1, as updated and supplemented by UML’s submittal dated April 10, 2019 (Ref. 73), the logarithmic (log) power/period channel also provides a control blade drive inhibit signal for a short period, downscale power, or high power indication. TS 3.2.6, which is discussed and found acceptable in SER Section 2.5.3, imposes operability requirements for the minimum interlocks required to inhibit rod withdrawal unless the TS 3.2.6-specified conditions are met.

According to SAR Sections 1.5, 4.2.2.3, and 4.5, the regulating rod is fabricated of a hollow, square tube that is 25 inches long and has a wall thickness of approximately 0.25 inch. Three sides of the tube are aluminum-clad BORAL (i.e., B_4C and aluminum composite), while the fourth side is an aluminum plate with no neutron poison material. The regulating rod is screwed to the servo regulating rod drive shaft. The regulating rod moves vertically within an approximately three inch square aluminum shell guide tube that is seated along the edge of the core grid, as shown in SER Figure 2-3. A cross-sectional view of the regulating rod, its drive shaft, and its guide tube, with dimensions in inches, is depicted in SER Figure 2-7, which is reproduced from SAR Figure 4-8. SER Figure 2-7 shows the aluminum side of the regulating rod facing toward the core fuel elements. However, as discussed in SAR Section 4.5, the regulating rod can be rotated such that the aluminum side can face toward or away from the fuel (when the aluminum side faces the fuel, the poison material is farther from the fuel, and the regulating rod reactivity is reduced), and since the HEU to LEU fuel conversion was completed in 2000, the regulating rod has been oriented such that the aluminum side faces away from the fuel.

As discussed in SAR Section 4.2.2.3, the servo drive mechanism for the regulating rod is similar to the control blade drive mechanisms, except that a solid coupling replaces the scram electromagnet assembly, meaning that the regulating rod cannot be scrammed. As discussed in SAR Section 4.2.2.3 and UML’s letter dated February 16, 2021 (Ref. 101), the servo control drive is operated by a servo motor and reducing gear train through a total stroke of 26 inches at a maximum travel speed of approximately 55 inches per minute. The servo-controlled regulating rod drive can automatically regulate reactor power within closer limits than those

attainable by using the control blades alone. Manual and automatic control of the reactor using the regulating rod is discussed in SAR Section 7.3.5. The automatic control system is designed such that the reactor may only be placed into or remain in automatic control if certain conditions are met (e.g., reactor period greater than 30 seconds and regulating rod not full-in or full-out).

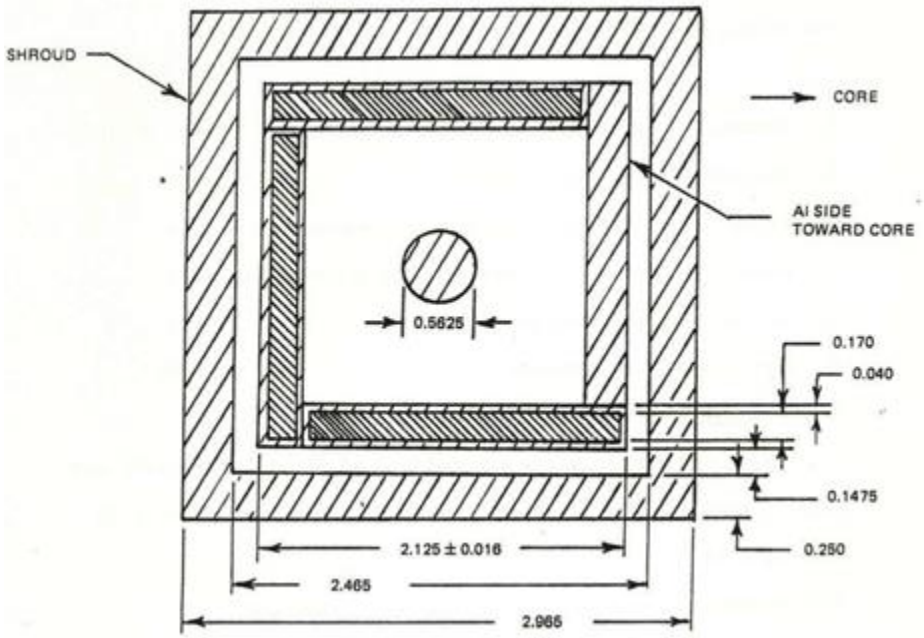

Figure 2-7 Regulating Rod Cross-Sectional View

TS 3.1.1(2), which is discussed and found acceptable in SER Section 2.5.3, would require that the reactor have a minimum shutdown margin (SDM), even with the most reactive control blade and the regulating rod fully withdrawn. TS 3.1.1(2) helps ensure that the control blades have sufficient reactivity to shut down the reactor even if one of the control blades fails in its most reactive position.

Renewed TS 3.2.1, “Control Blades,” would state:

Applicability:

This specification applies to the reactor control blades.

Objective:

To specify the minimum number of operable control blades and their maximum scram time to ensure the reactor can be shut down and the Safety Limit is not exceeded.

Specifications:

(1) All four control blades shall be operable when the reactor is operating.

(2) The time from initiation of a scram signal and movement of each control blade from the fully withdrawn position to 80% of the fully inserted position shall be less than one second.

TS 3.2.1(1) would require that all four control blades be operable when the reactor is operating. In its response to RAI-14.3.6 (Ref. 71), UML stated that TS 3.2.1(1) does not include the regulating rod because it is not scrammable. As discussed in the SAR, as supplemented, the intended function of the control blades includes both routine reactivity control during normal operation and scram capability. UML's core design (neutronics) analyses in SAR Section 4.5, as supplemented (which are discussed in SER Section 2.5), assume that the four control blades are banked (i.e., positioned at the same height), which helps minimize radial power peaking. The lower-worth regulating rod is not assumed to be banked with the control blades. The NRC staff finds that TS 3.2.1(1) helps ensure that all control blades are operable such that they are able to insert negative reactivity during a scram and also such that each control blade is able to be banked with the other control blades such that the reactor is operated consistent with the safety analyses. Although TS 3.2.1(1) would help ensure that all control blades are able to be banked, it would not require that the control blades be banked during any specific operation, since it is not always desirable or necessary to have the control blades banked (e.g., flux tilts are typically not an issue at low power). Although all control blades must be operable, UML's SDM requirement (TS 3.1.1(2)) and its transient analyses conservatively assume that one control blade is stuck fully withdrawn and does not scram with the others. The NRC staff finds that regulating rod operability does not need to be a TS requirement, because the regulating rod is not scrammable, it is not assumed to be banked with the control blades in UML's core design analyses, and its worth is small relative to the control blades. The NRC staff also finds that TS 3.2.1(1) appropriately implements guidance in NUREG-1537 and ANSI/ANS-15.1-2007 by specifying the number of operable control blades. Therefore, based on the above, and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.2.1(1) is acceptable.

TS 3.2.1(2) would requires that from the initiation of a scram signal, each control blade be movable from fully withdrawn to 80 percent inserted in less than one second. As discussed in SAR Section 4.5.5 and SER Section 5.2, although UML's transient analyses conservatively assume that the instrument delay time is separate from the one second it takes the blades to physically fall, such that it takes longer for a scram to occur and the blades to insert, the TS 3.2.1(2) maximum scram time of 1 second includes both any instrument delay time and the time it takes the blades to fall. The NRC staff finds that TS 3.2.1(2) helps ensure that UMLRR operation is within the bounds of the assumptions of the transient analyses discussed in SER Chapter 5, which show that credible transients will not cause a SL to be exceeded. The NRC staff also finds that TS 3.2.1(2) is consistent with the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 because it specifies the maximum scram time for all scrammable control elements. Therefore, based on the above, and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.2.1(2) is acceptable.

Renewal TS 3.2.2, "Maximum Reactivity Insertion Rate and Regulating Rod Worth," would state:

> Applicability:
>
> This specification applies to the maximum positive reactivity insertion rate by the most reactive control rod and regulating rod simultaneously.

Objective:

To ensure that the reactor is operated safety and the safety limit is not exceeded during any credible ramp reactivity insertion.

Specifications:

(1) The maximum reactivity insertion rate by the most reactive control blade and the regulating rod simultaneously shall not exceed 0.05% Δk/k per second.

(2) The total reactivity worth of the regulating rod shall be < 0.5% Δk/k.

(3) Only one control blade shall be able to be withdrawn at a time.

TS 3.2.2(1) would require that the maximum simultaneous reactivity insertion from the most reactive control blade and the regulating rod shall be 0.05 percent delta k over k (% Δk/k) per second. TS 3.2.2(3) would require that only one control blade be able to be withdrawn at a time (as discussed above, the control blade drives are hard wired with mechanical relays with logic preventing more than one control blade from being withdrawn at a time). Therefore, TSs 3.2.2(1) and 3.2.2(3) effectively limit the maximum reactivity insertion rate using control blades and the regulating rod to 0.05% Δk/k per second. UML's slow reactivity transient analyses (see SER Section 5.2) demonstrate that for transients in which 0.07% Δk/k per second of positive reactivity is inserted into the reactor, a SL will not be exceeded. The NRC staff finds that TSs 3.2.2(1) and 3.2.2(3) help ensure that the maximum rate at which reactivity can be inserted into UMLRR by the control blades and regulating rod combined will not exceed the assumptions in the transient analyses. The 0.02% Δk/k per second margin between TS 3.2.2(1) and the analyses helps account for any uncertainty in the analyses or in measured control blade or regulating rod worths. The NRC staff also finds that TSs 3.2.2(1) and 3.2.2(3) are consistent with guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 because they limit reactivity insertion rates. Therefore, based on the above, and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TSs 3.2.2(1) and 3.2.2(3) are acceptable.

TS 3.2.2(2) would require that the regulating rod be worth less than 0.5% Δk/k. SAR Section 4.2.2.3 states that the regulating rod worth is less than 0.5% Δk/k and that assuming this total worth, a regulating rod travel speed of 78 inches per minute, and that the maximum differential worth of the rod is twice the average, the rod has a maximum reactivity addition rate of 0.054% Δk/k per second. The NRC staff notes that this reactivity addition rate is greater than the TS 3.2.2(1) maximum rate for the regulating rod and highest-worth control blade. However, in its letter dated February 16, 2021 (Ref. 101), UML stated that the actual maximum regulating rod speed is approximately 55 inches per minute. Additionally, as illustrated by SER Table 2-4 in SER Section 2.5.1, actual measured total regulating rod worths for various operational cores have only been about 0.3% Δk/k. The NRC staff notes that based on these actual regulating rod parameters, and continuing to assume that the maximum differential worth of the rod is twice the average, the rod's maximum reactivity addition rate would be much less than 0.05% Δk/k per second, such that UML could reasonably meet TS 3.2.2(1) (when the maximum differential worth of the highest-worth control blade is also considered) as well as TS 3.2.2(2). The NRC staff finds that TS 3.2.2(2), in conjunction with TS 3.2.2(1), helps limit the total worth of the regulating rod, such that it is suitable for fine control of reactivity and such that any possible rapid failure of the regulating rod could not create a rapid reactivity transient greater than the 0.6% Δk/k assumed for UML's rapid reactivity transient analyses (see SER Section

5.2). Therefore, based on the above, and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.2.2(2) is acceptable.

Renewed TS 4.2.1, "Control Blades," would state:

> Applicability:
>
> This specification applies to the surveillance requirements for operability of the reactor control blades.
>
> Objective:
>
> To ensure the control blades meet the specified limiting conditions for operation.
>
> Specifications:
>
> (1) Prior to the first reactor start-up of the day, all the control blades shall be verified as operable.
>
> (2) The control blades shall be visually inspected annually.
>
> (3) Control blade scram times and drive times, and regulating rod drive time shall be determined annually, or if maintenance or modification is performed on the mechanism.

TS 4.2.1(1) would require that prior to the first reactor start-up of the day, UML verify that all control blades are operable. The NRC staff finds that TS 4.2.1(1) helps ensure that UML verifies that the control blades (including their drive and scram mechanisms) are operable as required by LCO TS 3.2.1(1) before the reactor is operated, such that the control blades are all available to provide routine reactivity control and scram capability. The NRC staff notes that for the purposes of TS 4.2.1(1), verifying control blade operability does not include verification of parameters that are checked by other surveillances required annually by TSs 4.2.1(3) and 4.2.2(1) (i.e., scram times, drive times, and reactivity worths). Based on the above, and based on its 10 CFR 50.36(c)(3) findings for TSs on SRs in SER Section 6.7, the NRC staff concludes that TS 4.2.1(1) is acceptable.

TS 4.2.1(2) would require that UML visually inspect the control blades annually. In SAR Section 4.2.2.2, UML stated that periodic inspection of the control blades provides detection of any abnormalities in individual control blades, as well as any potential generic control blade deficiencies. In its marked-up TSs submitted as an attachment to its RAI responses dated March 5, 2019 (Ref. 71), UML stated that a TS for regulating rod inspection is not necessary because the regulating rod is not part of the reactor safety system. The NRC staff finds that TS 4.2.1(2) helps ensure that the four control blades are inspected at a frequency that will allow UML to be aware of any issues with, or degradation of, the blades that could prevent the blades from performing their intended function. The NRC staff finds that TS 4.2.1(2) or other UMLRR TSs do not require regulating rod inspection, but that this is reasonable because the regulating rod does not perform a safety function and is not required to be operable per TS 3.2.1(1) (however, the regulating rod worth is measured as required by TS 4.2.2(1) to help ensure the regulating rod meets LCO reactivity limits). The NRC staff notes that TS 4.2.1(2) does not specifically require inspection of the control blade drive and scram mechanisms, as recommended by NUREG-1537, Appendix 14.1, Section 4.2(9), but the NRC staff finds that

other required surveillances in TSs 4.2.1 and 4.2.2 help ensure that the control blade drives and scram mechanisms function properly. The NRC staff finds that TS 4.2.1(2) appropriately implements guidance in NUREG-1537, Appendix 14.1, by requiring visual inspection of control blades, and requires a surveillance interval that is conservative compared to the recommended biennial interval for control rod poison sections in NUREG-1537, Appendix 14.1, Section 4.2(9). Therefore, based on the above, and based on its 10 CFR 50.36(c)(3) findings for TSs on SRs in SER Section 6.7, the NRC staff concludes that TS 4.2.1(2) is acceptable.

TS 4.2.1(3) would require that UML determine control blade scram times, control blade drive times, and the regulating rod drive time annually, or if maintenance or modification is performed on the corresponding scram or drive mechanism. As discussed earlier in this SER section, the scram time for each control blade is required to be less than one second per LCO TS 3.2.1(2). SAR Section 4.2.2, as updated by UML's letter dated February 16, 2021 (Ref. 101), states that the maximum drive speed for the control blades is 3.6 inches per minute, the maximum drive speed for the regulating rod is approximately 55 inches per minute, and the control blades and regulating rod each have a 26 inch stroke. The NRC staff finds that 4.2.1(3) helps ensure that UML verifies that the scram times of each control blade meet the requirement of TS 3.2.1(2). The NRC staff notes that the UMLRR TSs do not include any specific corresponding LCO TSs for control blade or regulating rod drive times or speeds. LCO TS 3.2.2(1) imposes a requirement on control blade and regulating rod reactivity insertion rate, which is related to speed, but compliance with TS 3.2.2(1) is verified by separate surveillance TS 4.2.2(1). However, the NRC staff finds that the TS 4.2.1(3) surveillance of drive times helps ensure that UML verifies that the control blades and regulating rod are operating as designed to allow routine reactivity control and that the control blade and regulating rod speeds remain appropriate to allow compliance with TS 3.2.2(1), particularly if maintenance or modification is done on the drive mechanisms; therefore, the NRC staff finds that UML's TS for surveillance of drive times is appropriate and that no additional LCO TSs are necessary. The NRC staff further finds that TS 4.2.1(3) appropriately implements guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 by requiring measurement of scram and drive times and by requiring a surveillance interval that is consistent with recommended intervals in NUREG-1537, Appendix 14.1, Sections 4.2(2) and 4.2(4), and ANSI/ANS-15.1-2007, Section 4.2, items (2) and (4). Therefore, based on the above, and also based on its 10 CFR 50.36(c)(3) findings for TSs on SRs in SER Section 6.7, the NRC staff concludes that TS 4.2.1(3) is acceptable.

Renewed TS 4.2.2, "Rod Reactivity Insertion Rate," would state:

> Applicability:
>
> This specification applies to the surveillance requirements for the reactivity insertion rates.
>
> Objective:
>
> To ensure the reactivity insertion rates do not exceed the specified limiting conditions for operation.
>
> Specifications:
>
> (1) The reactivity worth and maximum reactivity insertion rate of the regulating rod and each control blade shall be determined annually or following any significant core configuration change and/or change in a control blade or the regulating rod.

A significant core configuration change is defined as a change in reactivity greater than 0.2% Δk/k.

(2) Prior to the first reactor start-up of the day, the control blade drive system shall be tested to verify only one control blade can be withdrawn at a time.

TS 4.2.2(1) would require that the total reactivity worth (i.e., integral worth) and maximum reactivity insertion rate (i.e., maximum differential worth) of the regulating rod and each control blade be determined annually or following any significant (greater than 0.2% ∆k/k) core configuration change and/or control blade or regulating rod change. The NRC staff finds that TS 4.2.2(1) helps ensure that UML performs measurements necessary to allow it to verify its compliance with LCO TSs 3.2.2(1) and 3.2.2(2) for maximum reactivity insertion rate by the most reactive control blade and the regulating rod simultaneously, and total reactivity worth of the regulating rod, respectively (as well as LCO TSs 3.1.1(1) and 3.1.1(2) for excess reactivity and SDM, although those TSs have separate SRs). The NRC staff finds that UML's designation of a "significant" change as a change greater than 0.2% ∆k/k is reasonable because it is a small reactivity change that would be unlikely to have a significant effect on control blade or regulating rod worths and because it is comparable to requirements at similar research reactors. The NRC staff notes that TS 4.2.2(1) does not specifically require verification of worths following installation of, or changes to, in-core experiments; however, the NRC staff finds that this requirement is not necessary in TS 4.2.2(1) because, as discussed in SER Sections 6.3.7, 6.6.2, and 6.6.5, the UMLRR TSs require UML to evaluate all new experiments or classes of experiments to help ensure that experiments are designed and planned appropriately, including verification that experiments will not cause applicable TSs to be violated. Although TS 4.2.2(1) does not specifically require measurement of differential worths if maintenance or modification is done on a drive mechanism, the NRC staff finds that TS 4.2.1(3) separately requires surveillance of drive times if such maintenance or modification is performed, which helps ensure that control blade and regulating rod speeds remain appropriate to allow compliance with TS 3.2.2(1). The NRC staff further finds that TS 4.2.2(1) appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 by requiring measurements of control blade and regulating rod worths and requiring surveillance intervals that are consistent with the recommended intervals in NUREG-1537, Appendix 14.1, Section 4.2(1), and ANSI/ANS-15.1, Section 4.2, item (1). Therefore, based on the above, and based on its 10 CFR 50.36(c)(3) findings for TSs on SRs in SER Section 6.7, the NRC staff concludes that TS 4.2.2(1) is acceptable.

TS 4.2.2(2) would require that prior to the first reactor start-up of the day, UML verify that no more than one of the four control blades is able to be withdrawn at a time. The NRC staff finds that TS 4.2.2(2) helps ensure that each day, before the reactor is operated, UML verifies the operability of the LCO TS 3.2.2(3)-required interlock that prevents more than one control blade from being withdrawn at a time, such that reactor operation and possible reactivity transients remain within the bounds of UML's analyses. The NRC staff also finds that by requiring verification of the operability of a TS-required interlock, and also by requiring verification that reactivity insertion rates remain within TS limits, TS 4.2.2(2) appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007. Therefore, based on the above, and based on its 10 CFR 50.36(c)(3) findings for TSs on SRs in SER Section 6.7, the NRC staff concludes that TS 4.2.2(2) is acceptable.

Based on its review of the information provided by UML in the SAR, as supplemented, regarding the UMLRR control blades and regulating rod, the NRC staff finds that UML has adequately described the control blade and regulating rod systems. UML discussed the design bases for

the control blades and regulating rod and these design bases are derived from the planned operational characteristics of the reactor. UML included information on the materials, components, and fabrication specifications of the control blade and regulating rod systems, and these descriptions provide assurance that the systems conform with the design bases and can control and shut down the reactor safety from any operating condition. UML provided and justified appropriate TSs, including maximum scram times and reactivity insertion rates, which will help ensure that changes in reactivity caused by control blade and regulating rod dynamic characteristics are acceptable. TSs for minimum SDM (see SER Section 2.5.3) will also help ensure that the control blades can shut down the reactor for any permissible operating condition. Therefore, based on the above, the NRC staff concludes that the descriptions of the UMLRR control blades and regulating rods in the SAR, as supplemented, and the associated TSs, described above, are acceptable.

2.2.3 Neutron Moderator and Reflector

As stated in SAR Section 4.2.3, the UMLRR core utilizes the light water coolant from the reactor pool as a neutron moderator. The neutron reflectors are the water-filled (or water) radiation baskets (or water reflector elements) and the graphite reflector elements. SER Figure 2-3 illustrates the radiation baskets and graphite reflector elements in the UMLRR reference core.

A graphite reflector element is a graphite block contained in a 3-inch square aluminum can. The graphite log in the reflector element extends about 3 inches above and below the active length of the adjacent fuel assemblies. The graphite reflector elements are designed to allow for thermal expansion, good heat transfer to the pool water, and expansion and gas evolution due to irradiation (up to an integrated flux of 2E21 neutrons per square centimeter). UML stated that tests of the graphite at other reactors, in environmental conditions at least as severe as those at the UMLRR, have revealed no significant changes in the graphite that the graphite reflector elements are not designed for.

SAR Section 10.2.8 discusses the water-filled radiation baskets, which are made of aluminum. In addition to serving as a neutron reflector, these can also contain in-core experiments within a central tube that runs the length of the radiation basket. The central tubes also allow coolant to flow through the radiation baskets, although the tubes can be blocked to increase coolant flow through the fuel elements. The water-filled radiation baskets are primarily used in the core periphery. The flux trap position in the core (position D5 in SER Figure 2-3) also contains a radiation basket that can hold experiments. However, a water-filled radiation basket is not typically used in this position because it increases the local power peaking in the nearby fuel elements and has a relatively large reactivity effect on the core. Other types of radiation baskets (graphite and aluminum, which are similar to water-filled radiation baskets except that they contain graphite or aluminum, respectively, in place of water in the volume between the central tube and the outer shell), which cause less power peaking, are available for use in the flux trap position. However, as discussed in SER Section 2.5.1, UML's limiting core condition analysis uses a water-filled radiation basket in the flux trap position because this represents a bounding scenario and there is no restriction on the use of a water-filled radiation basket in the flux trap position.

SAR Section 4.2.3.1 discusses the lead void elements, which are located in the five core positions A3 through A7 as seen in SER Figure 2-3. These elements provide a 1.5-inch void sandwiched between two 0.5-inch-thick lead layers, and they are housed in a 3-inch square aluminum assembly. They are not specifically intended for core neutron moderation and

reflection, but are used to help facilitate the neutronic decoupling of the reactor core from the adjacent fast neutron irradiator facility.

SAR Section 4.2.3 states that the UMLRR is designed to allow the removal and/or replacement of any neutron reflectors. Graphite or water elements can be replaced on an individual basis if a breach of the aluminum cladding (for the graphite elements) or other damage is suspected.

The NRC staff reviewed the information in the SAR, as supplemented, regarding the design of the neutron moderator and reflectors and also compared the design of the reflectors with other reflectors at similar research reactors. Based on its review, the NRC staff finds that there is assurance that the moderator and reflector systems will continue to perform as necessary and will not adversely affect safe reactor operation or shutdown. The NRC staff also finds that the reflector elements are designed to withstand the conditions in the reactor core, and if damage is suspected, the reflector elements can be replaced. Therefore, based on the above, the NRC staff concludes that the descriptions of the neutron moderator and reflectors in the SAR, as supplemented, are acceptable.

2.2.4 Neutron Startup Source

As discussed in SAR Section 4.2.4, UML possesses (as authorized by LC 2.B.3 in the current UMLRR license and LC 2.B.3.a in the renewed license) a 5 curie (Ci) americium-beryllium (Am-Be) neutron source for routine startup of the reactor. This source is located inside a water-filled radiation basket seated in the reflector region of the core grid, in core position G5. The source is normally removed for reactor operation above 10 kWt to ensure that excess fission product production does not occur within the source. A handling line allows the neutron source to be inserted and removed, and this process is controlled via procedure.

SAR Section 9.5.2 states that UML also possesses (as also authorized by LC 2.B.3 in the current UMLRR license and LC 2.B.3.a in the renewed license) an antimony-beryllium (SB-Be) neutron source (up to 10 Ci) as a spare start-up source. This source has not been used or activated, and it is kept in storage in the reactor building. As discussed in SER Section 1.10, in conjunction with license renewal, UML also requested NRC approval to possess plutonium-beryllium (Pu-Be) neutron sources under Facility Operating License No. R-125 for the UMLRR, but UML stated that these sources would be used for detector checks and calibrations, not for reactor startup.

The NRC staff reviewed the information in the SAR, as supplemented, regarding neutron startup sources and finds that the sources are comparable to those used in other similar licensed non-power reactors. The NRC staff finds that the sources are adequate and appropriate for the intended use, will help ensure an acceptable count rate on reactor startup instrumentation and allow for monitored startup of the reactor. The NRC staff finds that avoiding use of the sources in the core at high power will help ensure that the sources will not be significantly degraded by reactor operation. Therefore, based on the above, the NRC staff concludes that UML's descriptions of its neutron startup sources in the SAR, as supplemented, are acceptable.

2.2.5 Core Support Structure

According to SAR Section 4.2.5, the reactor core support structure consists of a suspension frame bolted to, and suspended from, a movable bridge. SER Figure 1-1 includes an illustration of the reactor bridge, and SER Figure 2-2 shows the core suspension frame structure. The bridge spans the entire width of the reactor pool. The bridge consists of two separate sections of structural framework set horizontally, one above the other, and supported on each side of the pool by a two-wheel, rail-mounted truck assembly. The upper section of the bridge is designed to allow easy access to the reactor core, while the lower section supports the weight of the suspension frame structure and the core. The truck assembly allows the reactor bridge and the attached suspension frame to be moved, such that the reactor core can be positioned at the desired location within the reactor pool. The bridge is moved using a hand crank, and a brake assembly allows the bridge to be secured in the desired position.

SAR Section 4.2.5.1 states that the suspension frame is an aluminum rectangular column built of four-square corner posts forming a rigid structure. Neutron detectors for monitoring power and period are located in three of the suspension frame corner posts, and the startup counter is located in the fourth corner post. A locating plate, which spans the upper end of the suspension frame, serves as a mounting for the control blade and regulating rod drives. The core box is attached to the lower end of the suspension frame. Cross braces and stiffeners are utilized to provide structural rigidity and alignment in the upper half of the suspension frame, and coolant flow channels (the inlet channel or downcomer plenum and the outlet channel or riser plenum) provide this function in the lower half of the suspension frame. Stiffeners are provided on three sides of the frame while the fourth side is open to provide access to the core and allow manipulation of fuel assemblies during refueling operations.

The core suspension system is designed primarily to support the reactor core, to provide a means for moving it along the major axis of the pool, and to secure the core in any desired operating or service position, including one reproducible position (i.e., the high-power position) with respect to the coolant header and experimental facilities (there are also connections to allow the reactor to be operated with forced convection cooling in a specific position in the low-power (bulk) section of the pool, but UML does not typically use this configuration). As discussed in SER Section 2.5.3, there are reactor scrams for bridge movement (and primary coolant piping misalignment, when applicable) during reactor operation. The core suspension is also arranged to ensure positive alignment of the shafts between the control blade and regulating rod drive mechanisms and their respective driven control elements. As stated in SAR Section 4.2.2.1, large clearances are provided in guide bearings of the control blade shafts to minimize friction and possible binding, and as stated in SAR Section 4.2.2, a 0.125 inch clearance is provided within the control blade shrouds around the control blades themselves.

The aluminum core box, or core grid support structure, which includes the grid plate and grid box, is described in SAR Section 4.2 and UML's response to RAI-14.3.33 (Ref. 71). The core grid plate is installed at the bottom of the grid box, whose four sides are enclosed. The grid plate is a 9 by 7 grid, with the four corner positions occupied by the core suspension frame corner posts. The top of the grid box is open to the pool (via the unenclosed inlet plenum, which is directly above the core), and the bottom connects to an enclosed plenum (the riser plenum or core outlet channel) for coolant flow. The core box is illustrated in SER Figure 2-1. The confinement of the fuel and reflector elements within the core box confines the flow of water between and within the elements.

The NRC staff reviewed the information in the SAR, as supplemented, regarding the design of the core support structure, and also compared the design of the core support structure with those at similar research reactors. The NRC staff finds that UML adequately described the core support structure, including its design bases, and all functional and safety-related design bases can be achieved by the core support structure design. The NRC staff finds that the core support structure will help adequately position and align the core and fuel elements in a stable and reproducible configuration and includes acceptable guides and supports for core components other than fuel. The NRC staff further finds that the core support structure will help ensure sufficient coolant flow to the fuel and that the support structure is composed of materials, such as aluminum, that are compatible with the coolant and radiation environment. Therefore, based on the above, the NRC staff concludes that UML's descriptions of its core support structure in the SAR, as supplemented, are acceptable and that the core support structure is acceptable for the continued safe operation of the UMLRR.

2.3 Reactor Pool

SAR Section 4.3 describes the reactor pool. The reactor pool is illustrated in the reactor layout diagram in SER Figure 1-1. The reactor pool is divided into two interconnected sections: a high-power section (the stall section) and a low-power section (the bulk section). The entire reactor pool is approximately 31 feet (9.4 meters) deep by 32 feet long and holds approximately 76,000 gallons of high purity deionized water. The clean water minimizes corrosion and prevents the activation of impurities. As shown in SER Figure 1-1, the walls of the stall section of the pool are sloped such that the stall pool is narrower at the bottom where the core is located than near the top. The pool walls are aluminum-lined and are constructed of heavy aggregate concrete and ordinary concrete as necessary to provide adequate biological shielding. Fuel storage racks are attached along the inner surface of the pool liner.

An aluminum divider gate is available to isolate the two sections of the pool such that either pool section may be independently drained. A rubber gasket around the edges of the gate provides a watertight seal. The gate is moved into position using the overhead crane and two personnel. The reactor pool may be partially drained into an underground retention tank located adjacent to the reactor building. Drainage to this tank would occur from discharge lines connected to the primary coolant system or primary water cleanup system (i.e., not directly through a pool penetration).

The penetrations to the pool are seal-welded to the 0.25-inch-thick aluminum pool lining. Although each pool section is equipped with a set of primary cooling connections for operation above 100 kWt, only the stall pool connections are currently in service and used. Three beamports are visible in SER Figure 1-1. The three beamports visible in the figure have been removed to allow the UMLRR's fast neutron irradiator facility to be installed, but they remain in the figure for illustrative purposes to depict the symmetrically located remaining three beamports on the opposite side of the reactor core.

Penetrations in the pool wall are summarized in SER Table 2-2, which is adapted from SAR Table 4-2. SAR Chapter 5, including SAR Table 5-1, also provides information on pool penetrations (SAR Table 5-1 notes two additional core penetrations located at or just below the pool surface, the pool gutter and the hot cell transfer port fill line). There are no penetrations made in the pool floor or located below the beam ports located at core centerline (approximately 25 feet below the pool surface). As discussed in SAR Section 10.2.3, there is an additional 2 by 2 foot opening in the concrete portion of the pool wall, at core centerline (approximately 25 feet below the pool surface), associated with the gamma cave (not listed in SER Table 2-2);

however, the aluminum pool lining is continuous in this area and is also reinforced by a strengthened aluminum plate.

Table 2-2 Pool Penetrations

Reactor Pool Penetrations					
Penetration	Section of Pool	Number	Approximate Depth (feet)	Dimensions	Method of Isolation
Primary piping connections	Both	4	13	10-inch diameter	Anti-Siphon Break Valves (discussed in SAR Chapter 5)
8” Beam Port	Stall	2 (1 in use)	25	8-inch diameter	(1) Welded End Cap. (1) Bolted Closed
6” Beam Port	Stall	4 (2 in use)	25	6-inch diameter	(2) Welded End Cap. (2) Bolted Closed
Pneumatic Transfer Tubes	Stall	2 (1 in use)	5	2.5-inch diameter	Sealed End Cap
Pneumatic Transfer Tubes	Stall	2 (0 in use)	3	6-inch diameter	Bolted Closed
Pool Skimmer Inlet	Stall	1	2	2-inch diameter	Turn line valves to closed
Pool Skimmer Outlet	Stall	1	5	2-inch diameter	Turn line valves to closed
Medical Embedment	Bulk	1	25	3 feet by 3 feet	Bolted Closed
Hot Cell Transfer Port	Bulk	1	11	2 feet by 2 feet	Water Tight Gates

As discussed in SAR Sections 7.4.2.3 and 7.4.3.2, two diverse, independent sensors measure the pool water height above the core and will provide an alarm and scram if the pool water height drops 6 inches below its normal full level. As discussed in SAR Section 2.4.1, as supplemented (Ref. 44), if the reactor pool were to have a significant leak, the water would be confined to the reactor building, and even if water were released to the ground or sanitary sewer in an uncontrolled manner, the tritium concentrations would not be of radiological concern, based on the normal measured tritium concentration. As discussed in the 2017-2018 UMLRR annual report (Ref. 10.m), UML has successfully repaired past reactor pool leaks. In these previous pool leaks, leaked coolant water collected in the reactor building sump. Water that collects in this sump is pumped to the liquid waste holding tanks in the Pinanski building; water in these tanks is analyzed for radioactivity prior to being discharged to the sanitary sewer. TSs 3.3 and 4.3, which are discussed and found acceptable below, require routine testing of primary coolant water to help ensure that radioactivity in the water remains below specified limits and any issues can be addressed appropriately. As discussed in SAR Section 7.7, radiation monitors would also help detect any significant release of radioactive material to the primary coolant in the reactor pool.

UML’s analysis of a potential loss of coolant accident (LOCA) at the UMLRR, which demonstrates that no credible LOCA would cause fuel failure or result in unacceptable dose rates due to direct radiation shine from the exposed core, is discussed and found acceptable in SER Section 5.3.

Renewed TS 3.3, “Reactor Coolant Systems,” would state:

Applicability:

This specification applies to the reactor primary coolant system water quality requirements and pool configuration.

Objective:

The objectives are to minimize corrosion and radioactive contaminants, and to ensure the full volume of pool water is available in the event of a loss of coolant accident.

Specifications:

(1) The conductivity of the pool water shall be maintained at ≤5 µmho/cm when averaged over a period of 1 month.

(2) The pH of the pool water shall be maintained between 5.0 and 7.5 when averaged over a period of 1 month.

(3) The concentrations of radionuclides in the pool water shall be no greater than the values presented for water in 10 CFR Appendix B to Part 20 Table 2.

(4) The reactor shall not be operated with the pool divider gate in position to separate the bulk pool and the stall pool.

TSs 3.3(1) and 3.3(2) would require that the conductivity of the pool water averaged over 1 month shall be less than or equal to 5 micromhos per centimeter (µmho/cm) and that the pH of the pool water averaged over 1 month shall be between 5.0 and 7.5. SAR Section 4.3 states that the UMLRR pool water is kept clean (i.e., high purity and low conductivity) to minimize corrosion and activation of impurities. In its basis for TS 3.3, UML stated that the purpose of its pH limit is also to minimize corrosion. In response to RAI-14.3.15 (Ref. 71) and its letter dated September 30, 2020 (Ref. 98), UML stated that the TSs have required a conductivity of less than 5 µmho/cm (when averaged over 1 month) since 1974 and that there have been no reactor fuel or cobalt-60 (Co-60) source corrosion problems observed during this time. UML stated that conductivity and pH may occasionally and briefly exceed the specified TS values following regeneration of the water purification system, but that operating experience (also since 1974) has shown that the conductivity and pH values return to the specified ranges typically within a few days post-regeneration. UML stated that allowing an averaging time allows time for the conductivity and pH to return to normal levels following regeneration of the primary water demineralizer.

The NRC staff finds that the conductivity and pH limits in TSs 3.3(1) and 3.3(2) help minimize corrosion of fuel and other reactor components and also help minimize activation of impurities. The NRC staff has previously found (Ref. 63) that if the conductivity of open-pool research reactor pool water is controlled such that it remains below 5 µmho/cm, the pool water pH will remain between 5.6 and 5.8, but finds that TS 3.3(2) provides additional redundancy to help ensure that the pool pH is maintained in an appropriate range. The NRC staff also finds that controlling the pool conductivity and pH can help minimize any release of iodine from the pool during a potential fuel failure scenario (see SER Section 5.1). In addition, the NRC staff finds that allowing pool conductivity and pH to be based on a 1-month average is reasonable,

because it will allow for relatively brief fluctuations that may occur after demineralizer maintenance while ensuring that the long-term average conductivity and pH remain at an appropriate level. The NRC staff further finds that TSs 3.3(1) and 3.3(2) appropriately implement guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 by establishing pool water conductivity and pH limits. Additionally, TSs 3.3(1) and 3.3(2) include the specific conductivity and pH limits recommended in NUREG-1537, Appendix 14.1, Section 3.3(9). Therefore, based on the above, and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TSs 3.3(1) and 3.3(2) are acceptable.

TS 3.3(3) would require that the concentrations of radionuclides in the bulk pool water shall be no greater than the water effluent concentration limits in 10 CFR Part 20, Appendix B, Table 2, Column 2. In its basis for TS 3.3, UML stated that radionuclide analysis of the pool water allows for early determination of any significant buildup of radioactivity from operation of the reactor or Co-60 sources. The NRC staff notes that the limits in 10 CFR Part 20, Appendix B, Table 2, Column 2, are for releases of liquid radionuclide effluents directly to the environment (for example, from leaks or evaporation, rather than via a sanitary sewer system where it would be diluted). In its response to RAI-14.3.17 (Ref. 71), UML stated that given that there would be further dilution in the event of any leak from the pool water (i.e., primary coolant) to the secondary coolant system (secondary coolant is normally released directly to the environment by evaporation, as discussed in SAR Section 5.3), there is no need for a separate TS for secondary coolant radioactivity limits. In its response to RAI-14.3.16 (Ref. 71), UML stated that the intent of TS 3.3(3) is not to require that the total concentration of all radionuclides in the pool be within 10 CFR Part 20, Appendix B, Table 2, Column 2 limits (i.e., as determined using a "sum of ratios" calculation), but to require that the concentration of each radionuclide be within its individual limit independent of any other radionuclides; UML also stated that given the additional dilution in the secondary system, a requirement that the total concentration of all radionuclides in the pool be within 10 CFR Part 20, Appendix B, Table 2, Column 2 limits is not necessary.

The NRC staff finds that the limit in TS 3.3(3), in conjunction with the SRs in TS 4.3, and trending and analysis of radionuclide data (consistent with standard industry practice), helps ensure that radionuclides in the pool are maintained at a low level and that UML will be able to detect issues that could release radioactive material to the pool such as failed fuel, experiment malfunctions, or a Co-60 source leak. The 10 CFR Part 20, Appendix B, Table 2, Column 2, limits for iodine-131 and cesium-137 (likely fission product indicators of fuel failure) are 1×10^{-6} microcuries per milliliter (µCi/mL). Although the Table 2, Column 2, Co-60 limit (and thus the proposed renewal TS limit) of 3×10^{-6} µCi/mL is higher than the current (i.e., prior to this renewal) Co-60 TS limit of 1×10^{-6} µCi/mL, the NRC staff finds that the 3×10^{-6} µCi/mL limit is sufficient to help ensure that UML can detect any Co-60 source leak. Specifically, UML must continue to monitor or sample the Co-60 concentration in the pool at least weekly in accordance with TS 4.3(3) (which is discussed and found acceptable below), and the monitoring or sampling will help ensure that UML is able to detect and evaluate any trends in the Co-60 concentration that could indicate a source leak, even if the leak is minor and does not cause a 3×10^{-6} µCi/mL (or 1×10^{-6} µCi/mL) pool concentration to be reached. The NRC staff finds that UML's approach of basing compliance with the TS 3.3(3) limit on individual radionuclides reaching the limits instead of using a "sum of ratios" calculation is reasonable because the primary intent of the TS is for leak detection and not to make the water safe for direct release to the environment (given that pool water is not normally released directly to the environment and, as discussed above, any leakage from the pool is normally collected in the reactor building sump) and the limit provides a threshold for monitoring for failures that could release radionuclides to the pool. The NRC staff further finds that given the conservative pool water radioactivity limit in TS 3.3(3), given the

relatively limited number of different radionuclides that are likely to be released to the pool in significant quantities during most potential failures, and given the substantial dilution that would likely occur in the event of any leak to the secondary system, a TS for secondary coolant radioactivity monitoring is not necessary. The NRC staff also finds that, by limiting radioactivity in the primary coolant, TS 3.3(3) appropriately implements guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007. Therefore, based on the above, and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.3(3) is acceptable.

TS 3.3(4) would require that when the reactor is operating, the pool divider gate shall not be in position to separate the bulk and stall sections of the reactor pool. As discussed in SER Section 5.3, UML's LOCA analysis assumes that the pool has a surface area of 350 square feet, based on the combined size of both the bulk and stall sections of the pool. The larger pool surface area (and volume) available if the reactor is not enclosed in one section of the pool by the divider gate increases the time it would take to drain the pool during a LOCA. As discussed in SER Section 5.3, UML determined that in order to avoid exceeding a SL following a LOCA, the reactor core needs to remain covered with water for at least 3,947 seconds following reactor operation to allow for adequate reduction in decay heat. During the 2020-2021 audit, as documented in the NRC staff's audit report dated December 17, 2020 (Ref. 86), UML explained why it is not necessary for TS 3.3(4) to be applicable following reactor operation when the reactor is in the bulk pool. Per TS 3.8, in order for UML to open a beam tube plug when the corresponding beam tube shutter is also open, the reactor must not only be shut down, but also moved to the bulk pool; when the reactor is in the bulk pool, having the divider gate separating the bulk and stall portions of the pool would not be an issue from the standpoint of keeping the reactor core covered following a beam tube break, because the beam tubes are in the stall pool, and a damaged beam tube would only cause water to drain from the stall pool. In its letter dated September 30, 2020 (Ref. 98), UML states that it is also not necessary for TS 3.3(4) to prohibit the pool divider gate from being in place following reactor operation when the reactor is in the stall pool. UML states that a LOCA due to a sheared beam tube when the reactor is in the stall pool is highly unlikely, given the protection provided by the reactor bridge when the reactor is in the stall pool, which would make it extremely difficult for a falling object to damage a beam tube. Additionally, TS 3.8(4), which is discussed and found acceptable in SER Section 6.3.8, requires the beam tube shutters to be closed before the divider gate is placed in position (UML stated that the only purposeful operation in which the pool divider gate would be closed while the reactor is in the stall pool would be draining of the bulk pool, e.g., for maintenance). UML stated that, even in the unlikely event a beam tube is damaged while the reactor is closed in the stall pool, having the shutters closed would mitigate the leak (see SER Section 5.3).

The NRC staff finds that, by prohibiting the reactor from being enclosed in either section of the pool during reactor operation, TS 3.3(4) helps ensure that UML operates the reactor consistent with the assumptions of its LOCA analysis. Although placing the divider gate in place following reactor operation could, potentially, cause the pool to drain more quickly than the 3,947 seconds that UML determined the core needs to remain covered with water following extended full-power reactor operation (see SER Section 5.3), the NRC staff finds that it is not necessary for TS 3.3(4) to prohibit the gate from being in place following operation. This is because TS 3.8 separately requires the reactor to be in the bulk pool when both a beam tube shutter and its corresponding plug are open, and having the gate in place would protect the reactor from any LOCA caused by a beam tube rupture. In addition, TS 3.8 requires that before the gate is closed when the reactor is in the stall pool, UML must close all beam tube shutters to mitigate any (unlikely) LOCA that could occur due to a beam tube rupture. (Although TS 3.8 prohibits having both a shutter and its corresponding plug open when the reactor is in the stall pool, it

also allows a plug to be modified to have an opening smaller than the normal area of the plug. Therefore, the NRC staff notes that if the shutter were open and a beam tube were damaged, water could drain through the opening in the plug, even if the plug were in place.) The NRC staff also finds that, because the pool gate must be open during reactor operation and it would take some time for UML to close the pool gate following reactor shutdown, the core decay heat (and associated risk of the SL being reached) would be reduced at any time the gate is closed, compared to the decay heat immediately following reactor operation. As discussed in SER Section 5.3, UML could help mitigate a LOCA by moving the reactor to one end of the pool (that does not have a leak), and closing the divider gate, after the reactor is shutdown; the NRC staff also finds that, by not prohibiting the divider gate from being put in place following reactor operation, TS 3.3(4) appropriately allows this mitigating action to be taken if needed, including less than 3,947 seconds after reactor shutdown. Based on the above, and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.3(4) is acceptable.

Renewed TS 4.3, "Coolant Systems," would state:

> Applicability:
>
> This specification applies to verifying the quality of the primary coolant system water and the pool configuration.
>
> Objective:
>
> To ensure the primary coolant system limiting conditions for operation are met.
>
> Specifications:
>
> (1) The conductivity and pH of the pool water shall be measured weekly.
>
> (2) The radioactivity in the pool water shall be analyzed monthly.
>
> (3) The pool water shall be either monitored continuously for Co-60 or sampled once per week.
>
> (4) Prior to the first reactor start-up of the day, the pool divider shall be verified as open.
>
> (5) Prior to placing the pool divider gate in position to separate the bulk pool and stall pool, when the reactor is in the stall pool, the beam port shutters shall be verified to be in the down (closed) position.

TS 4.3(1) would require that the conductivity and pH of the reactor pool water be measured weekly. The NRC staff finds that TS 4.3(1) helps ensure that UML makes measurements to verify compliance with LCO TSs 3.3(1) and 3.3(2), such that pool conductivity and pH remain at appropriate levels. The NRC staff also finds that TS 4.3(1) appropriately implements guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 because it requires measurements to verify compliance with LCOs on pool chemistry and because it requires a surveillance interval that is consistent with the guidance in NUREG-1537, Appendix 14.1, Section 4.3(6), and ANSI/ANS-15.1-2007, Section 4.3, item (6). Therefore, based on the above, and based on its

10 CFR 50.36(c)(3) findings for TSs on SRs in SER Section 6.7, the NRC staff concludes that TS 4.3(1) is acceptable.

TS 4.3(2) would require that UML perform a monthly analysis of the radioactivity in the pool water to verify compliance with LCO TS 3.3(3) and to detect potential leakage of radioactive material into the pool from any source, including fuel. TS 4.3(3) would separately require that UML perform pool water sampling for Co-60 once per week, or continuously monitor the pool water for Co-60, to verify compliance with LCO TS 3.3(3) for the Co-60 radionuclide specifically and to detect any potential leakage from the Co-60 sources in the pool. In its response to RAI-14.4.15 (Ref. 71), UML stated that in its proposed license renewal TS 4.3(2), it increased the frequency for primary coolant radioactivity monitoring from weekly (in the current, i.e., pre-renewal, TS requirements) to monthly, but it maintained a separate weekly surveillance for Co-60 monitoring (in renewal TS 4.3(3)). Additionally, UML stated in its RAI-14.4.15 response that for Co-60 monitoring, renewal TS 4.3(3) provides flexibility to either perform weekly sampling (consistent with the current, i.e., pre-renewal, TS requirements), or conduct continuous monitoring, provided such continuous monitoring would provide the necessary sensitivity to determine if a Co-60 source were leaking (the NRC staff notes that any Co-60 measurement method would also need to have the sensitivity to properly demonstrate compliance with the limit specified in TS 3.3(3)). The NRC staff notes that TS 4.3(3) requires surveillances to verify TS 3.3(3) limits for Co-60 concentration only, and TS 4.3(2) requires surveillances to verify that the concentrations of all radionuclides meet TS 3.3(3); however, TSs 4.3(2) and 4.3(3) do not necessarily require that UML perform individual analysis for any radionuclide, if gross analyses are sufficient to demonstrate compliance with TS 3.3(3). The NRC staff notes that the weekly surveillance interval for Co-60 measurements in TS 4.3(3) is less conservative than the 10 CFR 36.59(b) requirement that water contamination checks for pool irradiators be conducted each day the irradiator operates. However, the NRC staff previously determined (Ref. 70) that because UML's Co-60 sources are located in a research reactor pool and the sources and gamma cave are part of a research reactor facility, the UMLRR is not subject to the regulations in 10 CFR Part 36, "Licenses and Radiation Safety Requirements for Irradiators."

The NRC staff finds that the TS 4.3(3) weekly surveillance interval for Co-60 monitoring is consistent with the interval approved by UMLRR License Amendment No. 5, issued January 15, 1982 (Ref. 74), which originally authorized UML to use the Co-60 sources as an irradiation source in the UMLRR pool. The NRC staff also finds that although weekly analyses have been used to evaluate for Co-60 leaks in the past, UML's use of an appropriate system for continuously monitoring the pool water for Co-60, if used, as would be allowed by renewal TS 4.3(3), would provide additional conservatism in helping detect a Co-60 source leak. Additionally, in its response to RAI-14.4.15 (Ref. 71), UML stated that since the UMLRR Co-60 sources were obtained, no source corrosion problems (that could potentially result in source leaks) have been observed. Therefore, the NRC staff finds that the TS 4.3(3) surveillance intervals for monitoring Co-60 in the pool (in conjunction with the TS 3.3(3) limit and UML's evaluation of pool radioactivity trends consistent with standard industry practice, as discussed earlier in this SER section) is sufficient to continue to help ensure early detection of any possible Co-60 source leak at the UMLRR. The NRC staff also finds that the monthly surveillance interval in TS 4.3(2) is conservative relative to the guidance in NUREG-1537, Appendix 14.1, Section 4.3(4), which recommends quarterly primary coolant radioactivity analyses, and the guidance in ANSI/ANS-15.1-2007, Section 4.3, item (4), which recommends that reactor coolant be analyzed for radioactivity annually, if necessary. The NRC staff finds that TS 4.3(2) is sufficient (in conjunction with TS 3.3(3) and UML's evaluation of pool radioactivity trends consistent with standard industry practice, as discussed earlier in this SER section) to help ensure early detection of leaks of radioactive material from sources other than Co-60 into the

pool. Additionally, the NRC staff finds that TSs 4.3(2) and 4.3(3) appropriately implement the overall guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 because they establish requirements to monitor primary coolant radioactivity. Therefore, based on the above, and based on its 10 CFR 50.36(c)(3) findings for TSs on SRs in SER Section 6.7, the NRC staff concludes that TSs 4.3(2) and 4.3(3) are acceptable.

TS 4.3(4) would require that prior to the first reactor start-up of the day, the pool divider shall be verified as open to ensure that LCO TS 3.3(4) is met when the reactor is operated. The NRC staff finds that TS 4.3(4) helps ensure that before any reactor operation, UML verifies that the pool divider gate is not in place to separate the bulk and stall pools and that TS 3.3(4) is met such that reactor operation is consistent with the assumptions of UML's LOCA analysis (see SER Section 5.3). Therefore, based on the above, and based on its 10 CFR 50.36(c)(3) findings for TSs on SRs in SER Section 6.7, the NRC staff concludes that TS 4.3(4) is acceptable.

TS 4.3(5) would require that when the reactor is in the stall pool, UML must verify that all beam port shutters are closed before it places the pool divider gate in place to separate the bulk pool and stall pool to ensure that LCO TS 3.8(4) is met when the divider gate is put in place while the reactor is in the stall pool. In its basis for TS 4.3, UML stated that verifying the beam port shutters are closed when the reactor is isolated in the stall pool ensures that the beam ports will be isolated (i.e., blocked) to inhibit stall pool drainage if a beam tube rupture occurs. The NRC staff finds that TS 4.3(5) helps ensure that before the divider gate is put in place while the reactor is in the stall pool, UML verifies that the beam tube shutters are closed such that TS 3.8(4) will be met and such that a potential (but unlikely) LOCA scenario involving rapid drainage of the stall pool and potentially exceeding UML's analyzed LOCA will be avoided (see discussion of TS 3.3(4) in this SER section, and SER Sections 5.3 and 6.3.8). Therefore, based on the above, and based on its 10 CFR 50.36(c)(3) findings for TSs on SRs in SER Section 6.7, the NRC staff concludes that TS 4.3(5) is acceptable.

Renewed TS 5.2, "Reactor Coolant System," would state:

> Applicability:
>
> These specifications apply to the reactor pool and primary coolant system.
>
> Objective:
>
> To specify the major design features of the reactor coolant system.
>
> Specifications:
>
> The reactor coolant system shall consist of the following:
>
> (1) An open pool containing approximately 75,000 gallons of demineralized water (H_2O).
>
> (2) A single primary cooling loop containing a heat exchanger, a circulation pump, and various valves.
>
> (3) All materials associated with the reactor coolant system shall be aluminum alloys, except for the heat exchanger which shall be comprised of stainless steel,

and small non-corrosive components such as gaskets, filters, and valve diaphragms.

TSs 5.2(1), 5.2(2), and 5.2(3) would describe the reactor coolant system. TSs 5.2(1) and 5.2(2) specify the basic components of the coolant system, including an open pool containing approximately 75,000 gallons of water (similar to the approximately 76,000 gallon pool capacity discussed in SAR Section 4.3) and a single primary cooling loop (connected to the pool) that contains a heat exchanger, primary coolant circulation pump, and various valves. TS 5.2(3) would require that all materials associated with the reactor coolant system be aluminum alloys, except for the heat exchanger (stainless steel) and small non-corrosive components. SAR Section 3.5.7 states that primary coolant system components are aluminum or stainless steel, and SAR Section 5.2 states that the primary coolant system includes the heat exchanger. The NRC staff finds that TSs 5.2(1), 5.2(2), and 5.2(3) impose requirements on the design features of the reactor coolant system that are consistent with information in the SAR, as supplemented. The NRC staff also finds that TSs 5.2(1) and 5.2(2) help ensure that design features of the primary coolant system are maintained to be consistent with assumptions in UML's thermal-hydraulic and accident (e.g., LOCA) analyses. The NRC staff finds that TS 5.2(3) helps ensure that materials in contact with primary coolant are constructed of appropriate materials that minimize the possibility of corrosion and/or significant primary coolant contamination or degradation. The NRC staff also finds that TSs 5.2(1), 5.2(2), and 5.2(3) appropriately implement the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 by describing important design features of the reactor coolant system. Based on the above, and based on its 10 CFR 50.36(c)(4) findings for TSs on design features in SER Section 6.7, the NRC staff concludes that TSs 5.2(1), 5.2(2), and 5.2(3) are acceptable.

The NRC staff reviewed the information in the SAR, as supplemented, regarding the design of the reactor pool, and also compared the design of the reactor pool with those at similar research reactors. The NRC staff finds that UML adequately described the reactor pool. The NRC staff finds that the pool is robustly constructed using materials, such as aluminum and concrete, that will help ensure the continued integrity of the pool and that are compatible with the coolant and radiation environment. The pool penetrations are designed to help prevent loss of coolant from the pool and to ensure safe reactor operation. The design of the reactor pool, the low levels of radioactivity in the pool water, and equipment such as pool level monitors would help ensure that if significant leakage from the reactor pool did occur, UML would be able to detect it and there would be no exposure to contaminated coolant that would cause 10 CFR Part 20 limits to be exceeded. Therefore, based on the above, the NRC staff concludes that UML's descriptions of its reactor pool in the SAR, as supplemented, are acceptable, and that the reactor pool is acceptable for the continued safe operation of the UMLRR.

2.4 Biological Shield

The biological shield is described in SAR Sections 4.4 and 11.1.5.1. The biological shields for the reactor are the pool water (approximately 25 feet (7.6 meters) of water above the core centerline) and the thick, reinforced concrete walls of the pool structure. The reactor pool structure was originally designed for a reactor that would operate at 5 MWt, while the UMLRR is only licensed to operate at 1 MWt. The concrete and water combination ensures that limited or no radiation fields exist at the periphery of the pool wall structure, aside from experimental facilities such as beam ports and thermal column where such fields are expected and are utilized for experiments. Normal external dose rates in most non-experimental areas of the normally-occupied portions of the reactor confinement building are less than 1 millirem (mrem) per hour. Dose rates in certain locations near ongoing experiments (e.g., certain beam tube

experiments) may be greater than this at times. However, in such cases, local shields, such as cement blocks and/or lead bricks, are utilized as required to reduce dose rates to acceptable levels.

SAR Chapter 10 discusses design features of beam port facilities and other experimental facilities that help prevent streaming or scattered radiation that could increase dose rates within the reactor building. These features include barrier walls to provide shielding around experimental facilities, labyrinthine entrances to experimental facilities, instrumentation conduit lines that are designed to not provide a direct streaming path, and shielding plugs that are wider than the ports they shield.

As discussed in SAR Section 11.1.5.1, gamma radiation levels in the holdup tank area of the pump room in the confinement building basement are normally high (several rem per hour) during forced flow operation at 1 MWt, but access to the pump room is restricted and appropriate supervision is required for all personnel operations in this area because of the dose rates there. SER Sections 4.1.2, 4.1.3, and 4.1.5 discuss UML's radiation protection program and administrative controls to minimize radiation doses at the UMLRR, including the as low as is reasonably achievable (ALARA) program.

During operation at 1 MWt, normal radiation levels above the reactor pool surface and beneath the reactor bridge are less than 10 mrem per hour.

Additional shielding between the interior of the reactor building and the external environment is provided by the thick concrete shadow shield that lines the confinement building walls. UML states that the overall design of the reactor, including its pool, pool walls, and confinement building, are adequate to provide more than sufficient shielding to ensure that the dose requirements in 10 CFR Part 20 are met.

The NRC staff reviewed the information in the SAR, as supplemented, regarding the design of the biological shield, and also compared the design of the biological shield with those at similar research reactors. The NRC staff finds that UML adequately described the biological shield. The NRC staff also finds that the descriptions in the SAR provide reasonable assurance that the shield designs (along with UML's radiation protection program and administrative controls to minimize radiation doses at the UMLRR, as discussed in SER Sections 4.1.2, 4.1.3, and 4.1.5) will limit exposures from the reactor and reactor-related sources of radiation, including streaming radiation from experimental facilities, so as not to exceed the limits of 10 CFR Part 20 and the guidelines of the UMLRR ALARA program. Therefore, based on the above, the NRC staff concludes that UML's descriptions of its biological shield in the SAR, as supplemented, are acceptable and that the biological shield is acceptable for the continued safe operation of the UMLRR.

2.5 Nuclear Design

SAR Section 4.5, as supplemented by the response to RAI-4.1 (Ref. 23), describes UML's neutronics analyses. The neutronics analyses establish part of the design bases for UML's thermal-hydraulic analyses discussed in SAR Section 4.6, as supplemented, and the transient and safety analyses discussed in SAR Chapter 13, as supplemented.

To demonstrate that its neutronics models are suitably predictive, UML presented calculations demonstrating the ability of the models to reproduce results from one another and to reproduce actual measurements from normal core operating configurations. In the SAR, as supplemented,

UML refers to one of these normal core configurations, which is a beginning-of-life (BOL) configuration (i.e., a configuration consisting of fresh fuel), as its “reference core” or its “BOL reference core” configuration. However, in this SER, to prevent confusion with the “reference core condition” as defined in the UMLRR TSs, the NRC staff refers to this BOL normal core configuration as the OCC. UML also refers to the OCC as the “M-2-5” or “M-2-5 BOL” configuration in the SAR, as supplemented. The OCC consists of all silicide fuel and is illustrated in SER Figure 2-3. In response to RAI-4.1(a) and (b), UML discussed two other normal core configurations, in addition to the OCC, for which it compared model outputs and actual measurements to help further demonstrate the validity of its neutronics models. The first of these is the initial startup core following the HEU to LEU conversion of the UMLRR. This configuration, which UML refers to as the “M-1-3” configuration, is a BOL configuration consisting of all silicide fuel. The second of these configurations is the M-2-5 configuration at 50 megawatt-days (MWD) burnup. (The NRC staff notes that in some instances in the SAR, as supplemented, UML also uses “reference core” to refer to non-BOL M-2-5 cores, i.e., M-2-5 cores with some amount of burnup.)

The OCC and other normal core configurations discussed above do not represent a worst-case configuration, i.e., the configuration that would maximize the power peaking within the core. UML investigated many other possible core configurations besides the OCC to develop an LCC, which is the core configuration that would produce the greatest peaking factors of any allowable core configuration. The LCC represents a core configuration that would not typically be used by the licensee, but could be used within the bounds of the UMLRR TSs. As discussed in SER Section 2.5.1, UML’s LCC includes both silicide and aluminide fuel, arranged to maximize peaking. UML uses the same methods and one of the same codes that it validated for its OCC to analyze its LCC in order to help provide confidence in the predictive results of the LCC analysis.

As discussed in SAR Section 4.5.1.1, the neutronics calculations performed for the reference core include 3-D models using both the VENTURE diffusion theory code and the Monte Carlo N-Particle Transport (MCNP) code. The two-group cross sections for VENTURE are generated using a variety of modules from the SCALE package. In general, VENTURE is used to obtain the power and few-group flux distributions within the UMLRR and for most routine reactivity evaluations, including control blade worths as well as excess reactivity and criticality estimations. In addition, VENTURE is used for most of the fuel depletion (burnup) analyses performed to date. In contrast, MCNP is typically used for general validation purposes, for the evaluation of specific detailed experiments, and for certain studies where MCNP may be better suited than VENTURE, for example, comparative analyses of certain core component changes. As discussed in SAR Section 4.5.3, UML used its VENTURE model for investigating the core configuration that would represent the LCC for the UMLRR and calculating the LCC peaking factors. As discussed in the response to RAI-4.1(a) and (b), UML’s updated VENTURE model allows the silicide and aluminide elements to both fit within the same model.

As described in SAR Section 4.5.1.1, UML performed the steady-state thermal-hydraulic analyses for natural convection and forced convection operation of the UMLRR using the NATCON and PLTEMP codes, respectively. For transient thermal-hydraulic analyses (evaluating the consequences of both rapid and ramped reactivity changes in the reactor core and evaluating how a loss of flow scenario affects reactor performance and safety), UML used Program for the Analysis of Reactor Transients (PARET), which is a transient analysis code that simulates the behavior associated with both reactivity- and flow-induced transients within the system. UML’s LCC peaking factors are used as inputs to its NATCON, PLTEMP, and PARET calculations, as well as to its maximum hypothetical accident (MHA) analysis. UML’s steady-

state thermal-hydraulic analyses, and accident and reactivity transient analyses, are discussed and found acceptable in Section 2.6 and Chapter 5 of this SER, respectively.

In SAR Section 4.5.4, UML provided calculated basic kinetics and reactivity coefficients data for the UMLRR. These data, which are also used as inputs to UML's PARET calculations, are discussed and found acceptable in SER Section 2.5.2.

2.5.1 Normal Operating Conditions

In response to RAI-4.1(a) and (b) (Ref. 23), UML provided comparisons between calculated VENTURE and MCNP results, and actual measurements, for three operational cores: M-2-5 BOL (the OCC), M-2-5 at 50 MWD burnup, and M-1-3. UML provided these comparisons to demonstrate that its VENTURE and MCNP models are reasonably predictive of core reactivity for varying core configurations and that its VENTURE model is suitable for calculations of its LCC. In response to RAI-4.1(c) and (d) (Ref. 23), UML described its LCC and provided the results of VENTURE calculations for this core.

Operational Core Eigenvalues

For the critical condition for each core (the known, measured control blade and regulating rod heights at which the core is critical), UML used both VENTURE and MCNP to calculate the eigenvalues (i.e., k-effectives (k_{eff})) to give an indication of the degree of agreement in estimated critical positions (ECPs) between the VENTURE and MCNP models and actual conditions. The eigenvalues indicate the degree by which the core is calculated to be greater than or less than critical, i.e., 1.010 is an eigenvalue that is 1% $\Delta k/k$ greater than critical and 0.990 is an eigenvalue that is 1% $\Delta k/k$ less than critical. Table 2-3 (adapted from Table 1 of UML's response to RAI-4.1(a) and (b)) lists UML's calculated eigenvalues for each critical operational core.

Table 2-3 Calculated Eigenvalues for Critical Operational Cores

Model Description	Control Blades 1-4 Banked Location (inches out)	Regulating Rod Location (inches out)	VENTURE k_{eff}	MCNP k_{eff}
BOL M-1-3	15.3	8.0	0.980	0.995
BOL M-2-5	14.9	10.0	0.978	0.999
M-2-5 at 50 MWD	16.3	7.7	0.975	0.996

UML stated that although there is a negative 2 to 2.5 percent bias for its VENTURE results, its calculated VENTURE and MCNP eigenvalues demonstrate that its models are reasonably predictive of reactivity in the UMLRR operational cores.

The NRC staff reviewed UML's calculated eigenvalue data for the UMLRR. The NRC staff finds that UML's methodology of calculating eigenvalues, in conjunction with other methodologies discussed later in this section, is appropriate for demonstrating the adequacy of its neutronics models. The NRC staff finds that MCNP-calculated eigenvalues are close to the expected value of 1.000 for an exactly critical core. The NRC staff notes that there is a negative bias in the VENTURE results, which may result from VENTURE's use of homogenization (rather than an exact geometric model). However, the bias is relatively consistent between the three cores analyzed, which tends to indicate that VENTURE is valid for making comparisons between different core configurations.

Operational Core Control Blade and Regulating Rod Worths and Excess Reactivity

UML also used both VENTURE and MCNP to calculate individual control blade worths, total control blade worth, regulating rod worth, and excess reactivity for each of the three operational cores. Table 2-4 (adapted from Table 1 of UML's response to RAI-4.1(a) and (b)) provides these calculations, as well as actual measurements for comparison.

Table 2-4 Calculated and Measured Control Blade and Regulating Rod Worths and Excess Reactivity

	M-1-3 BOL Core Reactivity (%Δk/k)			M-2-5 BOL Core Reactivity (%Δk/k)			M-2-5 at 50 MWD Reactivity (%Δk/k)		
Blade #	Meas.	VENTURE	MCNP	Meas.	VENTURE	MCNP	Meas.	VENTURE	MCNP
Blade 1	2.63	2.95	3.00	2.82	2.91	2.76	2.55	2.86	2.73
Blade 2	2.47	2.80	2.75	2.19	2.35	2.40	2.23	2.29	2.40
Blade 3	3.32	3.32	3.42	3.19	3.16	3.42	3.64	3.06	3.19
Blade 4	3.20	3.43	3.55	3.93	3.72	3.83	4.19	3.74	3.71
Total of Blades 1-4	11.6	12.5	12.7	12.1	12.1	12.4	12.6	12.0	12.1
Excess Reactivity of Blades 1-4	2.82	3.22	3.44	3.46	3.45	3.46	2.41	2.71	2.60
Regulating Rod	0.28	0.44	0.38	0.30	0.45	0.38	0.31	0.45	0.38

UML stated that considering the uncertainty inherent in the measured data, its calculations are reasonably predictive of measured control blade and regulating rod worths and excess reactivity for its three operational cores analyzed.

The NRC staff reviewed UML's calculated and measured control blade and regulating rod worth and excess reactivity data for the UMLRR. The NRC staff finds that UML's methodology of comparing calculated and measured data is appropriate for demonstrating the adequacy of its neutronics models and is consistent with the methodology used at other research and test reactors (RTRs). The NRC staff finds that the worths of blades 1 through 4, the regulating rod worth, and the excess reactivity of blades 1 through 4 are acceptably estimated by both MCNP and VENTURE. The discrepancies are reasonable, given the uncertainty in the measured data, and the results are comparable to those seen for similar RTRs. The NRC staff notes that although the regulating blade worth is significantly over-predicted by both codes on a percentage or ratio basis, the absolute difference is small. The differences in measured and calculated regulating rod worths may be explained by the approximate representation of the geometry of the regulating rod in the models, as well as the location of the regulating rod at the core periphery where flux gradients are higher. (The NRC staff notes that measured, rather than calculated, regulating blade worths are typically used to adjust ECPs.)

As discussed in SER Section 2.2.2, UML was (as of 2020) in the process of replacing its control blades with new BORTEC blades. Replacement of the control blades could affect the control blade worth. However, as discussed in SER Sections 2.2.2 and 2.5.3, UML is required by TSs to measure the reactivity of its control blades annually and after any core configuration changes (including control blade replacement) and to ensure that excess reactivity and SDM requirements are met.

Operational Core Power Distribution

In response to RAI-4.1(c) and (d) (Ref. 23), UML provided power distribution information for its OCC (the BOL M-2-5 core configuration). The core map in SER Figure 2-8 below illustrates this information. (The core map in SER Figure 2-9, later in this section, illustrates the LCC power distribution.) The axes of the core maps are labelled to indicate fuel element positions, e.g., the data at the intersection of row B and column 5 correspond to a fuel element in core position B5. The abbreviations in the SER Figure 2-8 and Figure 2-9 keys are explained as follows:

kWt:	power in the fuel element (kWt)
f_a:	power in the fuel element / average core element power
f_{za}:	axial peaking factor in the element
f_{xya}:	intra-element radial peaking factor (relative to the individual fuel element)
$f_a \times f_{xya}$:	total radial peaking factor (relative to the overall core)

	1	2	3	4	5	6	7	8	9
A									
B				28.656 0.602 1.513 1.751 1.054	32.076 0.674 1.504 1.696 1.142	31.773 0.667 1.515 1.684 1.124			
C			28.410 0.597 1.455 1.179 0.703	53.378 1.121 1.452 1.240 1.390	62.646 1.316 1.446 1.214 1.597	59.407 1.248 1.449 1.197 1.494	48.493 1.018 1.449 1.289 1.313		
D	Key:	kWt f_a f_{za} f_{xya} $f_a \times f_{xya}$	46.488 0.976 1.376 1.221 1.192	60.687 1.274 1.373 1.280 1.632		68.051 1.429 1.371 1.218 1.741	54.034 1.135 1.370 1.213 1.376	36.380 0.764 1.422 1.320 1.008	
E			29.755 0.625 1.468 1.221 0.763	56.334 1.183 1.465 1.284 1.519	66.145 1.389 1.459 1.183 1.643	62.558 1.314 1.462 1.241 1.630	50.645 1.064 1.462 1.334 1.419		
F				38.010 0.798 1.529 1.547 1.235	43.894 0.922 1.521 1.451 1.337	42.179 0.886 1.527 1.487 1.317			
G									

Figure 2-8 OCC Fuel Element Power Distribution

In response to RAI-4.1(a) and (b), UML provided plots showing MCNP- and VENTURE-calculated, and measured, axial thermal flux profiles for several locations within its M-1-3 core configuration (see Figure 3 in the RAI response). UML stated that these plots help demonstrate

the ability of its MCNP and VENTURE models to accurately predict flux profiles. The NRC staff reviewed these plots in Figure 3 of UML's response to RAI-4.1(a) and (b) and finds that the axial traverses taken at several locations for the M-1-3 core configuration help provide assurance that UML's MCNP and VENTURE neutronics models are reasonably accurate and predictive.

Fuel Burnup

As discussed in SER Section 1.6, total operation since the HEU to LEU conversion in 2000, through the reporting period for the 2019-2020 annual report, has been 85.84 MWD. In SAR Section 4.5.1.1, UML stated that because of the low burnup of the UMLRR fuel, for most core modeling applications it uses a BOL model with fresh fuel and with the blade positions adjusted to account for the current critical conditions. However, in response to RAI-4.1(a) and (b), UML provided information to show that its VENTURE model provides a reasonable treatment of the fuel depletion and fission product buildup within the UMLRR. This information includes the comparisons between VENTURE- and MCNP-calculated and measured eigenvalues and reactivities for the M-2-5 core with 50 MWD of burnup, that are discussed earlier in this SER section and shown in SER Table 2-3 and Table 2-4. Additionally, Figures 16 and 17 of the response to RAI-4.1(a) and (b) provide plots comparing how VENTURE-calculated reactivity and measured critical control blade height change with increasing core burnup, which illustrate the ability of the VENTURE model to adequately account for burnup.

The NRC staff reviewed the above information regarding UML's methodologies for considering burnup effects for the UMLRR and the ability of UML's VENTURE model to account for burnup. The NRC staff finds that UML's methodology of using a BOL model and adjusting the blade positions is generally reasonable, given the relatively low burnup of the UMLRR fuel. However, the NRC staff also finds that the information that UML provided in response to RAI-4.1(a) and (b) helps demonstrate that UML's VENTURE model can reasonably consider burnup effects. Therefore, the NRC staff finds that UML's methodologies for calculating burnup effects are acceptable.

Renewed TS 5.3, "Reactor Core and Fuel," would state, in part:

> Applicability:
>
> These specifications apply to reactor core and fuel.
>
> Objective:
>
> To specify design features of the reactor core and fuel and allowable fuel configurations.
>
> Specifications:
>
> …
>
> (7) The average fission density in a fuel element shall not exceed 2×10^{21} fissions/cm^3.

TS 5.3(7) would require that the average burnup in each UMLRR silicide or aluminide fuel element not exceed 2.0×10^{21} fissions per cubic centimeter of fuel matrix. In its basis for TS 5.3(7), UML stated that NUREG-1313 (Ref. 52) provides data indicating that fission densities up to 2.5×10^{21} are acceptable. The guidance in NUREG-1537, Appendix 14.1, Section 3.1(6)(c),

states that for research reactors with plate-type MTR-type fuel, there should be burnup limitations on the fuel to prevent fuel swelling and buildup of oxide on the fuel cladding. The NRC staff finds that TS 5.3(7) imposes a burnup limit that is conservative relative to NUREG-1313 data that are relevant for silicide or aluminide fuel to help demonstrate that unacceptable fuel swelling will not occur, and is also conservative relative to the NUREG-1537, Appendix 14.1, Section 3.1(6)(c), maximum recommended burnup limit of 2.3×10^{21} for aluminide fuels. The NRC staff finds that TS 5.3(7), in conjunction with surveillance TS 4.1(8) (see SER Section 2.2.1) that requires periodic fuel inspections, helps ensure that the UMLRR is not operated with fuel that is excessively swollen or degraded due to burnup, which helps reduce the possibility of any fission product release from the fuel. The NRC staff notes that fuel burnup may also result in changes in core power peaking; notwithstanding TS 5.3(7), UML is required (in accordance with the regulations in 10 CFR 50.59 and UMLRR TS 6.2) to evaluate any core configurations it uses, including configurations with fuel burnup, to verify that the configurations are appropriately bounded by analyses in the SAR, as supplemented, in addition to evaluating the configurations for compliance with the UMLRR TSs. The NRC staff estimated that UML's core burnup of 85.84 MWD as of 2020 corresponds to a core-average burnup of only approximately 2 percent of the TS 5.3(7) limit. The NRC staff also finds that TS 5.3(7) implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 by establishing an appropriate limit on fuel burnup. Therefore, based on the above, and also based on its 10 CFR 50.36(c)(4) findings for TSs on design features in SER Section 6.7, the NRC staff concludes that TS 5.3(7) is acceptable.

LCC Power Distribution

In SAR Section 4.5.3, as supplemented by its response to RAI-4.1(c) and (d), UML provided its LCC. UML stated that to determine its LCC, it analyzed many feasible core configurations to determine the configuration with the maximum power peaking. UML considered cores containing all silicide fuel, all aluminide fuel, and a mix of both fuel types, and also considered both fresh cores and cores with some accumulated burnup. UML determined the limiting configuration to be a BOL 21-element core, with the control blades positioned at 14.9 inches out, a water-filled radiation basket in core position D5 (see SER Section 2.2.2), eight silicide fuel elements immediately surrounding the D5 core position, and, to emphasize the power peaking in the inner ring, 13 aluminide fuel elements (which have lower uranium loading and are less reactive) in the outer ring.

In its response to RAI-4.1(e) (Ref. 23), UML provided discussion and analyses demonstrating that the silicide fuel elements are more reactive and more limiting from a safety analysis standpoint than the aluminide fuel elements. This supports UML's determination that core power peaking would be greatest in the silicide fuel. UML analyses also showed that the aluminide fuel has a greater margin to ONB than the silicide fuel, which supports UML's consideration of silicide fuel for its thermal-hydraulic analyses, which are discussed and found acceptable in SER Section 2.6.

In SAR Section 4.5.3, UML stated that previous reactor physics studies of the UMLRR have determined that more severe power peaking is often associated with higher reactivity cores (e.g., BOL cores), for which the control blades are inserted further into the core when the reactor is at an exactly critical condition.

SER Figure 2-9 shows the power distribution for the LCC. Based on the combination of the axial peaking factor (f_{za}) and total radial peaking factor ($f_a \times f_{xya}$) in each fuel element, the largest computed overall peaking factor (axial peaking factor multiplied by total radial peaking factor) is

in the element located in the D6 core position, which has an axial peaking factor of 1.383 and a total radial peaking factor of 1.993. Therefore, the element in D6 is considered the hot element for the purposes of UML's thermal-hydraulic analyses, transient analyses, and MHA. To obtain the peaking factors for its NATCON, PLTEMP, and PARET calculations, UML introduced additional conservatism by increasing its D6 axial peaking factor and total radial peaking factor by 5 percent, and then rounding up the result to obtain two significant figures. This results in an axial peaking factor of 1.5 and a total radial peaking factor of 2.1, which combined leads to a total overall peaking factor of over 3.1.

The total radial peaking factor of 1.993 or 2.1 may be considered to represent the hot channel or hot plate peaking factor (i.e., the peaking factor corresponding to the hottest entire channel or fuel plate in the core). For the UMLRR MHA (see SER Section 5.1), which is the loss of cladding from a single UMLRR fuel plate, it is assumed that the accident involves the hottest plate in the core, which contains twice the activity of the average core fuel plate (however, for the MHA, UML conservatively assumed that the core only contains 18 fuel elements, such that the activity in the hottest plate would be even greater than that of the LCC).

For the axial power profile used in all of its thermal-hydraulic and transient analyses, UML used the VENTURE-calculated profile for the fuel element in core location B5 of the OCC (see SAR Figure 4-15 and figure in UML's response to RAI-4.1(c) and (d)), because that fuel element also has an axial peaking factor of approximately 1.5. For a hot channel analysis, this entire profile is multiplied by the total radial peaking factor of 2.1.

The NRC staff reviewed UML's LCC and UML's determination of peaking factors and power profiles to use for its thermal-hydraulic and accident analyses. The NRC staff finds that the LCC appears to be a realizable core configuration within the bounds of the UMLRR TSs and is a core configuration that is designed to emphasize power peaking in the inner ring. The use of a water-filled radiation basket in core position D5 will significantly increase power peaking by increasing neutron moderation in the center of the core. The NRC staff finds that UML's determination that a BOL core would be the LCC is reasonable, because a BOL core is more reactive and has lower control blade heights for an exactly critical core, which can increase peaking. The NRC staff notes that the silicide partial fuel elements and aluminide removable plate fuel element have a lower uranium loading than the aluminide standard fuel elements and, therefore, that replacing an aluminide standard element with one of these types of elements in the periphery of the 21 element LCC could potentially increase radial peaking. However, given that the critical blade heights would also likely be higher for this scenario due to the lower uranium loading, which would decrease axial peaking, and also given that the LCC is not an operational core and is only intended to provide an approximation of worst-case peaking factors, the NRC staff finds that UML's use of standard fuel elements in its LCC is reasonable. As discussed in SER Section 2.2, UML is required (in accordance with the regulations in 10 CFR 50.59 and UMLRR TS 6.2) to evaluate any core configurations it uses, including configurations with partial and/or removable plate elements, to verify that the configurations are appropriately bounded by analyses in the SAR, as supplemented (TS 5.3(6) also specifically requires UML to perform appropriate analyses to verify that reactor operation would remain within appropriate SAR safety margins prior to any use of the removable plate element in the core). As can be seen by comparing SER Figure 2-8 for the OCC with SER Figure 2-9 for the LCC, the LCC power distribution is more limiting. The NRC staff finds that UML's methodology for determining its bounding peaking factors and power profiles to use for its thermal-hydraulic and accident analyses is reasonable, conservative, and consistent with research reactor industry practice. Based on its experience with similar RTRs, the NRC staff also finds that UML's peaking factor and power profile results are reasonable and conservative, given the design and layout of the

UMLRR core. The NRC staff further finds that UML's adding of 5 percent to its calculated peaking factors and rounding up adds additional conservatism that helps compensate for any possible error in the calculated factors.

	1	2	3	4	5	6	7	8	9
A									
B				27.393 0.575 1.522 1.664 0.957	29.467 0.619 1.513 1.622 1.004	29.067 0.610 1.523 1.627 0.993			
C			40.300 0.846 1.458 1.224 1.036	54.428 1.143 1.458 1.325 1.514	65.534 1.376 1.455 1.290 1.775	58.315 1.225 1.456 1.290 1.579	43.825 0.920 1.456 1.258 1.158		
D	Key:	kWt f_a f_{za} f_{xya} $f_a \times f_{xya}$	45.495 0.955 1.380 1.141 1.090	64.980 1.365 1.385 1.403 1.914		69.602 1.462 1.383 1.364 1.993	48.213 1.012 1.375 1.205 1.220	31.823 0.668 1.434 1.277 0.853	
E			42.186 0.886 1.470 1.269 1.124	57.425 1.206 1.471 1.287 1.552	69.196 1.453 1.467 1.251 1.818	61.429 1.290 1.468 1.254 1.618	45.794 0.962 1.468 1.301 1.251		
F				36.431 0.765 1.536 1.470 1.124	40.422 0.849 1.529 1.388 1.178	38.673 0.812 1.535 1.436 1.166			
G									

Figure 2-9 LCC Fuel Element Power Distribution

Startup Plan for Cores Containing Aluminide Fuel

The guidance in NUREG-1537, Part 2, Section 12.11, states, in part, that licensees should submit a startup plan when significant core modifications are being made. UML proposes to use aluminide fuel in its core, in addition to its existing silicide fuel. In its response to RAI-12.1 (Ref. 71), UML stated that for new core configurations, it has an existing standard operating procedure for loading the core and performing an approach to critical. Additionally, UML has existing procedures for performing reactivity evaluations (i.e., determining differential and integral control blade worth, excess reactivity, and SDM). UML stated that these procedures have been used for numerous core configuration changes, including the HEU to LEU fuel conversion in 2000, and will be used when any aluminide fuel is added to the core. Additionally, in its letter dated September 30, 2020 (Ref. 98), UML stated that for its initial startup of the reactor with aluminide fuel in the core, it will perform comparisons to verify the similarity of calculated and measured reactor parameters (e.g., reactivity and control blade height) to help confirm modeling and analysis predictions for cores containing aluminide fuel. UML further stated that for any new fuel configuration, even with the same type of fuel, it performs measurements of reactor parameters not just to ensure compliance with the TSs, but also to verify reasonable consistency with expected parameter values for the new configuration.

The NRC staff reviewed the above information and finds that UML's procedures will help ensure that UML measures important parameters for new cores, including cores containing aluminide fuel. Measuring these parameters will allow UML to make comparisons to its analyses of new cores and allow it to confirm the validity of its analysis predictions for new cores. These measurements will also allow UML to ensure that new cores comply with UMLRR TSs, such as TSs 3.1.1(1) and 3.1.1(2), which limit the core excess reactivity and SDM. TSs 4.1(1) and 4.2(2) also require that excess reactivity and SDM be measured to ensure compliance with TSs 3.1.1(1) and 3.1.1(2) when significant core configuration changes are made; these TSs are discussed and found acceptable in SER Section 2.5.3. As discussed in SER Section 2.2.1, the aluminide fuel is similar to the UMLRR's existing silicide fuel, and aluminide fuel has been successfully used in operation of similar research reactors. Based on the above, the NRC staff finds that UML's implementation of its proposed plans and procedures for the use of cores containing aluminide fuel will help ensure that the UMLRR is operating consistent with its TSs and its safety analyses in the SAR, as supplemented, for cores including aluminide fuel. Therefore, the NRC staff finds that UML's proposed plan for operation of the reactor with cores including aluminide fuel is acceptable.

TSs Relevant to Normal Operating Conditions

Renewed TS 3.1.2, "Maximum Power Level," would state:

> Applicability:
>
> This specification applies to the reactor thermal power level.
>
> Objective:
>
> To ensure the safety limit is not exceeded.
>
> Specification:
>
> The reactor shall not be continuously operated at a power level exceeding $1MW_t$.

TS 3.1.2 would require that the UMLRR not be continuously operated above 1 MWt. TS 3.1.2 would limit the steady-state operating power level to 1 MWt, but would not prohibit momentary drifts above 1 MWt or small variations in the power level due to measurement uncertainty. The limiting safety system setting (LSSS) TS 2.2.1(1) does not require automatic protective action until the reactor power level reaches a measured value of 1.15 MWt, and UML performed steady-state (SER Section 2.6), reactivity transient (SER Section 5.2), and loss of coolant flow (SER Section 5.4) analyses assuming a power level of up to 1.25 MWt, in part to account for uncertainty in the measured power level. The MHA (SER Section 5.1) and LOCA fuel integrity (SER Section 5.3) analyses effectively assume core power levels of greater than 1 MWt but less than 1.25 MWt, and the LOCA radiological consequence analysis (SER Section 5.3) assumes a core power of 1 MWt; however, these analyses also assume long-term (i.e., infinite) continuous operation at those power levels, which is a bounding assumption. While reactor power may momentarily drift above 1 MWt, TS 3.1.2 would prohibit continuous operation above 1 MWt. The NRC staff finds that TS 3.1.2 helps ensure that for forced convection operation, the UMLRR is operated at its licensed power level, with adequate margin to the LSSS and UML's analysis assumptions. Based on the above, and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.1.2 is acceptable.

Renewed TS 4.2.3, "Reactor Protection System Scrams," would state, in part:

> Applicability:
>
> This specification applies to the surveillance requirements for the Reactor Protection System.
>
> Objective:
>
> To ensure Reactor Protection System limiting conditions for operation are met.
>
> Specifications:
>
> …
>
> (4) Thermal power level shall be verified annually.
>
> ….

TS 4.2.3(4) would require that UML verify the thermal power of the UMLRR annually. In its response to RAI-7.14.a (Ref. 79), UML stated that TS 4.2.3(4) is intended to satisfy the ANSI/ANS-15.1-2007, Section 4.2, item (8), recommendation that thermal power verification be performed for RTRs with a power level of 2 MWt or less. UML also states that the purpose of the thermal power verification is to ensure that the UMLRR is not being operated in excess of its licensed limit of 1 MWt (or the TS 3.1.2 power level limit of 1 MWt) when the power measuring channels measure 100 percent of full power. During the 2020-2021 audit (Ref. 86), the NRC staff confirmed that the thermal power verification is accomplished by comparison of calorimetric measurements with neutronic power measuring channel outputs and a subsequent adjustment of indicated neutronic power to match calculated thermal power if the difference between them exceeds the procedural threshold (approximately two percent of rated thermal power). UMLRR TS 4.2.3(3), which is discussed and found acceptable in SER Section 2.5.3, separately requires that the reactor power level channels (linear and log) and the period channel be calibrated

annually. The NRC staff finds that the TS 4.2.3(4) annual thermal power verification, in conjunction with the TS 4.2.3(3) annual calibration of all UMLRR power channels, helps ensure that UML does not operate the reactor at steady-state power levels above 1 MWt, such that the reactor operation is bounded by analyses in the SAR, as supplemented, and also helps ensure that the requirement of TS 3.1.2 that the reactor shall not be continuously operated at a power level exceeding 1MWt is met. Additionally, the NRC staff finds that TS 4.2.3(4) appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 because it requires periodic thermal power verification to help ensure that measured values of reactor power from instruments (calibrated as required by separate TS 4.2.3(3)) are consistent with the thermal power calorimetric heat balance calculation and because it requires a surveillance interval that is consistent with the recommended interval in ANSI/ANS-15.1-2007, Section 4.2, item (8). Therefore, based on the above, and based on its 10 CFR 50.36(c)(3) findings for TSs on SRs in SER Section 6.7, the NRC staff concludes that TS 4.2.3(4) is acceptable.

Conclusion

The NRC staff reviewed the above information regarding UML's normal operating conditions and nuclear design, including its neutronics calculation methodology and models and its calculated peaking factors and flux profiles. Because the neutronics methodology is generally consistent with the methodology used at other RTRs, the NRC staff finds that UML's methodology is reasonable and appropriate for the UMLRR. The NRC staff finds that UML described possible operational core configurations and analyzed reactivity conditions for these configurations. The NRC staff finds that UML's neutronics calculations for its operational core configurations are sufficiently similar to measured results to demonstrate that its neutronics models are predictive over a range of possible UMLRR core configurations, including configurations with fuel burnup. The NRC staff finds that procedures that UML will use for loading new core configurations, including configurations with aluminide fuel, will also help ensure that measured core parameters are acceptably in-line with predicted values and in compliance with TSs. The NRC staff also finds that UML acceptably addressed an LCC that is designed to produce worst-case peaking factors suitable for use in UML's thermal-hydraulic and accident analyses. Therefore, based on the above, the NRC staff concludes that UML's neutronics calculations are suitable for UML's proposed normal operating cores and conditions and, thus, are acceptable.

2.5.2 Reactor Core Physics Parameters

In SAR Section 4.5.4, UML provided basic kinetics data and reactivity coefficients for the UMLRR. These values are used as inputs to UML's PARET transient calculations. The PARET code uses the point kinetics approximation to model the transient power and reactivity behavior during simulated off-normal transient scenarios. UML's PARET calculations are discussed and found acceptable in Chapter 5 of this SER.

Kinetics Parameters

As described in SAR Section 4.5.4, the neutron kinetics parameters applicable to UMLRR and used in the PARET analysis are the same as those calculated and used for the UMLRR HEU to LEU conversion studies. The kinetics parameters for the UMLRR (the effective delayed neutron fraction (β_{eff}), the neutron generation time (Λ), and the delayed neutron parameters for the six precursor groups) are summarized in SER
Table 2-5, which is adapted from SAR Table 4-4. UML stated that, in its PARET analyses, these parameters are treated as constants for its particular reactor design.

As discussed in SAR Section 4.5.1.1, UML performed validation of its PARET model, including its kinetics parameters, by comparing PARET results with measured results. UML provided plots summarizing its validation results in SAR Figures 4-13 and 4-14. These results showed that the PARET model for the UMLRR can simulate actual reactor behavior with reasonable accuracy.

Table 2-5 Point Kinetics Data Used in the PARET Models

β_{eff}	0.0078	Λ (microseconds)	65

Precursor Group (i)	Precursor Weights β_i/β_{eff}	Decay Constants, λ_i (seconds^{-1})
1	0.038	0.0127
2	0.213	0.0317
3	0.188	0.1160
4	0.407	0.3111
5	0.128	1.4000
6	0.026	3.8710

The NRC reviewed UML's kinetics parameters for the UMLRR. The NRC staff notes that it previously reviewed and approved UML's effective delayed neutron fraction and prompt neutron lifetime for the HEU to LEU conversion of the UMLRR, as documented in the NRC staff safety evaluation for the conversion (Ref. 37). Prompt neutron lifetime is approximately equal to prompt neutron generation time, or the neutron generation time excluding delayed neutrons, which is the neutron generation time typically used for transient analyses.

The NRC staff finds that UML's kinetics parameters appear reasonable for UML's core design, are reasonably similar to kinetics parameters calculated for similar RTRs, and have been validated by comparisons with measurements taken during UMLRR operation. The NRC staff also finds that UML's use of constant kinetics parameters to represent all allowed core configurations and conditions (i.e., burnup levels, temperature, etc.) is reasonable based on the limited variation expected in these parameters and is consistent with established practice at similar RTRs.

The NRC staff also reviewed the kinetics parameters (prompt neutron lifetime and effective delayed neutron fraction) calculated for the WPI aluminide fuel core during the WPI HEU to LEU conversion studies and discussed in WPI's application for NRC approval of the WPI reactor conversion (Ref. 47 and Ref. 48), which was reviewed and approved by the NRC (Ref. 42). WPI previously calculated the prompt neutron lifetime and effective delayed neutron fraction for the WPI LEU core to be 61.2 microseconds and 0.0077, respectively. Given (1) the similarities between these values and UML's corresponding calculated values for its initial LEU silicide fuel core and (2) the relatively similar core configurations and fuel parameters for the WPI core (see Ref. 47), the UML cores containing silicide fuel (see SER Section 2.2), and the UML cores that would contain aluminide fuel and silicide fuel (see SER Section 2.2), the NRC staff finds that UML's use of its kinetics parameters listed in SER
Table 2-5 for UML core configurations including aluminide fuel is also reasonable.

Coefficients of Reactivity

As described in SAR Section 4.5.4, as supplemented (Ref. 44), UML's PARET calculations consider reactivity feedback components. PARET inputs include reactivity coefficients for fuel temperature, moderator temperature, and moderator void, which have units of delta k over k per degree Celsius or delta k over k per percent void (Δk/k/°C or Δk/k/%void, respectively), as applicable. The reactivity change corresponding to an increase in fuel or moderator temperature or percent void is determined by multiplying the change in fuel or moderator temperature or percent void by the applicable reactivity coefficient.

The reactivity coefficients that UML generated for the UMLRR are listed in SAR Table 4-5 and SER Table 2-6. For added conservatism, and to account for any uncertainty in the coefficients, UML adjusted the coefficients shown in SER Table 2-6 by 25 percent (higher or lower, as appropriate to make calculations with the coefficients more conservative) to obtain the values it used for its PARET transient analysis calculations discussed in Chapter 5 of this SER. UML's PARET model assumes that the reactivity coefficients are independent of fuel or moderator temperature.

As discussed in SAR Section 4.5.1.1, UML performed validation of its PARET model, including its reactivity coefficients, by comparing PARET results with measured results. UML developed its reactivity coefficients based on a BOL core (the OCC). However, when the validation measurements were performed in 2014, the core was at about 70 MWD burnup. UML provided plots summarizing its validation results in SAR Figure 4-14. These results showed that the PARET model for the UMLRR can simulate actual reactor behavior with reasonable accuracy.

Table 2-6 Coefficients of Reactivity

Reactivity Coefficients	Values
Moderator Temperature Coefficient (Δk/k/°C)	-5.30×10^{-5}
Fuel Temperature Coefficient (Δk/k/°C)	-2.12×10^{-5}
Moderator Void Coefficient (Δk/k/%void)	-2.59×10^{-3}

The NRC staff reviewed UML's reactivity coefficients and finds that the coefficients are similar to kinetics parameters calculated for other RTRs with similar plate-type fuel, considering uncertainties and the differences between the RTRs. Although the coefficients were developed based on a BOL core, UML's validation of the coefficients against measurements taken during UMLRR operation with the core at about 70 MWD burnup tends to demonstrate that models using the coefficients can adequately simulate reactor behavior over varying burnup levels. However, because the LCC assumed for the transient analyses is a BOL core, the NRC staff notes that BOL conditions are most relevant for the transient analyses. The NRC staff also finds that UML's assumption that the coefficients are temperature-independent is reasonable and consistent with calculations and/or established practice for similar RTRs.

The NRC staff also reviewed the reactivity coefficients calculated for the WPI aluminide fuel core during the WPI HEU to LEU conversion studies and discussed in WPI's application for its reactor conversion (Ref. 47 and Ref. 48), which the NRC staff reviewed and approved (Ref. 42). The NRC staff finds that the water temperature, fuel temperature, and void coefficients calculated for the WPI aluminide fuel are reasonably similar to or more conservative (more negative) than the coefficients listed in SER Table 2-6. Based on these comparisons, and also based on the similarities between the silicide fuel and the aluminide fuel (see SER Section

2.2.1), the NRC staff finds that UML's assumption that its coefficients are valid for cores containing aluminide fuel instead of, or in addition to, silicide fuel is reasonable.

Based on the facts that UML's calculated reactivity coefficients are negative and are not expected to vary significantly with changes in reactor parameters, as well as experience with similar RTRs, the NRC staff finds that there is reasonable assurance that the reactivity coefficients for the UMLRR will remain negative over the credible range of reactor operation.

The NRC staff also finds that the 25 percent conservatism factor that UML applied to these coefficients in the UMLRR transient analyses will help ensure that any variation in the coefficients due to calculation uncertainty, burnup, the use of silicide versus aluminide fuel, or other factors is adequately accounted for in the PARET calculations.

As discussed in SER Section 5.2, the NRC staff performed a calculation of a ramp reactivity insertion transient that demonstrated that even when all reactivity feedback is conservatively ignored (i.e., reactivity coefficients are set to zero), the maximum fuel temperatures reached still remained well within acceptable levels and a large margin was maintained to temperatures at which fuel cladding failure could occur.

Although the TSs prior to the UMLRR license renewal included requirements related to reactivity coefficients, the NRC staff finds that such TSs are no longer necessary based on UMLRR operating experience and because, as discussed above, reactivity coefficients should not vary unacceptably (e.g., should remain negative) over the credible range of reactor operation. The NRC staff finds that this is consistent with the guidance in NUREG-1537, Appendix 14.1, which states that TSs on reactivity coefficients are only necessary if the parameters could vary unacceptably with reactor operation.

Conclusion

Based on its review of the above, the NRC staff finds that UML determined values for neutron lifetime, effective delayed neutron fraction, and coefficients of reactivity that have been validated with experimental measurements. The effects of changes in reactor characteristics are considered, as appropriate, in the analyses of the core physics parameters. The core physics parameters are specific to the design of the UMLRR and are acceptable for use in UML's analyses of reactor operation, including its PARET transient calculations. Therefore, the NRC staff concludes that UML's reactor core physics parameters are acceptable.

2.5.3 Operating Limits

Excess Reactivity and Shutdown Margin

Excess reactivity for the UMLRR is the amount of reactivity that would exist if all control blades and the regulating rod were moved to the fully withdrawn condition from the point where the reactor is exactly critical at reference core conditions (see TS definition for excess reactivity, which is discussed and found acceptable in SER Section 6.1.2). As discussed in SAR Section 4.5.1.1, UML generally uses the VENTURE code (with a BOL model, because of the low accumulated fuel burnup, and with the control blade positions adjusted to their current critical height) for most routine reactivity evaluations, including excess reactivity evaluations. According to SAR Section 4.5.3, the excess reactivity in the OCC (the M-2-5 BOL core) is approximately 3.5% $\Delta k/k$. Based on the information provided in Table 2 of UML's response to RAI-4.1(a) and (b) and reproduced in SER Table 2-4, the measured and calculated excess

reactivities for other operational cores are similar to or less than 3.5% Δk/k. TS 3.1.1(1), which is discussed and found acceptable later in this section, requires that the UMLRR excess reactivity be less than 4.7% Δk/k.

In SAR Section 4.5.3, UML stated that a practical lower limit on UMLRR excess reactivity to ensure that the reactor is readily operable is approximately 1.5 to 2.0% Δk/k, since sufficient excess reactivity is needed to override temperature and xenon feedbacks, to allow for additional fuel burnup, and to counter the negative reactivity effects of experiments.

As discussed in NUREG-1537, Part 2, Section 4.5.3, SDM is negative reactivity obtainable by control rods to ensure that the reactor can be shut down from any reactor condition. SDM for the UMLRR is the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems starting from any permissible operating condition and with the most reactive control blade and regulating rod fully withdrawn, and that the reactor will remain subcritical without further operator action (see TS definition for SDM, which is discussed and found acceptable in SER Section 6.1.2). As discussed in UML's response to RAI-4.1(a) and (b), the VENTURE code is typically used for UML's SDM calculations as well as its excess reactivity calculations.

SDM may be calculated by subtracting the total worth of the three least reactive control blades from the maximum allowed excess reactivity (the most reactive control blade and the regulating rod are not credited because they are assumed to be fully withdrawn). TS 3.1.1(2), which is discussed and found acceptable later in this section, requires that the UMLRR SDM be negative by at least 1.0% Δk/k.

The result of a SDM evaluation performed by the NRC staff for the OCC, based on UML's TSs and provided control blade worths, is shown in SER Table 2-7. The evaluation illustrates how UML's control blade worth measurements and calculations can be used to demonstrate that the OCC SDM is negative by at least 1.0% Δk/k, consistent with the TS 3.1.1(2) requirement. SER Table 2-7 shows that the UMLRR SDM for the OCC is -3.72% Δk/k or -3.50% Δk/k when calculated from VENTURE results or experimental measurements, respectively; these values are both significantly more negative than 1.0% Δk/k.

Table 2-7 NRC Staff Shutdown Margin Evaluation for the OCC

	Source	VENTURE Calculated Reactivity (% Δk/k)	Measured Reactivity (% Δk/k)
Maximum Excess Reactivity	TS 3.1.1(1)	+4.70	+4.70
Regulating Rod	Table 2 of response to RAI-4.1(a) and (b)	-0.45*	-0.30*
Control Blade 1		-2.91	-2.82
Control Blade 2		-2.35	-2.19
Control Blade 3		-3.16	-3.19
Control Blade 4		-3.72*	-3.93*
SDM	NRC staff calculation from TS 3.1.1(1) value and UML's data in Table 2 of response to RAI-4.1(a) and (b)	-3.72 (acceptable)	-3.50 (acceptable)
Minimum SDM	TS 3.1.1(2)	-1.00	-1.00
*Regulating rod and most reactive control blade are not considered in the SDM calculations.			

Renewed TS 3.1.1, "Reactivity and Core Configurations," would state, in part:

> Applicability:
>
> These specifications apply to the reactivity condition of the reactor, core configuration, and experiments.
>
> Objective:
>
> To ensure that the reactor can be safety operated and shutdown and maintained in a safe shutdown condition at all times such that the Safety Limit will not be exceeded.
>
> Specifications:
>
> When the reactor is operating, the following conditions shall exist:
>
> (1) The excess reactivity in the reference core condition shall be <4.7% Δk/k.
>
> (2) The shutdown margin shall be >1% Δk/k with the most reactive control blade and regulating rod in their fully withdrawn position; all installed experiments in their most reactive state; and the reactor in the reference core condition.
>
> ….

TS 3.1.1(1) would require that the excess reactivity in the reference core condition be less than 4.7% Δk/k. As discussed above, UML stated that the excess reactivity of its OCC was about 3.5% Δk/k, but the excess reactivity of other operational cores is typically lower. A maximum excess reactivity of 4.7% Δk/k gives UML the flexibility to operate with larger cores than the

OCC (i.e., cores with more than 21 fuel elements, as allowed by TS 5.3), provided that those cores meet other TSs and are within the bounds of UML's safety analyses in the SAR, as supplemented. However, as illustrated by the data in SER Table 2-7, even when excess reactivity is at this maximum, the UMLRR can still have a sufficient SDM to meet TS 3.1.1(2) by a significant margin.

As discussed in SER Section 5.2, UML analyzed various credible accident scenarios in which a portion of the available excess reactivity could be rapidly or slowly inserted into the reactor. Given the limits on reactivity of experiments (TS 3.7), the maximum rate of reactivity insertion by control blades and the regulating rod (TS 3.2.2), the design of the reactor and control blades and regulating rod, and standard industry practice for operating research reactors such as the UMLRR, the NRC staff finds that there is no credible scenario by which an entire available excess reactivity of 4.7% $\Delta k/k$ could be rapidly inserted into the UMLRR.

The NRC staff finds that TS 3.1.1(1) helps ensure that the UMLRR excess reactivity is limited such that an acceptable SDM can be achieved, while still allowing UML sufficient operational flexibility. The NRC staff also finds that the TS 3.1.1(1) requirement is applicable during any reactor operation and thus must be met during any configuration of reactor experiments (i.e., experiments installed or not installed, and experiments in their most reactive state or another state). The NRC staff further finds that credible accidents involving the insertion of a portion of the excess reactivity allowed by TS 3.1.1(1) would not cause a SL to be exceeded. Additionally, the NRC staff finds that TS 3.1.1(1) appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 by specifying an upper limit on excess reactivity. Therefore, based on the above, and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.1.1(1) is acceptable.

TS 3.1.1(2) would require that the SDM be greater than 1.0% $\Delta k/k$ when conservatively assuming that the most reactive control blade and the regulating rod are in their fully withdrawn (most reactive) positions, all installed experiments are in their most reactive state, and the reactor is in the reference core condition (i.e., negligible negative reactivity from xenon or temperature feedback). In its response to RAI-14.3.2 (Ref. 71), UML stated that a revised minimum SDM of 1.0% $\Delta k/k$ (as compared to the current, i.e., pre-renewal, TS-required minimum SDM of 2.7% $\Delta k/k$) is sufficient to ensure that the reactor remains subcritical under specified conditions, is easily measurable, and is consistent with similar MTR-fueled reactors. Based on the information in SER Table 2-7, the NRC staff finds that the 1.0% $\Delta k/k$ SDM in TS 3.1.1(2) is easily achievable for the UMLRR. The NRC staff also finds that a 1.0% $\Delta k/k$ SDM is large enough to be easily measurable. Given the conservative conditions included in the TS, the NRC staff also finds that TS 3.1.1(2) will help ensure that the UMLRR can be shut down, and remain shut down (following cool down, xenon decay, experiment removal, etc.), for any credible circumstance using the negative reactivity provided by the control blades. The NRC staff further finds that TS 3.1.1(2) appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 by specifying a minimum SDM under conservative conditions. Therefore, based on the above, and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.1.1(2) is acceptable.

Renewed TS 4.1, "Reactor Core Parameters," would state, in part:

> <u>Applicability</u>:
>
> This specification applies to surveillance requirements for the various reactor core parameters.

Objective:

To ensure the reactor core parameters meet the specified limiting conditions for operation.

Specifications:

(1) The reactor core excess reactivity at the reference core condition shall be verified annually or following any significant core configuration, control blade and/or regulating rod change. A significant core configuration change is defined as a change in reactivity greater than 0.2 % $\Delta k/k$.

(2) The shutdown margin shall be verified annually or following any significant core configuration and/or control blade change. A significant core configuration change is defined as a change in reactivity greater than 0.2 % $\Delta k/k$.

….

TS 4.1(1) and TS 4.1(2) would require that excess reactivity and SDM, respectively, be verified annually or following any significant (greater than 0.2% $\Delta k/k$) core configuration and/or control blade change. TS 4.1(1) would also explicitly require that excess reactivity be verified following any regulating rod change. The NRC staff finds that TSs 4.1(1) and 4.1(2) help ensure that UML performs verification of its compliance with LCO TSs 3.1.1(1) and 3.1.1(2) for excess reactivity and SDM, respectively, at intervals that are appropriate to account for potential core changes over time (e.g., burnup) and other changes (e.g., fuel movement and/or replacement). The NRC staff finds that it is not necessary for TS 4.1(2) to explicitly require SDM be verified following a regulating rod change (if no other changes are being made) because the regulating rod is not considered in determining SDM and because it has a small worth compared to the control blades. However, the NRC staff finds that a regulating rod change may be a core configuration change requiring SDM verification if it is a change in the type of regulating rod. The NRC staff notes that although UML performs reactivity calculations as discussed above, compliance with excess reactivity and SDM TSs for research reactors is typically verified by measurements conducted at low reactor power. The NRC staff finds that UML's designation of a "significant" change as one greater than 0.2% $\Delta k/k$ is reasonable because it is a small reactivity change relative to the excess reactivity and SDM limits and because it is comparable to requirements at similar research reactors. The NRC staff notes that TSs 4.1(1) and 4.1(2) do not specifically require verification of excess reactivity and SDM following installation of, or changes to, in-core experiments (as recommended in NUREG-1537, Appendix 14.1, Sections 4.1(1) and 4.1(2)); however, the NRC staff finds that this requirement is not necessary in TSs 4.1(1) or 4.1(2) because, as discussed in SER Sections 6.3.7, 6.6.2, and 6.6.5, the UMLRR TSs, including TSs 6.2.3 and 6.5, require UML to evaluate all new experiments or classes of experiments to ensure that the experiments are designed and planned appropriately, including verification that experiments will not cause applicable TSs to be violated. The NRC staff further finds that TSs 4.1(1) and 4.1(2) appropriately implement the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 by requiring measurement of SDM and excess reactivity and by requiring intervals that are consistent with the recommended intervals in NUREG-1537, Appendix 14.1, Sections 4.1(1) and 4.1(2) (except as related to in-core experiments, as discussed above), and ANSI/ANS-15.1, Section 4.1, items (1) and (2). Therefore, based on the above, and based on its 10 CFR 50.36(c)(3) findings for TSs on SRs in SER Section 6.7, the NRC staff concludes that TSs 4.1(1) and TS 4.1(2) are acceptable.

Scrams, Interlocks, and Channels Required for Operation

Renewed TS 3.2.3, “Reactor Protection System Scrams,” would state:

Applicability:

This specification applies to the reactor protection system.

Objective:

To stipulate the minimum number of reactor protection system scrams that shall be operable to ensure that the safety limit is not exceeded.

Specification:

The reactor shall not be operated unless the reactor protection system scrams described in Table 3.2.3-1 are operable.

Table 3.2.3-1
Minimum Reactor Protection System Scrams

		Forced Convection Mode		**Natural Convection Mode**	
	Scrams	Function	Minimum Required	Function	Minimum Required
1.	**Reactor Period**	Scram at ≤ 3 second period	1	Scram at ≤ 3 second period	1
2.	**Reactor Power Level**	Scram at ≥ 1.15 MW	2*	Scram at ≥115kW	2*
3.	**Primary Coolant Flow Rate**	Scram at ≤1400 GPM	1	n/a	n/a
4.	**Pool Water Level**	Scram at ≤ 24.25 ft above core centerline	1	Scram at ≤ 24.25 ft above core centerline	1
5.	**Pool Inlet Temperature**	Scram ≥ 108°F	1	n/a	n/a
6.	**Pool Temperature**	Scram ≥ 108°F	1	Scram ≥ 108°F	1
7.	**Control Room Manual Scram Button**	Scram if pressed	1	Scram if pressed	1
8.	**Detector High Voltage (each period and power channel)**	Scram ≤ 500 V	1	Scram ≤ 500 V	1
9.	**Process Controls Display Watch Dog Timer**	Scram for communication loss >10 second	1	Scram for communication loss >10 second	1

10.	**Drives Controls Display Watch Dog Timer**	Scram for communication loss >10 second	1	Scram for communication loss >10 second	1
11.	**Seismic Disturbance**	Scram on seismic motion	1	Scram on seismic motion	1
12.	**Bridge Movement**	Scram if moved > 1 inch	1	Scram if moved > 1 inch	1
13.	**Primary Piping Alignment**	Scram when alignment limit switches not met	1	n/a	n/a
14.	**Riser Coolant Gate Open**	Scram when gate opens in cross-pool mode	1	n/a	n/a
15.	**Coolant Gate Open**	Scram when either gate opens in downcomer mode	2	n/a	n/a

*one of which shall be the log power/period monitoring channel

TS 3.2.3, Table 3.2.3-1, item 1, would require that the wide range log period power module (PPM) provide a scram when reactor period is 3 seconds or less for either flow mode (forced convection or natural convection). As discussed in SAR Chapter 7, as supplemented, the reactor period scram is only provided by the log PPM channel, which also provides a reactor power level indication and scram. UML does not credit the period scram in its reactivity transient analyses because it conservatively assumes that the single required scram fails (see SER Section 5.2). Normally, however, the reactor will scram on a short period. The NRC staff finds that this scram helps to reduce the likelihood and severity of any reactor overpower event, or power increase that is faster than can be effectively controlled by the operator. The NRC staff also finds that TS 3.2.3, Table 3.2.3-1, item 1, appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 by specifying a minimum nuclear scram required for reactor operation. Therefore, based on the above, and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.2.3, Table 3.2.3-1, item 1, is acceptable.

TS 3.2.3, Table 3.2.3-1, item 2, would require two channels be able to provide scrams when the reactor is operating in either flow mode. The scram setpoints are required to be no greater than 1.15 MWt for forced convection mode and no greater than 115 kWt for natural convection mode. As discussed in SAR Chapter 7, as supplemented, there are three channels capable of providing power level scrams (the log PPM and two linear power channels). However, to satisfy the minimum reactor protection system scrams required by TS Table 3.2.3-1, the power level scram for the PPM and the power level scram of at least one of the two linear power channels are required to be operable to operate the UMLRR. In its letter dated January 30, 2021 (Ref. 99), UML clarified information in SAR Section 7.4.1.1.5 by stating that the linear channels do not operate in a 'one out of two mode,' but that only one linear channel is required and the second channel provides redundancy to the required channel, and the required log PPM channel provides redundancy and diversity to the single required linear channel. The NRC staff finds that TS 3.2.3, Table 3.2.3-1, item 2, helps ensure that the reactor will scram to terminate any potential overpower scenario, even if one of two required scrams fails to function correctly. The NRC staff finds that by requiring two different types of channels to provide power level scrams, TS 3.2.3, Table 3.2.3-1, item 2, also helps ensure diversity in the operable scrams. The NRC staff also finds that the required setpoints in TS 3.2.3, Table 3.2.3-1, item 2, are

consistent with the power level LSSSs and are less than the conservative 1.25 MWt and 125 kWt power levels assumed in UML's worst-case steady-state thermal-hydraulic analyses (see SER Section 2.6), reactivity transient analyses (see SER Section 5.2), and worst-case loss of flow analyses (see SER Section 5.4), helping ensure adequate safety margins and that actual UMLRR operation will be bounded by the analyses. The NRC staff further finds that TS 3.2.3, Table 3.2.3-1, item 2, appropriately implements guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 by specifying minimum nuclear scrams required for reactor operation. Therefore, based on the above, and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.2.3, Table 3.2.3-1, item 2, is acceptable.

TS 3.2.3, Table 3.2.3-1, item 3, would require a scram on low primary coolant flow rate, with a setpoint of no less than 1,400 gpm, during forced convection operation. As discussed in SAR Section 7.4.2.1, the primary coolant flow rate is measured by an orifice plate installed in the primary piping after the heat exchanger, and the signal from this measurement can provide a reactor scram. The NRC staff finds that TS 3.2.3, Table 3.2.3-1, item 3, helps ensure that the reactor will scram in any low flow scenario during UMLRR forced convection operation. The NRC staff also finds that the required 1,400 gallon per minute minimum setpoint is consistent with the forced convection flow rate LSSS and is greater than the conservative 1,370 gpm flow rate assumed in UML's worst-case steady-state thermal-hydraulic analyses (see SER Section 2.6), reactivity transient analyses (see SER Section 5.2) for forced flow operation, and worst-case loss of flow analyses (see SER Section 5.4), helping ensure adequate safety margins and that actual UMLRR operation will be bounded by the analyses. The NRC staff also finds that TS 3.2.3, Table 3.2.3-1, item 3, appropriately implements guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 by specifying a minimum process scram required for reactor operation. Therefore, based on the above, and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.2.3, Table 3.2.3-1, item 3, is acceptable.

TS 3.2.3, Table 3.2.3-1, item 4, would require a scram on low pool water level, with a setpoint of no less than 24.25 feet above core centerline, during both flow modes. As discussed in SAR Section 7.4.2.3, the water level is measured by a non-contact ultrasonic transducer, and the signal from this measurement can provide a reactor scram. The NRC staff finds that TS 3.2.3, Table 3.2.3-1, item 4, helps ensure that the reactor will scram in any low pool level scenario during UMLRR operation. The NRC staff also finds that the required 24.25-foot minimum setpoint is consistent with the pool level LSSSs and that TS 3.2.3, Table 3.2.3-1, item 4, will help ensure adequate safety margins and that actual UMLRR operation will be bounded by UML's worst-case steady-state thermal-hydraulic and accident analyses, including the LOCA analyses (see SER Section 5.3). The NRC staff also finds that TS 3.2.3, Table 3.2.3-1, item 4, appropriately implements guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 by specifying a minimum process scram required for reactor operation. Therefore, based on the above, and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.2.3, Table 3.2.3-1, item 4, is acceptable.

TS 3.2.3, Table 3.2.3-1, items 5 and 6, would require scrams on pool inlet temperature and pool temperature (i.e., bulk pool temperature) with setpoints of no greater than 108 °F (42.2 °C). The pool inlet temperature scram is only required for the forced convection flow mode. As discussed in SAR Section 7.4.2.2, primary coolant and pool water temperatures are measured by 3 resistance temperature detectors for the pool inlet, core outlet, and bulk pool. The measurements from any of these detectors can provide a reactor scram. However, TS 3.2.3, Table 3.2.3-1, items 5 and 6, only require the pool inlet temperature (during forced convection) and bulk pool temperature (during any operation) scrams, respectively. The NRC staff finds that

TS 3.2.3, Table 3.2.3-1, items 5 and 6, help ensure that the reactor will scram in a high coolant temperature scenario. The NRC staff finds that the 108 °F (42.2 °C) maximum setpoint is consistent with the pool inlet and bulk pool temperature LSSSs for forced convection and natural convection (bulk pool temperature is not an LSSS for forced convection operation and, therefore, the required scram on bulk pool temperature for forced convection operation provides additional conservatism). The NRC staff also finds that the 108 °F (42.2 °C) setpoint is less than the conservative core inlet temperatures UML used in its worst-case steady-state thermal-hydraulic analyses (see SER Section 2.6), reactivity transient analyses (see SER Section 5.2), and worst-case loss of flow analyses (see SER Section 5.4), helping ensure adequate safety margins and that actual UMLRR operation will be bounded by the analyses (for its analyses, UML assumed that the core inlet temperature is equal to the pool inlet temperature or bulk temperature, as applicable). Additionally, the NRC staff finds that TS 3.2.3, Table 3.2.3-1, items 5 and 6, appropriately implement guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 by specifying minimum process scrams required for reactor operation. Therefore, based on the above, and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.2.3, Table 3.2.3-1, items 5 and 6, are acceptable.

TS 3.2.3, Table 3.2.3-1, item 7, would require a control room manual scram. SAR Section 7.4.3.2 states that in addition to the manual scram button in the control room, emergency manual scram buttons are also located in strategic locations in the reactor building, and one is located outside the reactor building. In response to RAI-14.4.10 (Ref. 71), UML stated that while the other manual scram buttons outside the control room provide additional safety functionality, the reactor operator in the control room would be the primary initiator of a manual scram should conditions warrant such action. The NRC staff finds that by requiring a manual scram button in the control room be operable, TS 3.2.3, Table 3.2.3-1, item 7, would help ensure that the reactor operator has the ability to immediately shut down the reactor when necessary. The NRC staff also finds that TS 3.2.3, Table 3.2.3-1, item 7, appropriately implements guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 by specifying a minimum manual scram required for reactor operation. Therefore, based on the above, and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.2.3, Table 3.2.3-1, item 7, is acceptable.

TS 3.2.3, Table 3.2.3-1, item 8, would require a scram when the detector voltage for any required power or period channel is 500 volts or less. In SAR Table 7-5 in SAR Section 7.4.3.2, UML specifies that the nominal setpoint for the loss of high voltage scram for the linear and log/period channels is 700 volts or less (the startup channel also has a loss of high voltage scram with a setpoint of 600 volts or less, but this scram is not part of TS 3.2.3). The NRC staff finds that TS 3.2.3, Table 3.2.3-1, item 8, will help ensure that the reactor scrams if there is a loss of high voltage to a TS-required power or period channel that could affect the operability of that channel. The NRC staff also finds that TS 3.2.3, Table 3.2.3-1, item 8, appropriately implements guidance in NUREG-1537, Appendix 14.1, by specifying a minimum facility-specific (based on the SAR, as supplemented) scram required to ensure the minimum channels required for reactor operation are operable and to provide a fail-safe shutdown in the event of the loss of detector voltage. Therefore, based on the above, and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.2.3, Table 3.2.3-1, item 8, is acceptable.

TS 3.2.3, Table 3.2.3-1, items 9 and 10, would require that the process and drives controls displays watchdog timers (WDTs), respectively, each provide a scram for a communication loss of greater than 10 seconds. SAR Section 7.6.2.1 states that the process controls and drives controls displays are not of paramount concern because the main parameters of concern for the

reactor protection system (i.e., the parameters with associated LSSSs: reactor power level, primary coolant temperature, primary coolant flow, and pool height) can be monitored by the operator on the instrumentation panel located adjacent to the control console or using other stand-alone indicators available from the control room. Additionally, the scrams for these parameters are provided by separate stand-alone channels and the separate safety chain scram circuit. As discussed in SAR Section 7.6.1, the process and drives controls displays are the two human-machine interface (HMI) systems in the UMLRR control room that provide the graphical user interface to the various display and control functions of the reactor. For example, the process controls system (PCS) allows display of the status and control of the ventilation and coolant systems, and the drives controls system (DCS) display allows for display and control of the control blade and regulating rod positions. (The area radiation monitoring system (ARMS) HMI is a separate HMI in the control room that provides displays and alarms for the various area radiation monitors.) The three HMI systems in the control room, the PCS, the DCS, and the ARMS, are discussed further in SER Chapter 3.

In its response to RAI-7.2 (Ref. 79), UML stated that the ARMS, the PCS, and the DCS HMI displays may be considered as non-safety-related. SAR Section 7.6.2.1 states that, nonetheless, the PCS and the DCS HMIs employ a fail-safe WDT that activates a trip relay in the scram circuit. In its RAI-7.14 response (Ref. 79), UML stated that a communication loss of greater than 10 second between the boards, controller, or displays will scram the reactor to ensure that any computer or communication failure will result in a reactor shutdown condition. Additionally, the SAR states that following a scram due to failure of one or both of the PCS and the DCS HMI displays, the reactor operator would still be able to verify that the reactor was shut down by direct visual observation of the power measuring instruments (and, if necessary, visually verifying the control blades are fully inserted in the reactor core). UML also stated, in its response to RAI-7.14, that the 10-second setpoint for the WDT is an adequate response time for a failure of either the PCS or the DCS HMI, since the reactor protection channels specified in TS 3.2.5 operate independently of the PCS HMI and include independent displays (as described in SAR Sections 7.4.1 and 7.4.2, as supplemented) and since any control blade movement that could be related to a DCS HMI failure during the 10 second delay until a DCS WDT reactor scram would be bounded by the reactivity transient scenarios analyzed in the SAR, as supplemented (the NRC staff notes that TS-required reactor power and/or period scrams would also separately help ensure that any reactivity transient is terminated before a SL could be reached, as discussed earlier in this SER Section and in SER Section 5.2). As discussed above, the scram function for LSSS parameters is also independent of the HMIs.

The NRC staff finds that the PCS and the DCS HMI displays are not necessary to ensure that automatic protective action is initiated to prevent the SL from being exceeded, because separate channels provide scram functions for LSSS parameters. However, the WDTs ensure that failure of the PCS or the DCS HMI will result in a fail-safe shutdown of the reactor. Accordingly, the NRC staff finds that TS 3.2.3, Table 3.2.3-1, items 9 and 10, will help ensure that the reactor enters a safe shutdown if there are any issues with the HMI displays that could prevent normal operation, control, and monitoring of reactor systems. The NRC staff also finds that TS 3.2.3, Table 3.2.3-1, items 9 and 10, appropriately implement guidance in NUREG-1537, Appendix 14.1, by specifying minimum facility-specific (based on the SAR, as supplemented) scrams required for reactor operation. Therefore, based on the above, and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.2.3, Table 3.2.3-1, items 9 and 10, are acceptable.

Although, as discussed above, UML states that the ARMS HMI is not safety-related, the NRC staff notes that one TS 3.6.1-required function of the UMLRR radiation monitoring system,

specifically, the control room readout for the pool level constant air monitor (CAM), is only provided by the ARMS HMI display (see SER Section 4.1.4). However, as discussed in SER Section 4.1.4, the TS-required CAM alarm is still independent of the Area Radiation Monitoring Computer Data Acquisition System (ARM CDAS), including the ARMS computer and HMI, and there are control room alarms and readouts for other TS-required radiation monitors that are also independent of the ARM CDAS. Furthermore, the ability of the ARMS to provide an automatic confinement isolation function or reactor scram is not required by TSs, and reactor operators can manually isolate confinement or scram the reactor if needed, independently of the ARMS. Therefore, based on the above, the NRC staff finds that the ARMS HMI does not represent a single point of failure for the UMLRR radiation monitoring system, and it is acceptable that the UMLRR TSs, including TS 3.2.3, do not include a TS requirement for any ARMS HMI WDT scram. As discussed in SER Section 6.7, the NRC staff finds that UML has LCO TSs that satisfy the provisions of 10 CFR 50.36(c)(2).

TS 3.2.3, Table 3.2.3-1, item 11, would require a seismic disturbance scram that scrams the reactor on seismic motion. As discussed in SAR Section 3.4, this scram is generated by a seismic detector (switch) mounted on the biological shield. TS 3.2.3, Table 3.2.3-1, item 11, does not include any requirement that the seismic disturbance scram be associated with a specific level of motion, but SAR Section 3.4 states that the switch is set to scram at a Modified Mercalli Scale IV. UML references the guidance in ANSI/ANS-15.7-1977 (Ref. 72), which states that a research reactor should have a seismic alarm if the reactor is located in an area that could have seismic activity of Modified Mercalli Scale V or greater. In its letter dated September 30, 2020 (Ref. 98), UML stated that the specific setpoint was in the TSs prior to the renewal, but is being removed. However, UML stated that the setpoint need not be in the TSs since it is a nominal value and since the scram can continue to serve its purpose without a specific TS-required setpoint. As discussed in SER Section 5.8, the UMLRR is designed to withstand any seismic activity expected in the Lowell, Massachusetts area and is also designed such that an earthquake should not prevent a scram from occurring. SAR Section 3.4 states that if the UMLRR scrammed due to an earthquake, it would not be restarted until the reactor structure could be examined to ensure that no damage had occurred. The NRC staff finds that TS 3.2.3, Table 3.2.3-1, item 11, would help ensure that the reactor shuts down during seismic activity that is less than any seismic activity that could damage the reactor structure or systems. The NRC staff finds that it is not necessary for TS 3.2.3, Table 3.2.3-1, item 11, to require a specific setpoint for the seismic scram, because the TS requires that the scram will activate on seismic motion and the SAR provides an appropriate nominal setpoint that is used for the scram. The NRC staff also finds that TS 3.2.3, Table 3.2.3-1, is consistent with guidance in ANSI/ANS-15.7-1977 because it helps ensure that the reactor operator is made aware of significant seismic activity. The NRC staff further finds that TS 3.2.3, Table 3.2.3-1, item 11, appropriately implements guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 by specifying a minimum facility-specific (based on the SAR, as supplemented) scram required for reactor operation. Therefore, based on the above, and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.2.3, Table 3.2.3-1, item 11, is acceptable.

TS 3.2.3, Table 3.2.3-1, items 12 and 13, would require a reactor scram if the reactor bridge is moved more than 1 inch when the reactor is operating in forced or natural convection mode and would require a reactor scram when primary piping alignment is not achieved when the reactor is operating in forced convection mode, respectively. SAR Table 7-5 in SAR Section 7.4.3.2 lists the “Bridge Movement Limit Switch,” which provides the reactor bridge movement scram (SAR Table 7-5 indicates that the reactor bridge movement scram will only occur during forced convection mode, but in its letter dated September 30, 2020 (Ref. 98), UML clarified that the

scram occurs during any operation (forced and natural convection modes), consistent with the requirement of TS 3.2.3, Table 3.2.3-1, item 12). SAR Section 5.2 states that this switch is located near a cogwheel for bridge movement and prevents bridge movement while the reactor is operating. (SAR Section 5.2 also discusses other limit switches on the reactor bridge that have associated scrams to help ensure proper bridge position and alignment, but these are not included in TS 3.2.3, Table 3.2.3-1, item 12.) SAR Sections 5.2 and 7.4.3.2 and UML's letter dated January 30, 2021 (Ref. 99), also discuss various limit switch scrams related to primary coolant piping alignment or valve positioning. The single required primary piping alignment scram described in the TS actually consists of eight separate limit switches, any one of which can individually cause actuation of the primary piping alignment scram. The proper positioning for each limit switch (i.e., the "closed" position; the switch causes a scram if it is "open," or not in the proper position) depends on the position of the reactor in the pool (i.e., stall pool or bulk pool) and the forced convection mode the reactor is operating in (i.e., cross-pool or downcomer; see discussion of TS 3.2.3, Table 3.2.3-1, items 14 and 15, below). The limit switches and positions for each of the four combinations are described in UML's letter dated January 30, 2021. For each combination, there are two primary piping swivel joint alignment switches and four primary piping valve alignment switches (which ensure primary piping and valves are correctly aligned to direct flow through the core for a given combination), and two additional valve alignment switches (which are intended to ensure that piping is aligned to prevent possible siphoning of the pool during forced-convection operation). The NRC staff finds that TS 3.2.3, Table 3.2.3-1, items 12 and 13, help ensure that if the reactor bridge or primary coolant piping/valves are moved out of position during reactor operation such that proper flow through and cooling of the core could be affected, the reactor will be shut down. The NRC staff finds that the additional bridge alignment scrams that are discussed in the SAR, but not required by TS 3.2.3, Table 3.2.3-1, items 12 and 13, help provide additional redundancy to ensure proper alignment of reactor components for the applicable flow mode, but need not be required by the TSs because the scrams required by TS 3.2.3, Table 3.2.3-1, items 12 and 13, and other TSs such as TS 3.2.3, Table 3.2.3-1, item 3, discussed above, are sufficient to help ensure proper flow through the core. The NRC staff finds that although the reactor may be operated in any part of the pool when it is in natural convection mode (in forced convection mode, the bridge and reactor must be in a specific position to allow primary coolant system connections to be made) and no specific bridge or piping alignment is necessary, the TS 3.2.3, Table 3.2.3-1, item 12, requirement for a scram during bridge movement in natural convection mode would help ensure that there is no bridge movement that could otherwise affect reactor operations (e.g., cause power oscillations due to the core movement/vibration). The NRC staff also finds that TS 3.2.3, Table 3.2.3-1, items 12 and 13, appropriately implement guidance in NUREG-1537, Appendix 14.1, by specifying minimum facility-specific (based on the SAR, as supplemented) required scrams for reactor operation. Therefore, based on the above, and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.2.3, Table 3.2.3-1, items 12 and 13, are acceptable.

TS 3.2.3, Table 3.2.3-1, items 14 and 15, would require reactor scrams when the riser gate opens when the reactor is operating with forced convection flow in cross-pool (or cross-stall) mode and when either coolant gate (i.e., the riser gate or the downcomer gate) opens when the reactor is operating with forced convection flow in the downcomer mode, respectively. As discussed in SAR Section 5.2 and UML's response to RAI-14.3.33 (Ref. 71), two modes exist for reactor operations with forced convection: downcomer mode and cross-stall mode. In cross-stall mode, coolant that leaves the heat exchangers is fed into the pool section not occupied by the reactor, instead of being fed directly into the core inlet channel (or downcomer plenum) on the suspension frame. The downcomer plenum is effectively bypassed and the coolant is drawn from the opposite side of the pool into the unenclosed inlet plenum directly

above the core, and then down into the core. In downcomer mode, coolant is supplied from the heat exchangers directly to the inlet plenum above the core via the downcomer plenum. In either mode, coolant is returned from the core to the heat exchanges via the riser plenum. The riser gate and downcomer gate are located on the riser plenum and the downcomer plenum, respectively (approximately 2 feet below the couplings between the plenums and the corresponding primary coolant pipes). These hinged coolant gates provide openings between the enclosed flow channel plenums and the pool to assist the circulation of pool water through the core when the reactor is operating in natural convection mode. When the reactor is not operating with forced flow, these gates are normally open. During forced flow operation, the gates are designed such that the flow through each plenum forces the associated gate in each plenum shut (for forced flow operation in cross-stall mode, the downcomer plenum is effectively bypassed, so only the riser gate is forced shut). In the unlikely event one of these gates fails to be forced closed by the coolant flow during forced convection operation, some limited amount of flow could be diverted from the core through the open gate; therefore, both gates have limit switches that scram the reactor if they are open during forced convection operation (the limit switch scram for the downcomer gate is bypassed during cross-stall mode operation, because there is no primary flow through the downcomer plenum and no need for the downcomer gate to be closed). The NRC staff finds that by requiring that the reactor scram when a coolant gate that could divert flow from the core during forced convection operation is not closed, TS 3.2.3, Table 3.2.3-1, items 14 and 15, help ensure adequate flow through the core during forced convection operation and that forced convection operation is within the bounds of UML's steady-state and transient analyses. The NRC staff finds that TS 3.2.3, Table 3.2.3-1, items 14 and 15, in conjunction with TS 3.2.3, Table 3.2.3-1, items 3, 12, and 13, help provide additional assurance that there is a reactor scram if the reactor is operated with insufficient flow during forced convection operation. The NRC staff also finds that TS 3.2.3, Table 3.2.3-1, items 14 and 15, appropriately implement guidance in NUREG-1537, Appendix 14.1, by specifying minimum facility-specific (based on the SAR, as supplemented) scrams required for reactor operation. Therefore, based on the above, and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.2.3, Table 3.2.3-1, items 14 and 15, are acceptable.

The NRC staff notes that the UMLRR TSs prior to this license renewal included a requirement for a reactor scram when the reactor key switch is moved to the off position, but this TS is not in the renewal TSs. In response to RAI-14.4.12 (Ref. 71), UML stated that this switch energizes the scram circuit, but that it is not a protective scram and that similar research reactor facilities do not have this switch as a TS-required scram. The NRC staff finds that this switch is a necessary component because it helps ensure that the reactor can be secured per the TS definitions, but because the scram that would occur if this switch is turned off during reactor operation does not serve a safety function (other mechanisms, e.g., the manual scram button, would be available to allow the operator to scram the reactor if needed), no TS is required for this switch's scram function. As discussed in SER Section 6.7, the NRC staff finds that UML has LCO TSs that satisfy the provisions of 10 CFR 50.36(c)(2).

Renewed TS 3.2.5, "Minimum Channels Needed for Reactor Operation," would state:

> Applicability:
>
> This specification applies to channels in the reactor protection and control systems.

Objective:

To stipulate the minimum number of channels that shall be operable to ensure that the reactor operator has sufficient information for safe operation of the reactor.

Specification:

The reactor shall not be operated unless the channels in the Table 3.2.5-1 are operating.

Table 3.2.5-1
Minimum Reactor Protection Channels

	Channel	Operations Mode	Minimum Required
1.	Start-up Count Rate	Both	1
2.	Reactor Period	Both	1
3.	Reactor Linear Power Level	Both	1
4.	Reactor Log Power Level	Both	1
5.	Primary Coolant Flow Rate	Forced	1
6.	Pool Water Level	Both	1
7.	Pool Inlet Temperature	Forced	1
8.	Pool Temperature	Both	1

TS 3.2.5 would require that measuring channels for startup count rate, reactor period, reactor linear power level, reactor log power level, pool water level, and pool temperature (i.e., bulk pool water temperature) be operating during any reactor operation and that measuring channels for primary coolant flow rate and pool inlet temperature also be operating when the reactor is operating in the forced convection mode of operation. SAR Section 7.4 describes these measuring channels. As noted in UML's basis for TS 3.2.5, these channels ensure that measurements of reactor power level and process variables are adequately displayed during reactor operation. TS 3.2.5 stipulates the minimum number of reactor protection channels that are required to be operating to ensure that the reactor operator has sufficient information for safe operation of the reactor. (TS 3.2.5 differs from TS 3.2.3 because TS 3.2.5 specifies required measuring channels, while TS 3.2.3 specifies required scrams.) The UMLRR has a dedicated startup count rate channel with a boron-10 proportional counter detector, as discussed in SAR Section 7.4.1.3. However, as discussed in UML's response to RAI-14.3.13 (Ref. 71) and UML's letter dated April 10, 2019 (Ref. 73), the log power channel is also capable of functioning as a startup channel, and UML stated that having redundant channels capable of monitoring in the source range for reactor startup allows operational flexibility if a channel is inoperable (or not measuring enough counts to ensure adequate source range monitoring; see discussion of TS 3.2.6(2) below). The NRC staff finds that TS 3.2.5 helps ensure that important reactor operating parameters are measured, and that indications of these parameters are readily available to the reactor operator. The NRC staff finds that the requirement for at least one each of linear and log power level monitors helps ensure diversity. The NRC staff also finds that the two duplicate linear channels provide additional redundancy of power monitoring instrumentation that is beyond the UMLRR TS requirements. The NRC staff finds that the TS 3.2.5, Table 3.2.5-1, item 1, requirement for a startup channel may be satisfied by either the dedicated startup channel with a proportional detector or the log power channel, because either channel is capable of monitoring in the source range (the NRC staff notes that UML could operate the reactor with the dedicated startup channel inoperable but could not operate the

reactor with the log channel entirely inoperable, because the log channel is separately required by TS 3.2.5, Table 3.2.5-1, item 4, and also provides the reactor period channel function required by TS 3.2.5, Table 3.2.5-1, item 2). The NRC staff also finds that TS 3.2.5 appropriately implements guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 by specifying the minimum required measuring channels needed for safe reactor operation. Therefore, based on the above, and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.2.5 is acceptable.

Renewed TS 3.2.6, "Reactor Control System Interlocks," would state:

> Applicability:
>
> This specification applies to the reactor control system.
>
> Objective:
>
> To stipulate the minimum number of interlocks available to inhibit control blade withdrawal.
>
> Specifications:
>
> The following interlocks to prevent control blade withdrawal shall be operable when the reactor is operating:
>
> (1) Scram circuit not reset.
>
> (2) Start-up neutron count rate is ≤ 2 counts per second.
>
> (3) The reactor period ≤15 seconds.

TS 3.2.6(1) would require an interlock to prevent control blade withdrawal when the scram circuit is not reset. SAR Sections 7.2.2.1 and 7.3.3 state that the UMLRR withdrawal inhibit circuit design requires that limit switch contacts in the scram chain be in the normal safe position (i.e., not in the position that would cause a scram) and that relay contacts in the scram chain be reset and energized in the normal closed condition. UML stated that this ensures that the reactor operator has enabled the RPS and that operating conditions are normal prior to a reactor startup. (UML stated that this interlock is part of the control blade withdrawal inhibit circuit, but separately from this interlock, the control blade drive magnets are also not energized until the scram circuit is reset.) The NRC staff finds that by requiring that the scram circuit be reset, TS 3.2.6(1) helps ensure that the RPS is functioning properly (e.g., no issues with any scram relay contacts) and that reactor conditions are normal (i.e., no conditions are present that would result in any reactor scram) prior to control blade withdrawal. The NRC staff also finds that TS 3.2.6(1) appropriately implements the guidance in ANSI/ANS-15.1-2007 by specifying a minimum required interlock. Therefore, based on the above, and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.2.6(1) is acceptable.

TS 3.2.6(2) would require an interlock to prevent control blade withdrawal when the startup channel count rate is 2 cps or less. SAR Section 7.3.3 states that this inhibit function ensures that nuclear instrumentation for reactor startup is operating and that there is adequate neutron indication for monitoring the approach to critical. As discussed above, UML may use either its

dedicated startup channel or its log power channel for source range monitoring for reactor startup. As discussed in SAR Section 7.3.3 and UML's letter dated April 10, 2019 (Ref. 73), both channels have a control blade withdrawal inhibit function for low count rate. As discussed in UML's response to RAI-14.3.13 (Ref. 71), both of these channels have neutron sensitivity such that a count rate above 2 cps is sufficient to allow reactor startup. The NRC staff finds that TS 3.2.6(2) helps ensure that whichever channel is used for source range monitoring during reactor startup, there is adequate neutron indication for the startup. The NRC staff also finds that TS 3.2.6(2) appropriately implements guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 by specifying a minimum required interlock. Therefore, based on the above, and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.2.6(2) is acceptable.

TS 3.2.6(3) would require an interlock to prevent control blade withdrawal if the reactor period is 15 seconds or less. As discussed in UML's submittal dated April 10, 2019 (Ref. 73), this control withdrawal blade inhibit is provided by the proposed Thermo-Fisher Scientific (TFS) log power level channel, which also measures the reactor period. The interlock is used to initiate an alarm annunciator and open an inhibit relay to prevent control blade withdrawal. The NRC staff finds that requiring this interlock will help ensure that reactor power changes are gradual and that power is not increased on an excessively short period due to operator error or other issues. The NRC staff also finds that the rod withdrawal inhibit function for the reactor period channel helps limit the possible rate of power increase and provides additional redundancy to the TS 3.2.3, Table 3.2.3-1, item 1, required scram if the reactor period is decreased to 3 seconds or less. The NRC staff also finds that TS 3.2.6(3) appropriately implements the guidance in ANSI/ANS-15.1-2007 by specifying a minimum required interlock. Therefore, based on the above, and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.2.6(3) is acceptable.

Renewed TS 4.2.3, "Reactor Protection System Scrams," would state, in part:

> Applicability:
>
> This specification applies to the surveillance requirements for the Reactor Protection System.
>
> Objective:
>
> To ensure Reactor Protection System limiting conditions for operation are met.
>
> Specifications:
>
> (1) A channel check of each channel listed in Specification 3.2.5, specific to the operating mode, shall be performed daily when the reactor is in operation.
>
> (2) A channel test, including scram function where applicable, of each channel listed in Specification 3.2.5, specific to the operating mode, shall be performed prior to the first reactor start-up of the day.
>
> (3) A channel calibration of the reactor power level and period channels (Linear and Log PPM) shall be made annually.
>
> …

(5) A channel calibration of the following channels shall be made annually:
   a. Pool temperature
   b. Primary coolant flow rate
   c. Pool water level
   d. Pool inlet temperature

(6) The manual scram in the control room shall be verified to be operable prior to the first reactor start-up of the day.

(7) All scrams listed in Specifications 3.2.3 items 8 – 15 and 3.2.4 shall be verified operable annually.

(8) The interlocks listed in Specification 3.2.6 shall be verified operable annually.

TS 4.2.3(1) would require that a channel check of each channel required by TS 3.2.5 for the operating mode or modes (i.e., natural or forced convection modes) planned for each day the reactor is operating be performed during that day's reactor operation. The NRC staff finds that TS 4.2.3(1) helps ensure that UML verifies, at intervals to provide early indication of unusual channel behavior, by observation or comparison of each channel with other independent channels (i.e., by qualitative verification of channel indications), that the TS 3.2.5-required channels (discussed earlier in this SER section) are performing acceptably. Therefore, based on the above, and based on its 10 CFR 50.36(c)(3) findings for TSs on SRs in SER Section 6.7, the NRC staff concludes that TS 4.2.3(1) is acceptable.

TS 4.2.3(2) would require that a channel test (i.e., an introduction of a signal into the channel for verification that it is operable), including a test of scram function where applicable, of each channel required by TS 3.2.5 for the operating mode or modes (i.e., natural or forced convection modes) planned for each day of reactor operation, or for each operation extending more than one day, be performed prior to that day's operation or prior to the beginning of the operation extending more than one day. As discussed earlier in this SER section, TS 3.2.5 requires eight channels, designated as Table 3.2.5-1, items 1 through 8, during reactor operation. TS 3.2.3, Table 3.2.3-1, items 1 through 6, require scrams that are associated with channels designated items 2 through 8 in TS 3.2.5, Table 3.2.5-1 (the start-up count rate channel required by TS 3.2.5, Table 3.2.5-1, item 1, does not have any associated scram). The NRC staff finds that TS 4.2.3(2) helps ensure that before the reactor is operated, UML verifies the operability of all channels required by TS 3.2.5, as well as the scram functions required by TS 3.2.3, Table 3.2.3-1, items 1 through 6, which are associated with the channels required by TS 3.2.5, Table 3.2.5-1, items 2 through 8, such that the channels and scrams are capable of performing their intended indication and protective functions. The NRC staff also finds that TS 4.2.3(2) appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 by requiring channel tests and scram function tests of TS-required channels and scrams, and that its interval is consistent with the guidance in NUREG-1537, Appendix 14.1, Section 4.2(5), and ANSI/ANS-15.1-2007, Section 4.2, item (5). Therefore, based on the above, and based on its 10 CFR 50.36(c)(3) findings for TSs on SRs in SER Section 6.7, the NRC staff concludes that TS 4.2.3(2) is acceptable.

TS 4.2.3(3) would require an annual channel calibration of the linear and log reactor power and period channels required by TS 3.2.5, Table 3.2.5-1, items 2, 3, and 4. In its response to RAI-7.14.a (Ref. 79), UML stated that the channel calibration requirement in TS 4.2.3(3) applies to both of the General Atomics NMP-1000 linear power modules and the TFS Wide Range Log

Period Power Module, and that TS 4.2.3(3) is intended to satisfy the scram channel surveillance recommendation in ANSI/ANS-15.1-2007, Section 4.2, Item (5)(b). In its response to RAI-14.4.22 (Ref. 71), UML stated that, for the upgraded (Gen-2) NMP-1000 linear power channels, the calibration required by TS 4.2.3(3) includes a calibration of the NMP-1000 touchscreen, as described in the manufacturer's operations manual (UML also stated that the NMP-1000 calibration procedure also includes items performed for the previous (Gen-1) NMP-1000 channel, e.g., verification of the linearity of the response to an input signal, verification of the set points for the automatic switching of ranges, and verification of the alarm and scram set points). The NRC staff finds that TS 4.2.3(3) helps ensure that power and period channels required by TS 3.2.5, Table 3.2.5-1, items 2, 3, and 4, are periodically adjusted (if needed) such that their output sufficiently corresponds to known values of the parameters the channels measure. The NRC staff notes that TS 4.2.3(4), which is discussed and found acceptable in SER Section 2.5.1, separately requires a thermal power level verification to provide a surveillance for TS 3.1.2, but TS 4.2.3(3) specifically requires a calibration of all power level channels, which TS 4.2.3(4) does not. The NRC staff also finds that TS 4.2.3(3) appropriately implements the guidance in ANSI/ANS-15.1-2007 by requiring calibrations of channels that provide both measurement and scram functions and that its interval is consistent with the recommended interval in ANSI/ANS-15.1-2007, Section 4.2, item (5). Therefore, based on the above, and based on its 10 CFR 50.36(c)(3) findings for TSs on SRs in SER Section 6.7, the NRC staff concludes that TS 4.2.3(3) is acceptable.

TS 4.2.3(5) would require that a channel calibration of the pool temperature (i.e., bulk pool water temperature), primary coolant flow rate, pool water level, and pool inlet temperature channels be performed annually. LCO TS 3.2.5, Table 3.2.5-1, items 5, 6, 7, and 8, require primary coolant flow rate, pool water level, pool inlet temperature, and pool temperature channels, respectively. The NRC staff finds that TS 4.2.3(5) helps ensure that channels required by TS 3.2.5, Table 3.2.5-1, items 5 through 8, are periodically adjusted (if needed) such that their output sufficiently corresponds to known values of the parameters the channels measure. The NRC staff also finds that TS 4.2.3(5) appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 by requiring calibrations of channels that provide both measurement and scram functions and that its interval is consistent with the recommended interval in ANSI/ANS-15.1-2007, Section 4.2, item (5). Therefore, based on the above, and based on its 10 CFR 50.36(c)(3) findings for TSs on SRs in SER Section 6.7, the NRC staff concludes that TS 4.2.3(5) is acceptable.

TS 4.2.3(6) would require that the control room manual scram be verified operable prior to reactor operation on any day the reactor will be operated, or prior to the beginning of reactor operation extending more than one day. As discussed in UML's response to RAI-14.4.10 (Ref. 71), the reactor operator in the control room would be the primary initiator of a manual scram should conditions warrant such action. The NRC staff finds that TS 4.2.3(6) helps ensure that the control room manual scram required by TS 3.2.3, Table 3.2.3-1, item 7, is verified to be operable prior to any reactor operation, such that the reactor operator is able to quickly shut down the reactor if necessary during operation. The NRC staff also finds that the interval for TS 4.2.3(6) is appropriate because the manual scram is the primary mechanism that the operator would have to quickly shut down the reactor if needed (if it were not shut down automatically). Therefore, based on the above, and based on its 10 CFR 50.36(c)(3) findings for TSs on SRs in SER Section 6.7, the NRC staff concludes that TS 4.2.3(6) is acceptable.

TS 4.2.3(7) would require that scrams required by TS 3.2.3, Table 3.2.3-1, items 8 through 15, and TS 3.2.4 be verified operable annually. As discussed above, TS 3.2.3, Table 3.2.3-1, item 8, requires a scram when the detector voltage for any required power or period channel falls

below a minimum level, which could indicate a possible issue with the channel. TS 3.2.3, Table 3.2.3-1, items 9 and 10, require the WDTs to cause the reactor to scram if there is a loss of communications of greater than 10 seconds with either the PCS or the DCS display, respectively, as discussed in SER Section 3.6.1.1. TS 3.2.3, Table 3.2.3-1, item 11, requires a seismic disturbance scram. TS 3.2.3, Table 3.2.3-1, items 12 through 15, require scrams on limit switches that help ensure appropriate positioning or alignment of equipment to allow proper reactor cooling and/or otherwise ensure that reactor operation is not affected by an improper equipment configuration. As discussed in SER Section 4.1.5, TS 3.2.4 requires scrams on limit switches and/or airlock configurations, which provide radiation protection functions. The NRC staff finds that TS 4.2.3(7) helps ensure surveillance of detector high voltage, WDT, seismic disturbance, limit switch, and airlock integrity scrams at intervals that are appropriate given the importance of the scrams to facility safety and are consistent with research reactor industry practice and experience. The NRC staff notes that although NUREG-1537, Appendix 14.1 and ANSI/ANS-15.1-2007 do not provide specific surveillance interval recommendations for these types of scrams, ANSI/ANS-15.1-2007, Section 4.2, item (9), recommends an annual surveillance for interlocks, which can serve a similar purpose as certain scrams (e.g., limit switch scrams) required by TSs 3.2.3 and 3.2.4. Therefore, based on the above, and based on its 10 CFR 50.36(c)(3) findings for TSs on SRs in SER Section 6.7, the NRC staff concludes that TS 4.2.3(7) is acceptable.

TS 4.2.3(8) would require that the interlocks required by TS 3.2.6 be verified operable annually. As discussed above, TS 3.2.6 requires interlocks, which prevent control blade withdrawal under certain conditions, to help ensure the RPS and reactor channels are functioning properly when rods are withdrawn and to help ensure rods are not withdrawn when the reactor period is excessively short. The NRC staff finds that TS 4.2.3(8) appropriately implements the overall guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 by requiring periodic testing of interlocks and helps ensure surveillance of the TS 3.2.6-required interlocks at an interval that is consistent with the guidance in ANSI/ANS-15.1-2007, Section 4.2, item (9). Therefore, based on the above, and based on its 10 CFR 50.36(c)(3) findings for TSs on SRs in SER Section 6.7, the NRC staff concludes that TS 4.2.3(8) is acceptable.

Conclusion

The NRC staff reviewed the above information regarding UMLRR operating limits, including excess reactivity and SDM requirements, and scrams, interlocks, and channels required for operation. As discussed in SER Section 2.5.1, the NRC staff also reviewed the UMLRR power peaking factors.

The NRC staff finds that UML discussed and justified excess reactivity factors needed to ensure that the UMLRR is readily operable. Sufficient excess reactivity is available for normal operations, but the maximum allowed excess reactivity would not cause unacceptable risk to facility staff or the public because it would not preclude having sufficient SDM to shut down the reactor from any operating condition, helping to ensure that transients that could add reactivity to the reactor would not lead to loss of fuel integrity. UML also considered the design features of the control systems that ensure that excess reactivity is fully controlled under normal operating conditions.

The NRC staff finds that UML proposed and justified a minimum SDM that helps ensure that there is enough negative reactivity available to shut the reactor down from any operating condition, even if the most reactive control blade is inadvertently stuck in its fully withdrawn

position and the non-scrammable regulating rod is also fully withdrawn. The NRC staff finds that the UMLRR SDM is readily measurable.

The NRC staff finds UML's TSs for scrams, interlocks, and channels required for operation to be acceptable, as discussed above.

Additionally, as discussed in SER Section 2.5.1, the NRC staff finds that UML provided calculations of the highest power peaking factors achievable with any allowed core configuration and that these peaking factors are suitable for use in UML's thermal-hydraulic and accident analyses.

Therefore, the NRC staff concludes that UML's operating limits are acceptable.

2.6 Thermal-Hydraulic Design

Thermal-Hydraulic Methodology

SAR Sections 4.5.1.1 and 4.6, as supplemented by the response to RAI-4.2 (Ref. 23), describe UML's thermal-hydraulic analyses.

As described in SAR Section 4.5.1.1, UML performed the steady-state thermal-hydraulic analyses for natural convection and forced convection operating of the UMLRR using the NATCON and PLTEMP codes, respectively.

For transient thermal-hydraulic analyses (evaluating the consequences of both rapid and ramped reactivity changes in the reactor core and evaluating how a loss of flow scenario affects reactor performance and safety), UML used PARET, which is a transient analysis code that simulates the behavior associated with both reactivity- and flow-induced transients within the system. UML's transient thermal-hydraulic analyses are discussed and found acceptable in SER Chapter 5.

PLTEMP, as discussed in the PLTEMP manual (Ref. 45), is a code that obtains a steady-state flow and temperature solution for a reactor core, or a single fuel element. The code may be used to model fueled and non-fueled core regions. Each fuel element is modeled as one or more plates or tubes separated by coolant channels. The temperature solution begins with a one-dimensional solution across all coolant channels and fuel plate/tubes within a given fuel element, at the entrance to the element, and then the temperature solution is repeated for each axial node along the length of the element, effectively producing a two-dimensional temperature solution. PLTEMP may be used with a variety of thermal-hydraulic correlations to determine safety margins such as ONB, departure from nucleate boiling (DNB), and onset of flow instability. PLTEMP's intended use is for thermal-hydraulic analysis of research reactor performance in the sub-cooled boiling regime.

The PLTEMP code has been validated by Argonne National Laboratory (Ref. 49). The NRC staff reviewed the validation effort and finds that this effort is acceptable.

As discussed in SAR Section 4.5.1.1 and UML's response to RAI-4.2(d) and (f), UML used the PLTEMP code (modeling the entire core) to simulate pump flow through the entire reactor core to determine the amount of flow in the fuel versus the bypass channels (such as experiment positions and control blade locations). SAR Section 4.5.7 describes UML's flow distribution and fuel element flow rate calculations. For conservatism in these calculations, UML used the flow

rates assuming a 21-aluminide element core for its PLTEMP calculations, because the slightly smaller channels in the aluminide fuel result in slightly less flow through the fuel. Also, UML assumed the core contains five radiation baskets that allow flow bypass, the maximum allowed by TS 3.1.1(4), which is discussed and found acceptable in SER Section 2.2. (As also discussed in SAR Section 4.5.7, UML also calculated coolant mass fluxes through both silicide and aluminide fuel elements from its calculated aluminide element flow distributions and rates, because UML noted that PARET, the code used for UML's reactivity transient calculations, required mass fluxes. UML's PARET calculations are discussed and found acceptable in SER Chapter 5.)

In addition, once it defined the fraction of total pump flow rate that goes through the fuel channels, UML used PLTEMP (modeling only the hot element) to do a hot channel analysis at steady-state forced flow conditions to determine the thermal characteristics of the fuel, cladding, and coolant as a function of power level and flow rate. The hot channel analysis in PLTEMP was used to determine a power-to-flow map to identify, at a given power level, what total pump flow rate would lead to ONB conditions. Additionally, as discussed in UML's response to RAI-4.2(e), UML used PLTEMP to calculate the nominal and worst-case (i.e., more conservative power level, flow rate, and coolant inlet temperature than TS-required LSSSs) steady-state bulk fuel, cladding, and coolant axial temperature profiles along the length of the hot channel of the LCC.

As discussed in SAR Section 4.5.1.1, NATCON is a relatively simple natural convection steady-state analysis tool used to simulate the conditions of natural convection flow in a thin rectangular fuel channel. For a given power level, it balances buoyancy and friction forces to determine the steady-state flow rate in the channel for a given heat source. UML used NATCON to find the power level at which ONB occurs in the hottest coolant channel under steady-state natural convection conditions. Additionally, as discussed in UML's response to RAI-4.2(e), UML used NATCON to calculate the nominal and worst-case (i.e., power level and bulk pool temperature (assumed to be equivalent to core inlet temperature) both more conservative than TS-required LSSSs) steady-state bulk fuel, cladding, and coolant axial temperature profiles along the length of the hot channel of the LCC.

As discussed in SAR Section 4.6 and UML's response to RAI-4.2(e), for UML's NATCON and PLTEMP calculations for worst-case natural and forced convection scenarios, hot channel factors are used to account for fuel design tolerances and for uncertainties in various calculated and measured parameters. These factors are applied as three separate components: F_q, which accounts for heat flux uncertainties; F_b, which treats uncertainties in bulk flow or enthalpy change in a channel; and F_h, which quantifies the uncertainty in the heat transfer process. These factors are used to either increase or decrease, as appropriate, the nominal estimate of the heat flux, channel flow rate, and calculated heat transfer coefficient. Each hot channel factor is mathematically composed from several subfactors, which are estimated based on specified design tolerances and experience with various measurement devices and empirical corrections. In its steady-state thermal-hydraulic analyses, UML used hot channel factors used previously in the safety analyses for the UMLRR and WPI HEU to LEU conversions for the silicide and aluminide fuel elements, respectively. UML's hot channel factors and subfactors are listed in SER Table 2-8, which is adapted from SAR Table 4-11.

Table 2-8 Hot Channel Subfactors and Factors for UMLRR Silicide and Aluminide Fuel

	Silicide Fuel			Aluminide Fuel		
Subfactors	F_q	F_b	F_h	F_q	F_b	F_h
Fuel Matrix Thickness	1.08	-	-	1.07	-	-
U-235 Loading	1.05	1.05	-	1.02	1.02	-
U-235 Homogeneity	1.20	1.10	-	1.20	1.10	-
Channel Thickness	-	1.15	1.07	-	1.16	1.14
Power Measurement	1.05	1.05	-	1.05	1.05	-
Calculated Power Density	1.10	1.10	-	1.10	1.10	-
Coolant Flow Rate	-	1.10	1.10	-	1.10	1.10
Heat Transfer Coefficient	-	-	1.20	-	-	1.20
Factor (calculated from subfactors)	F_q	F_b	F_h	F_q	F_b	F_h
Hot Channel Factor	1.25	1.24	1.35	1.24	1.24	1.41

As discussed in SAR Section 4.6.1, other than the hot channel factors, other inputs to UML's NATCON and PLTEMP calculations include the fuel and channel geometry, reference pressure and temperature, the hot plate radial peaking factor ($f_a \times f_{xya}$ for the hot element (1.993), increased and rounded up to 2.1 for conservatism), and the axial power profile. As discussed in UML's response to RAI 4.2(c), the PLTEMP calculation also utilized the hot element peaking factor (f_a for the hot element (1.462), rounded up to 1.5 for conservatism) with the hot plate radial peaking factor in order to consider a variable power distribution within the hot assembly (in contrast, UML's NATCON and PARET calculations only consider the hot plate radial peaking factor, not the hot element peaking factor, and implicitly assume that neighboring plates to the hot plate produce the same amount of power). The radial peaking factor, hot element peaking factor, and axial power profile are discussed in SER Section 2.5.1.

As discussed in UML's response to RAI-4.2(a) and (b), UML's limiting criteria for steady-state operation is the ONB point. Since the pool has about 24 feet of water above the core, the ONB point would typically be reached at a plate surface temperature of 118 to 125 °C (244 to 257 °F), depending on the heat flux and flow conditions.

As stated in UML's response to RAI-4.2(a) and (b), with ONB as the limiting setpoint for steady-state operation, the only heat transfer correlations needed in UML's PLTEMP analyses were the Seider-Tate relationship for the turbulent-flow single-phase heat transfer coefficient and the Bergles-Rohsenow correlation for the ONB. (These correlations are also used in UML's PARET calculations.) For the natural circulation analysis within NATCON, an average of the analytical constant temperature and constant heat flux Nusselt number relationships for laminar flow in rectangular channels was used to obtain the single-phase heat transfer coefficient and the Bergles-Rohsenow correlation was used for prediction of the ONB point.

The NRC staff reviewed UML's methodology, discussed above, for UML's analysis of steady-state thermal-hydraulic conditions for the UMLRR. Based on the information provided by UML, the NRC staff finds that UML's methodology is appropriate and consistent with methodologies used for thermal-hydraulic analyses of other research reactor facilities. Therefore, the NRC staff finds that UML's thermal-hydraulic methodology is acceptable.

Thermal-Hydraulic Results

UML's PLTEMP power-to-flow map is shown in SER Figure 2-10 (taken from SAR Figure 4-17). The power-to-flow map identifies, for a given power level, what total pump flow rate would lead to ONB conditions in the hot channel. The map illustrates that at nominal power and flow conditions, or LSSS power and flow conditions, there is a large margin to ONB (regardless of whether hot channel factors are conservatively applied). All calculations for the map conservatively assume a pool inlet temperature of 110 °F (43.3 °C), slightly above the LSSS setpoint.

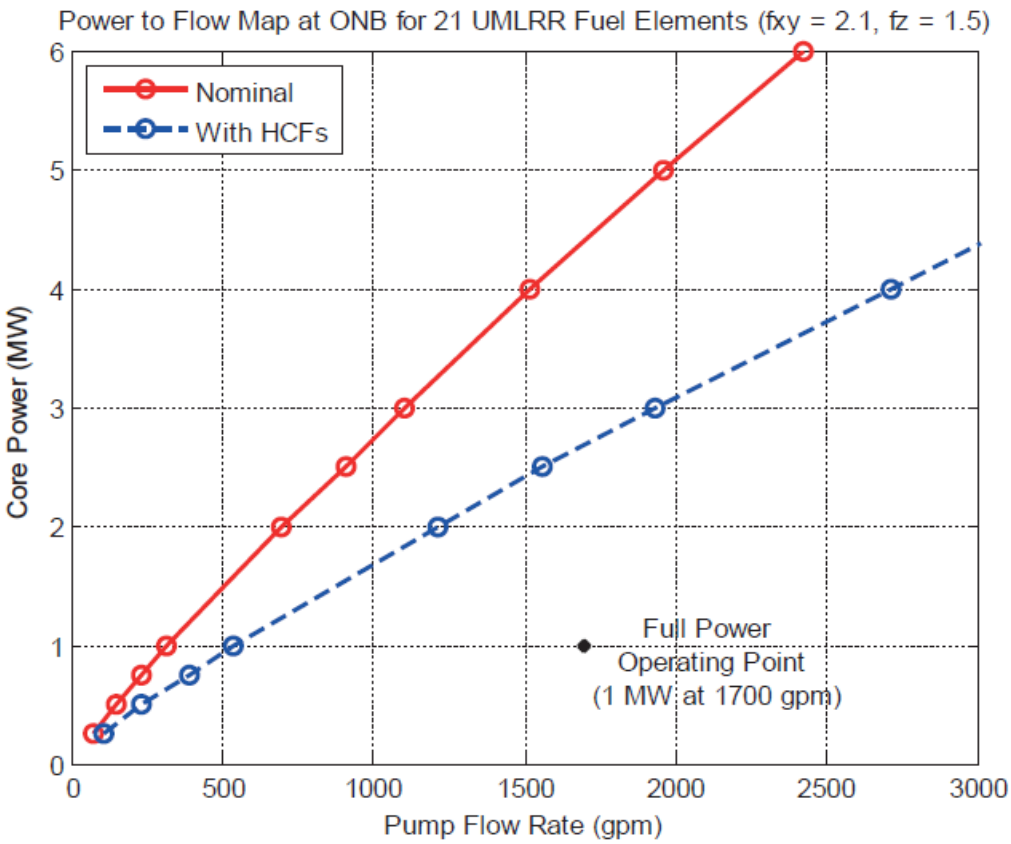

Figure 2-10 PLTEMP Power-to-Flow Map at ONB

In SAR Section 4.6.1, UML provided NATCON results which predict the reactor power level at which ONB would occur in the hot channel during natural convection operation. UML calculated a nominal best-estimate ONB-onset power level of 392 kWt and a worst-case (i.e., hot channel factors included) ONB-onset power level of 248 kWt. These ONB-onset power levels are well above the nominal and LSSS natural convection power levels of 100 kWt and 115 kWt, respectively, indicating a large margin to ONB for natural convection operation.

In its response to RAI-4.2(e), UML provided steady-state hot channel fuel, cladding, and coolant axial temperature profiles, in graphical form, for four cases: nominal forced flow (Case 1, calculated with PLTEMP), worst-case forced flow (Case 2, calculated with PLTEMP), nominal natural convection (Case 3, calculated with NATCON), and worst-case natural convection (Case 4, calculated with NATCON). The nominal calculations are best-estimate calculations assuming normal operating conditions and did not include hot channel factors (i.e., no increased

conservatism to account for uncertainties), while the worst-case calculations assumed beyond-LSSS conditions and did include hot channel factors. The power level, pump flow, and pool/core inlet temperatures assumed for each of the four cases are listed in SER Table 2-9. The axial temperature profiles for each of the four cases are shown in SER Figure 2-11, Figure 2-12, Figure 2-13, and Figure 2-14 (taken from UML's response to RAI-4.2(e)).

Table 2-9 Assumptions for PLTEMP and NATCON Axial Temperature Profile Calculations

Case No.	Description	Power (MWt)	Pump Flow (gpm)	Inlet Temp. (°C)
Case 1	Nominal forced flow	1.000	1700	30
Case 2	Worst-case forced flow	1.250	1370	43
Case 3	Nominal natural convection	0.100	0	30
Case 4	Worst-case natural convection	0.125	0	43

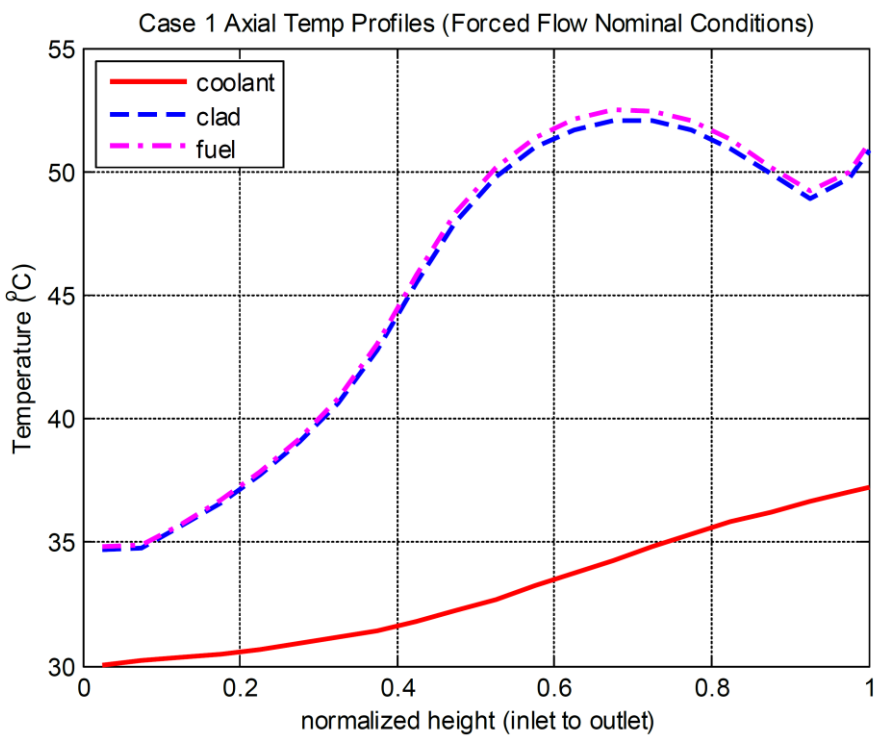

Figure 2-11 Case 1 Axial Temperature Profiles

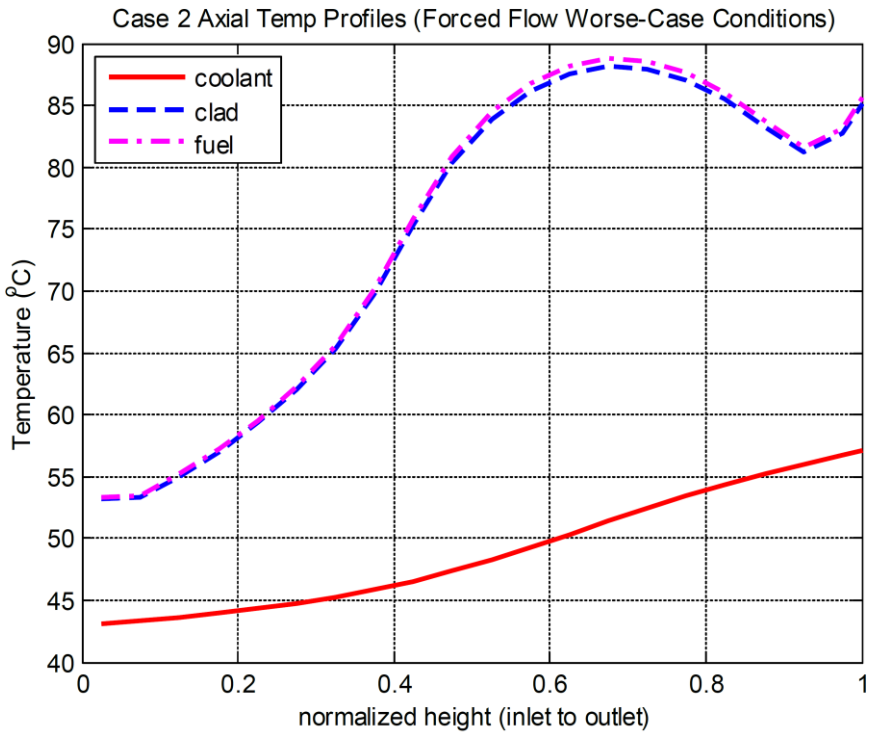

Figure 2-12 Case 2 Axial Temperature Profiles

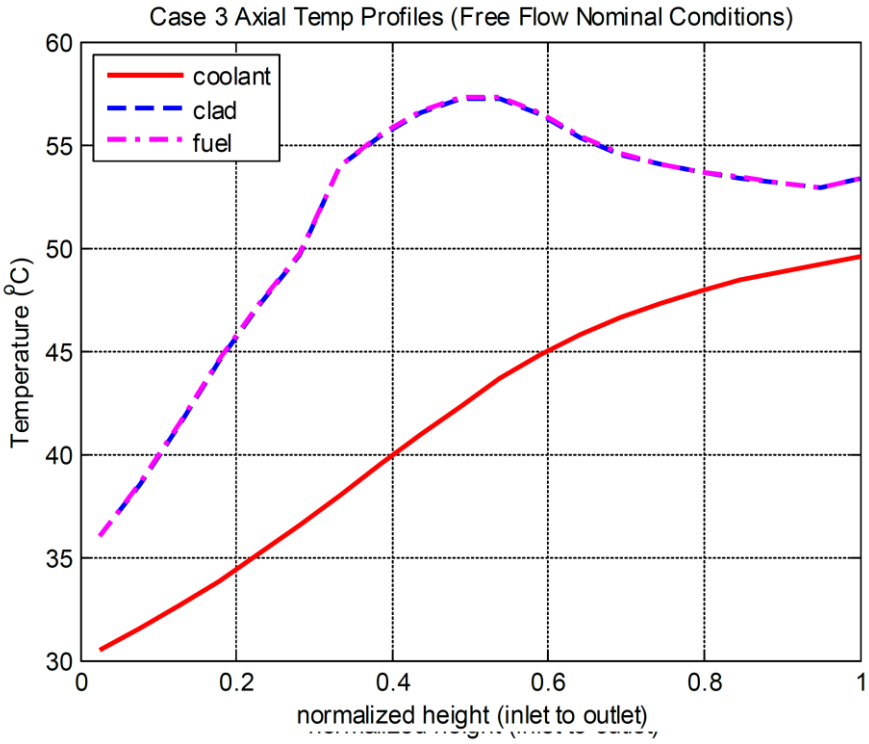

Figure 2-13 Case 3 Axial Temperature Profiles

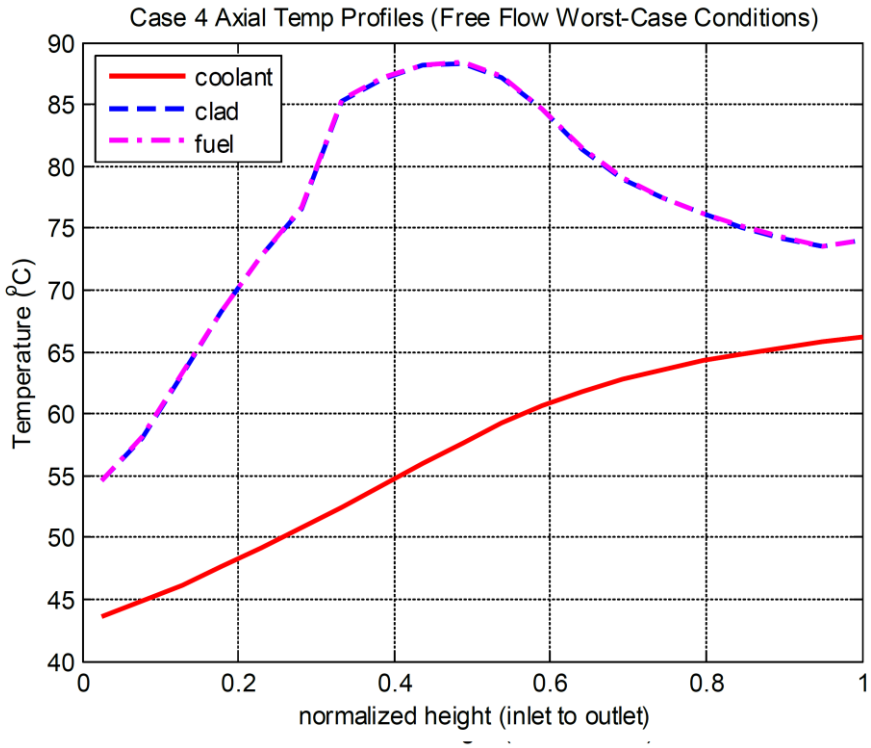

Figure 2-14 Case 4 Axial Temperature Profiles

The above axial temperature profiles indicate that for steady-state UMLRR operation, at nominal or beyond-LSSS conditions, the fuel and cladding temperatures remain well below both the TS 2.1 SL and the minimum ONB threshold of 118 °C (244 °F), confirming a large margin to ONB or possible fuel failure for any routine condition of forced flow or natural convection operation.

The NRC staff reviewed UML's thermal-hydraulic results, described above, and finds that the results demonstrate that the UML core would maintain a large margin to the SL and ONB for any allowed routine condition of forced flow or natural convection operation. Therefore, the NRC staff finds that UML's thermal-hydraulic results are acceptable.

Limiting Safety System Settings

The regulations in 10 CFR 50.36(c)(1) require licensees to specify LSSSs in their TSs. LSSSs for nuclear reactors are settings for automatic protective devices related to those variables that have significant safety functions. LSSSs for automatic protective devices must be chosen so that automatic protective actions will correct an abnormal situation before a SL is exceeded.

The principal physical barrier to the release of radionuclides for MTR-type reactors is the fuel plate cladding, and the most important parameter to maintain the fuel plate cladding integrity is the fuel and cladding temperature. A loss in the integrity of the fuel plate cladding may occur if the temperatures reach the point of allowing cladding to fail.

Renewed TS 2.2.1, “Forced Convection Mode,” would state:

Applicability:

This specification applies to the set points for the safety channels monitoring reactor thermal power, coolant flow rate, reactor coolant inlet temperature, and the height of water above the center line of the core under the condition of the forced convection mode of operation.

Objective:

To ensure that automatic protective action is initiated in order to prevent the Safety Limit from being exceeded.

Specifications:

(1) The Limiting Safety System Setting for the reactor power level shall initiate automatic protective action at or below a measured value of 1.15 MW_t.

(2) The Limiting Safety System Setting for the primary coolant flow shall initiate automatic protective action at or above a measured value of 1400 GPM.

(3) The Limiting Safety System Setting for the pool inlet temperature shall initiate automatic protective action at or below a measured temperature of 108 °F.

(4) The Limiting Safety System Setting for pool height above the core centerline shall initiate automatic protective action at or above a measured value of 24.25 ft.

TS 2.2.1(1) would require a maximum thermal power level LSSS of 1.15 MWt when the reactor is operated in forced convection mode.

TS 2.2.1(2) would require a minimum primary coolant flow rate LSSS of 1400 gpm when the reactor is operated in forced convection mode.

TS 2.2.1(3) would require a maximum pool inlet temperature LSSS of 108 °F (42 °C) when the reactor is operated in forced convection mode (as discussed in SAR Section 7.4.2.2, pool inlet temperature is measured in the primary coolant in the piping exiting the heat exchanger).

TS 2.2.1(4) would require a minimum pool height above the core centerline LSSS of 24.25 feet when the reactor is operated in forced convection mode.

UML’s steady-state forced-flow thermal-hydraulic analyses, which are discussed and found acceptable above, demonstrate that when the UMLRR is operated in forced-flow mode at LSSS conditions (or conditions that are more conservative than LSSS conditions, as discussed above) during routine operation, the UMLRR core will maintain a large margin to the SL (and to ONB). UML’s reactivity transient, LOCA, and loss of coolant flow analyses for forced flow operation, which are discussed and found acceptable in SER Sections 5.2, 5.3, and 5.4, respectively, further demonstrate that accidents when the LSSSs are at TS 2.2.1 limits (or in some cases conditions that are beyond, i.e., more conservative than, LSSS conditions) could not cause fuel temperatures to exceed the TS 2.1 SL. Therefore, because TSs 2.2.1(1), 2.2.1(2), 2.2.1(3), and 2.2.1(4) would help ensure that fuel failure would not occur for any routine or accident condition

under forced-flow operation and because these TSs appropriately implement the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1 by establishing LSSSs and based on its 10 CFR 50.36(c)(1) findings for LSSSs in SER Section 6.7, the NRC staff concludes that TSs 2.2.1(1), 2.2.1(2), 2.2.1(3), and 2.2.1(4) are acceptable.

Renewed TS 2.2.2, "Natural Convection Mode," would state:

> Applicability:
>
> This specification applies to the set points for the safety channels monitoring reactor thermal power, reactor pool temperature, and the height of water above the center line of the core under the condition of the natural convection mode of operation.
>
> Objective:
>
> To ensure that automatic protective action is initiated in order to prevent undesirable radiation levels on the surface of the pool.
>
> Specifications:
>
> (1) The Limiting Safety System Setting for the reactor power level shall initiate automatic protective action at or below a measured value of 115 kW_t.
>
> (2) The Limiting Safety System Setting for the pool temperature shall initiate automatic protective action at or below a measured temperature of 108 °F.
>
> (3) The Limiting Safety System Setting for pool height above the core centerline shall initiate automatic protective action at or above a measured value of 24.25 ft.

TS 2.2.2(1) would require a maximum thermal power level LSSS of 115 kWt when the reactor is operated in natural convection mode.

TS 2.2.2(2) would require a maximum bulk pool temperature LSSS of 108 °F (42 °C) when the reactor is operated in natural convection mode (as discussed in SAR Section 7.4.2.2, bulk pool temperature is measured in the primary coolant in the pool near the surface of the water above the reactor).

TS 2.2.2(3) would require a minimum pool height above the core centerline LSSS of 24.25 feet when the reactor is operated in natural convection mode.

UML's steady-state thermal-hydraulic analyses for natural convection operation, which are discussed and found acceptable above, demonstrate that when the UMLRR is operated in natural convection mode at LSSS conditions (or conditions that are more conservative than LSSS conditions, as discussed above) during routine operation, the UMLRR core will maintain a large margin to the SL and to ONB. UML's reactivity transient analyses for natural convection operation, which are discussed and found acceptable in SER Section 5.2, further demonstrate that reactivity transients for conditions that are beyond, i.e., more conservative than, LSSS conditions could not cause fuel temperatures to exceed the SL. Therefore, because TSs 2.2.2(1), 2.2.2(2), 2.2.2(3), and 2.2.2(4) would help ensure that fuel failure would not occur for any routine or accident condition under natural convection operation and because these TSs appropriately implement the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1 by

establishing LSSSs and based on its 10 CFR 50.36(c)(1) findings for LSSSs in SER Section 6.7, the NRC staff concludes that TSs 2.2.2(1), 2.2.2(2), 2.2.2(3), and 2.2.2(4) are acceptable.

The NRC staff notes that the current, i.e., pre-renewal, UMLRR TSs (including LSSSs) allowed operation of the reactor in "low pool level" mode, in which the pool level could be less than 24.25 feet for natural convection mode operation (the natural convection LSSS pool level was only 2.25 feet). However, this allowance has been removed from the renewed TSs.

NRC Staff Confirmatory Calculations of Thermal-Hydraulics

The NRC staff performed an analysis of the UMLRR steady-state thermal-hydraulics (for forced-flow and natural convection operation) to determine if it could confirm UML's calculations demonstrating the acceptability of the steady-state thermal-hydraulic performance of the UMLRR. For its analysis, the NRC staff used the TRAC/RELAP Advanced Computational Engine (TRACE) code, which allows for fluid dynamics, heat transfer, and point kinetics modeling of nuclear reactor cores (Ref. 50). The NRC staff also used TRACE for its confirmatory analyses of UMLRR reactivity transient, loss of coolant, and loss of flow accidents, as discussed in SER Sections 5.2, 5.3, and 5.4, respectively. The NRC staff prepared TRACE models of the UMLRR which fully replicate the applicable basic features of the UMLRR. A diagram of the TRACE model used for all confirmatory steady-state thermal-hydraulic, reactivity transient, and loss of flow analyses discussed in this SER is shown in Figure 2-15. This model is referred to as the non-LOCA TRACE model (a different TRACE model was used for the confirmatory LOCA analysis, as discussed in SER Section 5.3).

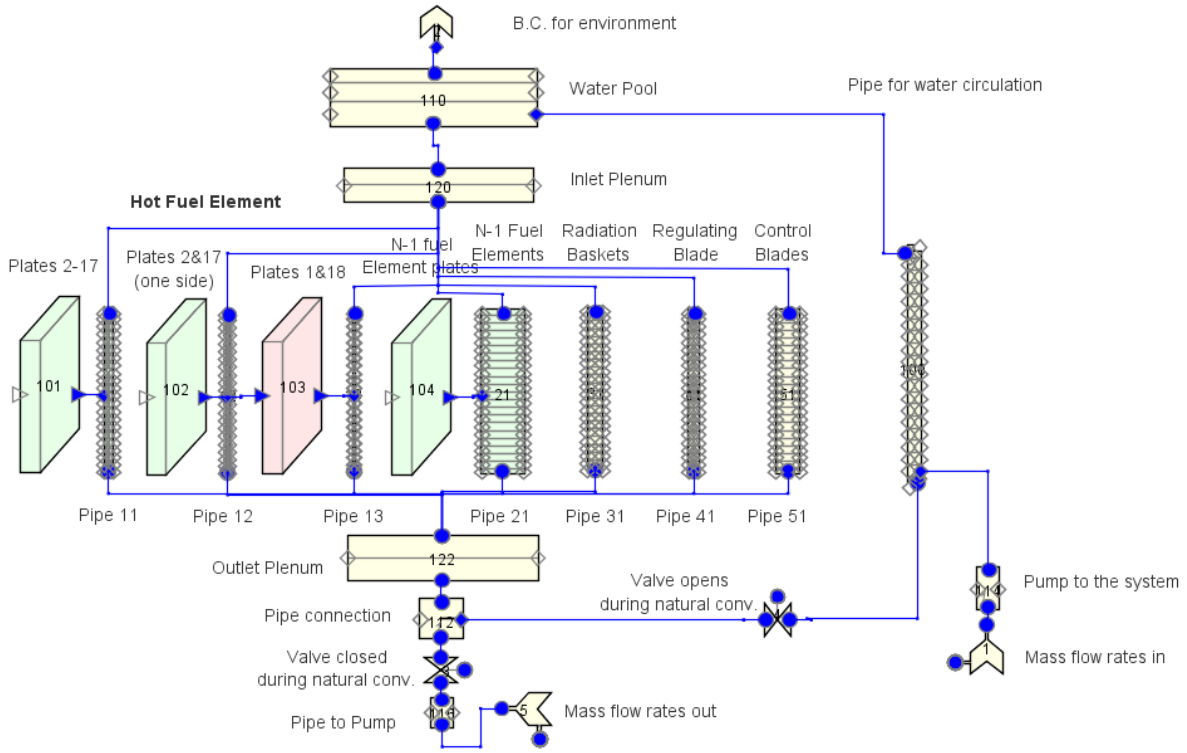

Figure 2-15 Non-LOCA TRACE Model for NRC Confirmatory Calculations

The non-LOCA TRACE model describes the silicide hot element in position D6 of the LCC using three coolant channels ("pipes") and their associated heat structures. Each heat structure has

its own coolant channel. Pipe 11 and its associated heat structure (101) represent the internal channels between the 16 fuel plates (plates 2 through 17) of the hot fuel element. Pipe 12 and its associated heat structure (102) represent the side channels formed between the end fuel plates (plates 2 and 17) and the non-fuel aluminum side plates (plates 1 and 18) of the hot fuel element. Pipe 13 and its associated heat structure represent the wall channels formed between the aluminum side plates (plates 1 and 18) and the walls of the hot fuel element. Pipe 21 and its associated heat structure (104) represent the remaining fuel elements in the core (the "N-1" fuel elements) collectively. Each heat structure in the fuel region is represented by 22 axial nodes and 30 radial nodes. Axial nodes 1 and 22 represent the unheated portions at the ends of the fuel elements. Each pipe associated with the fuel heat structures is represented by 22 axial cells, each of which is connected to its corresponding heat structure node.

Pipes 31, 41, and 51 represent the fuel bypass regions (the radiation baskets (the core is assumed to contain five radiation baskets), regulating rod, and control blades, respectively). Pipes 31, 41, and 51 are also represented by 22 axial cells. Upper (inlet) and lower (outlet) plenums and the coolant pumps are also represented to allow the model to consider forced flow, natural circulation, and transition flow conditions.

An additional "supplemental" heat structure (not explicitly shown in SER Figure 2-15) is defined to model the power in the limiting (hot) fuel plate of the hot element in position D6 of the LCC. Similar to UML's steady-state thermal-hydraulic analyses, the NRC staff's confirmatory analyses use peaking factors and hot channel factors for modeling the hot fuel plate. The NRC staff analyses used the peaking factors (specifically, the hot element peaking factor of 1.462 and the intra-element radial peaking factor of 1.364) indicated in SER Figure 2-9 for position D6 of the LCC. Also similar to UML's analyses, the NRC staff's analyses used UML's VENTURE-calculated axial power profile for the fuel element in core location B5 of the OCC (see SER Section 2.5.1). Additionally, the NRC staff analyses used the hot channel factors listed in SER Table 2-8 for silicide fuel.

The NRC staff analyzed 2 cases for its confirmatory steady-state thermal-hydraulic analysis: forced flow at LSSS conditions (Case 1) and natural circulation at LSSS conditions (Case 2). The cases are described in SER Table 2-10, and the results are presented in SER Table 2-11.

Table 2-10 NRC Staff Confirmatory Thermal-Hydraulic Analysis Cases

Case No.	Description	Power (MWt)	Pump Flow (gpm)	Inlet Temp. (°C)	Pool Water Level (feet)
Case 1	Forced flow at LSSS conditions	1.150	1400	43	24.25
Case 2	Natural convection at LSSS conditions	0.115	0	43	24.25

Table 2-11 NRC Staff Confirmatory Thermal-Hydraulic Analysis Results

Parameters	Case 1 (forced flow)	Case 2 (natural convection)
Maximum fuel cladding temperature (°C)	98	74
Maximum coolant temperature (°C)	50	58
Saturation temperature at the location of maximum fuel cladding temperature (°C)	116	116
Onset of Nucleate Boiling Ratio (ONBR)	>>2	>>2
Departure from Nucleate Boiling Ratio (DNBR)	>>2	>>2

Similar to UML's analyses for nominal and beyond-LSSS conditions, the NRC staff's confirmatory analyses for steady-state forced flow and natural convection operation at LSSS conditions demonstrate that, for any routine allowed condition of reactor operation, the maximum fuel cladding temperature will remain well below the SL. The NRC staff's confirmatory analyses also show that the maximum coolant temperature will remain well below the saturation temperature of the coolant during forced flow and natural convection operation. Additionally, the ONBR and DNBR are large (much greater than 2), indicating that there is a large margin to ONB and DNB.

The NRC staff notes that there is some variation between the results of the UML and the NRC staff thermal-hydraulic analyses. Compared to UML's calculations at beyond-LSSS conditions, the NRC staff's calculations at LSSS conditions show a higher maximum cladding temperature during forced flow, but a lower maximum coolant temperature during forced flow, and lower maximum cladding temperature and coolant temperature during natural convection. However, given the differences in models and assumptions used for the calculations, the NRC staff finds that the differences in results are reasonable.

Conclusion

The NRC staff reviewed the above information regarding the UMLRR thermal-hydraulic design and analyses. The NRC staff finds that the information in the SAR, as supplemented, includes thermal-hydraulic analyses that are specific to the UMLRR design and possible core configurations, that the analyses determine limiting conditions for UMLRR operation, and that UML justified the assumptions and methods for its analyses. The NRC staff performed confirmatory thermal-hydraulic calculations that produced results similar to UML's analyses and determined that given the differences in models and assumptions used for the calculations, the differences in the results were reasonable. The NRC staff finds that UML has a SL (see SER Section 2.2.1) and LSSSs that, based on the results of UML's steady-state thermal-hydraulic analyses and accident analyses (see SER Chapter 5), and based on the cladding failure thresholds of the UMLRR fuel (see SER Section 2.2.1), provide an acceptable margin to loss of fuel integrity for any condition of routine reactor operation and during any credible accident. Therefore, the NRC staff concludes that UML's thermal-hydraulic design and analyses for the UMLRR are acceptable.

2.7 Fuel Storage

SAR Sections 3.5.9 and 9.2, as supplemented by UML's response to RAI-9.1 (Ref. 43) and UML's letter dated September 30, 2020 (Ref. 98), describe fuel storage at the UMLRR. UML's response, dated August 11, 2016 (Ref. 51), to NRC Generic Letter 2016-01, "Monitoring of Neutron-Absorbing Materials in Spent Fuel Pools," (Ref. 53) also discusses fuel storage.

As discussed in SAR Section 9.2.1 and UML's letter dated September 30, 2020, and illustrated in SAR Figure 9-3, there are nine fuel storage racks submerged in, and located at two different heights along the walls of, the UMLRR pool. Racks can be in both the stall and bulk sections of the pool. As discussed in SAR Section 3.5.9, the upper racks are located such that any fuel in those racks is at least 14 feet below the normal pool water surface level. As discussed in UML's letter dated September 30, 2020, the vertical centerline of the lower racks is 1.5 inches above the core centerline. The racks are 26 inches high. As discussed in SAR Section 9.2.1, the elements are inserted into the racks via square slots in the top of the aluminum rack frame structure that are 0.25 inch wider than the elements; slots in the bottom of the frame structure hold the elements in place. The bottom end of the elements' end boxes do not protrude from the bottom of the racks, but because the elements are longer than 26 inches, the additional length of a stored element sticks out of the top of a rack.

SAR Section 9.2 states that the fuel storage racks are used to hold both new (i.e., unirradiated) and irradiated fuel elements. In its response to RAI-9.1(b), UML clarified that each rack is capable of holding up to nine fuel elements. As discussed in SAR Sections 1.3.2 and 4.1, spent fuel is stored in the racks in the bulk pool.

SAR Section 9.2.1 states that the fuel storage racks in use ensure that there can be no inadvertent criticality, and UML's response to NRC Generic Letter 2016-01 states that a criticality analysis for a rack fully loaded with fresh fuel elements has shown that without credit for the neutron-absorbing BORAL sheet incorporated in the rack, the fuel in the rack has a maximum k_{eff} of less than 0.7.

In Appendix A to its response to RAI-9.1(a), UML provided MCNP analyses supporting the statements in SAR Section 9.2.1 and in the response to NRC Generic Letter 2016-01. UML states that its fuel storage criticality analyses assume that all fuel elements in storage are silicide fuel elements. UML's analyses assumed that the fuel in the racks is completely water reflected (i.e., the aluminum rack structure and concrete pool wall were not considered in the analyses). To demonstrate that this assumption is valid, UML performed additional analyses using models that include the aluminum and concrete, which showed that the effect of including the aluminum and concrete is not significant. UML's analyses calculated the k_{eff} for a single rack containing a variable number of fuel elements (from zero to 25), for a variable number of rows of full nine-element racks (from zero to four), and for significant variations in fuel element spacing (from 0 to 2.5 centimeters (0 inches to 0.98 inches)) from the nominal design value of 1.83 centimeters (0.72 inches). UML's calculations for a single rack showed that in all cases, the k_{eff} is below 0.7. For the calculations for up to four rows of full racks, the k_{eff} is still below approximately 0.82.

The NRC staff reviewed UML's fuel storage criticality safety analysis. The NRC staff finds that UML's MCNP modeling and analysis methodologies are reasonable and consistent with standard nuclear industry practice and can, therefore, provide suitably predictive results for the analysis of the UMLRR fuel storage racks. The NRC staff also finds that UML's assumptions are conservative and would bound potential fuel storage conditions in the UMLRR pool,

because UML assumed that the modeled racks contain all fresh silicide fuel (which has higher uranium loading than the aluminide fuel), because UML evaluated fuel element configurations (including configurations with neutronic coupling between multiple fuel racks) that are more limiting than actual configurations, and because the analyses do not credit the BORAL sheets. The NRC staff finds that UML's analyses demonstrate that for any possible condition of fuel storage in the UMLRR reactor pool, the k_{eff} will be well below the value of 0.9 specified in the guidance in NUREG-1537, Part 2, Section 9.2, and in TS 5.4, which is discussed and found acceptable below.

Renewed TS 5.4, "Fissionable Material Storage," would state:

> Applicability:
>
> These specifications apply to the storage of reactor fuel when not in the core and the storage of other fissionable material.
>
> Objective:
>
> To ensure that stored fuel or other fissionable material does not become critical and will not reach an unsafe temperature.
>
> Specifications:
>
> (1) Fuel, including fueled experiments and fueled devices not in the reactor shall be stored in the reactor building and in a configuration that ensures adequate cooling and is designed to maintain k_{eff} less than 0.9 under all conditions of moderation and reflection.
>
> (2) Where a licensed shipping container is used, the k_{eff} and cooling design considerations of the container shall apply and TS 5.4(1) shall not apply.

TS 5.4(1) would require that UMLRR fuel elements, fueled experiments, and fueled devices (e.g., fission chambers) that are not in the reactor be stored in the reactor building, be stored such that adequate cooling will be ensured, and be stored in a criticality-safe configuration with a k_{eff} less than 0.9 under all conditions of moderation and reflection. TS 5.4(2) would require that if UMLRR fuel elements or other fissionable material are located in an NRC-licensed shipping container in accordance with the conditions of the container license, then the k_{eff} and cooling design considerations of the container apply in lieu of the k_{eff} and cooling requirements of TS 5.4(1). Additionally, per TS 5.4(2), the TS 5.4(1) requirement that fuel and other fissionable material be stored in the reactor building would not apply if the material is in a licensed shipping container.

In its basis for TS 5.4, UML stated that the adequate cooling requirement of TS 5.4(1) assures that stored fuel temperatures do not reach a level where damage could occur. As discussed earlier in this SER section, UML stores irradiated fuel (which generates decay heat) in the racks in the pool. As discussed in SAR Section 9.2.1, the racks have openings in the bottom to help facilitate any decay heat to be removed by natural convection to the pool water. However, in SAR Section 13.2.4, UML stated that based on historical studies of other plate-type fueled reactors (Ref. 93 and Ref. 94), it is reasonable to conclude that irradiated UMLRR fuel could also be adequately cooled by natural air convection. As discussed above, UML also uses the racks to store new (i.e., unirradiated) fuel, which does not require cooling because it does not

produce any decay heat. In its letter dated September 30, 2020, UML stated that for fuel being stored in the pool, it typically places higher-activity fuel (i.e., fuel that has been more recently irradiated and/or has higher burnup, which generates more radiation and decay heat) in the lower racks, which are deeper in the pool (see SER Figure 1-1 and SAR Figure 9-3; as discussed above, the vertical centerline of the lower racks is 1.5 inches above the core centerline), as a best practice. Also, as discussed earlier in this SER section, spent fuel is stored in the racks in the bulk section of the pool. UML stated in its letter dated September 30, 2020, that its fuel storage configurations and methods are intended to provide adequate cooling both for normal conditions and abnormal conditions such as a possible LOCA. In its basis for TS 5.4, UML also stated that TS 5.4(2) allows flexibility for shipments.

The NRC staff finds that TS 5.4(1) helps ensure that fuel or other fissionable material at the UMLRR is stored in criticality-safe configurations that will also provide appropriate cooling, if necessary. As discussed earlier in this SER section, UML provided analyses in the SAR, as supplemented, demonstrating that for any normal fuel storage configuration in the UMLRR reactor pool, as well as for conditions that are more criticality-favorable than the normal storage conditions, the k_{eff} remains well below 0.9.

Regarding cooling of stored fuel, the NRC staff notes that while the racks are designed for natural water convection, and natural convection water cooling of irradiated fuel in storage is good practice, natural convection water cooling may not necessarily be needed to protect the integrity of fuel in the racks (i.e., prevent fuel failure), given the typical power density and history of UMLRR fuel, and given that any fuel in the racks would have undergone some decay since having been in the reactor core during operation. Additionally, given the locations of the racks in the pool, natural convection water cooling may not be able to be maintained for stored fuel in racks for an abnormal scenario such as a severe unmitigated LOCA in which the pool drains through a beam tube. However, the NRC staff finds that by requiring UML to store fuel in a configuration that ensures adequate cooling, TS 5.4(1) helps ensure that any fuel that UML places in storage in the racks will have at least the necessary cooling to prevent fuel failure or damage, or the SL from being exceeded, for any condition. As discussed above, UML stated that it is reasonable to conclude that UMLRR fuel can be sufficiently cooled by natural air convection; the NRC staff finds that the design of the racks could facilitate natural convection air cooling (especially if the racks become completely uncovered with water), as well as natural convection water cooling. Additionally, the NRC staff notes that UML's storage of spent fuel in the bulk pool, as discussed above, can help ensure continued water cooling of the fuel in an abnormal scenario involving pool drainage through a beam tube, because UML could use the pool dam to isolate the spent fuel in the bulk pool (see SER Section 5.3). The NRC staff also notes that UML could allow decay time (i.e., cooldown time) before removing fuel from the core, if needed, consistent with standard industry practice, to help ensure adequate cooling of stored fuel for any condition, and also to reduce the fission product inventory of the fuel to minimize any possible release in the event of a fuel handling accident.

The NRC staff finds that there is reasonable assurance that UML can ensure sufficient cooling of stored fuel for any condition, as required by TS 5.4(1), based on the following: (1) UML's fuel storage racks are designed to provide natural convection water cooling under normal conditions; (2) in the extremely unlikely event (see SER Section 5.3) of an unmitigated LOCA in which the pool could drain below the primary piping connection penetrations (which are above all of the fuel storage racks) and natural convection water cooling would not be possible, the fuel in the racks would continue to be cooled by natural air convection and/or other mechanisms (e.g., conduction to the pool water heat sink in the case of the lower rack fuel since based on the rack dimensions discussed earlier in this SER section, the lower racks would remain partially

submerged to approximately the bottom of the elements' fuel plates/top of the elements' bottom end boxes even if the pool drained to the bottom of an 8-inch beam tube, 4 inches below core centerline); (3) the expected reduced fission product inventory and decay heat of the fuel in the racks; and (4) historical studies (e.g., Ref. 93 and Ref. 94) evaluating the ability of other reactors with plate-type fuel to withstand a LOCA without the fuel reaching temperatures that could cause fuel damage. The NRC staff finds that storing irradiated fuel in the pool also provides shielding to help minimize any personnel dose from the fuel and notes that UML's practice of storing higher-activity fuel in the lower racks, as discussed above, can ensure greater water shielding and cooling of the higher-activity fuel, even under abnormal conditions (e.g., a possible LOCA).

The NRC staff also finds that by requiring that fuel and fissionable material (outside of a licensed shipping container) be stored in the reactor building, TS 5.4(1) helps ensure that fuel and other fissionable material is stored in locations that are covered by the UMLRR physical security plan (PSP) and within the UMLRR licensed boundary. The NRC staff finds that TS 5.4(2) provides UML with flexibility to have fuel and other fissionable material in appropriately licensed shipping containers, while still helping ensure that the fuel or fissionable material in the container is in a criticality-safe, appropriately cooled (and shielded, if necessary) configuration. The NRC staff finds that it is not necessary for TS 5.4 to explicitly require that fuel or other fissionable material in a licensed shipping container be stored in the reactor building because the container ensures that the material is in a safe configuration regardless of location, and because this allows flexibility for shipments. However the NRC staff notes that fuel and other fissionable material on the reactor license must be stored and handled in accordance with the UMLRR PSP and 10 CFR 73.67, "Licensee fixed site and in-transit requirements for the physical protection of special nuclear material of moderate and low strategic significance," regardless of location.

The NRC staff also finds that by specifying requirements for fissionable material storage, TSs 5.4(1) and 5.4(2) appropriately implement the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007, including the recommended maximum TS-allowed k_{eff} in ANSI/ANS-15.1-2007, Section 5.4.

Based on the above, and based on its 10 CFR 50.36(c)(4) findings for TSs on design features in SER Section 6.7, the NRC staff concludes that TSs 5.4(1) and 5.4(2) are acceptable.

Although UML's TS 5.4 basis states that the requirements of 10 CFR 70.24(a) apply for special nuclear material stored, handled, or used outside the reactor pool, in its letter dated January 30, 2021 (Ref. 99), UML stated that any fissionable material it currently stores, handles, or uses outside of the pool or licensed shipping containers is less than the quantities specified in 10 CFR 70.24(a), which would require a criticality monitoring system.

The NRC staff reviewed the above information regarding fuel storage at the UMLRR. The NRC staff finds that UML demonstrated that fuel storage at the UMLRR will help ensure that inadvertent criticality cannot occur. The NRC staff finds that UML showed that even under optimum geometry conditions, the maximum k_{eff} for fuel storage could not exceed 0.90. The NRC staff finds that, because any fissionable material (e.g., fueled experiments or fueled devices) that UML has present outside of the reactor pool or licensed shipping containers is less than the quantities specified in 10 CFR 70.24(a), the requirements of 10 CFR 70.24 are not applicable to UML. The NRC staff additionally finds that methods and TSs for shielding, cooling, and storing irradiated fuel provide reasonable assurance that potential personnel doses from irradiated fuel will not exceed regulatory limits and will be ALARA and that irradiated fuel

can be cooled as necessary during storage. Therefore, the NRC staff concludes that the fuel storage at the UMLRR is acceptable.

2.8 Reactor Description Conclusions

Based on the above findings and conclusions, in conjunction with the findings and conclusions in SER Chapter 5, the NRC staff concludes that UML adequately described the bases and functions of the reactor design to demonstrate that the UMLRR can be safety operated and shutdown from any normal or accident condition assumed in the safety analysis. These conditions include the use of cores containing aluminide fuel elements from the permanently-shutdown WPI research reactor. The UMLRR systems provide an adequate control of reactivity, the containment of coolant, barriers to the release of radioactive material, and sufficient radiation shielding for the protection of facility personnel and the public. Nuclear and thermal-hydraulic design and operating limits, established in the TSs, will adequately provide for the protection of silicide and aluminide fuel integrity. For this reason, the TSs will help ensure that no cladding breach will occur when the UMLRR is operated in accordance with the TSs. Therefore, the NRC staff concludes that continued operation of the UMLRR, including operation with WPI aluminide fuel, within the limits of the TSs and facility operating license will not result in undue risk to the health and safety of facility personnel, the public, or the environment.

3. INSTRUMENTATION AND CONTROL SYSTEMS

3.1 Summary Description

The University of Massachusetts Lowell (UML) Research Reactor (UMLRR) instrumentation and control (I&C) systems provide functions for monitoring and controlling the reactor during startup, normal operation, shutdown, and abnormal conditions. The I&C systems provide the reactor operator with the required information and controls to keep the reactor within its operational safety envelope, initiate automatic protective action if the setpoint for reactor period, reactor power, primary coolant flow rate, pool water level, or coolant temperature is exceeded, and prevent the reactor from operating if required support systems are not in the proper operating configuration.

In its safety analysis report (SAR) (Ref. 1), UML stated that most of the original components of the I&C systems for the UMLRR were previously replaced and upgraded with newer systems that provide the same functions, but more reliably. In Section 3.5.2 of the SAR, the licensee described the current UMLRR I&C systems as mainly consisting of analog-digital hybrids, whereby an analog signal is transmitted to an analog-to-digital converter. The digital output is then made available for display or other uses, such as process control or alarm functions. In its request for additional information (RAI) responses (Ref. 79), the licensee stated that these changes were made in accordance with the licensee's procedures under Title 10 of the *Code of Federal Regulations* (10 CFR) 50.59, "Changes, tests and experiments," without prior U.S. Nuclear Regulatory Commission (NRC, the Commission) approval. These changes are briefly discussed in Section 1.8 of this safety evaluation report (SER) and listed in SER Table 1-1. The NRC staff examined the 10 CFR 50.59 evaluations and screens related to the UMLRR I&C systems during the 2017 audit (Ref. 86), and UML provided these evaluations and screens in an RAI response (Ref. 79). The NRC staff assessed the licensee's 10 CFR 50.59 evaluations and determined that in conducting the evaluations, the licensee generally provided sufficient detail to address the criteria in 10 CFR 50.59. As part of this renewal review, I&C system replacements and upgrades are discussed in this SER commensurate with the significance of the changes since the last license renewal.

In its application for license renewal, as supplemented, the licensee proposed upgrades to the linear power channels, Wide-Range Logarithmic Power/Period Channel, and the radiation monitoring system. The proposed upgrade of linear power channels would replace the current General Atomics (GA) NMP-1000 architecture with new (digital-based) NMP-1000 pulse monitoring channels manufactured by GA. The proposed NMP-1000 is described in SAR section 7.4.1 and in supporting documentation (Ref. 102 and Ref. 103). The licensee also proposed to replace the current GA Wide-Range Logarithmic Power/Period Channel with a wide range logarithmic (log) power and period module (PPM) manufactured by Thermo Fisher Scientific (TFS). The proposed PPM is described in UML's letter dated April 10, 2019 (Ref. 73). Additionally, the licensee proposed upgrading the radiation monitoring system by adding ratemeters and audible alarms. The radiation monitoring system upgrades are primarily discussed in SER Section 4.1.4 and are also discussed below in SER Section 3.7.

SER Sections 3.2 through 3.7 describe each UMLRR I&C subsystem and summarize the NRC staff's evaluation of the I&C systems, including the proposed upgrades. Additionally, for the proposed flux monitoring system I&C upgrades, SER Section 3.8 describes the NRC staff's evaluation of the licensee's process to upgrade the proposed I&C systems for the UMLRR,

including its configuration management (CM) and quality assurance (QA) program and procedures.

The current and proposed I&C systems employed at the UMLRR are similar to those used by other research reactors operating in the United States, such as those at the Rhode Island Nuclear Science Center (RINSC), North Carolina State University, Reed College, and the Missouri University Research Reactor (MURR). According to SAR Chapter 7, the UMLRR I&C systems associated with the reactor include the following:

(1) Reactor control system (RCS), including the control elements and interlocks

(2) Reactor protection system (RPS), including the nuclear instruments, process instruments, and the scram chain circuit

(3) Engineered safety feature (ESF) actuation system

(4) Control console and instrumentation panel displays (i.e., reactor operator interface)

(5) Area radiation monitoring system (ARMS)

SER Figure 3-1 (adapted from SAR Figure 7-1) is a simplified block diagram depicting the major subsystems of UMLRR I&C systems and the interconnections and relationships among the five subsystems that comprise the I&C systems.

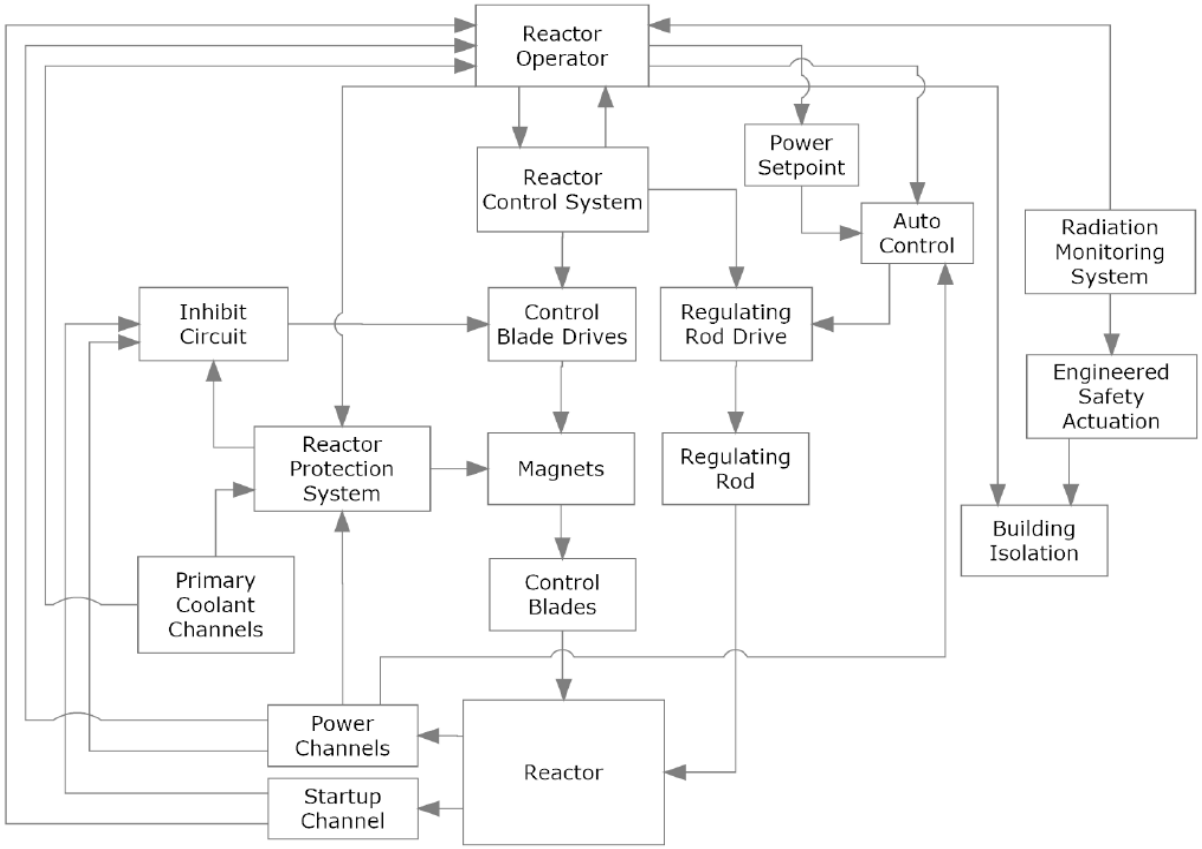

Figure 3-1 UMLRR Reactor Controls and Instrumentation

The RCS provides control of the reactor during all modes of reactor operation and shutdown. Control of the nuclear fission process is achieved using four control blades and one regulating rod that are moved in and out of the reactor core by mechanical drives. In the event of power failure, receipt of an automatic scram signal, or manual initiation by the operator, the control blades are disconnected from their drives by means of electromagnets and are allowed to fall, by gravity, into the reactor.

The RPS includes two subsystems: the nuclear instrumentation (NI) system and the process controls and instrumentation (PCI) system. The RPS is designed to place the reactor in a subcritical, safe shutdown condition (scram) if any of the monitored parameters exceeds its setpoint. UML stated that the RPS setpoints are more conservative relative to the technical specification (TS) limit values justified by the UMLRR SAR and used in both the analyses for steady-state operation and various transient conditions. The RPS scram circuits (see Ref. 79, Appendix A) interrupt power to the control blade drive magnets, allowing the control blades to drop by gravity into the reactor core.

The UMLRR accident analyses in the SAR, as supplemented, show that ESFs are not required, thus the facility RPS is not required to have ESF actuation systems. However, although not required, UML has controls and indications for the actuation of the confinement isolation system and associated ventilation, filtration, and emergency exhaust systems, which are described in SAR Chapter 6.

The control console displays consist of the digital panel indicators, chart recorders, and display monitors that provide information to the reactor operator. The nuclear and process instrumentation provides indication of process variables, reactor core nuclear parameters, and other parameters necessary to allow safe operation and shutdown of the reactor and protection of the operating personnel and the public.

The ARMS monitors and displays radiation readings (including radiation levels and effluent activity levels) at various locations within the reactor containment building and the building ventilation exhaust system. The ARMS initiates warning alarms inside and outside the reactor building and provides for automatic initiation of the confinement isolation.

The I&C subsystems are discussed in more detail in the following sections.

3.2 Design of Instrumentation and Control Systems

The I&C system design requirements for non-power reactors, such as UMLRR, are generally derived from the results of accident analyses of normal operating conditions and of accidents and transients that could occur. NUREG-1537, Part 2, Section 7.4, “Reactor Protection System,” states, in part, that the SAR should include the design bases, acceptance criteria, and guidelines used for design of the protection system, as well as an “[a]nalysis of adequacy of the design to perform the functions necessary to ensure reactor safety, and its conformance to the design bases, acceptance criteria, and the guidelines used.” NUREG-1537, Part 1, Section 7.2, provides guidance on design criteria to be considered, such as reliability, single failure protection, fail-safe design (i.e., design to fail into a safe state on loss of electrical power or exposure to extreme adverse environments), redundancy, diversity, and operability.

As stated in Chapter 7 of the SAR, the design basis requirements for the UMLRR I&C systems are that the I&C systems are designed with sufficient reliability and redundancy to ensure that the reactor can be operated safely and so that design basis events are detected and the reactor

can be automatically shut down under the condition of any single failure in the system. As described in the accident analyses in SAR Chapter 13, as supplemented, the shutdown reactor does not pose a radiological hazard so long as the reactor pool integrity is maintained and the core remains covered with water.

Per the design criteria for the I&C system in SAR Section 7.2.1, a non-power reactor, such as the UMLRR, is not expected to operate under adverse conditions such as loss of facility power, severe natural phenomena, a seismic event, or fire. The UMLRR administrative procedures require the reactor to be shutdown should any of these events occur. In addition, the UMLRR will shut down in the event of a loss of offsite power and is also equipped with a seismic sensor to provide for an automatic shutdown should a seismic event occur. Although the licensee anticipates that the UMLRR I&C systems will remain operable during such circumstances, UML stated that there is no safety reason for them to be operable once the reactor is shut down. With respect to the I&C systems and components, most of the I&C equipment is located in the control room, which is a separate enclosed structure within the reactor building. The control room is equipped with a smoke detector connected to the building fire alarm system and has a portable fire extinguisher. The licensee stated that the reactor building provides an additional protection boundary to shield the I&C systems from adverse external environmental conditions.

SAR Section 7.2 describes the design of the UMLRR I&C systems. The instrumentation at the UMLRR consists of the nuclear and process instrumentation, sensors, signal processing equipment, displays, recorders, interlocks, and actuation devices that provide the means to safely control the reactor and to avoid or mitigate accidents. The I&C systems provide audible and visual indications of key operating parameters such as neutron flux, pool level, confinement pressure, fuel and coolant temperature, primary flow, pH, and control blade position to allow the operators to monitor reactor operations. Actuation of the scram logic will occur automatically if any parameter reaches the predetermined TS setpoint. The operator also has the discretion and the means to manually scram the reactor. The I&C system also includes various interlocks that prevent specific actions from occurring unless all the prerequisites for that action are satisfied. Renewed TS Tables 3.2.3-1 and 3.2.4-1 list the minimum protection system scrams, and renewed TS Table 3.2.5-1 lists the minimum required reactor protection channels. Renewed TS 3.2.6 lists the minimum required interlocks. These TSs and the associated surveillance intervals for components and functions of the I&C systems are reviewed and found to be acceptable by the NRC staff as discussed in SER Sections 2.5.3, 4.1.5, 6.3, and 6.4. Additionally, definitions that apply to the UMLRR I&C systems are in Section 1.3 of the renewed TSs. These definitions are reviewed and found to be acceptable by the NRC staff in SER Section 6.1.2.

Evaluation Findings on the Design of the I&C Systems

The design bases for the UMLRR I&C systems are derived from the analysis presented in the UMLRR SAR and are specified in renewed TSs 2.0, “Safety Limit and Limiting Safety System Settings,” 3.1, “Reactor Core Parameters,” 3.2, “Reactor Control and Safety Systems,” 3.3 “Reactor Coolant Systems,” 3.4 “Confinement,” 3.5 “Ventilation System,” and 3.6, “Radiation Monitoring Systems and Radioactive Effluents.”

The NRC staff reviewed the UMLRR safety limit (SL), limiting safety system setting (LSSS), and limiting condition for operation (LCO) TSs listed above, and found them to be acceptable, as referenced in SER Sections 6.2 and 6.3. The NRC staff also reviewed the UMLRR surveillances and the corresponding surveillance intervals and found them to be acceptable, as referenced in SER Section 6.4.

During the 2017 audit (Ref. 86), the NRC staff observed the physical location of the I&C equipment and the layout of the reactor building and control room, as described in Chapter 7 of the UMLRR SAR. Based on the information provided in the SAR, as supplemented, and reviewed during the NRC audits, the NRC staff finds that the UMLRR I&C systems are located and protected consistent with the guidance in NUREG-1537, Part 1, Section 7.2, so that the effects of fires, explosions, and natural phenomena will not prevent the safety system from performing its safety function. As further discussed in SER Section 3.8.4, the NRC staff also finds that the physical location of the I&C equipment and UML's administrative controls to counteract unauthorized physical and electronic access to the I&C systems are acceptable. The NRC staff finds that the design of the I&C systems is derived from the results of analyses of normal operating conditions and of accidents and transients that could occur as discussed in SER Chapter 5. The NRC staff also finds that the design of the I&C systems provides the means to safely control the reactor and to avoid or mitigate accidents. Based on the above, the NRC staff concludes that the overall design of the UMLRR I&C systems is acceptable.

3.3 Reactor Control System

3.3.1 System Description of the Reactor Control System

Sections 3.5.2, 7.2.2.1, and 7.3 of the SAR describe the RCS. The primary functions of the RCS are to provide the reactor operator the information and control capability necessary to safely perform several functions including: system startup, system shutdown, maintaining a shutdown state, changing power levels, and maintaining operation at a set power level.

The RCS consists of an inhibit circuit, the withdraw/insert circuits for the control blades and the regulating rod, the regulating rod automatic control circuit, the startup counter drive system, and the drives control system (DCS) display. The I&C for these systems are mostly located in the control room. However, the instrument detectors are provided locally within the reactor building.

The RCS controls reactivity by means of a regulating rod and four shim safety control blades. The regulating rod has no scram capability. The four control blades can be scrammed manually or automatically by the RPS. The regulating rod and control blades are discussed in SER Section 2.2.2.

The RCS has two modes of operation: manual and automatic. Manual mode is used to start up and shutdown the reactor and to change power levels. Automatic mode is used for steady-state operation. While manual mode allows the reactor operator to manipulate all reactivity control devices (four shim safety control blades and the regulating rod), automatic mode only moves the regulating rod.

According to Section 7.1.1 of the SAR, with the exception of the isolated reactor power level signal used for automatic control, the instrumentation used by the RCS is separate and distinct from the RPS. In addition, the SAR describes several safety features designed into the RCS, which are separate from the RPS. These safety features include:

- Only one control blade can be withdrawn at a time
- All four control blades can be scrammed from any height in the reactor
- All four control drives are automatically inserted at normal speed upon receipt of a rundown signal

- The reactor operator can take manual control of the regulating rod (if in auto) by engaging the manual regulating rod control via the DCS graphical user interface (GUI)

3.3.1.1 Control System Interlocks

The SAR, as supplemented, discusses several interlocks associated with the RCS (see also SER Section 2.2.2; the first two interlocks on this list comprise the RCS inhibit circuit as discussed in SAR Section 7.3.3, while the third interlock provides an additional inhibit):

- Start-up interlock – this interlock ensures that the control blades cannot be withdrawn unless (1) the key-locked operate switch is unlocked; (2) the limit switch contacts in the scram chain are in the normal position; and (3) all relay contacts in the scram chain are reset and energized in the closed position

- Withdrawal inhibit – this interlock ensures that the control blades cannot be withdrawn unless (1) the neutron count is greater than two counts per second, and (2) the linear power channel indication is greater than five percent

- Log PPM Withdrawal inhibit – the log PPM channel also provides a control blade drive withdrawal inhibit signal for a short period, downscale power, or high-power indication

- Automatic regulating rod control – this interlock prevents placing the regulating rod in automatic control unless (1) reactor period is greater than 30 seconds and (2) the regulating rod is other than full in or full out

- Automatic rundown (control blade rundown circuit) – this is an automatic insertion of the control blade drives at normal speed after the control blades drop by gravity

As discussed in SER Section 2.5.3, both the startup channel and UML's proposed replacement log PPM channel, which are discussed in SER Section 3.4.1.1, are capable of providing the portion of the withdrawal inhibit interlock that prevents control blade withdrawal unless the neutron count rate is greater than two counts per second (cps). This is possible because both channels have a control blade withdrawal inhibit function for low count rate.

Renewed TS 3.2.6, which is discussed and found acceptable in SER Section 2.5.3, identifies the TS-required RCS interlocks. These TS-required interlocks are a portion of the actual full complement of the UMLRR interlocks listed above (e.g., TS 3.2.6(2) requires startup neutron count rate be greater than 2 cps, but, as stated above, the withdrawal inhibit interlock also prevents blade withdrawal unless the linear power indication is greater than five percent).

Based on the above list and description of interlocks, the NRC staff finds that the UMLRR has diverse safety interlocks that are in addition to the TS-required interlocks. In addition, TS 4.2.3(8) requires that the interlocks required by TS 3.2.6 be verified operable annually. TS 4.2.3(8) is discussed and found acceptable in SER Section 2.5.3. Accordingly, the NRC staff finds that, consistent with the guidance in NUREG-1537, Part 2, Section 7.3, the required UMLRR interlock components are periodically calibrated, inspected, and/or tested, as appropriate to ensure operability as analyzed in the UMLRR SAR. Additionally, the NRC staff finds that the types and kinds of RCS interlocks are consistent with those of other research reactors, such as the RINSC reactor, and that the interlocks meet the NUREG-1537, Part 2, guidance to limit personnel hazards or prevent damage to systems during the full range of

normal operations. Therefore, the NRC staff concludes that the UMLRR control system interlocks are acceptable.

3.3.1.2 Control Blade Drive System

The control blade drives, as described in SAR Sections 4.2.2.1, 7.3.2, 7.3.3, and 7.3.4, are mounted above the core on the underside of the bridge. The control blade drives are coupled to electromagnets such that they attach to the control blades when energized. The drive mechanisms include reversible direct-current (DC) servomotors. The control blade drives are hard wired with mechanical relays, whose logic prevents more than one control blade drive from operating in the up direction (blade withdrawal) at any one time. The control blade drives are controlled from the DCS GUI display on the control console.

Power to the electromagnets is controlled by the RPS. When a scram occurs, the rods are decoupled from the drives and rundown relays actuate an automatic insertion of all four control blade drives at normal speed to help ensure that all control blades are fully inserted. The operator can also activate the rundown relays at the control console to initiate a manual rundown to simultaneously insert all four control blade drives. The rundown circuit is designed to override the control blade drive inhibit and withdraw signals to force insertion of all four control blade drives.

According to SAR Sections 4.2.2.1 and 4.2.2.3 and UML's letter dated February 16, 2021 (Ref. 101), the maximum speed measured by UML for the regulating rod drive motor is approximately 55 inches per minute, and the maximum speeds for the control blade drive motors are approximately 3.6 inches per minute. Reactivity insertion rates for the control blades and regulating rod are a function of drive speed and blade or rod worth. TS 3.2.2, which is discussed and found acceptable in SER Section 2.2.2, requires that the maximum simultaneous combined reactivity insertion rate from the most reactive control blade and the regulating rod not exceed 0.05 percent delta k over k (% $\Delta k/k$) per second, which is significantly less than the assumed 0.07% $\Delta k/k$ per second reactivity addition accident in SAR Section 13.2.2, as supplemented. As stated in SAR Section 3.5.2, there is no conceivable malfunction of the reactivity control systems that could result in a reactivity accident worse than the conditions encountered during the startup accident (i.e., reactivity addition accident from continuous blade withdrawal occurring at low power during reactor startup). As described in SAR Chapter 13, as supplemented, neither continuous blade withdrawal nor loss of coolant will cause undue heating of the fuel.

SAR Section 3.5.2 states that the control blade, blade extension, and magnet armature are detached from the electromagnet and drop by gravity on a loss of power (i.e., a scram or loss of electric power). The UMLRR has four safety control blades and insertion of at least three of the control blades ensures reactor shutdown.

Integral to each control blade and regulating rod drive motor shaft is an optical encoder that generates an electronic pulse at each revolution of the encoder. The revolutions of the encoder are counted by an input module, converted to a measurement in inches, and displayed on the DCS GUI in 0.01-inch increments. According to the SAR, the accuracy of the electro-mechanical transmitter is 0.02 inches. Each of the control blades and the regulating rod have up and down position limits that will deenergize the drive motors to prevent damage to the drive mechanism when the drive components have reached the full-in or full-out position. The up and down limit sensors are through-beam photo-microsensors. A flexible metallic strip or "flag" on the lead screw ball nut intersects the beam to the sensor to inhibit further drive motion.

As stated in the SAR (see Figure 4-7), mechanical stops are also available as a redundant backup in case of limit sensor failure.

The NRC staff reviewed the information in the SAR, as supplemented, and finds that the control blade drive system and related TSs are similar to those at other research reactors. Renewed TSs 3.2.1 and 3.2.2, which impose requirements related to the control blades and regulating rod, are discussed and found acceptable in SER Section 2.2.2. Additionally, the surveillance requirements (SRs) in renewed TSs 4.2.1 and 4.2.2 related to the control blades and regulating rod are also discussed and found acceptable in SER Section 2.2.2. The NRC staff finds that these TSs require control devices, their driver and release devices, and display or interlock components to be calibrated, inspected, and tested periodically, as appropriate to ensure operability as analyzed in the SAR. The NRC staff finds that these requirements help ensure the availability and reliable operation of the RCS channels, control elements, and devices. Additionally, the range of operation is sufficient to cover the expected range of rod (blade) motion during reactor operation. The NRC staff also finds that the control blades drop by gravity during a scram (independent of the RCS) and that only three of the four control blades need to drop to ensure reactor shutdown. Therefore, no single malfunction in the RCS would prevent the RPS from being able to perform its function and safely shutdown the reactor. Based on the above, the NRC staff concludes that UML's control blade drive system is acceptable.

3.3.1.3 Servo-Controlled Regulating Rod Drive

As discussed in SAR Section 4.2.2.3, the servo drive mechanism for the regulating rod is similar to the control blade drive mechanisms, except that a solid coupling replaces the scram electromagnet assembly, meaning that the regulating rod cannot be scrammed. The servo-controlled regulating rod drive can automatically regulate reactor power within closer limits than those attainable by using the control blades alone. Manual and automatic control of the reactor using the regulating rod is discussed in SAR Section 7.3.5. As discussed in SAR Section 7.3.2, the regulating rod drive is controlled from the DCS GUI display on the control console.

The purpose of the servo-controlled regulating rod drive is to automatically hold the reactor power at a preselected flux level (when the reactor is operated in automatic control). The desired power level is selected by means of the power schedule set switch and is indicated on the DCS power level bar graph. An indicator informs the operator of the regulating rod control mode. The servo amplifier for automatic control is a proportional gain, integral, derivative (PID) controller. Initiation of automatic control is by the reactor operator via the DCS GUI. When the operator places the system in automatic mode, the PID controller will maintain the reactor power at a desired set-point level by controlling the output signal to the regulating rod drive servomotor. The power will be measured using one of the available optically isolated analog outputs from one of the proposed GA linear power monitors (NMP-1000). The difference between the set-point value that the operator inputs and the measured power level signal from the NMP-1000 generates an error value that generates an output signal to drive the servomotor, and increases or decreases reactor power by repositioning the regulating rod. The automatic servo control senses the position of the regulating rod and will receive a reactor period signal from the proposed TFS log PPM to prevent automatic control if the period is too short or the regulating rod is at a full-in or full-out limit. The proposed nuclear instruments (GA NMP-1000 and TFS log PPM) are discussed and found acceptable in SER Sections 3.4.1.1 and 3.4.2.

The automatic power level control may be activated by the operator at any time provided that reactor power is at or above a minimum value (typically 500 watts or greater), the reactor period

is greater than 30 seconds, and the regulating rod is in other than full-in or full-out position. Once placed in automatic control, the system remains in automatic until the operator presses the mode transfer (MAN) button to take manual control, the deviation between the power set-point and actual power is greater than two percent, the reactor scram relays are deenergized, or either of the automatic regulating rod control interlocks (period greater than 30 seconds and regulating rod other than full-in or full-out) are no longer satisfied.

The NRC staff reviewed the information in the SAR, as supplemented, regarding the design of the servo-controlled regulating rod drive and also compared the design of the regulating rod drive with that at similar research reactors. The NRC staff finds that the UMLRR servo-controlled regulating rod drive is similar to systems employed at other research reactors, such as the Missouri University of Science and Technology and the Purdue University research reactors. The NRC staff also finds that the servo-control is designed to limit reactor period and power oscillations and levels based on the values in renewed TS 3.2.2, which is discussed and found acceptable in SER Section 2.2.2, and that the design will not fail or operate in a mode that would prevent the RPS from performing its designed function or prevent safe reactor shutdown. Based on the above, the NRC staff concludes that the UMLRR regulating rod drive is acceptable.

3.3.1.4 Startup Counter Drive

As discussed in UML's letter dated February 16, 2021 (Ref. 101), the startup counter drive and its associated I&C allow remote control of the startup counter position with respect to the reactor core. Similar to the control blade and regulating rod drives (see SER Sections 2.2.2, 3.3.1.2, 3.3.1.3, and 3.6.1), the startup counter drive circuit is part of the RCS and includes two limit switches and a reversible control drive motor. The DCS GUI provides controls and position indications for the startup counter drive. The travel of the drive is established by the limit switches. The startup counter drive mechanism, which is geared to operate at a drive speed of approximately 6 inches per minute, enables the startup counter assembly to be positioned anywhere along its 31-inch stroke. The startup counter container, which is connected to the startup counter drive shaft, provides mounting for the startup counter (which is a proportional counter) and its related connecting hardware.

The NRC staff reviewed the information in the SAR, as supplemented, regarding the design of the startup counter drive. The NRC staff finds that the startup counter drive is adequate and appropriate for the intended use to position the startup counter and to reliably measure the position of the startup counter drive and therefore, is acceptable.

3.3.1.5 Drives Control System Display

The DCS is part of the control console described in Section 7.6 of the SAR. The DCS is a touch-screen, GUI display that provides the operator controls and indicators for the control blade drive selector switch, the control blade manual rundown switch, the control blade manual control switch, the regulating rod drive manual control switch, the start-up counter drive switch, the six position indicators for control blade, regulating rod, and startup counter position, the indicator lights for drive power ON, and drive limit switch indicator lights.

The control console GUI was installed by UML in 2003 using the OPTO 22 hardware and software that was used to upgrade the ARMS in 2000. The human-machine interfaces (HMIs) for the DCS, the process control system (PCS), and ARMS are evaluated and found acceptable in SER Sections 3.6 and 3.7.

3.3.2 Evaluation Findings on the Reactor Control System

The NRC staff evaluated the design of the UMLRR RCS in accordance with Section 7.3 of NUREG-1537, Part 2, based on the information provided in the UMLRR SAR, as supplemented, and reviewed during the NRC audits.

In its response to RAI-7.3 (Ref. 79), UML described two upgrades/modifications to the RCS since its last license renewal in 1985. This first change, made in 2003 to the RCS and DCS, included changes to the RCS drive motors, rod position sensors and limit switches, servo-controls, and the associated controls and displays on the control console. During and following the completion of the upgrade, UML stated that it conducted facility-specific testing of rod drop times, conducted scram function tests, and performed the reactor checkout procedure. These tests were intended to test all possible combinations of control sequences (e.g., control blade withdrawal/insertion, regulating rod withdrawal/insertion, startup counter withdrawal/insertion, and automatic/manual regulating rod control) to ensure the proper operation of the systems. The second upgrade of the RCS, made in 2010, improved the position indication for the drives. UML added relay logic for the position limit controls (full-in/full-out) and moved the encoder from the drive motor to the control drive shaft. Locating the encoder on the control drive shaft requires fewer revolutions per inch to be counted by the input/output (I/O) module, which allows for elimination of the pulse reduction circuit and improves positional accuracy. UML screened both changes under its 10 CFR 50.59 procedure, and UML performed a full 10 CFR 50.59 review for the 2003 change. The NRC staff assessed UML's 10 CFR 50.59 evaluations and determined that the licensee generally provided sufficient detail to address the criteria in 10 CFR 50.59. The NRC staff also reviewed the UML annual reports from 2010-2011 through 2019-2020 (Refs. 10.f through 10.o) and noted no issues with the reliable operation of the RCS since it was last upgraded in 2010.

During the 2017 audit, as documented in the NRC staff's audit report (Ref. 86), the NRC staff observed partial performance of the UMLRR reactor checkout procedure to confirm that the procedure included TS-required checks and surveillances associated with the UMLRR RCS I&C systems. This included witnessing checks of the RCS for magnet power supply voltages, control blade drive initialization and rundown checks, position indication for all blades and the regulating rod and the startup counter, and withdrawal interlock operation to verify RCS operability.

The NRC staff finds that the RCS meets the applicable design acceptance criteria in Section 7.3 of NUREG-1537, Part 2, which are that the system assume a safe state, enable safe reactor shutdown, and not prevent the RPS from performing its designed safety function in the case of control system action or inaction. Additionally, on the basis of its evaluation of the information presented in the licensee's SAR, as supplemented, this SER, and its audit observations, the NRC staff concludes as follows:

- The UMLRR RCS is a reliable and fail-safe system that will help ensure operation of the reactor without exceeding the SLs established in the TSs.

- The RCS has sufficient interlocks, redundancy, and diversity of subsystems to avoid total loss of operating information and control and to limit personnel hazards.

- UML analyzed the functions of the RCS and components designed to permit and support normal reactor operations, and confirmed that 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 normal reactor operations.

- The design of the automatic and manual control element, drives, and display systems are sufficient to monitor and limit reactor period and power to values UML found acceptable in the reactor analyses in Chapter 4 and the accident analyses in Chapter 13 of the SAR, as supplemented.

- The components and devices of the RCS are designed to sense all parameters necessary for RCS 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 operational characteristics of the UMLRR.

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

- Surveillance tests, checks, and calibrations, and associated intervals, as discussed in SER Section 6.4, give reasonable assurance that the RCS will function as designed.

3.4 Reactor Protection System

3.4.1 System Description of the Reactor Protection System

According to SAR Sections 7.1.2, 7.2.2.2, and 7.4, the RPS is designed to ensure that automatic protective action will correct an abnormal situation resulting from transients of the type discussed in Chapter 13 of the SAR before the SL is exceeded. For the UMLRR, these events are categorized as insertion of reactivity, loss of coolant, and loss of coolant flow. The primary parameters of concern are the reactor power level, the primary coolant temperature, the primary coolant flow, and the reactor pool water level. These reactor parameters are interrelated variables used to establish the values of the LSSSs. The SL and LSSS values ensure that the integrity of the fuel cladding is maintained by initiating an automatic protective action (scram) at or before the point where reactor power, coolant temperature, coolant flow, and coolant height reach LSSS values, either individually or in combination. The actual set-points at which RPS action occurs are typically set conservative to the TS-required LSSSs.

The RPS includes functions for monitoring nuclear parameters, such as reactor power and period, and process parameters, such as the cooling systems, the ventilation system, and experimental facilities. The RPS has two modes of operation: forced convection (1 megawatt-thermal (MWt) mode) and natural convection (100 kilowatt-thermal (kWt) or 0.1 MWt mode). In the forced convection mode, primary coolant flow scrams are active and linear power channel scrams are enabled on the highest current range of the electrometer (1 MWt range). In the natural convection mode, the primary coolant flow scrams are bypassed and the linear power channel scrams are enabled on the 100 kWt range.

SER Figure 3-2 (reproduced from SAR Figure 7-2) provides a functional block diagram that shows the linear and logarithmic power and period modules (electronic (fast) scrams) and the

safety chain (channel relay and limit switch scrams). The RCS is shown with dashed lines and is independent from the RPS.

In SAR Section 7.2.1, UML stated that the following criteria are applied to the RPS design:

(a) Single-failure, to ensure that no single failure or single maintenance action or any other single human action could disable its basic safety function

(b) Redundancy, to include functional and equipment diversity

(c) Independence, to ensure physical isolation such that any failure in one channel is isolated to the channel itself and does not affect other channels

(d) Reliability, through the use of technology that is qualified or proven by experience or testing or both

(e) Testability, to ensure the system is designed with the capability for periodic checks, tests, and calibration

(f) Manual Initiation, which provides a simple and direct means for the reactor operator to immediately shutdown the reactor if needed

(g) Access Control, by the use of physical provisions to prevent unauthorized use of the reactor systems

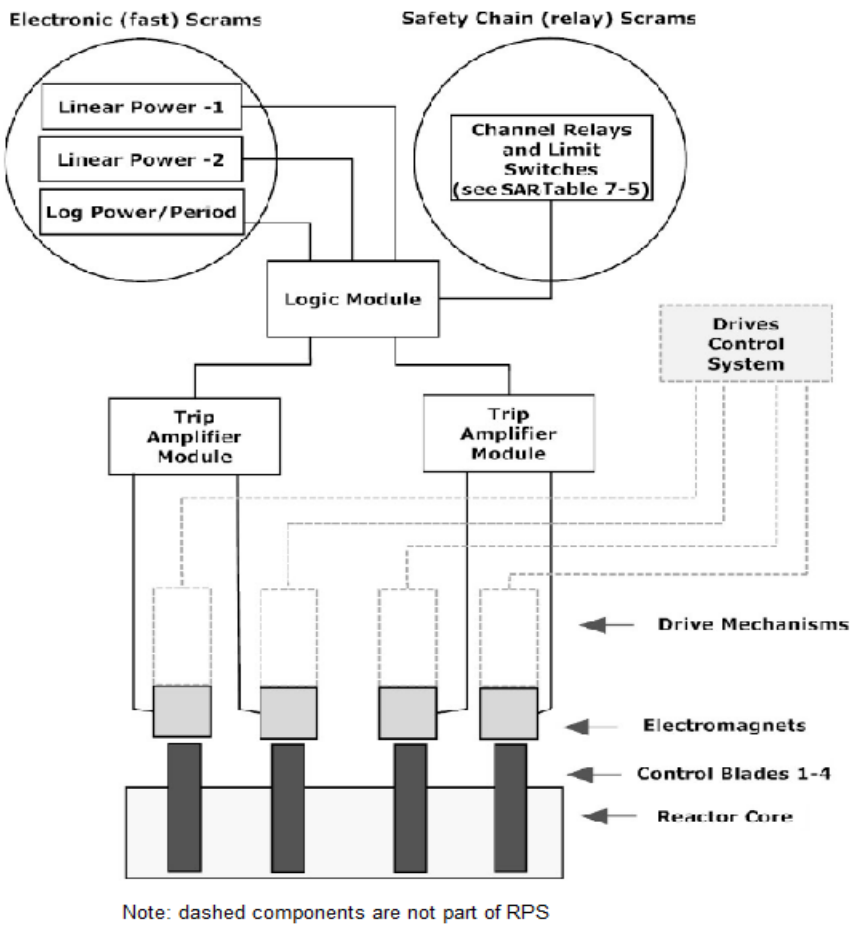

Figure 3-2 RPS Functional Block Diagram

The RPS consists of three subsystems: the NI system, the PCI system, and the scram chain circuit.

3.4.1.1 Nuclear Instrument System

Section 7.4.1 of the SAR describes the UMLRR NI system as consisting of a startup channel, a log power and reactor period channel, two linear power channels, and the linear percent power from the nitrogen-16 (N-16) detector. Each of the NI channels are connected to the scram systems to initiate a reactor shutdown (scram) in the event of an emergency or abnormal condition (e.g., overpower event). The NIs interface with an annunciator and alarm system (discussed in SER Section 3.6) to indicate if any monitored parameter reaches its predetermined setpoint.

In its response to RAI-7.3.b (Ref. 79), UML described two completed upgrades/modifications to the NIs since its last license renewal in 1985. The first change, made in 1997, replaced the original General Electric linear power and logarithmic power instruments with the current GA linear power (NMP-1000) and log N (NLI-1000) channels. The second upgrade, in December 1998, added mechanical height adjusters to the three existing compensated ion chamber (CIC) detectors for the two linear power and one logarithmic power channel. This height adjuster modification replaced a threaded compression fitting (which the licensee claims was difficult to adjust) on each CIC detector with a precision, rack and pinion, linear slide to

allow for more accurate positioning of each neutron detector during annual thermal power verification (required by renewed TS 4.2.3(4), which is discussed and found acceptable in SER Section 2.5.1).

In its renewal application, as supplemented, UML proposed to install new linear and logarithmic power channels. In place of the current analog NMP-1000 (referred to as generation one or Gen-1), UML proposed to install the next generation digital NMP-1000 (or Gen-2) for the linear power instruments, as described in the SAR. For the logarithmic power instrument, UML proposed to replace the GA NLI-1000 with an analog logarithmic power and period module manufactured by TFS, as described in UML's letter dated April 10, 2019 (Ref. 73). If these upgrades are approved by the NRC, the licensee intends to disconnect the existing GA NLI-1000 log/period amplifier and CIC to connect the fission chamber and TFS unit to the RPS and conduct its alignment and calibration procedure to ensure the TFS log PPM is operable prior to operating the reactor (Ref. 73). The licensee's proposed changes to the flux monitoring system are reviewed below and in SER Section 3.8.

UML's current startup channel (proportional counter) and N-16 detector are also discussed below. UML did not propose changes to either the startup channel or N-16 detector in its renewal application.

Startup channel

Sections 4.2.5.1 and 7.4.1.3 of the SAR state that the startup channel includes a boron-10 lined proportional counter detector (located in one corner post of the core suspension frame described in SER Section 2.1), preamplifier, linear pulse amplifier and single channel analyzer, log count rate meter, and high voltage power supply. The startup channel is used to monitor flux levels from a low range sensitivity of 0.12 thermal n/cm^2-s to a high of 5x10^4 n/cm^2-s. The pulses from the detector are amplified and sent to the analyzer for discrimination. The analyzer is capable of discriminating against strong gamma radiation to ensure that changes in neutron flux density are reliably measured. The pulses are then converted to a voltage logic signal processed by the ratemeter. This signal is provided as an analog output signal in counts per second to the PCS HMI display on the operator console to monitor reactor power during startup. The startup channel scaler can also provide a reading of the number of counts occurring during a preset time interval, or conversely, the time required to register a preset number of counts. The scaler is intended for indication at the lowest flux levels. As shown in the scram and inhibit circuit schematic provided by UML in its response to RAI-7.1.b (Ref. 79), the startup channel signal also drives an independent, series contact that powers the hard-wired relay in the control blade drive inhibit circuit (part of the withdrawal inhibit circuit described in SER Section 3.3.1.1). This relay will de-energize and prevent rod withdrawal if the count-rate drops below 2 cps or the startup channel fails.

The primary purpose of the startup channel is measurement and display of the reactor power level during reactor startup and during low-power natural convection operation. Additionally, the inhibit function for startup neutron count rate ensures that the required startup neutron source is sufficient for reactor startup, such that a minimum source multiplication count rate level is being detected. As discussed in SER Section 2.5.3, UML may use either its dedicated startup channel or its proposed log PPM channel for source range monitoring for reactor startup since both channels have sufficient range and both have a control blade withdrawal inhibit function for low count rate. Renewed TS 3.2.5, "Minimum Channels Needed for Reactor Operation," would require a startup count rate channel to be operable and renewed TS 3.2.6, "Reactor Control System Interlocks," would require an interlock for start-up neutron count rate to be greater than

2 cps when the reactor is operating. The required channel and its associated withdrawal inhibit interlock would be required to be tested periodically in accordance with renewed TS 4.2.3. TSs 3.2.5, 3.2.6, and 4.2.3 are discussed and found acceptable in SER Section 2.5.3.

The NRC staff reviewed the information in the SAR, as supplemented, and finds that the startup channel provides a redundant power level and period indication during reactor startup. The NRC staff also finds that the location and sensitivity of the startup channel, along with the location and emission rate of the neutron startup source (discussed in SER Section 2.2.4), will help ensure that changes in reactivity will be consistently displayed even with the reactor shut down. Additionally, the NRC staff finds that the startup channel and interlock meets the guidance of NUREG-1537, Part 2, to prevent reactor startup (increase in reactivity) without sufficient neutrons in the core to ensure that changes in reactivity will be reliably indicated. Therefore, the NRC staff concludes that the startup channel is acceptable.

Log Power and Period Meter

As noted in UML's letter dated December 19, 2019 (Ref. 81), TFS information on the TR-10 is marked as copyright and/or proprietary. UML stated that TFS (i.e., the owner of the information) has been unresponsive to repeated requests for TFS to supply its documentation to the NRC with an affidavit for withholding under the provisions of 10 CFR 2.390(b). During the 2020-2021 audit (Ref. 86), the NRC staff reviewed the TFS material made available to the NRC staff for viewing on the online audit portal in accordance with the requirements of 10 CFR 2.390 and, on the basis of its review, UML's statements, and document markings, the NRC staff determined that the TFS information appears to contain both proprietary information and material protected under copyright laws and, therefore, should not be placed in the NRC's Agencywide Documents Access and Management System (ADAMS) without the owner's express permission. Accordingly, the TFS information that has been docketed (Ref. 98.b, Ref. 98.e, and Ref. 98.f) consists only of the document coversheet under fair use to preclude copyright infringement on the exclusive rights granted to the copyright holder, such as the right to reproduce, distribute, or display the protected work.

In its updated SAR Section 7.4.1.2 provided by letter dated April 10, 2019 (Ref. 73), UML stated that the function of the logarithmic power/period channel is to monitor the neutron fluence rate (power level) in ranges that overlap the start-up channel and the linear power measuring channels. Currently, UML uses a GA NLI-1000 for the UMLRR logarithmic power and period monitoring. However, UML requested in its license renewal application that the NRC staff evaluate UML's replacement of the NLI-1000 with the TFS log PPM (model TR-10) (Ref. 73, Ref. 79, and Ref. 81). In its updated SAR Section 7.4.1.2 (Ref. 73), UML noted that there is widespread planned or current use of the log PPM at other nuclear reactors, including the North Carolina State University research reactor and the RINSC research reactor.

UML's updated SAR Section 7.4.1.2 states that the proposed log PPM channel consists of a fission chamber neutron detector and the rack mounted wide-range logarithmic neutron flux monitor. The pulses from the detector are amplified and shaped by a pulse height discriminator to remove unwanted signals due to alpha and gamma radiation and noise. The discriminator output signal is processed by the log count rate and rate (period) circuit board, the wide-range log count rate and rate circuit board, and the log amplifier and rate circuit board. The log count rate circuit provides an output that is proportional to the logarithm of the average count rate of detector signal pulses over a range of 0.1 to 10^5 cps. The wide-range log count rate circuit provides an output that is proportional to pulse count rate over the range of about 1E-8 percent to 3E-2 percent reactor power. The log amplifier circuit provides an output that is proportional to

the mean square voltage signal over the range of about five decades from 1E-3 percent to 200 percent of reactor power. The monitor chassis, which is proposed to be physically installed in the control room instrument panel (see SER Figure 3-9), contains the electronics for signal processing, bi-stable trips, isolated outputs, and the high voltage and low voltage power supplies. The front panel of the chassis displays bar and numeric readouts of source range counts and period, wide-range log power and period, and wide-range linear power, as well as light-emitting diode (LED) indicators for all trips and alarms. A remote readout module provides a signal-isolated display of counts, period, and linear power on the operator control console.

UML's updated SAR section 7.4.1.2 states that the source range, wide range, and power range circuits are combined by an auctioneer circuit such that the proposed log PPM can provide neutron flux measurement from reactor shutdown to reactor full power level spanning from 1E-2 n/cm^2-s to 1E10 n/cm^2-s. These power measurements are calibrated in units of watts-thermal to provide measurement indications from sub-critical levels up to 120 percent of UML's steady-state licensed power level of 1 MWt. In its updated SAR Section 7.4.1.2.4 (Ref. 73), UML's failure analysis states that failure of the power source or internal power supplies will cause all the trips to assume their tripped state. Thus, on loss of power, the log PPM trip circuits are deenergized, providing a fail-safe trip.

Table 7.4 of UML's updated SAR Section 7.4.1.2 lists the trips provided by the proposed log PPM, which are summarized below in SER Table 3-1. During the 2020-2021 audit (Ref. 86), the NRC staff reviewed UML's "Standing Order #5" (Ref. 98.a), which UML stated is currently used to document the setpoints for these trips for the current GA NLI-1000. UML also stated during the 2020-2021 audit that the proposed log PPM setpoints will also be controlled by UML procedure to be conservative relative to the TS 2.2 LSSSs and TS 3.2.3 required reactor protection system scram setpoints. This practice is consistent with the Section 2.2 guidance of American National Standards Institute/American Nuclear Society (ANSI/ANS)-15.1-2007 to achieve operational flexibility by setting actual trip points, where possible, more conservatively than specification values.

Table 3-1 Log PPM protective trips

TRIP DESCRIPTION
Short Period Alarm/Inhibit
Short Period Scram
Low Count Rate Inhibit
High Voltage Scram
High Power Alarm/Inhibit
High Power Scram
Natural Convection Power Alarm/Inhibit
Natural Convection Power Scram

As shown in SER Figure 3-3, which is reproduced from Figure 7.5 of UML's updated SAR Section 7.4.1.2 submitted by letter dated April 10, 2019 (Ref. 73), the log PPM (model TR-10) provides the capability to check the rackmount signal processor operation periodically by use of the front panel switches. The "TEST SELECT" switch provides seven calibration points for calibration of the level outputs. The front panel includes individual adjustments for calibrating the source, wide, and power ranges and positions to insert a positive, zero, or negative ramp signal for calibration of the rate circuits. In the "TEST" position, the "LIN ADJUST," "LOG

ADJUST," and "RATE ADJUST" potentiometers are actuated and inject a signal to test bi-stable trips, isolators, and displays. Whenever the "TEST SELECT" switch is not in the off position, the yellow "CH IN TEST" LED illuminates. Additionally, any action that takes the instrument out of operate mode (e.g., testing) activates the trip relay and results in a reactor scram. The red "NON-OP" LED illuminates if there is an improper detector high voltage or improper signal processor low voltage power supply output. Finally, two locking potentiometers provide the ability for adjustment of full-scale meter indications, which the TFS instruction manual recommends be completed to align the TFS at 100 percent reactor power following the channel calibration procedure.

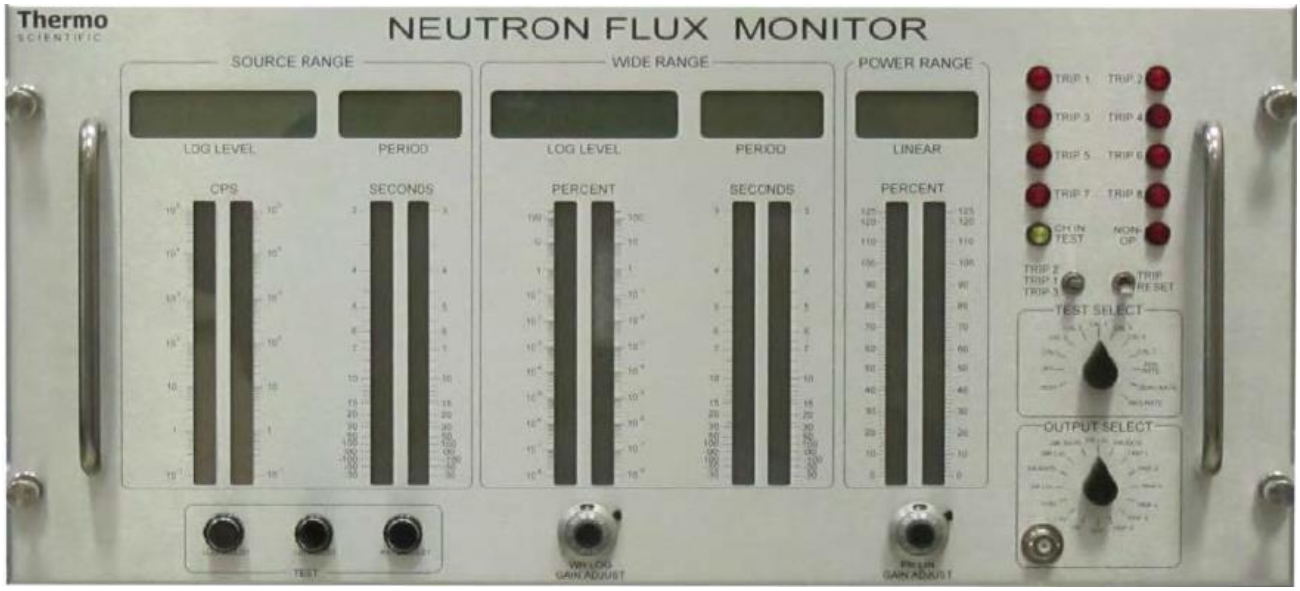

Figure 3-3 Wide Range Log Period Power Module (Reproduced from (Ref. 73))

The TFS instruction manual (Ref. 98.b) contains an annual calibration procedure for the log PPM. In its response to RAI-7.6 (Ref. 79), UML stated that a UMLRR procedure, which incorporates the TFS procedure, will be used to check the proper channel function and to provide the necessary steps for making calibration adjustments, when necessary, during the performance of surveillances. Renewed TS 4.2.3, which includes the required surveillances for the TFS log PPM, is discussed and found acceptable in SER Sections 2.5.1 and 2.5.3. During the 2020-2021 audit (Ref. 86), the NRC staff confirmed that UML used the TFS procedure to prepare UML's proposed Logarithmic Power Channel Check and Calibration procedure.

The NRC staff reviewed UML's descriptions of the detector current, sensitivity, linearity, and range of the proposed log PPM (in UML's updated SAR Section 7.4.1.2) and the existing NLI-1000 (in UML's RAI responses submitted October 18, 2019 (Ref. 79)). Additionally, the NRC staff reviewed the TFS instruction manual for the neutron flux monitoring systems (Ref. 98.b), which states that the neutron flux monitoring systems have demonstrated high immunity to electromagnetic interference and noise and employ modern, proven electronics for high reliability and low maintenance. Therefore, based on the information in UML's updated SAR Section 7.4.1.2, UML's RAI responses submitted October 18, 2019 (Ref. 79), and the TFS instruction manual, the NRC staff finds that the log PPM channel meets or exceeds the performance requirements of the UMLRR and of the existing NLI-1000. The NRC staff also finds that the power, temperature, and humidity specifications stated in the TFS instruction manual are compatible with the environmental conditions at the UMLRR. Specifically, the

NRC staff determined that the allowable operating ranges for the proposed equipment to be installed are well within the environmental range maintained in the reactor room by the heating, ventilation, and air conditioning system and are, therefore, acceptable. Further, the NRC staff finds that the proposed TFS log PPM channel provides suitable diversity and redundancy to both the startup and linear power channels. The TFS provides independent indication that overlaps both startup and linear channels by a minimum of one decade (during detector changeover) to help avoid a total loss of operating information and control, to limit hazards to personnel, and to ensure that a single isolated malfunction of equipment cannot prevent the reactor protection systems from performing necessary functions or prevent safe shutdown of the reactor. The NRC staff also finds that the TFS log PPM provides independent and isolated trip channels and is designed to assume a safe state on loss of electrical power. The NRC staff also finds that the proposed log PPM will provide reliable readings to a maximum power level that is in a range high enough to monitor possible excursions above the licensed power level of 1 MWt. Additionally, the NRC staff finds that the TFS log PPM is designed to operate reliably in the UMLRR ambient environment and that the equipment is designed to be readily tested and calibrated to ensure operability.

Based on the above, the NRC staff finds that the TFS log PPM meets the Section 7.3 guidance of NUREG-1537, Part 2, to be 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. The NRC staff also finds that the proposed log PPM, including the fission chamber, follows the guidance in NUREG-1537 for testing capabilities for periodic checks, tests, and calibrations, and related surveillances to verify the availability and reliable operation of the channel. Therefore, the NRC staff concludes that the proposed TFS log PPM (Model TR-10) is acceptable to replace the current NLI-1000 as the UMLRR wide range power and period monitor and to meet the requirements for reactor operation in TS 3.2.3, TS 3.2.5, and TS 3.2.6.

Linear Power Channels

Currently, as discussed in UML's responses to RAI-7.1.a and RAI-7.3 (Ref. 79), the UMLRR is operated with two GA NMP-1000 linear power channels, referred to as Generation 1 or Gen-1 channels. The Gen-1 NMP-1000 is an all-analog channel that was installed at UML under 10 CFR 50.59 in September 1997. The NRC staff reviewed the annual reports from 2005 through 2020 (Refs. 10.a to 10.o) and noted that the reports did not indicate any issues related to the Gen-1 NMP-1000 performance during the period from 2005 to 2010. However, during the period covered by the 2010-2011 annual report, the facility experienced 11 inadvertent, non-emergency automatic shutdowns due to an electronic problem associated with an NMP-1000 linear power monitoring channel. In the 2011-2012 reporting period, the facility experienced an additional 26 inadvertent shutdowns due to the Gen-1 NMP-1000. The facility continued to experience the following nuisance shutdowns due to linear power channel noise: 6 in 2012-2013; 7 in 2013-2014; approximately 12 in 2014-2015; approximately 8 in 2015-2016; approximately 12 in 2016-2017; approximately 14 in 2017-2018; 17 in 2018-2019; and 18 in 2019-2020. UML stated in the annual reports that there is no safety significance associated with any of these inadvertent scrams.

As stated in procurement documents submitted to the NRC by UML (Ref. 98.c), UML ordered replacement NMP-1000 linear power monitors in 2011, but it did not receive the equipment until February 2014. After delivery, UML discovered that the Gen-2 NMP-1000s it received contained digital processor technology. As a result of GA's unexpected change from analog to analog-digital hybrid, UML requested in its license renewal application that the NRC license

renewal review also evaluate the proposed replacement of the current analog Gen-1 NMP-1000 with the updated analog-digital Gen-2 NMP-1000.

In its “NMP-1000 System Requirements Specification” (Ref. 102.b), GA stated that the Gen-2 requirements are based on the existing functionality and operation of the existing analog NMP-1000 Module (Gen-1) and incorporate all of the features and functions of the existing analog NMP-1000 Module. The differences between the proposed Gen-2 NMP-1000 and the current Gen-1 version include:

- a microprocessor that can automatically select range,
- a liquid crystal display (LCD) touchscreen interface, and
- an ethernet connection (disabled for UMLRR, as discussed below).

Section 7.4.1.1 of the UMLRR SAR states that the function of the proposed linear power channel is to monitor the neutron fluence rate (power level) in ranges that overlap the start-up channel and the logarithmic power measuring channels. The proposed linear power channels are calibrated in units of watts and are capable of measuring power levels from 0.1 watt-thermal up to 1.2 MWt (120 percent of licensed power), which is the same range as the current Gen-1 NMP-1000. The channels provide reactor power level output to the reactor console, reactor power level signal input to the flux regulator for automatic control (channel 2 only), and multiple bi-stable trip functions for alarms and scrams. The proposed NMP-1000 front panel, shown in SER Figure 3-4, which is reproduced from the “NMP-1000, Multi-Range Linear Module User Manual” (Ref. 103.b), includes a monochrome LCD that includes an integrated touchscreen display (TSD) with integrated digital backlight and contrast controls. The display has four user selectable screens: “Main,” “Gain,” “Test/Cal,” and “About.”

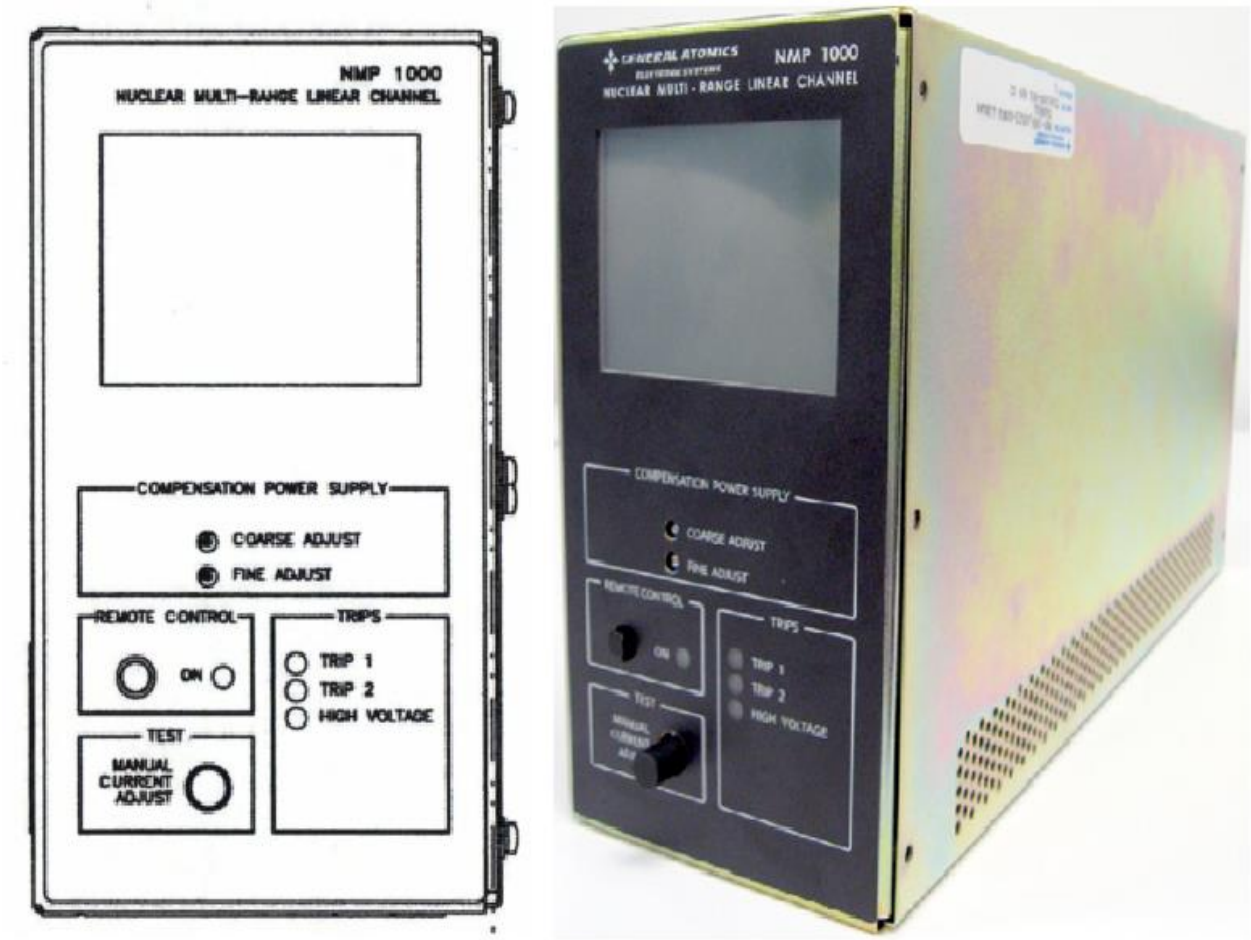

Figure 3-4 Generation 2 NMP-1000 Linear Power Monitor (Reproduced from (Ref. 103.b))

The NMP-1000 front panel is described in detail in the "NMP-1000, Multi-Range Linear Module User Manual" (Ref. 103.b). The main menu, shown in SER Figure 3-5, displays parameter readings (detector high voltage, compensating voltage, the selected gain range, and percent reactor power), a bar graph of reactor power (on a linear scale of 0 to 120 percent), a trip reset button, and a flashing heartbeat indication of the watchdog.

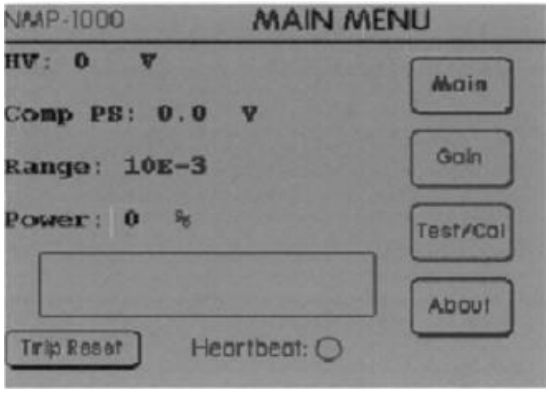

Figure 3-5 NMP-1000 Main Menu (Reproduced from (Ref. 103.b))

The gain menu displays percent reactor power, the selected detector measurement range, and buttons to select other ranges and to select automatic software range mode (the gain menu screen is shown in SER Figure 3-6). The Test/Cal menu displays percent reactor power, the selected detector gain range, and buttons to allow the user to select between operate and four test modes. The four test modes are: calibrate high and calibrate low (high/low for Trip 1 and for Trip 2); a manual current adjust to test the automatic range selection; and a test of the detector high voltage trip. Each of the four screens (the three screens discussed above and an "About" screen) also provides navigation buttons to select between the four screens. The "About" screen provides the software version information for the microprocessor and display, the Internet Protocol address, adjustments for contrast and backlight, and the means to calibrate the TSD. In its response to RAI-7.7 (Ref. 79), UML stated that it will verify that the proper software version is checked and verified against the configuration log during the TS 4.2.3(3)-required calibration for the equipment.

Two red LED indicators illuminate for high-power trips: a logic high power trip (Trip 1) and a relay high power trip (Trip 2). A third red LED illuminates when the detector compensating voltage goes below the high voltage setpoint, which will result in a detector "high voltage" scram. The front panel of the chassis also includes a manual current adjust test circuit to allow the user to test the proper performance of the electrometer and to ensure that the functionality of all trip circuits using the knob on the front panel to manually control the current in test mode. Two recessed potentiometers, accessible via the front panel with an adjustment tool, allow the user to set the neutron detector compensation voltage.

A remote-control button is also present on the front panel. During the 2017 audit, as documented in the audit report (Ref. 86), the NRC staff asked how the remote control is used for the UMLRR, and UML responded that the button is only used with the GA software for a remote LCD associated with a Training, Research, Isotopes, General Atomics (TRIGA) reactor console. UML also stated that the button, and the associated green "ON" LED indicator, have no function for the UMLRR. To confirm UML's statement, the NRC staff reviewed the systems requirement specification for the NMP-1000 (Ref. 102.b). The NRC staff verified that the remote port is electrically isolated from the other NMP-1000 circuitry with an independent power and ground. Additionally, the NRC staff noted that the system requirements specification states that when in local operation, the NMP-1000 ignores all command signals coming from the remote connections, which is tested during verification and validation (V&V) testing (Ref. 102.e). Also, during the 2017 audit, the NRC staff noted that the current NMP-1000 (Gen-1) also has the remote-control button and LED, which are not being used. In its response to RAI-7.9.c (Ref. 79), UML stated that it is part of the current startup checkout procedure (and will continue to be the case for the new Gen-2 NMP-1000) for the operator to verify that the remote-control button is in the off position. During the 2017 audit, the NRC staff observed the UML operator verify that the remote-control switch is "off" during the reactor checkout using the current UMLRR Pre-Startup Check Sheet (Forced Convection). In its response to RAI-7.9.a (Ref. 79), UML stated that the wiring to both the J8 and J9 connections within the instrument chassis, which are for communications with a TRIGA console that UML does not possess, have been disconnected by UMLRR staff for both proposed NMP-1000 units. UML also stated that its approved UMLRR calibration procedure for the NMP-1000 will include a procedural step to verify that J8 and J9 are disconnected when the instrument chassis is opened for calibration or maintenance purposes.

SAR Section 7.4.1.1 states that each proposed NMP-1000 channel consists of a CIC neutron detector and a multi-range power monitor. The current linear power monitors are physically mounted in the control room instrumentation panel (see SER Figure 3-9) in a nuclear

instrumentation module (NIM) bin.[1] UML indicated that the proposed monitor chassis for both Gen-2 linear power channels would also be mounted in the existing NIM bin. A signal-isolated analog output will continue to provide a remote display of linear power from each channel on the operator control console (see SER Figure 3-8).

The major subassemblies of the proposed Gen-2 NMP-1000 channel include:

- NMP-1000 motherboard
- analog amplifier
- digital interface board (DIB)
- trip/alarm board
- isolation amplifier
- LCD module
- front panel
- power supply module
- compensation power supply module

As stated previously, the Gen-2 NMP-1000 proposed by UML is an analog-digital hybrid. In the "NMP-1000, Multi-Range Linear Module User Manual" (Ref. 103.b), GA described the Gen-2 as a microprocessor-based wide-range linear power monitor. A high input impedance operational amplifier board measures the incoming current signal from the CIC detector and converts it into a linear analog DC voltage (0 to +10 DC volts (Vdc)) in nine one-decade ranges. For every decade of current, a relay switches the appropriate feedback resistor into the circuit to generate the expected output signal, as determined by the microprocessor. Every range has an adjustment potentiometer that allows for calibration of the circuit. The Gen-2 NMP-1000 also contains a DIB, local liquid crystal TSD, and the software to control and operate both, as discussed in GA's "NMP-1000 Software Requirements Specification" (Ref. 102.a). The DIB has an interface to the analog power level operational amplifier and an interface to the LCD touchscreen.

According to GA's "NMP-1000 Software Requirements Specification" (Ref. 102.a), neither the DIB nor the TSD, which uses the commercially-available NetBurner software and Amulet TSD software, respectively, will have any internal software interfaces (i.e., interfaces between multiple applications running on either the TSD or the DIB). The software on each is produced per the GA software requirements specification and GA's software QA plan (Ref. 103.d) for its specific application, and is the only software package installed on each. There will be no other software applications on the TSD or the DIB. According to the GA "NMP-1000 Failure Modes and Effects Analysis" (Ref. 102.c), the NetBurner module (including the embedded firmware) and the Amulet display kit have both been validated by GA for the intended use within the proposed Gen-2 NMP-1000. Software configuration for the proposed NMP-1000 is formally controlled according to the "GA Software Configuration Management Plan" (Ref. 103.a) and applicable UML CM operating procedures, which are discussed in SER Section 3.8.

The DIB also supports an interface to a maintenance computer and an ethernet interface to an external system. However, in its response to RAI-7.9.a (Ref. 79), UML stated that the maintenance and ethernet interfaces are not installed for the UMLRR because they are only applicable to a TRIGA console. UMLRR disables these interfaces as part of the Gen-2 linear

[1] NIM is a standard that defines an instrumentation system consisting of standard modular instruments, the bins in which they are housed, and associated power supplies. Standard NIM Instrumentation System; https://www.osti.gov/biblio/7120327/; Date accessed: January 5, 2022; Date published: May 1, 1990.

power channel installation plan provided in Appendix C to UML's response to RAI-7.6 (Ref. 79). The DIB includes the functions of a watchdog timer, programmable interrupt controller (to assign higher priority to user defined tasks), and both volatile and non-volatile memory for boot, code and data storage, and code execution. The specific functions of the DIB software are to monitor the NMP-1000 inputs, perform auto ranging (when selected) of the range switching relays, and interface to the TSD. The TSD software displays the data obtained from the DIB and provides user control of the display's contrast/brightness, touchscreen calibration, and DIB operating modes (auto/manual).

According to the NMP-1000 User Manual (Ref. 103.b), the range switching relays of the proposed NMP-1000 may be selected manually through the gain menu (via the TSD) or through the auto ranging logic circuitry. The gain menu display, shown in SER Figure 3-6, provides buttons to select the detector measurement range (decade) and for operator selection of automatic software range mode.

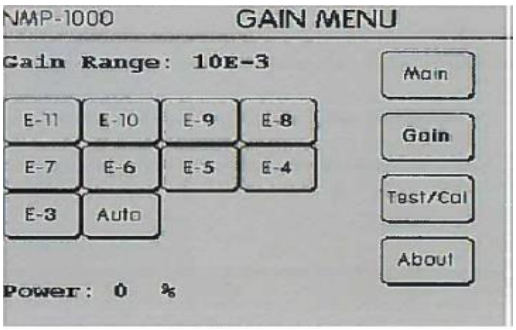

Figure 3-6 Gain menu display (Reproduced from (Ref. 103.b))

As discussed in the NMP-1000 user manual, while the NMP-1000 is in auto mode, a software algorithm controls the amplifier gain settings. The algorithm determines and enables the proper electrometer range based on the highest power trip current settings and enables the relay for the appropriate electrometer feedback circuit. The switching is triggered by a comparator circuit that compares the output of the electrometer voltage to a setpoint for the up-level switch point established by a voltage divider network. The switch points are nominally adjusted for 9.00 Vdc (95 percent of the meter full-scale) for the UP-level comparator and 0.667 Vdc (8 percent of full-scale) for the DOWN-level comparator. Since the analog voltage of each of the nine one-decade ranges returns a 0 to +10 Vdc signal, spurious scrams must be prevented at the lower eight scales when the comparator voltage approaches the setpoint. When in auto mode and also not in the highest range (i.e., 1E-3), the percent power signal is pulled low, essentially preventing a power trip from occurring. If the NMP-1000 is taken out of auto mode, or the highest range has been reached, the percent power signal is released (not pulled low) and both power trip circuits are enabled. GA discussed the possible failure modes for the auto mode in its "NMP-1000 Failure Modes and Effects Analysis" (Ref. 102.c). The analysis states that an auto mode software component either fails in such a way that the trips would never come on or the trips would never go off. The trips never turning off results in the system being inoperable due to the scram signal not clearing (fail-safe state). The failure modes and effects analysis further states that the trips never turning on are tested at prestart tests. During the 2017 audit (Ref. 86), the NRC staff observed performance of the testing for the trips failing to turn on as part of the UMLRR pre-start operability tests. The test demonstrated that the software algorithm determines the proper electrometer range based on the highest power trip current settings and enables the relay for the appropriate electrometer feedback circuit. The test also confirmed that

the NMP-1000 enables the power trips only when the NMP-1000 is measuring power in the highest current range, as designed.

Section 2.2.4, “Trip/Alarm Board,” in the user manual for the proposed NMP-1000, describes the optically-isolated trip/alarm circuits. There are six identical bi-stable circuits and a corresponding relay to generate the trips and alarm indications for a loss of high voltage and for the high power. A comparator monitors incoming voltage from the amplifier circuit and compares it to the reference voltage. The reference voltage (trip setpoint) is user configured by adjustment of an internal potentiometer on the circuit card. The NMP-1000 trip circuits latch in the tripped state. In order to unlatch the circuit, the parameter(s) causing the trip must have cleared and the operator must apply a reset signal. The bi-stable trip circuits also trip when deenergized, providing a fail-safe trip on loss of power. UML’s failure analysis in SAR Section 7.4.1.1.5 also states that an equipment failure will result in a scram. The user manual also describes the proposed NMP-1000 test modes that are enabled via the TSD. These test modes allow the user to test the proper performance of the electrometer and to ensure the functionality of all trip circuits. The test modes are “HV” (high voltage), calibrate high, calibrate low, and manual current. The HV and calibrate high test modes cause the bi-stable trips to alarm; current low gives fixed power indication in the highest range; and manual test lets the user vary the current over all ranges with a front panel potentiometer. Additionally, any action that takes the instrument out of operate mode (e.g., testing) will immediately activate all trip relays and result in a reactor scram.

During the 2020-2021 audit (Ref. 86), the NRC staff reviewed UML’s “Standing Order #5” (Ref. 98.a), which UML stated during the audit is currently used to document the trip setpoints for the current Gen-1 NMP-1000. UML further stated during the 2020-2021 audit that the proposed Gen-2 NMP-1000 setpoints will also be controlled by UMLRR procedure to be conservative relative to the TS 2.2 LSSSs and the TS 3.2.3 required reactor protection system scram setpoints. The NRC staff notes that this practice is consistent with the Section 2.2 guidance of ANSI/ANS-15.1-2007 to achieve operational flexibility by setting actual trip points, where possible, more conservatively than specification values.

A watchdog timer (WDT) provides continuous monitoring of the microprocessor activity for failure detection. The heartbeat LED on the NMP-1000 main menu display (see SER Figure 3-5) indicates the activity of the microprocessor and will blink about once per second as long as all software tasks are executing properly. Although the NMP-1000 WDT is not a UMLRR TS-required scram, if the WDT does not receive input from the microprocessor for more than 1.6 seconds, the WDT generates an alarm and causes the trip relay outputs to go to fail-safe. When the WDT alarm occurs, the heartbeat LED will stop blinking, the loss of communications will cause high power trip (Trip 1), high power trip (Trip 2), and the detector high voltage trip to open, any one of which results in a scram. Once activated, a WDT alarm requires deliberate action to be deactivated and can only be reset by cycling power (or, if the instrument cover is off and the instrument is in Test Mode, by pressing “S2” on the DIB).

In its letter dated September 30, 2020 (Ref. 98), UML stated that GA confirmed that the architecture of the Gen-2 NMP-1000 is fundamentally the same as the UML Gen-1 modules except that the proposed NMP-1000 modules require modification to enable the proper trip setpoints that correspond to the selected UMLRR mode (i.e., 1 MWt or 0.1 MWt) for either forced or natural circulation. Additionally, in correspondence with UML (Ref. 98.g), GA stated that the Gen-2 system is currently operating at a U.S. Department of Energy facility (Idaho National Laboratory). The documentation states that the units GA shipped to UML are identical to the Idaho National Laboratory NMP-1000 units, including the manuals and documentation,

except for the natural circulation modification for UML. In its response to RAI-7.4 (Ref. 79), UML confirmed that the Idaho National Laboratory documents are also applicable to the proposed NMP-1000 upgrade for the UMLRR. The natural circulation modification is discussed in SER Section 3.4.1.4.

Based on the information provided in the SAR, as supplemented, and reviewed by the NRC staff, and based on the NRC staff's audit observations, the NRC staff finds that the proposed NMP-1000 linear channels meet the NUREG-1537 design acceptance criteria for nuclear flux monitoring channels. Specifically, the NRC staff finds that the proposed NMP-1000 linear channels have sufficient range to provide reliable readings to a maximum power level that is high enough to monitor the licensed power level of 1 MWt (and above). The NRC staff also finds that the proposed NMP-1000 channels meet the design criteria for redundancy and independence. Each channel provides an independent and redundant signal isolated output of linear power on the instrumentation panel and on the operator control console. The corresponding isolated protective system trips help avoid a total loss of operating information and control to limit hazards to personnel and to ensure that a single isolated malfunction of equipment cannot prevent the reactor protection systems from performing necessary functions, or prevent safe shutdown of the reactor. The range of the proposed NMP-1000 overlaps the range of the log PPM channel and its power trips operate independently from each other and independent of the log PPM trips. The NRC staff notes that, as described in the SAR, as supplemented, the NMP-1000 is designed for ease of testing. As is shown in SER Figure 3-4, the proposed NMP-1000 has built-in capability for periodic checks, system tests, and calibration of the channel by use of the front panel switches and dials for testing channel operability. Accordingly, the NRC staff finds that the NMP-1000 is designed to operate reliably and that the equipment is designed to be readily tested and calibrated to ensure operability and is designed to assume a safe state on loss of electrical power.

The NRC staff reviewed information in GA's data sheets provided in the NMP-1000 user manual (Ref. 103.b) on the detector current, response time constants, linearity, and range of the proposed Gen-2 NMP-1000, as compared to these specifications for the existing Gen-1 NMP-1000, as provided in Appendix B to UML's response to RAI-7.3.b (Ref. 79), and finds that the proposed Gen-2 channel has performance specifications equivalent to the existing Gen-1 NMP-1000 and meets the performance requirements of the UMLRR. The NRC staff also reviewed the GA Gen-2 NMP-1000 user manual and the operations and maintenance manual for the Gen-1 NMP-1000 (Ref. 103.b and Ref. 103.c, respectively), and finds that the power, temperature, and humidity specifications are similar and compatible with the environmental conditions at the UMLRR. GA's "NMP-1000 System Requirements Specification" (Ref. 102.b) states, regarding electromagnetic shielding, that the NMP-1000, including its communication links, analog interfaces, and power leads, is designed and fabricated to achieve a noise resistant system suitable for operation within a research reactor facility. The NRC staff finds that the allowable operating ranges and equipment qualification for the proposed equipment to be installed are well within the environmental range maintained in the reactor room by the heating, ventilation, and air conditioning system and are, therefore, acceptable. The proposed NMP-1000 includes isolated alarm and trip circuits and the system V&V testing and failure mode analysis shows that the equipment is designed to perform its safety function after a single failure and to meet the requirements for seismic and environmental qualification. The design criterion for manual initiation is discussed in SER Section 3.4.1.3 and access control is evaluated in SER Section 3.8.4.

The NRC staff notes that the two NMP-1000 systems are potentially susceptible to common cause failure based on running identical internal software kernels. However, the NRC staff

finds, based on UML's engineering evaluation of the proposed NMP-1000 failure modes in the SAR, as supplemented by GA's "NMP-1000 Failure Modes and Effects Analysis" (Ref. 102.c), that any potential common cause failures will not degrade RPS functionality. Foremost, the internal WDT monitors the activity of the microprocessor and if no input is received for more than 1.6 seconds, the WDT sends an alarm and takes the module out of operate mode, which results in a scram. Therefore, if both NMP-1000 units failed simultaneously due to a common cause, the WDT would cause the reactor to scram. Second, the log PPM provides diverse protection from a reactor overpower scenario. Additionally, the log PPM and the N-16 Power Module provide independent power level indication that can be used by the operator to help detect any problems with the NMP-1000 channels. Finally, the operator can manually initiate a scram (see SER Section 3.4.1.3) to accomplish the protective function that might be defeated by a potential common-cause failure.

Based on the above, the NRC staff finds that the proposed NMP-1000 linear power channels provide suitable redundancy and diversity to both the startup and log PPM channel to avoid total loss of operating information and control, to limit hazards to personnel, and to ensure that single isolated malfunctions of equipment cannot prevent the reactor protection systems from performing necessary functions or prevent safe shutdown of the reactor. Accordingly, the NRC staff finds that the proposed NMP-1000, in conjunction with the existing CIC detector, meets the Section 7.3 guidance of NUREG-1537, Part 2, to be designed to sense the nuclear parameters necessary for facility operation with acceptable accuracy and reliability and to transmit that information with high accuracy in a timely fashion. The NRC staff also finds that the proposed NMP-1000 follows the guidance in NUREG-1537 for testing capabilities for periodic checks, tests, and calibrations, and related surveillances to verify the availability and reliable operation of the channel. Therefore, the NRC staff concludes that the proposed Gen-2 NMP-1000 is acceptable to replace the current Gen-1 NMP-1000 as the UMLRR linear power channel and to meet the requirements for reactor operation in TS 3.2.3, TS 3.2.5, and TS 3.2.6.

Nitrogen-16 Power Module

As indicated in SAR Table 7-7, the N-16 power module, and its associated high voltage power supply for the N-16 ion chamber detector, are mounted in the instrumentation panel (see SER Figure 3-9). As indicated in SAR Table 7-7 and discussed in SAR Section 10.2.6.1, the N-16 power module measures current from an ion chamber detector located above the primary coolant pipe exiting the core (prior to entrance to the N-16 holdup tank; see SER Section 4.1.1.1). N-16 is an isotope of nitrogen generated by neutron activation of oxygen contained in the primary water. It has a short half-life of 7.1 seconds and it decays via beta decay. This decay, by emission of very energetic gamma rays (6 mega electron-volts), can readily penetrate the wall of the piping and is easily measured by the ion chamber located on the UMLRR primary piping. The concentration of N-16 present in the primary coolant is in radioactive equilibrium and it is directly proportional to the fission rate in the core and thus to the reactor power. Linear percent power indication from the N-16 power monitor is displayed on the PCS, as stated in SAR Section 7.6.1.4. As discussed in SAR Chapter 5 and SER Section 4.1.1.1, primary coolant does not flow through the primary piping or N-16 holdup tank when the reactor is operated in natural convection mode. Therefore, the NRC staff notes that because coolant only flows through the primary piping when the reactor is operated in forced convection flow mode, the N-16 power module is only applicable for forced flow mode.

The N-16 power module is not a TS-required system. However, during the 2020-2021 audit (Ref. 86), UML stated that the N-16 power monitor is calibrated by applying calorimetry techniques. Additionally, during the audit, UML confirmed that although the trip is not described

in the SAR, as supplemented (and therefore not listed in SER Table 3-2), and does not provide a required safety function, the N-16 power module provides a high power trip (1.15 MWt) via the PCS scram contacts in the scram safety chain. In this manner, the N-16 power module provides both a redundant and diverse backup to the other NIs.

Evaluation Findings for the Nuclear Instruments

Based on the NRC staff's review of the information in the SAR, as supplemented, and its observations during the NRC staff audits (Ref. 86), the NRC staff finds that the design of the UMLRR nuclear instruments meets the design acceptance criteria in the guidance in Section 7.3 of NUREG-1537, Part 2, that the instrumentation provides continuous indication of the neutron flux over the expected range up to the licensed maximum power and that suitable alarms and/or indications are provided. The NRC staff further finds that the detector channels of the nuclear instruments directly monitor neutron flux for reactor power level and power rate-of-change and that suitable interlocks are in place to prevent reactor startup without a sufficient neutron count rate in the core or other unsafe conditions. The NRC staff also finds that the nuclear instruments provide redundant, independent, and isolated indications and trip channels and are designed to assume a safe state on a loss of electrical power. As described previously in this SER, the NRC staff finds that the proposed log PPM and proposed Gen-2 NMP-1000 channels incorporate all of the features and functions of the existing systems, including monitoring the neutron flux and reactor power either at the same range or over a larger range than the current Gen-1 NMP-1000 and NLI-1000 systems. The NRC staff also finds that the proposed channels continue to include appropriate compensation of the CIC detectors for the gamma induced current to allow for sensitivity to neutrons even in the presence of intense high gamma radiation and yield only the measurement of neutrons. The NI channels provide information to the reactor operator, the RCS, and the RPS portions of the UMLRR I&C systems to monitor reactor conditions during normal and abnormal conditions. The UMLRR is designed such that if the power or change rate are determined to be at unsafe levels, the nuclear instruments will generate a scram signal to the RPS, which will perform its protective scram action by interrupting power to the control blade drive magnets and scramming the reactor. Each channel is capable of individually scramming the reactor based on the parameters each measures (i.e., change rate, power level, and loss of detector high voltage) and the operator has the ability to manually initiate a scram at any time. In addition, the NIs transmit signals to the RCS to generate a rod withdrawal inhibit signal, when appropriate. In this manner, the proposed systems continue to provide diverse means to protect the reactor. Both proposed systems also provide the capability for periodic testing, channel checks, and calibration to help ensure reliable operation. Therefore, the NRC staff finds that the UMLRR NI systems are acceptable and that the TFS log PPM and two Gen-2 GA NMP-1000 linear power channel upgrades requested in conjunction with license renewal are acceptable.

3.4.1.2 Process Control and Instrumentation

Section 7.4.2 of the UMLRR SAR discusses the PCI system at the UMLRR and states that the system consists of the following channels: (1) primary coolant system flow; (2) reactor core flow; (3) primary coolant inlet temperature; (4) primary coolant outlet temperature; (5) pool water temperature; and (6) pool water height. All six channels have readouts on individual digital panel meters located on the upper left side of the proposed instrumentation panel layout (as shown in SER Figure 3-9). These channels are interconnected with the scram chain (discussed in SER Section 3.4.1.3) to initiate reactor shutdown (scram), and an annunciator and alarm system (discussed in SER Section 3.6) to indicate if any monitored parameter reaches its predetermined setpoint. In addition, an isolated analog output is sent to the display screen on

the control console for the display of the respective process parameters. The console display screen also provides the user interface for on/off controls for the various UMLRR pump motors, ventilation valves, and ventilation fans. The trip relays are configured to be de-energized (fail-safe) for the trip condition. To test the proper performance of the meters and to ensure the functionality of the trip relays, each meter has an external test circuit consisting of a spring-loaded momentary toggle switch and a potentiometer.

In its response to RAI-7.3 (Ref. 79), UML described three completed upgrades/modifications to the PCI system since its last license renewal in 1985. The first change, made in 2001, replaced the process control cabinet with an industrial PCS manufactured by OPTO 22, Inc. that uses a 32-bit microprocessor for program control and communications (the process control cabinet and the PCS display are discussed further in SER Section 3.6). The second upgrade, in 2014, added flow, temperature, and pool height panel indicators to provide an additional indicator for each of these variables in the control room and to augment the upgrades made in 2001. The third upgrade, finished in 2016, replaced the control room alarm panel. This upgrade included six additional alarm indicators to provide additional information for the operator. Since all of these upgrades are related to the instrumentation panel displays and the HMI, they are discussed in more detail in SER Section 3.6.1 and evaluated in SER Section 3.6.2

Primary coolant system flow

The primary coolant system flow rate is measured using a stainless-steel orifice plate installed in the primary piping after the heat exchanger. A differential pressure transmitter provides an analog output current signal that is displayed in gallons per minute on the primary flow indicator on the control room instrumentation panel. An indication of flow is also sent to the recorder, to the PCS, and to provide relay outputs for alarm and scram. The scram and alarm relays are configured to de-energize in the trip condition (fail-safe). The trip contact for the scram is used in the scram safety chain and the trip contact for the alarm is used to actuate the annunciator.

Reactor core flow

A turbine flow meter is mounted in a two-inch diameter sampling pipe located between the discharge header beneath the core and the primary flow riser plenum. Measurement of the reactor core flow provides a diverse and redundant method of monitoring the primary coolant flow rate. A magnetic pick-up sensor above the turbine rotor produces signal pulses that are converted to logic pulse signal and displayed as a percentage of the nominal flow on the control room instrumentation panel. The scram and alarm relays are configured fail-safe similar to the primary coolant system flow rate meter.

Primary coolant temperature

The temperature measuring channels consist of temperature compensated resistance temperature detectors (RTD). There are three independent temperature channels for the primary coolant system:

- The primary inlet temperature RTD measures the temperature of the primary coolant in the piping exiting the heat exchanger. This channel is referred to as pool inlet temperature in TSs renewed 3.2.3 and 3.2.5, which are discussed and found acceptable in SER Section 2.5.3.

- The primary coolant outlet temperature (also referred to as core outlet temperature) RTD measures the temperature of primary coolant in the piping exiting the reactor core.

- The pool temperature RTD measures the temperature of the primary coolant in the pool near the surface of the water above the reactor.

Each channel has a temperature indicator that displays the respective temperature in degrees Fahrenheit on the instrumentation panel in the control room. The associated scram and alarm relays are configured to de-energize in the trip condition (fail-safe). The trip contact for the scram is in the scram safety chain and the trip contact for the alarm actuates the annunciator.

Pool water height

The pool water height above the core is measured by an ultrasonic transducer. The pool water height signal is sent to the control room instrumentation panel indicator where the height is displayed in feet above the core center line. The associated scram and alarm relays are configured to de-energize in the trip condition (fail-safe). The trip contact for the scram is used in the scram safety chain and the trip contact for the alarm is used to actuate the annunciator.

Additionally, an independent magnetic float switch having contacts connected in series to the reactor safety chain provides a diverse and redundant method of sensing pool water height. If the float drops to a predetermined level, the switch contacts will open causing a scram.

Evaluation Findings for the Process Control Instrumentation

Based on the NRC staff's review of the information in the SAR, as supplemented, the NRC staff finds that the UMLRR PCI design is similar to that of many other non-power reactors operating in the United States, such as those at RINSC, North Carolina State University, Reed College, and MURR. The NRC staff finds that the PCI provides redundant, independent and isolated indications and trip channels that are designed to fail in a safe state on a loss of electrical power. The NRC staff also finds that consistent with the guidance of NUREG-1537, Part 2, Sections 7.1 and 7.2, the PCI is designed to display important process parameters, such as coolant flow, pool temperature, and pool level during the reactor's modes of operation (startup, low-power natural convection, and high-power forced convection) to ensure that the reactor operator has sufficient information for safe operation of the reactor. Additionally, the NRC staff finds that, consistent with ANS/ANSI-15.1-2007, Sections 3.2, 3.3, and 4.0, UMLRR TSs specify the minimum equipment, operating limits for the process parameters, and surveillance tests and intervals, needed to safely and reliably operate the facility. Renewed TS 3.2.3 would specify the minimum reactor protection system scrams and their setpoints, renewed TS 3.2.5 would specify the minimum number of channels, and renewed TS 4.2 would provide the surveillances that help ensure the availability and reliable operation of the PCI to ensure that automatic protective action will be initiated in order to prevent the SL from being exceeded. Renewed TSs 3.2.3, 3.2.5, and 4.2.3, which are based on analyses in SAR Sections 4.6 and 13.2.2, as supplemented, are evaluated and found acceptable in SER Section 2.5.3.

3.4.1.3 Scram Circuit

As discussed in SAR Section 7.4.3, automatic shutdown of the reactor (scram) can be initiated by the relays and associated contacts listed in SER Table 3-2 and the electronic scram inputs from the proposed linear and log PPM channels. The electronic scram circuit includes the logic unit and two trip actuator amplifiers (TAAs). The logic unit monitors the voltage output of the

two linear power modules and the log PPM. The TAAs provide the DC current to energize the scram electromagnets to allow the control blades to be latched. Automatic shutdown (scram) of the reactor can be initiated by the various sensors and independent contacts wired in series that form the RPS safety chain scram circuit (relay scram) or by redundant and diverse logic level trips from the log PPM and each linear power nuclear instrument channel (electronic scram).

Relay scrams

The safety chain scram circuit, which is also referred to as the "safety chain, seismic trip, blade withdrawal & inhibit" circuit is a series of relays and contacts with each contact corresponding to a scram condition (Ref. 79, Appendix A). The switches listed in SER Table 3-2 (based on SAR Table 7-5 and SAR Table 7-6, as supplemented by UML's responses to RAI-14.3.33 (Ref. 71) and RAI-7.14.b.iii (Ref. 79) and its letters dated September 30, 2020 (Ref. 98), and January 30, 2021 (Ref. 99)) are connected in series to form a safety chain scram bus, which provides power to the two master scram relays also connected in series with the scram chain contacts. The master scram relays provide 120 alternating-current volts (VAC) power to the TAAs through a set of electrical contacts energized by the scram relays. If any scram chain contact opens, the scram relays are deenergized and electric power to the four control blade electromagnets from the TAAs is interrupted, causing a reactor scram (relay scram). The RPS actuating logic is such that any one sensor indication exceeding its setpoint will cause a scram and all contacts fail open on loss of power (fail-safe). SAR Sections 4.5.5 and 7.4.3.2 state that the release time for a relay scram is approximately 185-190 milliseconds (msec).

Electronic scram

The electronic scram circuit includes the logic unit and two TAAs to monitor the electronic scram inputs from the log PPM and linear channels. As discussed in SAR Section 7.4.3.2, in addition to the relay trips, the log PPM and two linear power channels provide an independent, optically-isolated, 12 Vdc logic-level signal to the logic unit. This trip signal drops to 0 Vdc when tripped (fail-safe). The logic unit has two redundant comparators for each logic input (the two linear channels and the one log PPM channel). If the input to any one of these six comparators is lost, a 12 Vdc output from the logic unit to the TAA bi-stable drops to 0 Vdc, which in turn opens contacts that directly interrupts current to the electromagnets. SAR Section 7.4.3.2 states that the release time for a logic level fast scram is approximately 5 msec.

Table 3-2 RPS Safety Chain Scrams

Scram Condition	Scram Setpoint
Key Operated Switch	Switch to Off or Test
Manual Scram Pushbuttons (Qty 9)	Pushbutton Depressed(1)
Linear Module 1 High Power Relay	1.1 MW (forced flow) or 110 kW (natural convection)
Linear Module 2 High Power Relay	1.1 MW (forced flow) or 110 kW (natural convection)
Log PPM High Power Relay	1.1 MW
Log PPM Short Period Relay	< 7 seconds
Linear Module 1 High Voltage Relay	< 700 volts
Linear Module 2 High Voltage Relay	< 700 volts
Log PPM High Voltage Relay	< 700 volts
Startup Counter High Voltage Relay	< 600 volts
Primary Coolant Flow Indicator Relay	<1450 GPM
Reactor Core Flow Indicator Relay	<85 % of nominal flow
Pool Height Float Switch	>6 inches below full (24.5 feet above core CL)
Pool Height Sensor Transmitter Relay	<24.5 feet
Pool Temperature Indicator Relay	>104°F
Core Outlet Temperature Indicator Relay	>108°F
Pool Inlet Temperature Indicator Relay	>104°F
Temperature Recorder Relay	Same as Pool, Outlet, and Inlet above
Seismic Sensor Switch	>IV Modified Mercalli Scale(2)
Primary Plenum Outlet Gate Switch	Switch Open (Forced Convection)(3)
Primary Plenum Inlet Gate Switch	Switch Open (Forced Convection)(3)
Bridge Movement Limit Switch	Switch Open (Forced and Natural Convection) (3)
Bridge Position Limit Switch	Switch Open (Forced Convection) (3)
Inlet Pipe Swivel Limit Switch	Switch Open (Forced Convection)(4)
Outlet Pipe Swivel Limit Switch	Switch Open (Forced Convection) (4)
Primary Piping Valve Limit Switches (Qty 6)	Switch Open (Forced Convection) (4)
Process Control HMI WDT Relays (Qty 2)	> 10 second
Drive Control HMI WDT Relay	> 10 second
Area Radiation Monitoring System Relay	*See SER Section 3.7*(5)
Beamport & Thermal Column Limit Switch	Switch open(5)
Airlocks and Truck Door Seal Switch	Door unsealed(5)

Notes:

(1) Manual Scram – Manual scram can be initiated at operator discretion by actuation of the manual scram push button. Emergency manual scram pushbuttons are also located in strategic locations in the reactor building and one is located outside the building in the Reactor Supervisor's office.
(2) Seismic Disturbance - Relay scram occurs when a seismic disturbance closes the seismic trip detector contact which short circuits the seismic trip relay coil.
(3) Bridge, Coolant Gate - Relay scram occurs if the bridge is moved out of position or if the riser or downcomer plenum coolant gate opens under forced convection.
(4) Primary Piping - Relay scram occurs if primary piping is out of alignment or if pool inlet/outlet valves are unseated.
(5) SAR Table 7-6 lists radiation protection scrams, which UML stated are part of the RPS safety chain circuit, but which have no function in reactor protection.

Evaluation findings for the scram circuit

The NRC staff reviewed the information in the SAR, as supplemented, and finds that the scram circuit is designed with diversity and redundancy such that any single malfunction in its components, either analog or digital, would not prevent the RPS from performing its safety function and that the scram circuit is designed to assume a fail-safe state on loss of electrical power. Renewed TSs 3.2.1(2) and 4.2.1(3), which are discussed and found acceptable in SER Section 2.2.2, require the time from initiation of a scram signal and movement of each control blade from the fully withdrawn position to 80 percent of the fully inserted position to be less than one second, and that this scram time be verified annually or whenever maintenance or a modification is performed. SAR Sections 4.5.5 and 7.4.3.2 state that the release time for a relay scram is approximately 185-190 msec. The NRC staff finds that the actual release time for scrams is less than the required TS time (one second, which includes both the release time and the time for the blade to fall) and also less than the conservative RPS response time (210 msec) used by the licensee in the Chapter 13 safety analyses. Accordingly, the NRC staff finds that the scram circuit meets the RPS guidance in Section 7.4 of NUREG-1537 that time delays and actuated equipment response should be consistent with the SAR analyses, and that this information should be included in the TSs. Therefore, the NRC staff finds that the UMLRR scram circuit is acceptable.

3.4.1.4 100 kWt Scram

As discussed above and in SER Sections 2.2 and 2.6, and also in SAR Section 7.4.5, the UMLRR may be operated up to a steady-state power level of 100 kWt when the primary pump is off and core cooling is maintained by natural convection. To enable natural convection mode, SAR Section 7.4.5 states that the operator must place the Power Level Selector Switch on the instrument panel in the 0.1 MWt position. In natural convection (0.1 MWt) mode, +24 Vdc energizes the natural convection relays, which bypasses the primary coolant flow alarm and scrams and enables the linear power channel scrams on the 100 kWt decade range. If the +24 Vdc signal is not present, the set of contacts on natural convection control relay reverts to the normally open position, deactivating the logic status inputs and enabling the scram relays and alarm indicators. This is a fail-safe response since the reactor will scram on low flow.

During the 2017 and 2020-2021 audits (Ref. 86), the NRC staff reviewed the procedure to modify the proposed GA NMP-1000 modules to enable the proper trip setpoints that correspond to the selected mode (i.e., 1 MWt or 0.1 MWt) and allow the UMLRR to operate in both forced and natural circulation modes. This modification to the proposed Gen-2 NMP-1000 is unique to the UMLRR linear power channels and is not included in the GA manuals. The NRC staff notes that the current (Gen-1) UMLRR linear power channels, which also have both forced and natural circulation modes, also required modification to enable the natural circulation mode operation.

During the 2017 audit, the NRC staff also reviewed the specific instructions that GA provided to UML on how to complete the modification. As discussed in SAR Section 7.4.5, to implement the low-power (100 kWt) scram when operating in natural convection mode, the linear power modules will utilize two spare trips designed into the trip circuitry (SER Sections 3.4.1.1 and 3.8 discuss UML's flux monitoring system in further detail). Additionally, a make-before-break contact design will ensure that switching from natural convection to forced convection does not produce an open circuit fault and the associated scram. Each of the four trips (alarm and scram for both modes) will have independent trip set points. During the 2020-2021 audit and in its letter dated September 30, 2020 (Ref. 98), UML stated that the modification to accommodate dual-mode operations will be completed by a UMLRR engineer, using the instruction provided

by GA. UML also stated during the audit and in its letter dated September 30, 2020, that information related to the 100 kWt mode modifications will be added to the Gen-2 NMP-1000 Linear Power Channel Installation Plan provided in Appendix C to UML's response to RAI-7.6 (Ref. 79).

Based on the NRC staff's review of the information in the SAR, as supplemented, and its observations during the NRC staff audits (Ref. 86), the NRC staff finds that the design of the UMLRR 100 kWt scram provides independent, fail-safe operation that will cause an automatic protective action to ensure safe operation of the UMLRR. The NRC staff finds that the 0.1 MWt mode modifications to incorporate the 100 kWt scram will be properly tested and verified operational since renewed TS 4.0, item B., which is discussed and found acceptable in SER Section 6.4, requires that appropriate surveillances be conducted after replacement, repair, or modification before LCO-required equipment is considered operable. In addition, renewed TS 4.2.3(3), which is discussed and found acceptable in SER Section 2.5.3, requires that applicable surveillances (i.e., calibrations, which include a channel test) be performed for the RPS channels. This is in addition to the required operational tests and checks of the RPS required by renewed TSs 4.2.3(1) and 4.2.3(2), which are also discussed and found acceptable in SER Section 2.5.3, for the operating mode (i.e., natural or forced convection mode) planned for each day of reactor operation. Based on the above, the NRC staff finds that the modifications to incorporate the required scram into the proposed Gen-2 NMP-1000 instruments for the 0.1 MWt mode are consistent with the NUREG-1537 guidance to ensure that natural-convection flow can dissipate decay heat (Section 5.2) and that automatic protective action will help ensure that the peak fuel temperature does not reach an undesirable value (Chapter 13). Therefore, the NRC staff finds that the modifications are acceptable.

3.4.2 Evaluation Findings on the Reactor Protection System

Guidance for evaluation of the nuclear instrument design criteria is discussed in various sections of NUREG-1537, Part 2. Section 3.1, "Design Criteria," Section 7.3, "Reactor Control System," and Section 7.4, "Reactor Protection System," of NUREG-1537, Part 2, apply to the UMLRR nuclear instrument design criteria. The NRC staff used this guidance to evaluate the design of the UMLRR RPS including an in-depth evaluation by the NRC staff of UML's proposal to upgrade the nuclear instruments with a new TFS log PPM and two new GA digital NMP-1000 linear power channels, as discussed in SER Sections 3.4.1 and 3.8.

The NRC staff performed two regulatory audits (in 2017 and 2020-2021) to confirm information in the licensee's application, including the information on the proposed upgrades to the nuclear instruments submitted as part of the application. During the 2017 audit (Ref. 86), the NRC staff observed performance of the UMLRR reactor checkout procedure to confirm the ability of the TS-required checks and surveillances associated with the UMLRR RPS I&C systems to verify RPS operability.

Based on the information provided in the SAR, as supplemented, and reviewed by the NRC staff, and also based on its audit observations, the NRC staff finds that the RPS, including the proposed nuclear instrumentation upgrades, meets the design acceptance criteria in NUREG-1537, Part 2, Sections 3.1, 7.3, and 7.4, such that the system has reliable protection capability for all UMLRR operating modes and conditions, is fail-safe, and has sufficient redundancy, diversity, and independence to help ensure that a single failure or malfunction cannot prevent the RPS from performing its intended safety function, as analyzed in the SAR, as supplemented. Additionally, based on the above findings and conclusions, in conjunction

with the findings and conclusions referenced in this SER, and its audit observations, the NRC staff concludes the following:

- UML analyzed the design and operating principles of the RPS for the UMLRR. The protection channels and protective responses are sufficient to ensure that no SL, LSSS, or RPS-related LCO discussed and analyzed in the SAR will be exceeded.

- The design reasonably ensures that the design bases can be achieved, the system is of high-quality, and the system can be readily tested and maintained in the designed operating condition.

- The RPS design is sufficient to provide for all isolation and independence from other reactor subsystems required by SAR analyses to ensure that malfunctions or failures caused by the other systems do not adversely impact RPS operation.

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

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

- The RPS incorporates provision for manual initiation, which provides a simple and direct means for the reactor operator to immediately shut down the reactor if needed.

- The RPS I&C systems at the UMLRR, including the proposed NI upgrades, are well designed and maintained. Redundancy in the important ranges of power measurements by nuclear instrumentation is ensured by overlapping ranges of the log and linear power channels and the startup channel. All of the important nuclear process variables are monitored and displayed at the reactor console.

3.5 Engineered Safety Features Actuation Systems

In SAR Section 7.5, UML stated that the credible design basis events analyzed for the UMLRR in SAR Chapter 13, as supplemented (and discussed and found acceptable in Chapter 5 of this SER), show that ESFs are not required. The NRC staff finds that UML's analyses, supported by the NRC staff's confirmatory analyses (also discussed in SER Chapter 5), demonstrate reasonable assurance that no credible accident postulated by UML would result in significant fuel damage or require an active emergency core cooling system, and that no unacceptable radiation doses would result from any postulated credible accident, even without consequence mitigation by ESFs designed to prevent or mitigate the consequences of fuel damage, or gain control of any radioactive material released by accidents.

Although not an ESF required to ensure that no unacceptable radiation doses would result from any postulated credible accident, the actuation of confinement (manual or automatic), including as part of a local radiation emergency alarm (LREA) or general radiation emergency alarm (GREA), is discussed in detail in SER Sections 3.7.1, 4.1.4, and 6.3.4.

3.6 Control Console and Instrumentation Panel

3.6.1 System Description of the Control Console and Instrumentation Panel

Section 7.2.2.4 of the UMLRR SAR states that the design basis of the control console display instruments is that the reactor operator be provided adequate and reliable information at a central location from which to observe reactor performance and adjust operating parameters to varying requirements, as needed. The UMLRR controls and instrumentation associated with the flux monitoring systems, primary coolant systems, and auxiliary systems allow the operator to safely monitor and operate the reactor and take appropriate actions as necessary. User interfaces in the control room also provide the operator with the ability to start and stop equipment throughout the facility. SAR Sections 7.1.4 and 7.6 describe the I&C grouping and SAR Section 7.1.6 describes the HMI. UML stated that the instruments for monitoring parameters are both diverse and redundant. An annunciator alarm panel is provided to alert the operator to an abnormal condition and to facilitate both the diagnosis of the abnormal condition in the facility and the selection of the appropriate response to the condition.

As is shown in SER Figure 3-7 (reproduced from SAR Figure 7-7), the I&C systems are assembled in three cabinets: (1) the control console; (2) the instrumentation panel; and (3) the area radiation monitoring cabinet. All three enclosures are positioned in the reactor control room, located on the third floor of the reactor building.

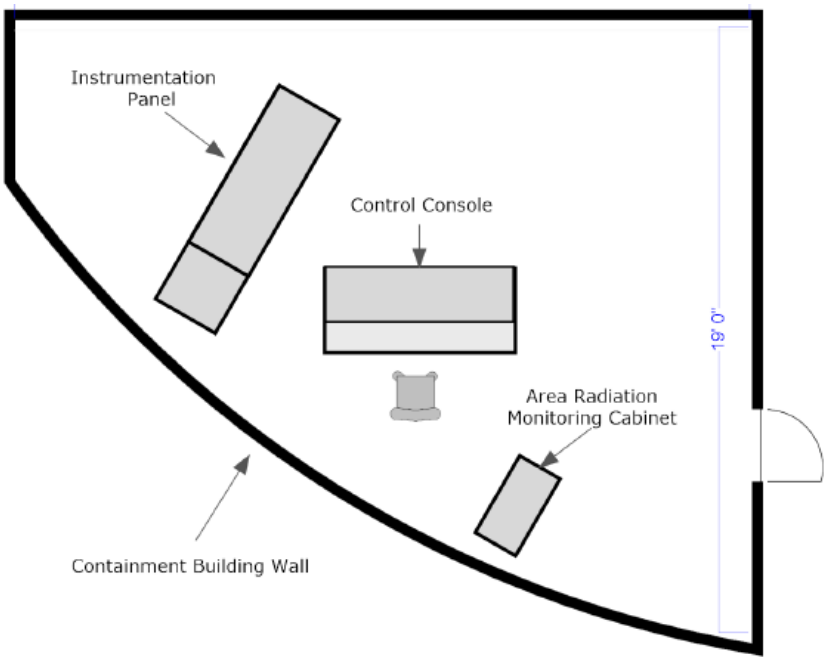

Figure 3-7 Layout for Control Console and Instrument Panel

In its RAI-7.3 response (Ref. 79), UML described seven implemented upgrades/modifications to the control console, instrumentation panel, and ARMS cabinet since its last license renewal in 1985, that were implemented under 10 CFR 50.59 without prior NRC approval. These modifications are listed in SER Table 3-3. The control console modifications (process control cabinet and drives control system) and instrumentation panel modifications (chart recorder, panel indicators, and annunciator panel) are discussed in further detail in SER Sections 3.6.1.1 and 3.6.1.2, respectively. The ARMS modifications are discussed in SER Section 3.7.

Table 3-3 Changes related to the Control Console, Instrumentation Panel, and ARMS Cabinet for which a 10 CFR 50.59 Evaluation or Screen was Performed

Year	Description	Type	Implemented
2016	Control Room Annunciator Panel Replacement	Screen	Yes
2014	Addition of Panel Indicators (Flow, Temperature, Pool Height)	Screen	Yes
2012	Chart Recorder Replacement	Full	Yes
2003	Drives Control System	Full	Yes
2001	Upgrade of UMLRR Process Control Cabinet	Full	Yes
2000	Area Radiation Monitoring System Upgrade (*from 1997*)	Full	Yes
1997	Flux monitoring and Radiation Monitoring System Upgrades	Full	Partial[a]

[a] - The flux monitoring upgrade associated with a 10 CFR 50.59 review conducted by UML in 1997 was implemented following the review, but the implementation of the Radiation Monitoring System upgrade associated with this same review was deferred until after completion of a new 10 CFR 50.59 review in 2000.

3.6.1.1 Control Console

Section 7.6 of the UMLRR SAR describes the UMLRR control console as a desk-type panel with two main TSDs. A simplified layout of the UMLRR control console is shown in SER Figure 3-8. The center of the console contains the TSD for the PCI system, which is designated the PCS display. The PCS display provides indicators for power level, various temperatures, flow rates, pressures, and water purity and on/off controls for various motors, valves, and fans. The process control instrumentation is discussed in SER Section 3.4.1.2. The TSD on the right side of the console is for the drives control display system and is designated as the DCS display. The DCS display provides displays (e.g., position indicators) and controls for the RCS, including the drive mechanisms associated with the control blades, regulating rod, and start-up counter. The RCS is described in SER Section 3.3.1.

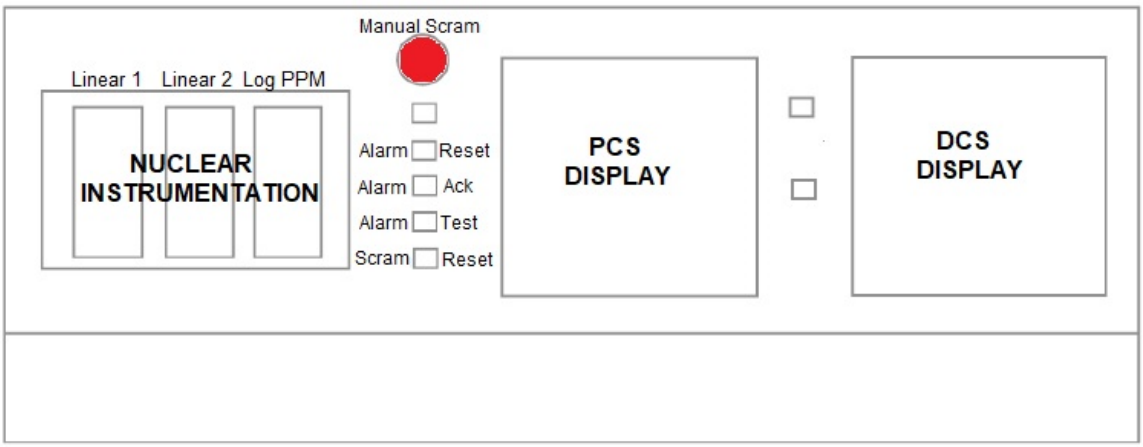

Figure 3-8 UMLRR Control Console

The left panel of the console contains the nuclear instrument indicators for the proposed linear power monitoring channels and the proposed log PPM channel described in SER Section 3.4.1.1. The panel also has alarm indicators and control switches, including a manual scram button to allow the operator to initiate a rector shutdown (scram). An alarm condition is annunciated by a buzzer and lighted indicator on the alarm panel (the alarm panel is on the instrumentation panel to the left of the control console). The operator uses the Alarm Reset switch to clear the indicator when the alarm condition is cleared. The Alarm Acknowledge switch is used by the operator to turn off the buzzer that annunciates an alarm condition. The momentary Alarm Test switch turns on all the alarm indicators and buzzer to periodically validate their functionality. The Scram Reset switch energizes the scram relays and scram circuit chain if all scram conditions are cleared.

As discussed in UML's responses to RAI-7.2 and RAI-7.3 (Ref. 79), a total of three upgrades/modifications that involved the addition of HMI displays were made to the UMLRR since the previous license renewal in 1985. The new HMI displays were added for the PCS, DCS, and ARMS. Two of these three upgrades/modifications, which are listed in SER Table 3-3, were made to the control console HMI. The 2001 and 2003 HMI changes, made by UML under the provisions of 10 CFR 50.59, are discussed in the following paragraphs under the PCS display and DCS display headers. The HMI changes to the ARMS, completed in 2000, are discussed in SER Section 3.7.1. All three upgrades (PCS, DCS, and ARMS) followed the same engineering and implementation process described below for the PCS. SAR Section 7.6.1.1 states that the PCS, DCS, and ARMS each have separate hardware and software and operate independently.

Process Controls System Display

As discussed in UML's responses to RAI-7.2 and RAI-7.3 (Ref. 79), in 2001, the reactor control room process control cabinet was removed and UML installed a computer-based PCS with a TSD. The upgraded PCS was designed and installed using the OPTO 22, Inc. hardware and integrated software package associated with the display. UML stated that this OPTO 22 technology is the same digital control technology applied to its 2000 upgrade of the ARMS control panel (discussed in SER Section 3.7). The PCS display, described in SAR Sections 7.6.1.3 and 7.6.1.4, consists of a 19-inch TSD (and manual keyboard terminal) to provide the HMI for process control, process monitoring, data acquisition, and for manual control of process functions. The associated computer contains the programmed display configuration developed for the PCS application to provide reactor power, coolant systems, and ventilation systems display screens. SAR Section 7.6.1.3 states that the PCS, DCS, and ARMS display configuration settings are password protected to prevent unauthorized changes. During the 2020-2021 audit, as discussed in the NRC staff's audit report dated December 17, 2020 (Ref. 86), UML also stated that the embedded PCS programming from the OPTO 22 manufacturer is not alterable by the operator.

SAR Section 7.6.1.4 describes the PCS screen displays as follows: the "reactor power" screen displays the count rate from the start-up count rate channel, the linear percent power from the N-16 detector, and a calorimetric determination of reactor power, which is based upon the primary coolant flow rate and core differential temperature. The "reactor power" screen will also display the percent power from proposed linear power channels 1 and 2 and the logarithmic power and period outputs from the proposed log PPM (replacing the current linear and log power and period indications). The "coolant systems" screen provides indicators for coolant flow rates, coolant temperature, differential temperature across the core, conductivity, the heat exchanger temperatures and differential pressures, pool water height, make up water control

status, pump room sump level, and temperature and humidity outside containment. The "coolant systems" screen also allows the operator to operate, and to view the controls status of, the primary coolant pumps and the secondary cooling tower fans with corresponding indicators that display the status of the associated motors and controls. The "ventilation system" screen displays information about the reactor building ventilation and isolation system and allows the operator to control the operation of the ventilation fans and associated air duct valves. The PCS computer also allows the operator to perform data trending (charts) and archiving, according to Appendix B of UML's response to RAI-7.3.b (Ref. 79).

As previously stated, the software development package used to upgrade the PCS HMI was manufactured by OPTO 22, Inc. OPTO 22 develops commercial off-the-shelf software products, which are used in conjunction with its hardware products for industrial automation, remote monitoring, data acquisition, and process control.[2] The software is Microsoft® Windows-based and consists of two main development programs. One program provides for the development of graphical, flow chart-based programming for interaction with I/O hardware. The other program provides for the development of a graphical operator interface. In its response to RAI-7.2 (Ref. 79), UML stated that the contractor used both programs to develop the UMLRR HMI applications for the ARMS, DCS, and PCS.

In Appendix B of its response to RAI-7.3.b, UML described the PCS hardware as consisting of the power supplies, the I/O controllers, an ethernet switch, and two personal computers. Two types of I/O modules are in use by UML: digital I/O modules are used for two-stage devices (e.g., motor control and valve position) and analog I/O modules are used for ranging inputs (e.g., temperature and flow). The optically-isolated I/O modules interface with the microprocessor, which communicates the information to/from the main controller. The controller, running the OPTO 22 software, interprets the information from the microprocessor using the vendor's programmed set of instructions to perform control actions and relay information to the HMI computer display. The HMI computers display the information received from the controller and provide a touchscreen and a keyboard terminal for manual control of process functions.

UML described the HMI development and implementation process for the ARMS, PCS, and DCS in its response to RAI-7.2 (Ref. 79). In its response to RAI-7.2.b, UML stated that its staff worked closely with the contractors during each phase of the HMI development for the ARMS, PCS, and DCS. This included site visits by the contractors, follow-up communications as needed, and providing descriptions of the existing systems, augmented with additional detail provided by operations staff. The existing hardware interface and SAR descriptions were used to communicate the desired functional requirements to the contractors for designing the hardware and software interface. Similar activities occurred during the actual design of the hardware and software configuration by the contractors. The design process was iterative, again with UMLRR staff working closely with the contractors, to ensure that each application would perform as expected to meet the functionality of the existing systems. In its response to RAI-7.2.c, UML stated that the HMI systems were physically integrated into the existing systems by both the contractors and the UMLRR operations staff. Acceptance checklists were developed to test the operability of the displays and controls. In its response to RAI-7.2.d, UML stated that V&V processes ensured that the HMIs performed as intended using specifically developed checklists for each system (the PCS, DCS, and ARMS). In addition, UML stated that prior to operation of the reactor with each of the systems, the applicable TS surveillances (e.g.,

[2] "OPTO 22 Programmable Industrial Controllers." Opto 22, Inc., https://www.opto22.com/

calibrations) were performed, in addition to the required pre-critical operational tests and checks of the RCS and RPS.

During the 2017 audit, the NRC staff observed the performance of the pre-startup reactor checkout procedure (Ref. 86). During the audit, UML demonstrated and explained that, in addition to performing the daily channel checks and tests required by TS 4.2.2, the operator performs administratively required checks that compare and record readings from both the individual equipment indicators on the instrumentation panel and on the HMI displays. During the audit, UML also stated that the PCS has two computers (primary and backup) since some ventilation valves and fans would have to be locally controlled if the primary PCS computer were to malfunction. The backup computer is a password protected laptop running an independent version of the same control system software that is completely redundant to the primary computer and is installed and used only if needed (i.e., failure of the primary computer).

As described in Sections 7.4.3.2 and 7.6.2.1 of the SAR, as supplemented and updated by UML's response to RAI-7.14.b (Ref. 79), if communication is lost between a controller and the associated HMI computer display, or between a controller and the I/O modules within an I/O rack on the instrumentation panel, within 10 seconds a command is given for the WDT scram relay to open. The PCS WDT scram (within 10 seconds following communication loss) is required by renewed TS 3.2.3, Table 3.2.3-1, item 9, which is discussed and found acceptable in SER Section 2.5.3. As stated in SAR 7.6.1.4 and the RAI 7.3b response, the PCS does not replace any safety system functionality, but it does provide additional diversity and redundancy by monitoring independent and isolated outputs from the process variables. If a setpoint is reached, the PCS controller sends a signal to the I/O rack trip relay which deenergizes to open a contact in the safety chain scram circuit described in SER Section 3.4.1.3. Additionally, TS-required primary system parameters (i.e., channel indications) are also displayed on the instrument panel immediately adjacent to the operator console as discussed in SER Section 3.6.1.2.

Drives Control System Display

The DCS display, described in SAR Sections 7.6.1.3, and 7.6.1.5 through 7.6.1.10, consists of a 19-inch flat panel TSD that displays the control blade, regulating rod, and startup counter position and status indicators and provides the operator interface for selecting, withdrawing, and inserting the drives for the control blades, regulating rod, and the startup counter. The SAR describes the following specific DCS status indicators and additional operator control interface functions:

a) Control blade and regulating rod drive indicators that provide:

    (1) Status of control blade engagement to the control drive magnet
    (2) Digital position indication of the blade/rod in inches and as a bar graph
    (3) Full-in/full-out status indicators
    (4) Indication of a drive motor malfunction
    (5) Inhibit indication if withdrawal of rods is prevented by an interlock

b) Regulating rod mode indicator that provides:

   (1) Indication of manual/auto mode
   (2) Power level digital indicator and two bar graphs. One bar graph provides actual reactor power and the setpoint bar graph provides the programmed reactor power control setpoint
   (3) Auto increase/decrease controls for adjusting the programmed setpoint in increments of approximately two percent

c) Startup counter drive indicators that provide:

   (1) Digital position indication of the startup counter in inches and as a bar graph
   (2) Full-in/full-out status indication
   (3) Indication of a drive motor malfunction

d) "Auto blade position schedule" control to allow the operator to automatically withdraw individual control rods to a preset position (one at a time) in a semi-automatic mode, or a "reset" control to drive all six drives (for four control blades, the regulating rod, and the startup counter) to full in and reset the digital position indicators to zero.

In its response to RAI-7.2 (Ref. 79), UML stated that the DCS is built upon the same OPTO 22 hardware and software used in the ARMS and PCS HMI upgrades. The DCS hardware upgrades and specific software development was also completed by an engineering consultant. Similar to the process described under the PCS upgrade, the UMLRR staff worked closely with the engineering consultant during the design, installation, and testing phases to verify that the system performed to the required design specifications to fulfill the system SAR and TS requirements. As discussed in SAR Section 7.6.1.3, similar to the PCS and ARMS, the DCS configuration settings are password protected to prevent unauthorized changes. Additionally, during the 2020-2021 audit, as discussed in the NRC staff's audit report dated December 17, 2020 (Ref. 86), UML stated that the DCS embedded programming by the OPTO 22 manufacturer is not alterable by the operator.

As described in Section 7.4.3.2 and 7.6.2.1 of the SAR, as supplemented and updated by UML's response to RAI-7.14.b (Ref. 79), similar to the PCS, the DCS also has a WDT that monitors communication between the I/O modules, the controller, and display. In SAR Section 7.6.2.1, UML stated that a failure of the DCS that results in lost control or positional information of the drives will result in a reactor scram. The operator can visually verify that the reactor is shut down by observation of the power measuring instruments and, if necessary, visually verifying that the control blades are in the reactor core. The DCS WDT scram (within 10 seconds following communication loss) is required by renewed TS 3.2.3, Table 3.2.3-1, item 10, which is discussed and found acceptable in SER Section 2.5.3.

3.6.1.2 Instrumentation Panel

As is shown in SER Figure 3-7, the instrumentation panel is located in the control room adjacent to the control console and angled toward the operator to facilitate viewing of and access to the instrument readouts and controls. The proposed layout of the UMLRR instrumentation panel is shown in SER Figure 3-9, which is reproduced from SAR Figure 7-8. As discussed in SAR Section 7.3.3, the key operated switch (also referred to as the master key switch or the master control switch) on the instrumentation panel prevents unauthorized reactor operation because

the RCS may not be operated unless the key-lock is unlocked and switched to “test” or “on” (see also SER Section 2.2.2).

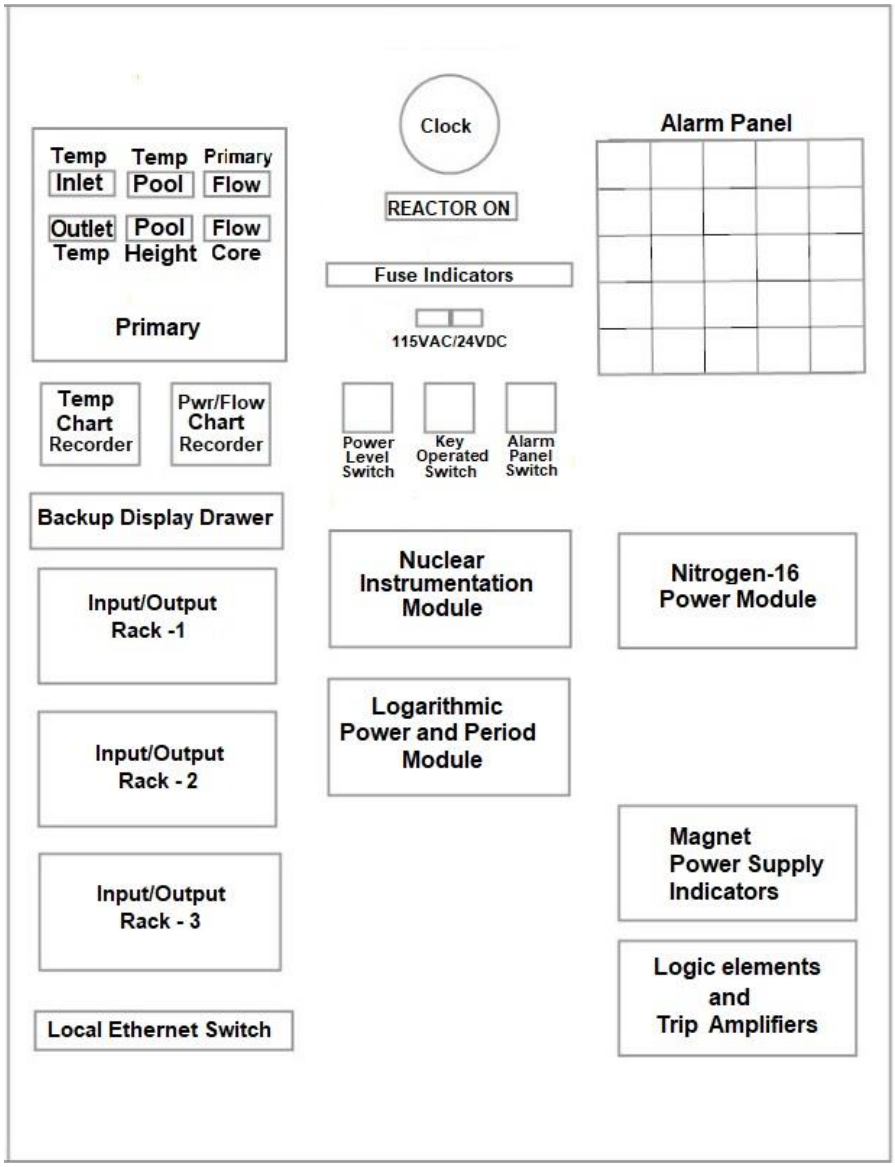

Figure 3-9 Proposed Instrumentation Panel Layout

In its response to RAI-7.3.b (Ref. 79), including the Appendix B to its response, and its response to RAI-7.10.a (Ref. 79), UML provided detail on three upgrades/modifications to the instrumentation panel (the first three changes listed in SER Table 3-3 above). The 2012 modification, described below, replaced the chart recorders. The 2014 modification added panel meters for the primary coolant variables of flow and temperature, and another panel meter for pool water level height, to the control room instrumentation panel to add redundancy and diversity to the existing display indications. The 2016 replacement of the control room annunciator panel added 6 additional indicators. The original alarm panel had 17 individual alarm indicators (2 by 3 inches each), each illuminated by two 120 VAC miniature incandescent light bulbs. In its response to RAI-7.10.a.ii, UML clarified that the six additional alarm indicators were added to provide additional information for the operator and that an unused indicator

labeled as “High Radiation” on the original panel that was never wired to an alarm output was eliminated to avoid operator confusion. Accordingly, UML stated that the replacement panel has 22 alarm indicators (2.25 by 2.75 inches each), each illuminated by multiple LEDs. Although the replacement panel has a 5 by 5 configuration with 25 indicator spaces, there are two unused indicators and the 25th space is occupied by a local alarm acknowledge switch that performs the identical function to the alarm acknowledge switch on the control console described in SER Section 3.6.1.1.

Chart Recorders

As described in SAR Section 7.1.6 and SAR Table 7-7, two chart recorders are mounted in the instrumentation panel (see SER Figure 3-9). The temperature chart recorder records primary coolant inlet temperature (i.e., pool inlet temperature; see SER Sections 2.5.3 and 3.4.1), outlet temperature (i.e., core outlet temperature), and pool temperature (i.e., bulk pool temperature). The power and flow chart recorder records reactor power and primary coolant flow rate. In its renewal application, UML proposed that the recorder inputs be modified to record indicated power of the proposed GA Gen-2 NMP-1000 Linear Power Module 1 (as stated in SAR Table 7-7) and the proposed TFS log PPM (as stated in UML’s updated SAR Section 7.4.1.2 (Ref. 73)). Renewed TS 3.2.5, which is discussed and found acceptable in SER Section 2.5.3, only requires that one of the two linear power channels be operating and, therefore, Linear Power Module 1 is not strictly required to be operating. However, the log PPM is always required to be operating by TS 3.2.5 when the reactor is operating; therefore, the output from at least one power monitoring channel will be recorded by the power and flow chart recorder while the reactor is operating.

In its response to RAI-7.3 (Ref. 79), UML described its 10 CFR 50.59 evaluation for the 2012 upgrades to the chart recorders to replace the analog chart recorders with hybrid (analog and digital) chart recorders. In the evaluation, provided as part of the RAI-7.3 response, UML stated that the replacement chart recorders conform to Underwriter’s Laboratory safety standards and were obtained from an ISO-9001 certified manufacturer. Additionally, the signal transmitter that was required for the old temperature chart recorder was eliminated in the replacement recorder. UML stated that the accuracy and resolution for the replacement (Omega) chart recorders is equivalent to or better than the previous (Leeds-Northrop) chart recorders. As required by TS 4.2.6, any reactor safety system instrument replacement must undergo a channel check prior to installation and a calibration before routine operation after the installation. In its RAI-7.3 response, UML also stated that the replacement recorders were subjected to factory QA testing and site integration calibration and testing. The recorders were bench tested and checked for proper response and then installed and the calibration checked to be accurate over the intended range of the recorder using the existing UMLRR detectors.

SAR Table 7-7 and UML’s 10 CFR 50.59 evaluation (Ref. 79) for the chart recorders indicate that both chart recorders have relay outputs for alarm and scram functions. SAR Table 7-5 specifies a temperature recorder relay that provides scrams for bulk pool, pool inlet, and core outlet temperature, which are redundant to the separate TS-required indicator relay scrams (see SER Table 3-2). Although SAR Table 7-7 describes a scram relay output for the power and flow chart recorder, SAR Table 7-5 does not appear to specify any recorder relay scrams for the power and flow recorder in the scram chain. During the 2020-2021 audit (Ref. 86), the NRC staff verified that the scram chain (diagram provided in Appendix A to UML’s response to RAI-7.1.b (Ref. 79)) included contacts for recorder low flow and the three recorder high temperature scrams. The recorder scrams are diverse and redundant relay scram trips to the scram relays that provide TS-required scram functions. The setpoints for the temperature

recorder scrams are the same as those for the TS-required indicator relay scrams, as shown in SER Table 3-2 in Section 3.4.1.

3.6.2 Evaluation Findings on the Control Console and Instrumentation Panel

Based on the information provided in the SAR, as supplemented, the NRC staff evaluated the design of the UMLRR control console and instrumentation panel displays, in accordance with the guidance provided by Section 7.6 of NUREG-1537, Part 2. The NRC staff compared the general arrangement and types of controls and displays provided by the control console and proposed instrumentation panel (which includes the proposed log PPM and linear power channels) to those at similar research reactors and finds that the designs are comparable. The NRC staff also finds that the UMLRR controls and instruments are designed and located to promote ease and efficiency in the performance of operations necessary for the safe control of the reactor and provide an acceptable means for operator interaction with the controls and indicators. The NRC staff also reviewed the UML annual reports from 2010-2011 through 2019-2020 (Refs. 10.f through 10.o), and identified no issues with respect to the reliable operation of the control console or instrument panel for any of the control console or display systems. Additionally, UML reported that all surveillance test results were found to be within specified limits and that surveillance inspections revealed no abnormalities.

UML did not propose any changes to the licensed power level or operating modes. As discussed in SER Sections 3.2, 3.3, and 3.4, the NRC staff continues to find that the designed range of operation of each control and indicator remains sufficient for the expected range of variation of monitored variables under the analyzed conditions of UMLRR operation. The NRC staff observed the control console and instrumentation panel during the 2017 audit (Ref. 86) and confirmed that both the control console, and the instrumentation panel, including proposed changes for the new NIs, will provide the reactor operator with the types of information, monitored variable ranges, and controls necessary to facilitate reliable and safe operation of the reactor. The NRC staff also finds that the UMLRR has diverse and redundant displays co-located within the control room to provide defense against a single failure and to prevent a loss of any information vital to safe operation of the reactor.

The NRC staff reviewed the OPTO 22 software QA plan (Ref. 98.d) and finds that the plan describes the managerial and administrative controls used to control development of the display software throughout its life cycle in a traceable, planned, and orderly manner, which includes factory testing, documentation, and quality reviews associated with each software development phase. During the 2020-2021 audit (Ref. 86), the NRC staff reviewed the data sheets for major components used in the OPTO upgrade, including the vendor data sheets for the main controller and the analog and digital I/O modules (Ref. 98.h), and confirmed that the components are designed and fabricated to equipment qualifications suitable for operation within a research reactor facility, consistent with the OPTO 22 software QA plan. Further, the NRC staff finds that parameters important to the operation of the control console and display instruments are adequately managed and protected from inappropriate configuration or unauthorized access. The NRC staff also finds that the design of the control console and instrumentation panels address potential display and control system vulnerabilities to software common mode failure and that any such failures, including loss of power, would scram the reactor.

Based on the NRC staff's review of the information in the SAR, as supplemented, and its observations of the actual layout of the indicator panel in the control room, and the UML staff's demonstration of the panel operation during the performance of the procedure for reactor

checkout during the 2017 audit (Ref. 86), the NRC staff finds that the replacement alarm indicator panel continues to provide the appropriate visible and audible annunciation of an alarm condition to the operator since the replacement alarm indicators have the same functionality and physical location on the instrumentation panel in the control room, but are slightly larger to improve readability. Additionally, the replacement alarm indicator panel changed the incandescent light indicators to a more reliable LED technology. Accordingly, the NRC staff finds the replacement annunciator panel acceptable.

Based on its review of the information UML provided as discussed above and in SER Section 3.6.1, and based on its audit observations, the NRC staff finds that the control console and proposed instrumentation panel meet the human factors guidance identified in NUREG-1537, Part 1, and the control console and display instrument design acceptance criteria of NUREG-1537, Part 2, by including readily available indication of UMLRR parameters and safety variables. The NRC staff's audit observations confirmed that the indications and locations of these parameters are designed in accordance with design acceptance criteria of NUREG-1537, Part 2, such that the operator is able to continuously view important reactor parameters, easily and with minimal operator action. Accordingly, the NRC staff finds that the control console and proposed instrumentation panel are acceptable. Specifically, the NRC staff concludes as follows:

- UML showed that all nuclear and process parameters important to safe and effective operation of the UMLRR will be displayed at the control console. The required nuclear and process parameters on the console and on the proposed instrumentation panel are easily understood and readily observable by an operator positioned at the reactor controls. The control console design and operator interface, including grouping, orientation, and location, are sufficient to promote safe reactor operation.

- The display instrumentation and the controls in the control console and/or proposed instrumentation panel are 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 alarms and indications on the control console and the annunciator alarm panel give assurance of the operability of systems important to adequate and safe reactor operation.

- The systems that digitally process information to the operator on status of the reactor have readily available redundant displays on the proposed instrumentation panel co-located in the reactor control room and the PCS and DCS HMI systems have built-in protection to automatically shut down the reactor in the unlikely event that a console display should fail.

- The locking system on the proposed instrumentation panel reasonably ensures that the reactor facility will not be operated by unauthorized personnel and facilitates secured shutdown, when required.

3.7 Area Radiation Monitoring System

3.7.1 System Description of the Area Radiation Monitoring System

The SAR, as supplemented and updated by UML's letters dated September 30, 2020 (Ref. 98), and January 30, 2021 (Ref. 99), describes the ARMS as a system which provides information on radiological conditions for both routine operations and possible emergency situations associated with the reactor. SAR Section 7.7.2 states that the design basis of the ARMS is to continuously monitor gamma and beta radiation levels at strategic locations in the UMLRR facility, as well as gaseous and particulate emissions from the facility, to ensure that the reactor operator and other personnel (e.g., other UML staff or experimenters) have adequate and reliable information to evaluate radiation levels in the building and air effluent leaving the building.

As discussed in SAR Sections 7.2.2.5, 7.2.3, and 7.7, and SER Section 4.1.4, the ARMS consists of multiple and various radiation detectors and includes a cabinet in the control room that houses the remote readout indicators for the detectors and visual alarm indicators for high alarms, alert (i.e., "warning" or "elevated") alarms, and detector failure. Functions are provided for various additional audible and visible alarms, for building isolation, and reactor scram (see discussion below and in SER Sections 4.1.4 and 6.3.4).

SAR Section 7.2.2.3 describes how certain combinations of ARMS radiation detectors reaching their setpoints will cause automatic reactor confinement building isolation. However, the overall confinement system is fail-safe in that loss of electrical power to the building, or the loss of either the main intake fan motor or main exhaust fan motor, will cause building isolation independent of radiation levels within the building. Furthermore, the reactor operator can manually initiate building isolation, as discussed in SER Sections 4.1.4 and 6.3.4.

As discussed in SER Section 4.1.4, the ARMS is interconnected with the scram chain and can provide a scram on high radiation levels. However, this is not a TS-required scram function. It is also not an automatic scram; it requires operator action to generate an ARMS scram after high radiation levels on certain combinations of monitors are detected.

Renewed TS 3.6.1(1), which is discussed and found acceptable in SER Section 4.1.4, specifies a minimum requirement for five radiation monitors: a stack gaseous monitor; a stack particulate monitor; a constant air monitor on the reactor pool level; an area monitor over the pool; and an area monitor on the experimental level. Per renewed TS 3.6.1(2), which is discussed and found acceptable in SER Section 4.1.4, an additional monitor may be required when a gamma irradiation facility is in use. However, the ARMS includes many radiation monitors beyond the minimum TS-required monitors. The ARMS, as described in the SAR, as supplemented, currently consists of nine individual ratemeters, each connected to multiple radiation monitors, as listed in SAR Table 7-10 and SAR Table 7-11, plus additional monitors that are connected directly to the Area Radiation Monitoring Computer Data Acquisition System (ARM CDAS).

Renewed TSs 4.6(1), 4.6(2), and 4.6(3), which are discussed and found acceptable in SER Section 4.1.4, provide SRs for TS-required radiation monitors.

In its license renewal application, as supplemented, UML proposed to add an additional ratemeter connected to the stack gas and particulate monitors, as discussed in UML's letters dated September 30, 2020, and January 30, 2021. UML would make this change during the implementation period of a renewed license. This change is discussed in detail and found

acceptable in SER Section 4.1.4. UML stated that the ARMS, including changes proposed in conjunction with license renewal, provides redundancy for radiation monitoring.

Area Radiation Monitoring Computer Data Acquisition System

The ARM CDAS is described in SAR Sections 7.6.1, 7.7.2, 7.7.3, and 7.7.4, and UML's response to RAI-7.3 (Ref. 79). Outputs from the constant air monitors (CAMs) and stack exhaust air monitors are hardwired to the ARMS chassis (cabinet) in the control room (which is located next to the reactor control panel as illustrated in SAR Figure 7-7) and transmitted directly to the ARM CDAS controller. Signals from the nine currently existing ratemeters (both the six control room panel-mounted and three remote ratemeters) are acquired using optically isolated I/O modules. This data is then processed by a stand-alone computer connected to the ARM CDAS controller.

The ARM CDAS computer within the chassis (cabinet) is programmed with software logic that determines alarms and actions for multiple combinations of radiation monitors reaching their setpoints (i.e., potential-LREA (P-LREA) and potential-GREA (P-GREA); see SER Sections 4.1.4 and 6.3.4). A computer monitor also provides displays for outputs, alarms, and test functions for the various radiation monitors. These are in addition to local and/or ratemeter output, alarm, and other functions of certain radiation monitors.

During the audit, as documented in the audit report dated March 4, 2021 (Ref. 100), UML stated that the proposed connection of a new (seventh) control room ratemeter to the stack monitor would not replace the existing separate connection from the stack monitor to the ARM CDAS. As discussed later in this SER section, this additional ratemeter would provide additional independent control room readouts and alarms for the stack gas and particulate channels.

Constant Air Monitors

Section 7.7.2.2 of the SAR states that monitoring for airborne contamination inside the facility during operations is performed by a CAM on the third floor of the reactor building. The CAM monitoring is required by renewed TS 3.6.1(1), specification b., which is discussed and found acceptable in SER Section 4.1.4. An additional CAM, not required by TSs, is located on the first floor. The CAM uses dual passivated implanted planar silicone detectors. A second detector in the CAM is mounted directly downstream of the first detector such that it only senses the background radiation of ambient gamma and cosmic rays. The signal of the second detector is electronically subtracted from the first to ensure that the CAM is only indicating actual gamma, beta, and alpha radiation that is trapped in the sensor filter. The CAM output is sent directly to the ARM CDAS for both indication and alarm logic related to the P-LREA and P-GREA. Local audible and visual alarms also alert the operator to abnormal air flow through the CAM equipment. UML stated that CAM setpoints are not readily modified, and can only be changed by directly connecting a stand-alone computer to the CAM. The UMLRR CAMs are also discussed in SER Section 4.1.4.

Stack Exhaust Air Monitor

Section 7.7.2.3 of the SAR describes the stack exhaust air monitor. The stack exhaust air monitor continuously monitors the airborne radioactivity of air exhausted from the facility and is a two-channel radiation detection system (to measure both radioactive gas and particulate concentrations). The stack exhaust monitor operation is discussed further in SER Section 4.1.4.

Area Monitors

SAR Sections 7.7.2.1, 7.7.3, and 11.1.4.2 describe fixed area radiation monitors that are part of the ARMS and that are also located throughout the reactor building. These monitors are primarily designed to detect gamma radiation fields. The area monitors are discussed further in SER Section 4.1.4.

Previous Upgrades/Modifications to the ARMS

In its response to RAI-7.3 (Ref. 79), UML described three upgrades/modifications related to the ARMS since the last license renewal in 1985.

UML implemented the first change following a full 10 CFR 50.59 review in 2000. As discussed in UML's RAI-7.3 response and SER Sections 1.8 and 3.6.1, this change included some upgrades of detectors and readout modules (ratemeters) that were based on a 10 CFR 50.59 review that was conducted in 1997, but not implemented at that time. The 2000 change upgraded the original 1970s detectors and readout modules and replaced the ARMS cabinet, which had hardware and relay logic that generated indication of a P-LREA or P-GREA, with a new chassis (cabinet), analog and digital I/O modules, and a computer/software-based logic to emulate the previous relay logic. The change also included the addition of additional detector readouts on a computer display above the new ARMS chassis. A second remote readout display was also added by the first-floor airlock. In its responses to RAI-7.2 and RAI-7.3, UML stated that the upgraded ARMS integrated software/hardware systems, including the ARM CDAS controller, computer (emulating the previous relay logic), and HMI display were installed by an independent contractor using hardware manufactured for industrial process control, process monitoring, and data acquisition, and an integrated suite of industrial control and automation software provided by OPTO 22, Inc. In its response to RAI-7.3, UML stated that although the P-LREA and P-GREA relay logic was replaced, the relays for ventilation isolation and reactor scram were not changed and manual initiation of either a LREA or GREA system isolation by the operator was unaffected. By design, operator actuation of the ALARM pushbutton on the chassis/cabinet (to initiate either an LREA or GREA) will scram the reactor. However, the upgrade did add a feature for automatic ventilation closure for a P-GREA. The upgrade also added new indicators for P-LREA, LREA, P-GREA, and GREA on the computer display, in addition to the P-LREA and P-GREA indicators on the cabinet/chassis, which remained unchanged. (See SER Sections 4.1.4 and 6.3.4 for detailed discussion of P-LREA, LREA, P-GREA, and GREA.) UML completed the OPTO 22 upgrades for the ARMS under the provisions of 10 CFR 50.59, and the upgrades are similar to those discussed for the PCS and DCS in SER Section 3.6.1. The NRC staff reviewed the OPTO 22 upgrades to the HMI, operator displays, and controls as discussed in SER Section 3.6.1.1 and 3.6.2. As stated in UML's responses to RAI-7.2.c, RAI-7.2.d, and RAI-7.3.b, a V&V process to ensure that the ARMS HMI, as well as other computer hardware and software, including the programmed logic for combinations of radiation monitors to produce a P-LREA or L-GREA, performed as intended was completed using specifically developed checklists for the ARMS. In addition, UML performed the applicable TS 4.6 surveillances prior to operation of the reactor with the upgraded ARMS systems.

The second change, which UML screened for 10 CFR 50.59 applicability in 2012, as discussed in SER Section 1.8, was a replacement of the stack effluent monitor. According to the UMLRR 2012-2013 annual report (Ref. 10.h), the stack effluent monitor was replaced by a comparable monitor.

The third change related to the ARMS was UML's replacement of the process control cabinet with the computerized PCS, which is discussed in detail in SER Section 3.6.1.1. The NRC staff notes that this change is related to the ARMS since it affected the ventilation system operation and the reactor building isolation valve closure initiation and time, and the reactor building isolation and ventilation operation could be affected by signals from the ARMS. As discussed in SER Section 6.3.4, however, operators can also manually secure ventilation, isolate the building, and initiate the emergency exhaust system, as needed, independent of the ARMS.

ARMS Upgrades Proposed in UML's License Renewal Application

In its letter dated January 30, 2021 (Ref. 99), UML proposed to make changes to its radiation monitoring system, which it stated it would implement during the implementation period of a renewed license (see SER Section 1.10). UML stated that the changes would consist of (1) adding an additional control room ratemeter for the stack gas and particulate monitors that would provide additional readouts and individual audible alarms in the control room and (2) configuring the existing control room ratemeters for TS 3.6.1-required area monitors to provide for individual audible alarms in the control room. The existing ratemeters already provide control room readouts independent of the ARMS HMI, and visual alarms. These changes are discussed in detail and found to be acceptable in SER Section 4.1.4.

3.7.2 Evaluation Findings on the Area Radiation Monitoring System

The NRC staff evaluated the design of the UMLRR ARMS in accordance with Section 7.7 of NUREG-1537, Part 2, based on the information provided in the UMLRR SAR, as supplemented, and summarized above in SER Section 3.7.1. The NRC staff also considered the results of its review of general radiation monitoring and surveying at the UMLRR, documented in SER Section 4.1.4, and the observations and information reviewed during the 2017 and 2020-2021 audits.

In SER Section 4.1.4, with respect to general radiation monitoring and surveying at the UMLRR, the NRC staff finds that the types and placement of radiation monitors used, and the radiation surveys conducted, at the UMLRR are appropriate for the types and intensities of radiation likely to be encountered at the UMLRR. In SER Section 4.1.4, the NRC staff finds that renewed TSs 3.6.1 and 4.6 related to radiation monitoring systems are acceptable to ensure that the systems will be operable and will perform their designed functions, and that the surveillances (tests and calibrations) performed on radiation monitors will help ensure that the monitors function correctly and accurately.

As also discussed in SER Section 4.1.4, the NRC staff finds that UML's proposed changes to its ARMS are acceptable because they add redundancy to the ARMS, eliminate the ARM CDAS (including the ARMS computer and HMI) as a single point of failure for the ARMS, and provide control room audible alarms for individual TS-required monitors to help ensure that operators will be appropriately alerted to high radiation levels and that TS 3.6.1(1) can be met. The NRC staff also finds that there is reasonable assurance that appropriate types of radiation in significant intensities will be detected, and that the radiation monitoring and surveying at the UMLRR, including the proposed changes to the ARMS (discussed in SER Section 4.1.4 and also summarized above in SER Section 3.7.1), will help ensure compliance with 10 CFR 20.1501(a) and 10 CFR 20.1501(c).

With respect specifically to the I&C systems for radiation monitoring, including the ratemeters and ARM CDAS that are part of the ARMS chassis (cabinet), the NRC staff finds that the ARMS

I&C systems provide radiation monitor readouts and alarms, are interconnected with the scram chain (discussed in SER Section 3.4.1.3) to initiate a reactor scram on high radiation levels, and are connected to relays to provide a building isolation function. The NRC staff also finds that the ARM CDAS HMI display of radiation levels and its automatic scram function provide additional redundancy and diversity to TS-required equipment and functions (e.g., ratemeter readouts and alarms and the operators' ability to manually scram the reactor or isolate the reactor building). However, the NRC staff notes that these additional functions involving the ARM CDAS are not required by TSs, with the exception of the third floor CAM's readout on the ARMS HMI display as discussed previously in SER Section 2.5.3.

The NRC staff finds that the components of the ARMS I&C system (i.e., the ARM CDAS components) that have scram or building isolation functions are designed to fail-safe because they cannot fail or operate in a mode that would prevent the RPS from performing its safety function or prevent safe reactor shutdown, nor fail or operate in a way that would preclude the building isolation function. The NRC staff also finds that, even if the ARMS failed to perform a scram or building isolation function as designed, the operator could still manually isolate the building and/or scram the reactor as needed.

UML made changes to the digital/computerized components of the ARMS I&C systems, as well as other components (new ratemeters and monitors) under its 10 CFR 50.59 process during the ARMS upgrade in 2000. The NRC staff reviewed the UMLRR annual reports from 2005 through 2020 (Refs. 10.a to 10.o) and found no indication of issues or unreliable operation of the ARMS during that period after being upgraded in 2000. In addition, as discussed in SER Section 4.1.4, the NRC staff also reviewed the UMLRR inspection reports for the years 2006 through 2019 (Ref. 11) and found no significant issues related to radiation monitoring and surveying at the UMLRR.

Renewed TSs 3.6.1 and 4.6, which are discussed and found acceptable in SER Section 4.1.4, require a minimum set of radiation monitors (including alarms and readouts) and that UML verify that these monitors are operating prior to the commencement of activities for which they are required. During the 2017 audit (Ref. 86), the NRC staff observed UML's performance of the UMLRR pre-startup reactor checkout procedure and confirmed that UML performed TS-required checks and surveillances associated with the UMLRR I&C systems. These included checks to verify ARMS operability.

Based on its review of the information UML provided as discussed in SER Sections 3.7.1 and 4.1.4, and based on its audit observations, the NRC staff finds that the ARMS (including changes proposed in conjunction with license renewal) meets the design acceptance criteria in Section 7.7 of NUREG-1537, Part 2, by providing appropriate information on facility radiation levels and alerting the operator to elevated radiation levels as appropriate. Accordingly, the NRC staff finds that the UMLRR ARMS I&C systems are acceptable. Furthermore, in addition to its conclusions in SER Section 4.1.4, the NRC staff concludes that UML described the designs and operating principles of the I&C for the radiation detectors and monitors in the SAR, as supplemented, and demonstrated that those designs and principles are applicable to the anticipated sources of radiation at the UMLRR.

3.8 Upgrade Process for the Flux Monitoring Systems

This section of the SER documents the NRC staff review and evaluation of the design and development process of UML's proposed flux monitoring upgrades to the Gen-2 GA NMP-1000 and the TFS log PPM systems for the UMLRR. Specifically, the NRC staff evaluated QA, CM,

V&V testing, and access control using the design acceptance criteria guidance in Chapter 7 of NUREG-1537, Part 2, including acceptance criteria from the guidance and industry standards referenced in Chapter 7 of NUREG-1537, and also referenced in this SER. The technical evaluation of the design of these systems as part of the RPS to allow reliable and safe operation and shutdown of the reactor during normal reactor operation and abnormal conditions is discussed in Section 3.4.1.1 of this SER.

In the renewal application, UML proposed replacing its current flux monitoring systems to improve operation and reliability of the UMLRR. For this evaluation, the NRC staff reviewed the design and development processes established by the licensee and its I&C systems vendors, GA and TFS, who developed the proposed flux monitoring systems.

3.8.1 Quality Assurance

A robust QA program and managerial and administrative controls are necessary to help ensure that the system can perform its required functions. Section 50.34(b)(6)(ii) of 10 CFR requires a description in the SAR of managerial and administrative controls to be used to help ensure safe operation. Section 7.2.1 of NUREG-1537, Part 1, recommends that all systems and components of the I&C systems should be designed, constructed, and tested to quality standards commensurate with the safety importance of the functions to be performed. Section 12.9 of NUREG-1537, Part 1, states that the applicant should consider the guidance in NRC Regulatory Guide (RG) 2.5, "Quality Assurance Program Requirements for Research and Test Reactors" (Ref. 95), and ANSI/ANS-15.8-1976 (Ref. 96) in developing QA programs for non-power reactors. The general criteria for establishing and executing a QA program for the testing, modification, and maintenance of research reactors in ANSI/ANS-15.8, which is endorsed by RG 2.5, provide an acceptable method for complying with the quality requirements of 10 CFR 50.34. However, ANSI/ANS-15.8 recognizes that the described controls are integral to the management of a facility and that it is not necessary to establish a separate QA program for a facility upgrade such as an upgrade to the I&C systems. The design criteria for the UMLRR (SAR Section 3.1) state, in part, that failure of the RPS does not have the potential for causing offsite exposure. Section 3.1.5 of the UMLRR SAR states:

> Structures, systems, and components important to safety were designed, fabricated, constructed, and tested to the original design specifications and associated codes and standards. All design and construction work was monitored by the contractors to assure that the specifications incorporated appropriate standards and that the design and construction was in accordance with these specifications. Modifications to the facility have been made in accordance with existing standards and requirements.

During the 2017 and 2020-2021 audits, the NRC staff verified that UML followed its QA program for UMLRR components procured from vendors. In its responses to RAI-7.9.a, RAI-7.11, and RAI-7.12 (Ref. 79), and its letter dated September 30, 2020 (Ref. 98.g), UML provided evidence of UML/Vendor collaboration to ensure that the NMP-1000 met the requirements for the UMLRR. GA developed the NMP-1000 following its "NMP-1000 System Requirements Specification" (Ref. 102.b), "Software Quality Assurance Plan" (Ref. 103.d), "Software Requirements Specification" (Ref. 102.a), and "Software Configuration Management Plan" (Ref. 103.a). Additionally, the GA documentation states that the NMP-1000 was developed and tested in accordance with ASME NQA-1-2000, "Quality Assurance requirements for Nuclear Facility Applications" (Ref. 89). UML also provided a document from TFS, the log PPM vendor, specific to UML. The TFS certificate of compliance (Ref. 98.e) certifies that the log PPM

equipment purchased by UML complies with all requirements of the initial purchase order and applicable specifications. Additionally, final inspections and tests to verify compliance were performed according to Revision 22 of the TFS Quality Assurance Program, dated October 11, 2018 (Ref. 98.f). In its response to RAI-7.1.a (Ref. 79), UML stated that the TFS Quality Assurance Program Revision 22 (Section III) documents that the TFS Nuclear Systems and Services products meet the objectives of 10 CFR Part 50, Appendix B, and ASME NQA-1-2000 and ASME NQA-1-2008, 1a-2009 (Ref. 108). ASME NQA-1 is generally approved by incorporation by reference in 10 CFR 50.55a(a)(1)(v) for power reactor quality programs, but NRC regulations do not require the use of NQA-1 at research reactors, such as UMLRR. The NRC staff finds that these vendor qualifications for both the Gen-2 NMP-1000 and the log PPM meet or exceed the applicable non-power reactor guidance of RG 2.5, ANSI/ANS-15.8, and NUREG-1537 with regards to the proposed flux monitoring equipment.

The NRC staff compared the referenced versions of NQA-1 used by the vendors to the 10 CFR Part 50, Appendix B, regulations and to the guidance in RG 2.5 and ANSI/ANS-15.8-1995. The NRC staff finds that the NQA-1 criteria cited by the vendors provide an acceptable method for complying with the quality requirements of 10 CFR 50.34(a)(7) and (b)(6)(ii) because they meet or exceed the guidance of RG 2.5 and ANSI/ANS-15.8-1995. Based on the information provided and reviewed, the NRC staff finds that the quality process and design process followed by both GA and TFS are adequate for the design, manufacture, and testing of the proposed NMP-1000 and log PPM, respectively.

The NRC staff finds that UML demonstrated that it followed a QA program for all procured components proposed for the UMLRR flux monitoring systems. Additionally, based on the information provided, the NRC staff finds that UML's QA program complies with the program requirements of 10 CFR 50.34(a)(7) and that the QA provisions meet the 10 CFR 50.34(b)(6)(ii) requirement that managerial and administrative controls be used to assure safe operation. The NRC staff also finds that the UML QA program is commensurate with the guidance of RG 2.5 and ANSI/ANS-15.8 for QA in the design, construction, and operation of the UMLRR. Accordingly, the NRC staff finds the UML QA program to upgrade the flux monitoring equipment to be acceptable.

3.8.2 Configuration Management

ANSI/ANS-15.8, which is referenced in Section 12.9 of NUREG-1537 and endorsed by RG 2.5, provides the following general guidance on configuration control: (1) that equipment that requires configuration control is identified; (2) that management is responsible for establishing and maintaining proper configuration and should authorize any changes to safety-related items; and (3) that all configuration changes to safety-related items should be documented. ANSI/ANS-15.8, Section 2.3.6, specifically states that changes to verified designs (including configuration controlled settings) must be documented, justified, and subject to design control measures commensurate with those applied to the original design. Additionally, the guidance in NUREG-1537, Part 2, Section 7.4, states that, "[h]ardware and software for computerized systems should meet the guidelines of [Institute of Electrical and Electronics Engineers (IEEE)] 7-4.3.2-1993, "IEEE Standard Criteria for Programmable Digital Devices in Safety Systems of Nuclear Power Generating Stations" (Ref. 107).

During the 2017 audit, the NRC staff noted that the calibration and checkout procedures described settings and parameters that should be under configuration control (i.e., under a CM program) and asked UML to provide documentation that a configuration management program appropriately traces changes to safety systems.

In its response to RAI-7.7 (Ref. 79), UML stated that the UMLRR CM program incorporates the following requirements:

- Any change in either the hardware or software configuration of the UMLRR TS-required equipment will be treated as a facility design change and subject to a 10 CFR 50.59 review as required by TS 6.2.3. The 10 CFR 50.59 review is performed and documented in accordance with an approved UMLRR administrative procedure. Additionally, as required by TS 4.0.B, the applicable surveillance testing will be conducted after replacement, repair, or modification before the equipment is considerable operable and returned to service.

- Operations personnel will receive training on the hardware or software change and such training is documented.

- A log of the current configuration of TS-required equipment software will be documented and maintained in the control room. The TS-required equipment software version will be checked and verified against the configuration log during the TS calibration for the equipment. Finally, if there is a discrepancy, the equipment will be taken out of service and the discrepancy immediately reported to the Radiation Laboratory Director (Level 2 authority per TS 6.1.1).

During the 2020-2021 audit, the NRC staff reviewed UML's software CM procedure (Control Doc. No. AP-7-00 (Ref. 98.j)) to confirm the UML program, described in UML's RAI-7.7 response, for checking, documenting, and reviewing changes to instrumentation software and trip point settings for the NMP-1000 linear power channels. Additionally, during the 2020-2021 audit, the NRC staff reviewed "Standing Order #5" (Ref. 98.a) from the UMLRR Chief Reactor Operator's book, which documents the current UML setpoints. As stated in the standing order, if these settings need to be adjusted to some other value, permission from the Reactor Supervisor or Chief Reactor Operator is required.

In its response to RAI 7-7 (Ref. 79), UML identified that configuration changes for the flux monitoring hardware and also the NMP-1000 software/firmware changes are not readily accomplished (by UML staff) and normally would be performed by the manufacturer. During the 2017 and 2020-2021 audits, the NRC staff reviewed the software CM plan, QA plan, and software development plan. The NRC staff's audit observations (from its document review) confirmed UML's RAI-7.9.b response (Ref. 79) that states that the UMLRR staff will perform calibrations of the NMP-1000 and simple repairs where practical, and where repairs cannot be made, or if the unit will not calibrate or perform as intended, the unit would be sent back to manufacturer.

During the 2017 audit, the NRC staff requested additional documentation specific to software requirements, development, QA, CM, and testing for the NMP-1000s. During the 2020-2021 audit, this request was expanded by the NRC staff to include the TFS log PPM based on UML's letter dated April 10, 2019, proposing to install the TFS log PPM (Ref. 73). However, UML stated during the 2020-2021 audit and in its RAI-7.5.b response (Ref. 79) that the TFS log PPM is entirely analog, as specified in Section 2.0, "Generic Requirements," in the procurement specifications for the log PPM (Ref. 98.i). The NRC staff review of the log PPM users' manual (Ref. 98.b) confirmed that the TFS log PPM is composed entirely of analog circuitry and, with the exception of LCD displays, the log PPM does not have digital signal processing or software.

During the 2017 and 2020-2021 audits, UML provided documentation that the software CM methods being used at GA (Ref. 103.a) ensure that software CM establishes a baseline version of the software configuration which serves as the basis for further development. After a baseline has been established, changes can only be performed through a formal change request process. UML's CM procedure (Ref. 98.j) requires that any change in software configuration be reviewed and evaluated as a design change. The UML CM procedure, and UML's response to RAI-7.7, state that UML will verify that the proper software version is checked and verified against the configuration log during the TS 4.2.3(3)-required calibration for the equipment. Additionally, in its RAI-7.7 response, UML stated that, if the manufacturer should neglect to communicate a change was made during service or repair, the surveillance required by TS 4.0.B and the UMLRR CM program would disclose that the software/firmware version number has changed, triggering a UML staff review. In its RAI-7.7 response, UML also stated that verification of the NMP-1000 software version will be checked and verified against the configuration log during the TS calibration for the equipment.

Based on the information in the SAR, as supplemented, and its audit observations, the NRC staff finds that the design and development of the proposed flux monitoring upgrades uses configuration control consistent with the ANSI/ANS-15.8 guidance. Specifically, the NRC staff finds that the UMLRR procedures: (1) require UMLRR management to be responsible for establishing and maintaining proper configuration and authorizing any changes to safety-related items and (2) that all configuration changes to safety-related items are documented. Additionally, the NRC staff finds that TSs 6.2 and 6.4 will help ensure that, before placing equipment into operation, a system is properly calibrated or checked, as appropriate, and that any deficiencies in the equipment or the current configuration of the system are documented and corrected.

The NRC staff also finds that changes to the proposed Gen-2 NMP-1000 system software are appropriately traced from point of origin to implementation and that the licensee has a program to ensure that the correct version of the software is installed. The NRC staff also finds that the software configuration for the NMP-1000 is formally controlled according to the GA Software Configuration Management Plan (Ref. 103.a) and by the applicable UML CM procedures consistent with the guidance of IEEE Standard 7-4.3.2 to help ensure that any software modifications, including firmware modifications, during the design process, and after acceptance of the software for use, will be made to the appropriate version and revision of the software.

Therefore, the NRC staff concludes that the UML CM of the proposed flux monitoring equipment meets the acceptance criteria for CM in ANSI/ANS-15.8 to identify items requiring configuration control and to document design changes (including configuration controlled setpoints associated with the proposed NMP-1000 and log PPM channels), and that design changes are subject to management review and control.

3.8.3 Verification and Validation Testing

The guidance in NUREG-1537, Part 1, Section 7.2.2, "Design-Basis Requirements," states that software for computerized systems should meet the guidelines of ANSI/ANS-10.4-1987, "Guidelines for the Verification and Validation of Scientific and Engineering Computer Programs for the Nuclear Industry" (Ref. 97), to provide adequate confidence that the requirements are correctly implemented and to minimize the potential for deficiencies that may result from the software programs in the I&C systems. Section 9 of ANSI/ANS-10.4 further recommends that the test results for the V&V activities be documented and reported as specified in the V&V plan

and, if the findings necessitate any retesting or revision of the test report, the updated test results should be verified again before final program acceptance.

During the 2017 audit and in its response to RAI-7.7 (Ref. 79), UML indicated that any modifications to the proposed Gen-2 NMP-1000 software are normally only made by GA. Thus, the NRC staff's software review related to the NMP-1000 for the UMLRR is based on the software development processes, software verification processes, and testing processes by GA for the NMP-1000. UML noted in its RAI-7.5.a response (Ref. 79) that GA provided UML with several proprietary documents to support the design and development process for the proposed Gen-2 NMP-1000 (Ref. 102.a, Ref. 102.b, Ref. 102.c, Ref. 102.d, and Ref. 102.e) and proprietary and redacted versions of the software development plan (Ref. 103.e):

a. General Atomics, "NMP-1000 Software Requirements Specification," Drawing No. T9S900D941-SRS, Revision A, dated January 24, 2014. (non-public)

b. General Atomics, "NMP-1000 System Requirements Specification," Drawing No. T9S900D940-SYR, Revision A. (non-public)

c. General Atomics, "NMP-1000 Failure Modes and Effects Analysis," Drawing No. T9S900D980-FME, Revision A. (non-public)

d. General Atomics, "NMP-1000 Traceability Matrix," Document No. T3401000-TRM. (non-public)

e. General Atomics, "NetBurner-MOD54415 Ethernet Core Module Validation Summary Report," Document No. 20130207001-RPT, Revision 2. (non-public)

f. General Atomics, "TRIGA INL Channels Software Development Plan," Drawing No. T9S900D970-SWP-2, Revision A. (redacted version publicly available)

During the 2017 audit, the NRC staff noted that some of these documents (c. through f. above) were specifically prepared for installation of the Gen-2 NMP-1000 at the Idaho National Laboratory as part of the Neutron Radiography Reactor TRIGA instrumentation replacement project. However, UML provided correspondence from GA (Ref. 98.g) confirming that the Gen-2, NMP-1000 documentation and correspondence submitted previously by GA to the NRC (Ref. 102 and Ref. 103.e) are all applicable to the UMLRR. During the 2017 audit, the NRC staff noted that unused code was detected and subsequently removed by GA as noted in Section 4.2.1.2 of GA's "NMP-1000 Failure Modes and Effects Analysis" (Ref 102.c), and the NRC staff requested that UML verify and document the removal of this unused code from subsequent NMP-1000 units. By letter dated September 30, 2020, UML provided evidence of an electronic communication dated July 14, 2017, in which a GA representative confirmed that the unused code was removed from all product releases, including the Gen-2 NMP-1000 modules procured by UML (Ref. 98.g).

The GA documentation describes GA-performed factory acceptance tests of the linear channels delivered to UML. As discussed above, GA provided documentation for the NMP-1000 Acceptance Test Procedure (Ref. 102.e) and the NMP-1000 requirements Traceability Matrix (Ref. 102.d) that traces safety system requirements from design implementation to implementation and vendor testing. Also, per the GA QA Program manual (Ref. 103.d), personnel conducting software testing functions report to the GA Director of Quality.

Additionally, QA personnel are responsible for witnessing software testing and certifying the quality of deliverable software (Ref. 103.d).

UML stated that it will also perform site acceptance tests on each Gen-2 NMP-1000. In Appendix C to its response to RAI-7.6 (Ref. 79), UML provided its draft "Gen-2 NMP-1000 Linear Power Channel Installation Plan" as an example of its required installation and testing of the Gen-2 NMP-1000 linear channel. In Appendix C to its response to RAI-7.6 (Ref. 79), UML also provided draft check and calibration documents for the proposed Gen-2 NMP-1000 linear channel. Renewed TSs 3.2.3 and 3.2.5 specify that at least one of the two NMP-1000 channels be operating to provide a scram function, and also to ensure that the reactor operator has sufficient information for safe operation of the reactor. TSs 3.2.3 and 3.2.5 are discussed and found acceptable in SER Section 2.5.3. Operability of the NMP-1000 linear channels is checked prior to any reactor operations under the surveillance requirements of renewed TS 4.2.3. TS 4.2.3 is also discussed and found acceptable in SER Section 2.5.3.

UML provided the following documents associated with the TFS log PPM for reference during the NRC staff 2020-2021 audit: (1) "Neutron Flux Monitoring Systems Instruction Manual: 1126 for UMASS Lowell" (Ref. 98.b; copyright and proprietary) and (2) "Quality Assurance Program Manual," Revision 22 (Ref. 98.f; copyright), in which TFS states its Nuclear Systems and Services products, including the log PPM, meet the objectives of 10 CFR Part 50, Appendix B, and ASME NQA-1-2000 (Ref. 89) and ASME NQA-1-2008, 1a-2009 (Ref. 108). UML also provided a Certificate of Conformance (Ref. 98.e), including a Configuration Record and Test Reports, for the log PPM. UML stated that the TFS test report noted in UML's response to RAI-7.4.d has an extensive acceptance procedure and test results from TFS. Additionally, Chapter 5 of the TFS instruction manual (Ref. 98.b) provides surveillance and calibration procedures, which UML stated in its response to RAI-7.6 that it is using to establish the UMLRR test and calibration procedures for the log PPM. Additionally, the TFS QA program manual (Ref. 98.f, pp. 5 and 6) discusses how employees performing quality activities (e.g., test technicians) must be sufficiently independent and cannot be responsible for performance of the work being verified.

In its RAI-7.6 response, UML stated that the UMLRR TSs require Reactor Safety Subcommittee review of proposed changes to the facility systems or equipment, procedures, and operations, including the reactor check out procedure (procedure No. RO-7) verifying operability of the systems prior to operations. During the 2017 audit (Ref. 86) and in its RAI-7.14 response, UML stated that the proposed TS-required verifications include confirming that surveillance test and self-test features address failure detection, that appropriate self-test features (e.g., monitoring memory and memory reference integrity, using WDTs or processors, monitoring communication channels, monitoring central processing unit status, and checking data integrity) are operable and that appropriate actions occur upon failure detection (related operator alarm and reactor scram).

Based on the information in the SAR, as supplemented, and its audit observations, the NRC staff finds that the UML V&V activities, including those conducted by the flux monitoring equipment vendors, are consistent with the guidance in ANSI/ANS-10.4 to have well defined systems requirements expressed in a formal V&V document that describes a development methodology to guide the production of the software and includes comprehensive testing procedures with provisions for independence of the V&V team from the software developers. Additionally, the NRC staff finds that these V&V activities for flux monitoring systems meet the intent of IEEE 7-4.3.2 (Ref. 107) to help ensure that the requirements are complete and correct and that the final system complies with the specified system requirements. Accordingly, the

NRC staff finds that the V&V activities conducted by UML and its vendors are consistent with the guidance in Section 7.2.2 of NUREG-1537, Part 1, because they help verify that the software will not cause unintended effects under some combinations of circumstances or conditions, or some malfunctions. Therefore, the flux monitoring V&V is acceptable.

3.8.4 Access control

This section documents the NRC staff review to confirm that the access control features of the upgraded flux monitoring I&C equipment, including the hardware, software, firmware, and interfaces, are adequate to protect the UMLRR systems from unauthorized access. Specific to digital devices in safety systems, the guidance in NUREG-1537, Part 2, Section 7.4, states that hardware and software for computerized systems should meet the guidelines of IEEE 7-4.3.2-1993 (Ref. 107) to control access to safety system equipment.

Section 7.2.1 of the UMLRR SAR states that the design criteria for access control in the I&C systems includes the use of physical provisions to prevent unauthorized use of the RCS. SAR Section 7.6.1 states that the reactor control room serves as a centralized management point for monitoring and interfacing with the reactor controls and instruments. The three I&C cabinets, (1) the control console, (2) the instrumentation cabinet, and (3) the radiation monitoring cabinet are all located in the control room. Access to the control room is via a door with access controls. Unescorted access to the facility and any facility controls is limited to personnel verified trustworthy by UML's designated, NRC-approved reviewing official per the regulations in 10 CFR 73.57. Personnel access to the confinement doors leading to the control room is controlled by the Reactor Supervisor, using biometrics and access cards as described in UML SAR Section 11.1.5.4. Additionally, as discussed in SER Section 4.1.5, unescorted access to the UMLRR is only given to trained and approved individuals.

During the 2017 audit, the NRC staff observed measures to ensure that access to the reactor controls is limited to authorized personnel, specifically, the physical locations of the equipment and the physical access controls that the licensee has incorporated into the UMLRR facility. In addition to limiting physical access, the licensee has two equipment-level access restrictions to prevent unauthorized access to the reactor controls—key locks and password control.

Section 7.3.3 of the SAR describes the key-locked master control switch located on the instrumentation panel that prevents unauthorized operation. In order to energize the control blade electromagnets, the master control switch must be unlocked and turned to the "On" position to power the control blade electromagnets. Additionally, the licensee has physical key locks on the ratemeters (e.g., the ARMS ratemeters, as described in SAR Section 7.7.2.1) that prevent any setpoint changes to the channel. UML stated that the key-lock reasonably ensures that an inadvertent or unmonitored adjustment of the trip point is highly unlikely.

UML's password controls are discussed in SAR Sections 7.4.2 and 7.6.1.3 and in UML's procedures, which include password protection for configuration settings and embedded programming by the manufacturer that is not alterable by the operator. During the 2017 audit, UML stated that facility procedures require password management (to include having passwords that are strong and that are changed when people are no longer on staff), key management, and chain of custody. In its RAI-7.6 response, UML provided an example, in its CP 2-04, "Linear Power Channel Check and Calibration," procedure, of required administrative controls to prevent/limit unauthorized access through supervisory notification for equipment changes. This procedure requires the Chief Reactor Operator or Reactor Supervisor to perform or supervise changes to NMP-1000 calibration parameters.

During the 2017 audit, the NRC staff visually confirmed that the flux monitoring systems are closed systems that do not connect to any external network Additionally, in its response to RAI-7.9.a, UML stated that the maintenance port on the NMP-1000 will be disabled and that there is no firmware/software that is modified by UML staff. As discussed in SER Section 3.4.1.1, UML also stated during the 2020-2021 audit that its current (Gen-1) processes and controls for controlling setpoints will not change for the Gen-2 NMP-1000 and that these processes and controls will also incorporate the proposed TFS log PPM.

Based on its review of the SAR, as supplemented, its audit observations, and its evaluation in SER Sections 3.4.1.1 and 3.4.2, the NRC staff finds that UML provides suitable access control, by the use of physical provisions to prevent unauthorized use of the reactor systems. The NRC staff also finds that potential I&C access control and cyber security vulnerabilities (physical and electronic) are adequately addressed for the I&C safety systems and software, including the upgraded flux monitoring systems, and that the administrative/management controls prevent/limit unintended and unauthorized physical and electronic access to the I&C systems. Accordingly, the NRC staff finds that the UMLRR I&C access controls are acceptable.

3.8.5 Conclusion for Flux Monitoring Systems Upgrade process

The NRC staff reviewed the design bases and criteria for the Gen-2 NMP-1000 and TFS log PPM flux monitoring systems using the design acceptance criteria identified in NUREG-1537, Part 2, including the industry standards referenced by Chapter 7 of NUREG-1537.

The NRC staff requested that UML provide information for the flux monitoring systems consistent with the guidance in NUREG-1537, Section 7.2, to include accuracy, reliability, adequacy, and timeliness of I&C system action, trip setpoint drift, quality of components, redundancy, independence, and single failure criteria with functional and environmental design information that compares the system design requirements and describes how they are met by the proposed flux monitoring components of the I&C system. In its response to RAI-7.1 (Ref. 79), UML stated that (as opposed to using a graded approach) it selected new equipment for the UMLRR that is in wide use at other similar research reactors. For example, UML procured a wide range logarithmic power/period instrument (model TR-10) from TFS. In its response to RAI-7.1.a (Ref. 79), UML stated that TFS indicated that the TR-10 is installed and used at several NRC-licensed non-power reactors, including: MURR, RINSC, North Carolina State University, Ohio State University, Pennsylvania State University, Texas A&M, Oregon State University, Reed College, Missouri University of Science and Technology, and University of California-Irvine. Similarly, UML stated that the proposed replacement NMP-1000 is a second-generation version of the NMP-1000 currently in use. The replacement NMP-1000 was purchased to the same performance specification as that which is currently installed and in use at the UMLRR.

In its SAR, as supplemented, UML provided a description of how the proposed upgrades for the flux monitoring systems, as part of the RPS, meet the applicable design criteria guidance of NUREG-1537, Sections 7.2, 7.3, and 7.4. Based on its review of the application, its audit observations, and its evaluation in SER Sections 3.4.1.1 and 3.4.2, the NRC staff finds that the design of the NMP-1000 and the log PPM (flux monitoring systems) follows the design acceptance criteria of diverse, redundant, fail-safe, and independent systems to help ensure operation of the reactor without exceeding the SL established in the TSs. Additionally, the NRC staff finds that UML demonstrated that (1) the QA, CM, and acceptance testing provisions of the proposed flux monitoring system designs are appropriate to validate that the upgrades are

reliable systems, built with high quality components and (2) the flux monitoring systems are designed with the capability for periodic checks, tests, and calibration to help ensure that the RPS continues to be capable of performing its intended function when needed to safely shut down the reactor when a scram condition exists. The NRC staff also finds that the flux monitoring systems will help ensure that the operators will receive adequate indications of the status of the reactor.

Specifically, the NRC staff concludes:

- Design criteria supporting the design bases are specified for the portions of the flux monitoring systems that are assumed in the SAR to perform an operational or safety function.

- The licensee included design criteria and provided references to relevant up-to-date standards, guides, and codes.

- The proposed flux monitoring systems provide operable protection capability in all operating modes and conditions, as analyzed in the SAR for the complete range of normal reactor operating conditions and for coping with anticipated transients and potential accidents.

- The design of the proposed flux monitoring systems reasonably ensures that the design bases can be achieved, that the systems are built of high-quality components using accepted engineering and industrial practices, and that the systems can be readily tested and maintained in the designed operating condition.

- The flux monitoring systems are designed for reliable operation in the normal range of environmental conditions anticipated within the facility.

- The design of the proposed flux monitoring systems provides sufficient redundancy to protect against unsafe conditions in case of single failures within the reactor protection/control system.

- The flux monitoring systems are designed to facilitate inspection, testing, and maintenance to provide reasonable assurance that component failures and degradation will be detected in a timely manner. TS intervals for calibration are adequate to prevent significant drift in instrument setpoints and detection ranges.

- The use of physical provisions to control access to the proposed flux monitoring systems is sufficient to prevent unintended and unauthorized use of the reactor systems.

4. RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

4.1 Radiation Protection

The University of Massachusetts Lowell (UML) Radiation Safety Program (RSP) is described in safety analysis report (SAR) Chapter 11. The SAR states that the RSP provides radiation protection oversight of the UML research reactor (UMLRR), as well as other activities associated with the licensed use of radioactive materials at UML. Since the RSP is the radiation protection program for the UMLRR, it must meet the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) 20.1101, "Radiation Protection Programs." That regulation requires, in part, that each licensee develop, document, and implement a radiation protection program and use, to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA).

The U.S. Nuclear Regulatory Commission (NRC, the Commission) inspection program routinely reviews radiation protection and radioactive waste management at the UMLRR. The NRC staff finds that the licensee's historical performance in these areas, as documented in NRC inspection reports and in the annual operating reports for the UMLRR, and the SAR, as supplemented, provide documentation that measures are in place to minimize radiation exposure to UMLRR staff and the public and to provide adequate protection against operational releases of radioactivity to the environment.

4.1.1 Radiation Sources

The NRC staff reviewed the descriptions of potential radiation sources in each physical form (airborne, liquid, or solid) presented in the SAR, as supplemented, including the inventories and the location of the sources. The review of radiation sources included identification of potential radiation hazards and verification that the hazards are accurately depicted and comprehensively identified.

4.1.1.1 Airborne Radiation Sources

Airborne radiation sources are primarily described in SAR Sections 5.6, 11.1.1 and 11.1.5.1.1, as supplemented by the response to request for additional information (RAI)-11.2 (Ref. 43). SAR Section 11.1.1 states that the airborne radioactive materials of principal concern at the UMLRR are argon-41 (Ar-41) and nitrogen-16 (N-16). Ar-41 is produced by neutron activation of stable argon-40, a natural component of atmospheric air. Ar-41 is mainly produced from neutron activation of the air in experimental facilities (e.g., the reactor beam ports) and the air dissolved in the reactor coolant. N-16 is generated by the neutron activation of stable oxygen-16 in the reactor coolant water as it passes through the reactor core.

SAR Section 11.1.1.1 states that the UMLRR airborne waste handling systems are designed to prevent production and/or release of airborne wastes in amounts that would result in doses to individuals in the reactor confinement building or in the external environs above the regulatory limits in 10 CFR Part 20. As discussed in SAR Section 9.1.1, the reactor confinement building exhaust system is designed to provide a slightly negative pressure with respect to the outside of the building to ensure that any airborne radioactive effluents are released through the 100 foot (30.4 meter) exhaust stack.

Doses from Nitrogen-16

As discussed in SAR Section 5.6, the primary control for N-16 is the holdup tank, which is part of the primary coolant system and is located in the pump room. The holdup tank has a 3,000-gallon (11,356 liter) capacity. The holdup tank delays the return of coolant to the reactor pool after it passes through the core, allowing sufficient time for N-16 to decay given its short (approximately 7 second) half-life. Although significant radiation levels from N-16 occur near the holdup tank during reactor operation with forced cooling, concrete shielding is used to reduce the dose rates and to ensure that facility personnel can access the pump room as needed during forced cooling operation.

During reactor operation at 100 kilowatts-thermal (kWt) or below, the primary coolant system is not required to be operating (i.e., the reactor may be operated with natural convection cooling). When the reactor is operated with natural convection cooling, no coolant passes through the holdup tanks. In SAR Section 4.6.1, UML stated that although its thermal-hydraulic analyses suggest that the UMLRR may be operated with natural convection cooling at power levels above 100 kWt (see safety evaluation report (SER) Section 2.6), it limits natural convection operation to 100 kWt to reduce the production and subsequent escape of N-16 from the pool surface. SAR Section 5.2 states that the level of N-16 detected at the surface of the pool is negligible for natural convection operation at 100 kWt.

The NRC staff reviewed UML's information regarding N-16. The NRC staff finds that the design of the UMLRR and the controls used by UML will help limit UMLRR facility staff doses to levels that are ALARA and within regulatory limits because the design and controls will limit the amount of N-16 escaping from the pool surface. The NRC staff also finds that, because of the short half-life of N-16 and the time that it would take for N-16 to escape the facility through the ventilation system, any public dose from N-16 will be negligible.

Occupational Doses from Argon-41

SAR Section 11.1.1.1.3 states that the Ar-41 release rate has been measured inside the stack, with the reactor operating at full power (1 megawatt-thermal (MWt)), and with and without experiment exhaust fans (the facilities exhaust blower, which serves the beam ports and thermal column, and the pneumatic tubes exhaust blower) operating. UML stated that the Ar-41 release rate through that stack is 15.7 microcuries (µCi) per second (µCi/s) with both fans operating and 11.9 µCi/s without the fans running.

For its SAR Section 11.1.1.1.3 calculation of the steady-state Ar-41 concentration in the reactor building during full-power reactor operation, UML used the Ar-41 production rate of 11.9 µCi/s, the main ventilation exhaust rate of 7.1 cubic meters per second (15,044 cubic feet per minute), the technical specification (TS) 5.1 minimum reactor room free volume of 9,486 cubic meters (3.35×10^5 cubic feet), and the Ar-41 half-life of 109 minutes. The Ar-41 half-life is used because the calculation considers radioactive decay. The Ar-41 production rate with the fans off is used, because the additional Ar-41 exhausted through the stack when the fans are on represents Ar-41 generated in experimental facilities that is not released to the reactor room, and is exhausted directly to the stack (see SAR Sections 9.1.1 and 9.1.2, and SAR Figure 9-1).

UML calculated that the steady-state Ar-41 concentration in the reactor building during full-power reactor operation is 1.4×10^{-6} µCi per cubic centimeter. UML stated that because this is below the 10 CFR Part 20, Appendix B, Table 1, Column 3, derived air concentration (DAC) of

3.0×10^{-6} µCi per cubic centimeter for Ar-41, the maximum annual occupational Ar-41 doses at the UMLRR will be below the 5,000 millirem (mrem) occupational dose limit of 10 CFR 20.1201.

The NRC staff performed an analysis to determine if it could confirm UML's analysis demonstrating that the Ar-41 concentration in the reactor building during full-power operation would not result in unacceptable worker doses, using similar assumptions as UML but conservatively ignoring radioactive decay. The NRC staff calculated a concentration of 1.68×10^{-6} µCi per cubic centimeter, similar to UML's value and also below the 3.0×10^{-6} µCi per cubic centimeter DAC. The NRC staff notes that, for an occupational worker present in the reactor room during full-power operation for 2,000 working hours in a year, this Ar-41 concentration would result in a dose of 2,800 mrem, which is below the 5,000 mrem occupational dose limit of 10 CFR 20.1201.

The NRC staff reviewed UML's occupational Ar-41 dose calculation and finds that UML's assumptions and methodology are reasonable. The NRC staff notes that UML's calculation and the NRC staff's confirmatory calculation are conservative because they are based on semi-infinite cloud models that overestimate the external Ar-41 dose, given the actual geometry of the reactor room. The NRC staff also notes that the calculations are conservative because they assume full-power operation of the UMLRR during the entire 2,000 hours that a worker could be present in the reactor room. Although the UMLRR license contains no restriction on reactor operating frequency, historically, the UMLRR has been operated much less than 2,000 hours a year (see SER Section 1.6). Additionally, the occupational personnel exposure data in SAR Table 11-7 help provide confirmation that historical occupational doses at the UMLRR, from Ar-41 and other sources combined, have been low (20 mrem per year or less). Based on the above, the NRC staff finds that occupational Ar-41 doses at the UMLRR are within 10 CFR Part 20 limits and, therefore, acceptable.

Public Doses from Argon-41

SAR Section 11.1.1.1.3, as supplemented by UML's response to RAI-11.2, provides calculations of doses to members of the public from Ar-41 at the UMLRR. UML calculated doses for two scenarios: individuals standing in the publicly accessible locations directly outside the reactor building truck door, reactor building wall, and reactor building personnel access doors exposed to external Ar-41 gamma radiation passing through the wall or doors/airlocks (Scenario B in SAR Section 11.1.1.1.3) and individuals at other publiclyaccessible locations outside the reactor building exposed to external gamma radiation from the Ar-41 released from the UMLRR stack (Scenario C in SAR Section 11.1.1.1.3). For all scenarios, UML conservatively assumed that the receptors were in the same location for an entire year and that the reactor was operated continuously and at full power for the entire year.

For its Scenario B calculations, UML used a Monte Carlo N-Particle Transport model of the reactor building structure that assumed that the entire reactor building was filled with Ar-41 at a concentration of 1.4×10^{-6} µCi per cubic centimeter. The highest dose UML calculated, 1.42 mrem for an individual located adjacent to the reactor building truck door for an entire year, is well below the 100 mrem public dose limit of 10 CFR 20.1301.

For its Scenario C calculations, UML used an Ar-41 production and release rate of 15.7 µCi/s (conservatively assuming that experiment exhausts are operating) and the main ventilation exhaust rate of 7.1 cubic meters per second (15,044 cubic feet per minute). For calculating atmospheric dispersion (chi over Q (X/Q) values) at the receptor locations considered, UML used the ARCON96 computer code, which calculates X/Q values for 95th percentile weather

conditions (i.e., weather conditions that would give a more conservative estimate of dose than the weather conditions that would be expected to occur 95 percent of the time). UML also used the 100 foot (30.5 meter) facility stack height, the 67 foot 10 inch (20.7 meter) height of the reactor building above finished grade (used in the determination of building wake effects), and wind rose data for Hanscom Air Force Base for 2013, which UML considers to be representative of conditions at the UMLRR site. Because it did not have historical Pasquill atmospheric stability class data for the UMLRR site, UML provided calculations for three different Pasquill atmospheric stability classes, assuming that each class occurs 100 percent of the time, in order to determine the class that would result in the highest dose for each receptor location. UML used the following classes: A (extremely unstable), D (neutral), and F (moderately stable). UML used air submersion external dose conversion factors from Federal Guidance Report No. 12 (FGR 12), "External Exposure to Radionuclides in Air, Water, and Soil" (Ref. 58). UML's Scenario C calculation conservatively did not consider radioactive decay. UML provided dose calculations for a maximum dose location 10 meters west of the stack; for a location in the parking lot 36 meters east of the stack; and for a location at the nearest residence to the UMLRR, 196 meters west-northwest of the stack. The results of UML's calculations of public doses from Ar-41 released from the UMLRR stack are summarized in SER **Table 4-1**.

The NRC staff performed calculations to assess UML's estimates of the public dose due to Ar-41 released from the UMLRR stack and determine if the NRC staff could confirm UML's results demonstrating that Ar-41 released from the UMLRR stack would not result in unacceptable public doses. For the receptor at the nearest residence, the NRC staff used the COMPLY computer code, with input values similar to UML's values for Ar-41 release rate, stack height, and building height. The NRC staff assumed a building width of 24.4 meters (80 feet), equivalent to the width of the reactor confinement building (this is conservative because it ignores the additional width of the attached Pinanski building, which would be expected to cause increased air mixing and Ar-41 dilution near the release point). The NRC staff used the default COMPLY assumptions that the average wind speed is 2 meters per second and that the wind blows in the direction of the receptor 25 percent of the time. For close-in receptors, the NRC staff performed a simple calculation to produce bounding estimates of doses to members of the public in outdoor locations close to the stack (such as the area near the truck door or the parking lot). For the close-in receptor calculation, the NRC staff used UML's value for Ar-41 release rate with an extremely conservative generic X/Q of 0.01 seconds per square meter, equal to the value assumed for accident conditions in NUREG/CR-2387, "Credible Accident Analyses for TRIGA and TRIGA-Fueled Reactors" (Ref. 55); an assumption that the wind blows in the direction of the receptor 11 percent of the time, which is the greatest percentage of time that the wind blows in any one direction based on the RAI response Figure 3 wind rose referenced in UML's response to RAI-11.2.d; a conservative 5-percent occupancy factor for an outdoor location; and an Ar-41 air submersion dose conversion factor from FGR 12. The results of the NRC staff's confirmatory calculations of public doses from Ar-41 released from the UMLRR stack are also summarized in SER **Table 4-1**, for comparison with UML's results.

Table 4-1 Calculations of Annual Public Dose from Ar-41 Released from the UMLRR Stack

Receptor Location	UML Calculation (mrem)	NRC Confirmatory Calculation (mrem)	10 CFR 20.1301 limit (mrem)
194 meters WNW from stack (nearest residence)	3.2[a]	8.2	100
36 meters E from stack (parking lot)	13.2[b]	6.5	100
10 meters W from stack (maximum dose location for UML calculation)	14.5[b]		100

[a] Assuming Pasquill Stability Class A (worst-case stability class for this location) occurs 100 percent of the year
[b] Assuming Pasquill Stability Class F (worst-case stability class for this location) occurs 100 percent of the year

SER **Table 4-1** shows that the UML- and NRC-calculated Ar-41 doses are well below the 100 mrem public dose limit in 10 CFR 20.1301 for all receptor locations. Although the UML-calculated doses are above the 10 mrem ALARA constraint on public dose from airborne releases of radioactive material in 10 CFR 20.1101(d), these calculated doses are very conservative, in part, because they are based on 95th percentile weather conditions, instead of average weather conditions, which would be more typical for a release distributed over an entire year, and they assume that a worst-case stability class occurs the entire year (which, in actuality, makes them effectively based on greater than 95th percentile weather conditions). The calculated doses are also conservative because they assume 100 percent occupancy for an outdoor location that is not continually occupied. UML provided additional calculations using "frequency averaged" stability classes based on historic atmospheric stability data from the Pilgrim Nuclear Power Station in Plymouth, Massachusetts, which produced results that are less than the 10 mrem constraint. Additionally, the NRC staff notes that, if a 5 percent occupancy factor for outdoor locations were applied to UML's calculated doses, the results would be less than 1 mrem.

The NRC staff notes that UML did not appear to specify the averaging interval (i.e., the time interval that the calculated 95th percentile X/Q values are determined for; shorter averaging intervals produce larger (more conservative) X/Q values) for its ARCON96 Ar-41 calculations. However, the NRC staff finds that, because routine airborne release calculations are typically based on average weather conditions, the use of any of the standard ARCON96 averaging intervals would produce conservative results for UML's Ar-41 calculation.

The NRC staff reviewed UML's public Ar-41 dose calculations, as described in the SAR, as supplemented, and finds that UML's assumptions, methodology, and results appear to be reasonable and conservative. The NRC staff also performed confirmatory calculations of public doses from Ar-41 released from the UMLRR stack, as discussed above. The NRC staff finds that UML's calculations, and the NRC staff's confirmatory calculations, demonstrate that doses to the public from Ar-41 produced at the UMLRR will be less than the 100 mrem dose limit in 10 CFR 20.1301. Additionally, based on the UML and NRC calculations, the NRC staff finds that expected public doses from Ar-41 airborne releases from the stack will be less than the 10 mrem constraint in 10 CFR 20.1101(d). Therefore, the public doses from Ar-41 are acceptable.

Historical Argon-41 Releases from the UMLRR

In SAR Section 11.1.1.1.3, UML stated that the measured UMLRR Ar-41 release rate (with experiment exhausts running) of 15.7 µCi/s used for UML's public dose calculations, with the assumption that Ar-41 is being released continuously at that rate during the entire year, is equivalent to an annual release of 494.6 curies (Ci) of Ar-41. The NRC staff reviewed the measured quantities of Ar-41 released through the stack from 2008 through 2019, which are documented in UMLRR annual reports (Ref. 10). The NRC staff noted that the quantities of Ar-41 released ranged from a low of 1.72 Ci in 2009 to a high of 6.66 Ci in 2014, well below the 494.6 Ci per year in UML's calculations in the SAR, as supplemented. The 2008-2009 through 2019-2020 annual reports also confirmed that no reactor-produced radionuclides other than Ar-41 were released from the UMLRR stack in significant quantities.

TSs Relevant to Airborne Radiation Sources

Renewed TS 3.6.2, "Effluents," would state, in part:

> Applicability:
>
> This specification applies to the monitoring and control of radioactive effluents from the reactor building.
>
> Objectives:
>
> To ensure that releases of liquid and airborne effluents are within 10 CFR Part 20 limits.
>
> Specifications:
>
> …
>
> (2) The concentration of argon-41 at the location of the maximum exposed individual in the unrestricted area shall not exceed the unrestricted area effluent concentration limit in 10CFR Part 20 Appendix B, Table 2, Column 1 for argon-41 when averaged over 1 year.

TS 3.6.2(2) would require that the concentration of Ar-41 released from the UMLRR (after dilution between the point of release (typically the stack) and the location of the maximally exposed individual) be less than the air effluent concentration limit in 10 CFR Part 20, Appendix B, when averaged over one year. The NRC staff notes that for submersion radionuclides including Ar-41, the air effluent concentration limits in 10 CFR Part 20, Appendix B, correspond to the concentration that would result in a dose of 100 mrem (the 10 CFR Part 20 public dose limit) if a person were exposed to that concentration during an entire year (8,760 hours). As discussed above, UML provided calculations demonstrating that even if the reactor were operated continuously for an entire year, and even if a 100-percent occupancy factor is assumed, the public dose from Ar-41 released from the stack to the unrestricted area (at ground level outside the facility where a member of the public could potentially be present) would be well below the 100 mrem public dose limit and also below the 10 mrem ALARA constraint in 10 CFR 20.1101(d). Historical Ar-41 releases have been much lower than the annual Ar-41 release assumed for UML's calculations. As also discussed above, the reactor building is maintained at a negative pressure such that potentially contaminated air released from the building is directed through the stack, and Ar-41 is the only reactor-produced radionuclide

released from the UMLRR stack in significant quantities. The NRC staff finds that TS 3.6.2(2) helps ensure that the public dose from airborne radionuclides generated by UMLRR operation will be less than the public dose limit in 10 CFR Part 20. The NRC staff also finds that TS 3.6.2(2) appropriately implements the guidance in NUREG-1537, Appendix 14.1, and American National Standards Institute/American Nuclear Society (ANSI/ANS)-15.1-2007 by establishing a limit on effluents. Therefore, based on the above and also based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.6.2(2) is acceptable.

Conclusion

The NRC staff reviewed the above information regarding airborne radiation sources at the UMLRR and UML's Ar-41 dose calculations. The NRC staff finds that UML adequately described airborne radiation sources at the UMLRR and that the airborne radiation sources are reasonable for a research reactor of the UMLRR's type and size and comparable to similar research reactors. Based on the information and analyses that demonstrate that the UMLRR's routine gaseous effluent releases are within the limits in 10 CFR Part 20, as well as the NRC staff's review of historical Ar-41 releases at the UMLRR as described in UMLRR annual reports showing that historical releases have been small, the NRC staff concludes that UML's production and control of airborne radioactive sources are acceptable.

4.1.1.2 Liquid Radiation Sources

The liquid radiation sources are described in SAR Chapter 5 and Sections 11.1.1.2, 11.2.1, and 11.2.3.2, as supplemented by UML's responses to RAI-11.1 (Ref. 23), RAI-11.3 (Ref. 43), and RAI-11.6 (Ref. 71) and by UML's letter dated February 1, 2018 (Ref. 44). The reactor coolant water, including water in the pool and piping, and liquid radioactive waste are liquid radioactive sources. As discussed in SER Section 4.1.1.1, the coolant water contains dissolved N-16 and Ar-41, although most exposure from N-16 and Ar-41 occurs after they have been evolved from the coolant. The coolant water also contains impurities, such as corrosion products, that become activated by neutrons as the water passes through the reactor core. Because the reactor piping carries water that has been circulated through the reactor core, activated impurities may be capable of causing exposure to personnel near the piping (e.g., in the pump room), especially when the reactor is operating at high power. However, as discussed in SAR Section 11.1.5.1, access to the pump room is restricted and appropriate supervision is required for all personnel operations in this area.

The primary coolant is continuously filtered by a mixed-bed deionizer unit to remove dissolved impurities in the water, including radioactive impurities. The coolant is also radiologically monitored both by direct continuous measurement with radiation detection equipment and by periodic sampling and analysis as required by renewed TSs 3.3 and 4.3, which help ensure that the coolant radioactivity remains within acceptable levels and which are discussed and found acceptable in SER Section 2.3.

The primary coolant loop, including the pool and piping, is essentially a closed system with no continuous liquid effluent release to the environment.

Any liquid effluent that is produced at the UMLRR (e.g., from pump gland leaks, washings from demineralizers, laboratory sink effluent, etc.) that is intended for eventual sanitary sewer disposal is transferred to liquid waste hold-up tanks before being released to the sewer. In its response to RAI-11.3, UML stated that when liquid waste is transferred to the hold-up tanks in the basement of the Pinanski building connected to the reactor confinement building, it is

considered to be transferred onto UML's broad scope radioactive material license issued by the Commonwealth of Massachusetts (License No. 60-0049). In accordance with renewed TS 5.1, which is discussed and found acceptable in SER Section 6.5.1, these tanks are still within the reactor licensed boundary (which includes the entire Pinanski building), but the NRC staff notes that the reactor licensed boundary may contain radioactive material on the state license in addition to radioactive material on the reactor license. UML stated that, before any UMLRR liquid radioactive waste is transferred from the reactor sump (in the reactor building) to the hold-up tanks in the Pinanski building, it is cycled and analyzed for the presence of radioactivity to ensure that the transfer would not cause the possession limits of the broad scope radioactive material license to be exceeded. UML also stated that, before the hold-up tanks in the Pinanski building are discharged to the sanitary sewer, they are also cycled, analyzed, and inspected to ensure that the radioactive material concentrations are within the limits for release to the sewer and that the radioactive material in the effluents is soluble in accordance with 10 CFR 20.2003.

The NRC staff reviewed the actual quantities of liquid radioactive material released to the sanitary sewer from the hold-up tanks from 2008 through 2019, as documented in the UMLRR annual reports (Ref. 10). The NRC staff noted that, on an activity basis, the quantities of liquid radioactive material released to the sewer per year (which the NRC staff notes could also include material from the Pinanski building in addition to the UMLRR, because other areas can also discharge liquid waste to the hold-up tanks) were very small (approximately 2 µCi or less).

Certain liquid radioactive waste, such as liquid scintillation waste, may be transferred to an authorized recipient (i.e., licensed disposal facility) for disposal, or decayed-in-storage, rather than being disposed of by release to sanitary sewerage. Long-lived liquid waste could be transferred directly to an authorized recipient or be first transferred to another UML license (e.g., UML's broad scope state license) and then transferred to an authorized recipient from that license, similar to how some long-lived solid waste generated at the UMLRR is handled (see SER Section 4.1.1.3). Any decay-in-storage of short-lived liquid waste would be done under the UMLRR license, similar to short-lived solid waste that is decayed-in-storage (see SER Section 4.1.1.3). Radioactive waste management and disposal is also discussed in SER Section 4.2.

Renewed TS 3.6.2, "Effluents," would state, in part:

> Applicability:
>
> This specification applies to the monitoring and control of radioactive effluents from the reactor building.
>
> Objectives:
>
> To ensure that releases of liquid and airborne effluents are within 10 CFR Part 20 limits.
>
> Specifications:
>
> (1) The discharge of licensed material into sanitary sewerage shall meet the requirements of 10CFR 20.2003(a), "Disposal by Release into Sanitary Sewerage."
>
> ….

TS 3.6.2(1) would require that UML's disposal of UMLRR licensed material into sanitary sewerage comply with 10 CFR 20.2003(a). The regulation, 10 CFR 20.2003(a), includes limits on radionuclide concentration and quantity for liquid waste and also requires that radioactive material disposed of in liquid waste be readily soluble in water. As discussed above, UML performs analyses of liquid waste in the hold-up tanks before it is discharged to the sewer to verify that the waste meets these requirements. Although, as discussed above, UML disposes of liquid radioactive waste generated at the UMLRR (other than liquid waste transferred to an authorized recipient for disposal as discussed in SER Section 4.2.1) by first transferring it to its broad scope license and then releasing it to the sewer under that license, the NRC staff finds that TS 3.6.2(1) helps ensure that any liquid waste containing radioactive material from the UMLRR and ultimately released to sanitary sewerage is disposed of properly and in compliance with applicable NRC regulations. The NRC staff finds that TS 3.6.2(1) helps ensure that any public dose from liquid radioactive waste from the UMLRR that is released to the sewer is within the 10 CFR Part 20 limit. The NRC staff also finds that TS 3.6.2(1) appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 by establishing a limit on effluents. Therefore, based on the above and also based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.6.2(1) is acceptable.

The NRC staff reviewed the above information regarding liquid radiation sources at the UMLRR. The NRC staff finds that UML has adequately described liquid radiation sources at the UMLRR and that the liquid radiation sources are reasonable for a research reactor of the UMLRR's type and size. The NRC staff also finds that UML has implemented procedures and controls to help ensure that doses from liquid radioactive sources associated with the normal operation of the UMLRR are small (i.e., less than 10 CFR Part 20 dose limits, and ALARA). The NRC staff additionally finds that UML's procedures for the disposal of liquid radioactive wastes will help ensure compliance with 10 CFR Part 20. Therefore, the NRC staff concludes that liquid radioactive sources at the UMLRR do not present a significant hazard to the public, UML staff, or the environment.

4.1.1.3 Solid Radiation Sources

The solid radiation sources are described in SAR Sections 3.1.2, 9.5, 10.2.3.1, 11.1.1, 11.1.5, and 11.2.3, as supplemented by UML's response to RAI-11.6 (Ref. 71). The primary solid radiation sources at the UMLRR are the reactor itself (i.e., the irradiated fuel in the reactor core, which contains fission products); other stored, irradiated reactor fuel; and the cobalt-60 (Co-60) irradiation source. The reactor core, all irradiated fuel stored outside the core, and the Co-60 irradiation source are submerged in the open reactor pool and surrounded by a concrete biological shield that protects personnel from the radiation produced by these solid radiation sources. Other solid radiation sources at the UMLRR include unirradiated fuel elements in storage; fission chambers; standard and check sources; neutron check, calibration, and/or startup sources (americium-beryllium, antimony-beryllium, and plutonium-beryllium (Pu-Be)); activated reactor components; irradiated samples; ion exchange resins and filters used for removal of radioactive material from reactor coolant and air effluents; and solid radioactive waste.

As discussed in SER Section 1.10, UML requested that, in conjunction with the renewal of the UMLRR license, the reactor license's special nuclear material and byproduct material possession limits be amended to allow for the possession of Pu-Be neutron sources and byproduct material with atomic numbers 3 through 83 (for checks, calibrations, and characterizations of radiation monitoring instruments), highly enriched uranium in the form of fission chamber linings, and uranium of any enrichment for reactor experiments, under the

reactor license (as opposed to under UML's broad scope material license, as it is currently possessed). UML also requested that the reactor license's possession limit for Co-60 be reduced from 1,500,000 Ci to 100,000 Ci because, as listed in SAR Table 11-1, as of 2014, the activity of UML's Co-60 irradiation source was less than 100,000 Ci. The NRC staff found these requested licensed possession limit changes acceptable, as discussed in SER Section 1.10.

Renewed TSs 3.3 and 4.3, which are discussed and found acceptable in SER Section 2.3, require monitoring of the UMLRR pool water chemistry and radioactivity and would help provide an early indication of leakage from, or problems with, solid radioactive sources in the reactor pool such as fuel or the Co-60 irradiation source.

The UML RSP oversees the safety of all radiation sources at the UMLRR, including solid radioactive sources and radioactive waste. UML incorporates access controls at the UMLRR facility that help ensure that personnel will not receive excess doses from solid radiation sources.

Solid radioactive waste at the UMLRR includes spent ion exchange resins, contaminated air filters, disposable clothing, sample transfer rabbits, contaminated paper and laboratory items, miscellaneous hardware, and other items that are produced as a result of experimental and maintenance operations. Solid waste with a half-life of less than or equal to 120 days may be held for decay in storage at the UMLRR under the UMLRR license until the radioactivity of the material is statistically indistinguishable from background radioactivity, and then released. Quantities of solid waste with a half-life of greater than 120 days (i.e., long-lived waste) may also be temporarily stored at the UMLRR until a sufficient quantity of the waste has been generated to warrant a transfer for disposal. Solid waste with a half-life of greater than 120 days is either transferred directly to an authorized recipient (i.e., a disposal facility licensed to receive such waste via a licensed waste vendor or broker) for disposal or is transferred first to another UML radioactive material license (e.g., UML's broad scope material license), then transferred to a disposal facility under that license. If the waste is first transferred to another UML license, then the transfer to the other license occurs when the waste material leaves the UMLRR licensed boundary. These transfers are internal to UML and no shipping is involved. Additionally, prior to these transfers, UML verifies the isotopes and activity of the waste, the quantities of radioactive material on the other UML license, and the limits of the other UML license to ensure that the other license is able to receive the UMLRR waste material. Radioactive waste management and disposal is also discussed in SER Section 4.2.

The NRC staff reviewed the above information regarding solid radiation sources at the UMLRR. The NRC staff finds that UML adequately described solid radiation sources at the UMLRR and that the solid radiation sources are reasonable for a research reactor of the UMLRR's type and size. The NRC staff also finds that UML implements procedures and controls to help ensure that doses from solid radioactive sources and radioactive waste associated with the normal operation of the UMLRR are small (i.e., less than 10 CFR Part 20 dose limits, and ALARA). Therefore, the NRC staff concludes that solid radioactive sources at the UMLRR do not present a significant hazard to the public, UML staff, or the environment.

4.1.2 Radiation Protection Program

The regulation, 10 CFR 20.1101(a), requires each licensee to develop, document, and implement a radiation protection program.

SAR Section 11.1.2 discusses the UML RSP, which encompasses activities conducted on UML's campus under UML's Commonwealth of Massachusetts broad scope radioactive material license, as well as activities conducted under NRC Facility Operating License No. R-125 for the UMLRR. The RSP governs the policies and procedures related to the safe use of all licensed radiation sources at UML. The UML Chancellor is ultimately responsible for the RSP, but delegates management authority for the program to the UML Director of Radiation Safety (DRS), who heads the UML Radiation Safety Office and also serves as the radiation safety officer (RSO) for the UML campus, including the UMLRR. Therefore, the DRS/RSO is responsible for developing and managing the RSP within the limits set forth by federal and state regulations.

Although SAR Section 11.1.2 states that the UML Chancellor is ultimately responsible for the RSP, and delegates management authority for the RSP to the DRS/RSO, renewed TSs 6.1.1 and 6.1.2(5), which are discussed and found acceptable in SER Section 6.1.1, require that the DRS/RSO report to and be directly responsible to the Level 1 authority (an individual in the Chancellor's office, designated by the Chancellor, at a position of Associate Vice Chancellor or higher, who has overall responsibility for the UMLRR and the conduct of UMLRR activities in accordance with NRC regulations) in matters pertaining to radiation safety.

As discussed in SAR Section 11.1.2.4, specific duties of the DRS/RSO include reviewing applications related to the generation, purchase, use, or disposal of radioactive material at UML and also reviewing (along with the UMLRR Reactor Supervisor) applications for routine reactor use. The DRS/RSO also serves as a permanent member of the UML Radiation Safety Committee (RSC), and all of its subcommittees, including the Reactor Safety Subcommittee (RSSC), which is specific to the UMLRR.

As discussed in SAR Section 11.1.2.4.1, the UML Radiation Safety Office also has a radiation safety specialist, who reports directly to the DRS/RSO. Specific duties of the radiation safety specialist include performing radiation surveys; maintaining and calibrating radiation survey and analytical equipment; maintaining the radioactive source inventory; maintaining the radioactive waste storage and disposal program; analyzing and packaging radioactive material being shipping to and from UML; assisting the DRS/RSO in providing training to radiation workers; and collecting and analyzing environmental samples as required.

As discussed in SAR Section 12.1.5, UML requires personnel using radioactive materials or radiation sources to become familiar with the RSP and to conduct activities in accordance with the RSP. However, the UML radiation safety staff have the authority to interdict or terminate the use of radioactive materials or radiation sources if adequate health physics support is not available or if significant deviations from established procedures have occurred or are likely to occur.

Renewed TS 6.3, "Radiation Safety," would state:

(1) The Radiation Safety Program shall be designed to achieve the requirements of 10 CFR 20 and should use the guidelines in American National Standard, ANSI/ANS-15.11 "Radiation Protection at Research Reactor Facilities." (R2016 or later revision).

(2) The Radiation Safety Program shall be the responsibility of the Radiation Safety Officer, having line authority as indicated in Figure 6-1.

(3) The Radiation Safety Program shall include management commitment to maintain exposures and releases as low as reasonably achievable.

TS 6.3(1) would require that the UML RSP, which is applicable to the UMLRR, be designed to meet 10 CFR Part 20 requirements. TS 6.3(1) would also recommend that the UML RSP conform to the guidance in ANSI/ANS-15.11-2016 (Ref. 69) (or a later revision of this guidance). TS 6.3(2) would require that the RSO be responsible for the RSP, and TS 6.3(3) would require that the RSP include a management commitment to ALARA. UML's response to RAI-14.6.12 (Ref. 71) states that the "RSP" terminology used at UML is intended to be equivalent to "radiation protection program" as used in NUREG-1537 and ANSI/ANS-15.1-2007. The NRC staff finds that TS 6.3(1) helps ensure that the RSP complies with 10 CFR Part 20 and that the TS recommends that the RSP be consistent with the appropriate standard for radiation protection at research reactors, which provides recommendations for implementing radiation protection programs that help ensure that radiation exposure and release levels are ALARA. The NRC staff finds that TS 6.3(2) is consistent with TS 6.1.2(4) and helps ensure that the individual responsible for implementing the RSP (i.e., the RSO) is clearly delineated and that the line authority to the RSO is separate from that of the reactor operations staff (see renewed TS Figure 6-1, which is discussed and found acceptable in SER Section 6.6.1). The NRC staff finds that TS 6.3(3) helps ensure UML management commitment to ALARA. Additionally, the NRC staff finds that TSs 6.3(1), 6.3(2), and 6.3(3) appropriately implement the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 by designating the responsibility for, and requirements of, the radiation protection program. Therefore, based on the above and also based on its 10 CFR 50.36(c)(5) findings for TSs on administrative controls in SER Section 6.7, the NRC staff concludes that TSs 6.3(1), 6.3(2), and 6.3(3) are acceptable.

Renewed TS 6.2, which is discussed and found acceptable in SER Section 6.6.2, requires that an RSSC, which must have the RSO and Reactor Supervisor as two of its at least five members, have oversight functions for UMLRR operations to ensure that the facility is operated in a manner consistent with public safety. The scope of audits (required by TS 6.2.4) must include, among other things, the conformance of facility operation to the UMLRR TSs and license, and the results of action taken to correct any deficiencies in equipment or methods of operation that affect safety, including radiation safety.

Renewed TSs 6.2.3 and 6.4, which are discussed and found acceptable in SER Sections 6.6.2 and 6.6.4, respectively, require that the RSSC review new procedures, and substantive changes to existing procedures, including radiation protection procedures. Renewed TS 6.5, which is discussed and found acceptable in SER Section 6.6.5, requires that the RSSC review new classes of experiments to be performed at the UMLRR; these reviews are required to include radiation safety aspects of experiments.

The regulation, 10 CFR 20.1101(c), requires that licensees periodically (at least annually) review the radiation protection program content and implementation. As discussed in SAR Section 11.1.2.6, the UML RSC meets at least quarterly to review the status of the RSP; these meetings include a review of all audit findings, incidents, and emergency response actions, as applicable. In addition, the UML RSC audits the UML RSP at least annually. In response to RAI-14.6.11 (Ref. 71), UML stated that the UML RSC's reviews and audits of the overall RSP are intended to satisfy the 10 CFR 20.1101(c) requirement. Additionally, TS 6.3(1), which is discussed above, requires that the RSP be designed to achieve all applicable requirements of 10 CFR Part 20, including 10 CFR 20.1101(c).

As discussed in SER Sections 4.1.4, 4.1.5, and 4.1.6, UML conducts radiation and contamination surveys at the UMLRR on a regular basis and also monitors personnel radiation exposure. UML also monitors liquid and gaseous effluents for compliance with renewed TS 3.6.2, which is discussed and found acceptable in SER Sections 4.1.1.1 and 4.1.1.2. Additionally, UML conducts environmental monitoring outside the UMLRR reactor confinement building, as discussed in SER Section 4.1.7. As required by renewed TS 6.8, which is discussed and found acceptable in SER Section 6.6.8, records of facility radiation and contamination surveys are retained for 5 years, and records related to gaseous and liquid effluents, personnel radiation exposures, and environmental monitoring required by TSs are retained for the life of the facility.

The NRC inspection program also routinely reviews the UML RSP as it relates to and is implemented at the UMLRR, and the results of these reviews are documented in inspection reports. The NRC staff reviewed the UMLRR inspection reports for the years 2006 through 2019 (Ref. 11) and found no significant issues with radiation safety or the implementation of the RSP at the UMLRR.

The NRC staff reviewed the above information and finds that the UML RSP, as it relates to and is implemented at the UMLRR, complies with 10 CFR 20.1101, paragraphs (a) and (c); is implemented in an acceptable manner; and provides reasonable assurance that for all UMLRR facility activities, UMLRR staff, members of the public, and the environment will be protected from unacceptable radiation exposures. Therefore, the NRC staff concludes that the UMLRR has an acceptable radiation protection program.

4.1.3 ALARA Program

The regulation, 10 CFR 20.1101(b), requires that licensees use, to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are ALARA.

Renewed TS 6.3, which is discussed and found acceptable in SER Section 4.1.2, recommends that the UMLRR RSP conform to the guidance in the most current revision of ANSI/ANS-15.11, “Radiation Protection at Research Reactor Facilities” (the current revision of ANSI/ANS-15.11 is ANSI/ANS-15.11-2016 (Ref. 69)). The guidance in ANSI/ANS-15.11-2016 discusses ALARA programs and states that each research reactor facility “shall have an ALARA program that establishes appropriate ALARA goals and reviews ALARA accomplishments.”

SAR Section 11.1.3 discusses the program at UML, including the UMLRR, to keep occupational doses ALARA. UML stated that the object of an occupational ALARA program is to reduce occupational exposures as far below the specified limits as is reasonably achievable by means of good radiation protection planning and practice, as well as a management commitment to policies that foster vigilance against departures from good practice. UML stated that aspects of its program to keep occupational doses ALARA include:

- Training to allow workers to recognize and protect themselves from sources of radiation;
- A dosimetry program, which includes badge monitoring and bioassays;
- Investigation by the UML Radiation Safety Office of any occupational radiation exposures greater than 10 percent of 10 CFR Part 20 limits;

- Radiation and contamination surveys; and

- RSO and RSC or RSSC oversight of activities involving radiation at UML, including the UMLRR (the NRC staff notes that renewed TS 6.2, which is discussed and found acceptable in SER Section 6.6.2, requires RSSC review and audit of UMLRR activities, and renewed TSs 6.4 and 6.5, which are discussed and found acceptable in SER Sections 6.6.4 and 6.6.5, respectively, require the RSSC to review UMLRR procedures and experiments).

SAR Section 11.1.5 discusses design aspects specific to the UMLRR, such as shielding and entry control devices for areas that could have elevated radiation levels, which also help reduce occupational radiation doses.

The UMLRR is also designed to minimize any radiation dose to the public from its operation. For example, as discussed in SER Sections 2.3, 2.4, and 5.8, the reactor is located in a thick concrete-walled pool of water within a confinement building that similarly has thick concrete walls, providing shielding to help ensure that any external dose in publicly accessible areas outside the reactor building is minimal. Additionally, as discussed in SER Sections 4.1.1.1, 4.1.1.2, and 4.1.4, gaseous and liquid radiological effluents from the UMLRR are released from the facility through controlled, monitored pathways that are designed to minimize the effluents' impact on members of the public and the environment. Renewed TSs 3.6(4) and 4.6(4), which are discussed and found acceptable in SER Section 4.1.7, require a UMLRR environmental monitoring program, which helps ensure that the facility is operated in a manner that minimizes radiation exposure to the public, and that the radiation exposure to the public resulting from the operation of the UMLRR facility is maintained ALARA.

SAR Section 11.1.2.1 states that the UML Chancellor is knowledgeable as to the overall UML RSP (and audits of the program). The chancellor grants both the UML Radiation Safety Office and the RSC the authority to immediately stop radiation-related activities in any UML laboratory (including the UMLRR) as necessary to ensure radiation safety. As discussed in SAR Section 11.1.2.2, the RSC, whose members are appointed by the Chancellor, includes at least two members of UML upper management (e.g., deans, directors, and the RSC chair, who all report directly to either the UML Chancellor, a vice chancellor, or the Provost), is responsible for reviewing the UML ALARA program (including personnel exposure, survey results, and any radiation-related incidents), conducting periodic audits of the program, and acting on audit findings. Additionally, renewed TS 6.3(3), which is discussed and found acceptable in SER Section 4.1.2, requires that the UML RSP include management commitment to ALARA. The NRC staff finds that this information helps demonstrate the commitment of upper levels of UML management and administration to maintaining doses ALARA at the UMLRR.

The NRC inspection program also routinely reviews radiation safety at the UMLRR, including the implementation of practices to maintain worker and public doses ALARA. The NRC staff reviewed the UMLRR inspection reports for the years 2006 through 2019 (Ref. 11) and found no significant radiation safety or ALARA issues at the UMLRR. Additionally, the NRC staff reviewed the occupational (whole-body and extremity) and public (from gaseous and liquid effluents) doses reported for the years 2008 through 2018 in UMLRR annual operating reports (Ref. 10) and noted that the doses are all well below 10 CFR Part 20 limits as well as the ALARA goals (10 percent of 10 CFR Part 20 limits) stated in the reports. The NRC staff finds that the reported occupational and public doses in the annual operating reports help demonstrate the effectiveness of UML's efforts to maintain UMLRR doses ALARA.

The NRC staff reviewed the above information and finds that the ALARA program, including practices and design features, implemented at the UMLRR complies with 10 CFR 20.1101(b). Additionally, as discussed in SER Section 4.1.1.1, based on the UML and NRC calculations, the NRC staff finds that realistic public doses from Ar-41 airborne releases from the stack will be less than the 10 mrem ALARA constraint in 10 CFR 20.1101(d). Therefore, the NRC staff concludes that the ALARA program implemented at the UMLRR will help limit occupational and public doses so that they are ALARA, and is acceptable.

4.1.4 Radiation Monitoring and Surveying

The regulation, 10 CFR 20.1501(a), states that licensees shall make, or cause to be made, surveys of areas, including the subsurface, that—

(1) May be necessary for the license to comply with the regulations in this part; and

(2) Are reasonable under the circumstances to evaluate—

(i) The magnitude and extent of radiation levels; and
(ii) Concentrations or quantities of residual radioactivity; and
(iii) The potential radiological hazards of the radiation levels and residual radioactivity detected.

The regulations in 10 CFR 20.1501(c) require licensees to ensure that instruments and equipment used for quantitative radiation measurements (e.g., dose rate and effluent monitoring) are calibrated periodically for the radiation measured.

Radiation monitoring and surveying at the UMLRR is discussed in SAR Sections 7.7, 11.1.4, and 11.1.6, as supplemented by UML's responses to RAI-11.4 (Ref. 43), RAI-14.3.23 (Ref. 71), RAI-14.3.24 (Ref. 71), RAI-14.3.25 (Ref. 71), and RAI-14.4.21 (Ref. 71) and UML's letters dated September 30, 2020 (Ref. 98), and January 30, 2021 (Ref. 99).

SAR Section 7.7.1 states that the system of fixed radiation detection monitors at the UMLRR is designed to measure the radiation expected to be encountered in a research reactor environment and helps ensure that all radiation sources may be appropriately identified and evaluated. As discussed in SAR Sections 7.7.2 and 11.1.4.2, the radiation monitors that are part of the UMLRR radiation monitoring system include area dose rate monitors, continuous air monitors (CAMs), and the stack effluent monitoring system, as well as other monitors. As listed in SAR Section 7.7.3 and 11.1.4.2, there are many area dose rate monitors, which are Geiger-Mueller detectors or ion chambers (used primarily to detect gamma radiation fields), located throughout the reactor building. To monitor any airborne radioactivity within the reactor building, there are also two CAMs, which use passivated implanted planar silicon detectors. One of the CAMs is located on the reactor experimental level (third floor of the reactor building), and the other is located on the reactor pool level (first floor of the reactor building). To monitor airborne radioactivity in air exhausted through the stack, the stack effluent monitoring system has two scintillation monitors (one measuring gaseous effluents and one measuring particulate effluents) located on the fourth floor of the Pinanski building attached to the reactor building (the NRC staff notes that the Pinanski building has three full above-grade floors, but there is a small "fourth floor" area containing this equipment).

As discussed in SAR Sections 7.7.2, 7.7.3, and 7.7.4, the various area monitors throughout the facility are connected either to one of six rack-mounted ratemeters located in the control room or

one of three remote ratemeters located on the first or third floor of the reactor building (there are also two neutron dose rate monitors connected to the ratemeters). The ratemeters display the dose rates measured by their connected area monitors and also display visual alarms when a lower “warning” or “elevated” setpoint is reached as well as when a “high” radiation level alarm setpoint is reached (certain area monitors that are connected to the control room ratemeters also have local readouts). The CAMs and the stack effluent monitors also have local visible and audible alarms when a high radiation setpoint is reached. The nine ratemeters, as well as the CAMs and the stack effluent monitoring system, output to the Area Radiation Monitor Computer Data Acquisition System (ARM CDAS) digital controller and a computer system that processes and graphically displays data from the controller and provides for the transfer of operator commands and pre-programmed system responses (see also SER Section 3.7). The computer display, which is on the radiation monitoring system cabinet in the control room, can display readings from all of the area monitors (including neutron monitors) connected to the nine ratemeters, as well as the CAMs and the stack effluent monitoring system.

As discussed in SAR Sections 7.7.5, 7.7.6, and 7.7.7, UML’s digital radiation monitor output processing system has a pre-programmed logic that causes alarms and/or actions to occur when certain combinations of radiation monitors (area, CAM, or stack gas or particulate) reach their “high” alarm setpoints. These are combinations that UML stated may indicate the potential for either a local (i.e., for a specific area within the reactor building) or general (i.e., could correspond to a radioactivity release outside the reactor building) radiation hazard. UML referred to the alarm conditions associated with a potential local or general radiation hazard as a potential local radiation emergency alarm (potential LREA or P-LREA) and potential general radiation emergency alarm (potential GREA or P-GREA), respectively.

The combinations that would produce a P-LREA include, for example, both CAMs (indicating an airborne radiation hazard within the reactor building), or the reactor bridge and control room area monitors (indicating a wide area external radiation hazard on the third floor of the reactor building), and are discussed in detail in SAR Section 7.7.6. The combinations that would produce a P-GREA include, for example, a reactor bridge area monitor and the stack gas monitor (which could be associated with significant radioactivity in and above the pool that is also being released from the stack), and are discussed in detail in SAR Section 7.7.7.

As discussed in SAR Section 7.7.5, the occurrence of either a P-LREA or P-GREA causes a “squee” alarm to activate in both the reactor building and the attached Pinanski building, and a P-GREA additionally causes a general reaction in the ventilation system (GRVS) (see SER Section 6.3.4).

UML stated that a P-LREA or P-GREA condition is considered a potential emergency until the operator can verify the high radiation conditions. The reactor operator must press a button on the control room radiation monitoring system cabinet or click on the radiation monitoring system display screen to declare an actual alarm (i.e., an actual LREA or GREA). When an actual LREA or GREA occurs, the reactor automatically scrams. Additionally, for an LREA, evacuation alarm horns activate in the reactor building and a GRVS occurs (a GRVS does not occur for a P-LREA); for a GREA, evacuation horns activate in both the reactor building and the attached Pinanski building.

Renewed TS 3.6.1(1), which is discussed and found acceptable below, requires a minimum set of radiation monitors with readouts and alarms in the control room. In its letter dated September 30, 2020 (Ref. 98), UML stated that its monitors that would be required by TS 3.6.1(1) do not currently have individual remote audible alarms in the control room. Additionally, UML stated

that for the stack gas and particulate monitors and the TS-required third floor CAM, the only control room readout is currently on the area radiation monitoring system (ARMS) human-machine interface (HMI) computer display. Therefore, because an individual TS-required radiation monitor reaching its setpoint would not necessarily cause an audible alarm in the control room, and because the ARMS computer system or HMI could also potentially represent a single point of failure, UML stated in its letters dated September 30, 2020, and January 30, 2021 (Ref. 99), that during the implementation period of a renewed license (see SER Section 1.10), it would change its radiation monitoring system by adding an additional control room ratemeter for the stack gas and particulate monitors, which would provide additional independent readouts and individual audible alarms in the control room; and configuring the existing control room ratemeters for TS-required area monitors to provide for individual audible alarms in the control room (as discussed above, the ratemeters already provide control room readouts independent of the ARMS HMI display, and visual alarms). UML stated that the new radiation monitor control room audible alarms (for individual radiation monitors reaching their alarm setpoints) will be provided by the existing radiation monitor alarm panel, which currently provides audible alarms when certain combinations of monitors reach their alarm setpoints (i.e., when a P-LREA or P-GREA occurs). The NRC staff confirmed that the new individual ratemeter audible alarms will be independent of the current pre-programmed alarm logic and the ARMS computer system because, during the 2020-2021 audit, as documented in the audit report dated March 4, 2021 (Ref. 100), UML stated that the audible alarms for the new and existing ratemeters will be provided by the buzzer in the existing radiation monitor alarm panel (on the control room radiation monitoring cabinet) via a hard-wire connection from the ratemeter relays, and that the actuation of the audible alarms will not involve the ARM CDAS.

In its letter dated September 30, 2020, UML stated that a new control room alarm is not necessary for the TS-required third floor CAM and that UML does not plan to make any changes related to this monitor, because this monitor is near the control room and has a loud audible alarm that is easily heard by the operator, even if the control room door is closed. UML also stated that the monitor additionally has a visible alarm (bright light) that is easily visible to the operator.

UML stated that its proposed radiation monitoring system configuration, including UML's proposed changes that will provide individual control-room-audible alarms for all TS 3.6.1(1)-required radiation monitors, and additional control room readouts, will be sufficient to ensure that the reactor operator is immediately alerted and can evaluate the situation and take immediate action if any single TS-required radiation monitor reaches its setpoint.

Other ARMS changes, that UML implemented since the previous license renewal in 1985, are described in SER Section 3.7.1.

In response to RAI-11.4, UML provided information regarding setpoints for its radiation monitors. UML stated that, for area dose rate monitors, the lower "warning" setpoints are at or below 5 mrem per hour, and the "high" alarm setpoints are at or below 100 mrem per hour. For CAMs, both setpoints are below the 10 CFR Part 20, Appendix B, Table 1, Column 3, DAC for cesium-138 (Cs-138); the Cs-138 DAC is used because Cs-138 is a particulate decay product of xenon-138 and is, therefore, considered to be a likely fission product release indicator that the CAMs would detect. For the stack particulate monitor, both setpoints are below the Cs-138 air effluent concentration in 10 CFR Part 20, Appendix B, Table 2, Column 1. For the stack gas monitor, both setpoints are below an Ar-41 activity concentration that UML states, if released continuously, could correspond to a dose of 100 mrem per year to a member of the public (equal to the annual public dose limit in 10 CFR 20.1301).

Renewed TS 3.6.1, “Radiation Monitoring,” would state, in part:

Applicability:

This specification applies to the availability of radiation monitoring equipment which must be operable during reactor operation.

Objective:

To ensure that radiation monitoring equipment is available for evaluation of radiation conditions in restricted and unrestricted areas.

Specifications:

(1) For any of the conditions in specification 3.4.1, the following minimum radiation monitors shall be operating with readouts and alarm indicators in the control room:

a. Stack gaseous and stack particulate radiation monitors.

b. A constant air monitor, located on the reactor pool level (third floor).

c. An area radiation monitor on the reactor experimental level (first floor).

d. An area radiation monitor over the reactor pool.

(2) Each gamma irradiation facility shall have an operating area radiation monitor having a local readout and alarm indicator capable of alerting personnel at the gamma irradiation facility when irradiations are performed.

(3) If a required radiation monitor becomes inoperable, operations may continue only if the monitor is repaired or replaced with a monitor of similar function within 1 hour of discovery.

….

TS 3.6.1(1) would require that during conditions when confinement is required (see renewed TS 3.4.1, which is discussed and found acceptable in SER Section 6.3.4), at least one each of a stack gaseous monitor; stack particulate monitor; CAM; area radiation monitor on the first floor of the reactor building; and area radiation monitor over the reactor pool shall be operating with readouts and alarms in the control room. As discussed above, following the changes UML proposes to make to its radiation monitoring system, all TS 3.6.1(1)-required monitors will have individual alarms that are audible in the control room. UML stated that the audible alarms will be sufficient to ensure that the reactor operator is immediately alerted and can take appropriate action (as needed) if any single monitor reaches its setpoint. Additionally, based on the above discussion, TS 3.6.1(1)-required area monitors and the stack gas and particulate monitors will have at least two readouts in the control room (from the ratemeters and ARMS HMI) following the changes. The third floor CAM will continue to have a readout on the ARMS HMI. As discussed above and in SER Section 5.1, the purpose of these monitors, readouts, and alarms

is to measure radiation levels at the UMLRR and to ensure that the operator is aware of any radiation conditions that could indicate a potential problem at the facility or require action.

Although the overall UMLRR radiation monitoring system is designed such that additional alarms (control room alarms as well as the facility-wide “squee” alarm and evacuation horn discussed above) and/or actions occur when certain combinations of radiation monitors reach “high” alarm setpoints, TS 3.6.1(1) only requires readouts and alarm indicators for the individual monitors listed in TS 3.6.1(1), items a. through d. (the individual alarm that is just outside, but audible in, the control room in the case of the third floor CAM, and individual alarms in the control room for the other monitors). As discussed above, some facility radiation monitors have local readouts and/or alarms in addition to the control room alarms (and facility-wide “squee” alarm and evacuation horn). In its response to RAI-14.3.25 (Ref. 71), UML stated that it considers the radiation alarm system discussed in the SAR, combined with the operator’s ability to manually initiate alarms or announcements to warn reactor staff of radiation hazards in the reactor building as needed, sufficient to alert reactor staff to potential radiation hazards. In its letter dated September 30, 2020 (Ref. 98), UML also stated that specific TS requirements for local readouts and/or alarms for TS 3.6.1(1)-required monitors are not necessary (other than as may separately be required by TS 3.6.1(2)), because of the operators’ ability to warn other staff in the facility and the multiple redundant local alarms in the facility, which are in addition to the portable survey meters routinely used per procedure in evaluating radiological conditions. For some of the monitor types listed in TS 3.6.1(1), items a. through d., there are multiple monitors that may serve as the one required monitor of that type (see discussion above).

The NRC staff finds that the required monitors, readouts, and alarms in TS 3.6.1(1) help ensure that the reactor operator is made aware of any abnormal radiation conditions at the UMLRR and can take necessary actions (if not completed automatically by the UMLRR systems discussed above) such as shutting down the reactor, securing facility ventilation, activating the emergency exhaust system, warning other staff, and/or declaring a LREA or GREA, as appropriate. As a result, the NRC staff finds that the monitors help ensure that the reactor operator can take necessary actions assumed in the MHA analysis (see SER Section 5.1). The NRC staff also finds that having a CAM and area radiation monitor near the reactor pool (TS 3.6.1(1), items b. and d., respectively) helps provide early warning of any radioactive release from the pool that could be indicative of a fission product release from fuel failure, or that could be caused by an experiment failure in the pool. The NRC staff further finds that having stack gaseous and stack particulate monitors (TS 3.6.1(1), item a.) also helps provide warning of any radioactive material release within the reactor building, and allows for routine effluent monitoring. The NRC staff finds that the TS 3.6.1(1), item c., requirement for an area radiation monitor on the reactor experimental level, in conjunction with TS 3.6.1(2), helps provide warning of any excessive radiation fields near experimental facilities such as the beam tubes, gamma cave, or medical embedment.

The NRC staff finds that the changes UML proposes to make to its radiation monitoring system, discussed above, will provide individual alarms in the control room to satisfy the TS 3.6.1(1) requirement for alarm indicators in the control room for the stack and area radiation monitors. The NRC staff also finds that, although the CAM is outside the control room, because the CAM provides existing audible and visible alarms that are easily heard and seen in the control room, the CAM provides alarm indication in the control room that satisfies the TS 3.6.1(1) requirement for alarm indicators in the control room for the CAM. The NRC staff finds that the third floor CAM normally has at least one control room readout (on the ARMS) to satisfy the requirement for a control room readout for this monitor. The NRC staff also finds that, following its proposed radiation monitoring system changes, UML will normally have at least two redundant control

room readouts for the other TS 3.6.1(1)-required monitors (on the ratemeters and ARMS), either of which would satisfy the requirement for a control room readout for those monitors. The NRC staff finds that the additional radiation monitors, alarms and readouts (including local alarms and readouts, and facility-wide alarms), and automatic functions of the overall UMLRR radiation monitoring system discussed above, other than the specific monitors, alarms, and readouts required by TS 3.6.1, provide additional redundancy to ensure the effectiveness of the UMLRR radiation monitoring system. The staff, however, finds that additional TSs are not required for this equipment given the sufficient scope, discussed above, of the TS-required equipment and the operator's ability to monitor the TS-required minimum equipment and take actions as needed. The NRC staff notes that an operator may not necessarily be required (per renewed TS 6.1.3, which provides control room staffing requirements, and is discussed and found acceptable in SER Section 6.6.1) to be present in the control room for some circumstances where radiation monitors are required by TS 3.6.1(1), such as when radioactive material with potential for significant airborne release is being handled but the reactor is secured, as defined in the TS definitions (see renewed TS 6.1.2, which is discussed and found acceptable in SER Section 6.1.2). However, the NRC staff finds that local radiation monitor alarms and readouts, facility-wide alarms, the use of procedures for radioactive material handling, and the use of portable radiation monitors (consistent with standard industry practice, and as described in UML's letter dated September 30, 2020 (Ref. 98)) will help ensure that UMLRR staff are aware of any unusual radiation conditions and able to take appropriate action during any such circumstance. In addition, TS 3.6.1(2) separately requires an appropriate area monitor with a local readout and alarm when a gamma irradiation facility is in use (and there may not necessarily be an operator in the control room). The NRC staff also finds that by specifying minimum air and area radiation monitors needed to detect fission products and/or high radiation levels, TS 3.6.1(1) appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007. Therefore, based on the above and also based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.6.1(1) is acceptable.

TS 3.6.1(2) would require that each gamma irradiation facility have an operating area radiation monitor having a local readout and alarm indicator capable of alerting personnel near the facility when irradiations are being performed in that facility. The SAR, as supplemented, discusses two gamma irradiation facilities, the gamma cave and the medical embedment (or "ELDRS"), which are located on opposite sides of the reactor experimental level (reactor building first floor). Although TS 3.6.1(1), item c., requires an operating area monitor on the experimental level during certain conditions, per TS 3.6.1(1), these conditions would not necessarily include instances where irradiations not directly involving the reactor (e.g., gamma irradiations using the Co-60 sources or irradiated fuel elements) are taking place. Also, TS 3.6.1(1), item c., does not require that the experimental level monitor necessarily be associated with or near either of the two gamma irradiation facilities. Furthermore, TS 3.6.1(1), item c., does not require that the experimental level monitor have a local readout or alarm (when gamma irradiation facilities are in use, there would not necessarily be a reactor operator in the control room to monitor the control room alarms and readouts). Therefore, as discussed in UML's response to RAI-14.3.23.f, TS 3.6.1(2) requires area monitors at each gamma irradiation facility that can provide appropriate radiation monitoring any time a given gamma irradiation facility is in use. As discussed in UML's basis for TS 3.6.1, the purpose of the area monitor requirement in TS 3.6.1(2) is to ensure that personnel are aware that a gamma irradiation source is in use. The NRC staff notes that the same radiation monitor could, potentially, be used to meet TS 3.6.1(1), item c., and TS 3.6.1(2) simultaneously. The NRC staff finds that TS 3.6.1(2) helps ensure that an appropriate area monitor is operating (and able to alert UMLRR staff or experimenters as needed) when either gamma irradiation facility is in use, even when reactor or facility conditions

do not otherwise require (per TS 3.6.1(1)) operating radiation monitors. The NRC staff also finds that by requiring minimum area radiation monitors to detect high radiation levels, TS 3.6.1(2) appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007. Therefore, based on the above and also based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.6.1(2) is acceptable.

TS 3.6.1(3) would require that if a radiation monitor required by TS 3.6.1(1) or TS 3.6.1(2) becomes inoperable, operations for which the monitor is required may continue only if the required monitor is repaired or replaced with a monitor of similar function within 1 hour of discovery that the required monitor is inoperable. As discussed above, the UMLRR has multiple installed fixed monitors that could potentially serve as some of the TS-required monitors. During the 2020-2021 audit, as documented in the NRC staff's audit report dated December 17, 2020 (Ref. 86), UML stated that it would not need an allowance to replace TS-required fixed monitors with portable monitors, given the redundant fixed monitors that it has available. The NRC staff finds that TS 3.6.1(3) helps ensure that if a required monitor fails, it will be promptly repaired or replaced by another monitor providing a similar function (e.g., readout and alarm in the control room). The NRC staff finds that the allowance of 1 hour during which operations may continue before a monitor is repaired or replaced allows UML sufficient time to attempt to repair a monitor or find a replacement, while also minimizing the time without an operable monitor required by TS 3.6.1(1) and TS 3.6.1(2). The NRC staff finds that even if one TS 3.6.1(1)-required monitor is inoperable for up to an hour, other remaining operable TS 3.6.1(1) monitors, and other installed radiation monitors would continue to help meet the function of the overall radiation monitoring system (i.e., alerting the operator to elevated or unusual radiation levels or potential events such as a fuel failure resulting in a fission product release). In the case of gamma irradiation facility monitors required by TS 3.6.1(2), the NRC staff finds that if the one TS-required monitor is inoperable for a short period, other facility features such as interlocks (see SER Section 4.1.5) and other non-TS required monitors, including portable monitors, which are typically used in addition to fixed monitors at gamma irradiation facilities, consistent with standard industry practice, will help ensure that personnel are aware of any elevated radiation levels. The NRC staff also finds that by specifying allowances for repair and replacement of TS-required radiation monitors, and stating that operations may continue only if a monitor is repaired or replaced by a substitute monitor, TS 3.6.1(3) appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007. Therefore, based on the above and also based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.6.1(3) is acceptable.

Renewed TS 4.6, "Radiation Monitoring Equipment," would state, in part:

> Applicability:
>
> This specification applies to the surveillance requirements for the area radiation monitoring equipment and systems for monitoring airborne radioactivity.
>
> Objective:
>
> To ensure radiation monitoring equipment limiting conditions for operation are met.

Specifications:

(1) A channel test of the radiation monitoring channels in TS 3.6.1(1) shall be made prior to the first start of any TS 3.4.1 operation of the day.

(2) A channel test of the radiation monitoring channels in TS 3.6.1(2) shall be made prior to the first start of operation of the associated gamma irradiation facility of the day.

(3) The radiation monitoring channels required by TS 3.6.1(1) and TS 3.6.1(2) shall be calibrated and the trip set points verified when initially installed and annually thereafter.

….

TS 4.6(1) would require that prior to the first start of any operation specified in TS 3.4.1 (i.e., conditions when confinement is required) of the day, UML shall perform a channel test of each of the radiation monitoring channels required by TS 3.6.1(1), items a. through d. TS 4.6(2) would require that prior to the first start of any operation of a gamma irradiation facility of the day, UML shall perform a channel test of the radiation monitoring channel required by TS 3.6.2(2) and associated with that gamma irradiation facility. In the basis for TS 4.6, UML stated that the channel tests verify operability by introduction of a test signal (use of a check source). Additionally, in the basis for TS 4.6, UML stated that it does not intend TSs 4.6(1) or 4.6(2) to require re-testing of monitors each day if the activity requiring monitors is continuous over more than one day (e.g., if reactor operation or a gamma irradiation is continuous over more than 24 hours). The NRC staff finds that by requiring UML to perform channel tests of its required radiation monitors, TSs 4.6(1) and 4.6(2) help ensure that UML verifies its compliance with TSs 3.6.1(1) and 3.6.1(2) and that appropriate radiation monitoring equipment is available to warn reactor operators and staff of any potential radiation danger and thus enable operators and staff to take appropriate action or precautions. The NRC staff also finds that TSs 4.6(1) and 4.6(2) appropriately implement the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 by requiring periodic testing of radiation monitors. In addition, the staff finds that TSs 4.6(1) and 4.6(2) are conservative relative to the intent of the ANSI/ANS-15.1-2007, Section 4.7.1, item (1), recommendation that TSs require that radiation monitoring system operability be verified (where possible, including source checks) monthly to quarterly and the NUREG-1537, Appendix 14.1, Section 4.7.1 recommendation that TSs require that a channel check be performed daily prior to reactor startup and (where physically possible) a channel test using a radiation source be performed at least monthly. Therefore, based on the above and also based on its 10 CFR 50.36(c)(3) findings for TSs on surveillance requirements (SRs) in SER Section 6.7, the NRC staff concludes that TSs 4.6(1) and 4.6(2) are acceptable.

TS 4.6(3) would require that radiation monitoring channels required by TSs 3.6.1(1) and 3.6.1(2) be calibrated and the channels' alarm setpoints verified when the channels are initially installed and annually thereafter. UML discussed calibration of its fixed radiation monitors in SAR Section 11.1.4.2. In its basis for TS 4.6(3), UML stated that the calibration provides a complete verification of the performance of the channels and that an annual calibration is based upon manufacturer recommendations and is sufficient to ensure the required reliability. The NRC staff finds that the TS 4.6(3) requirement to calibrate radiation monitoring channels helps ensure that the channels are operating properly, providing an accurate indication of radiation levels, and fulfilling their intended function. The NRC staff finds that, although the UMLRR TSs do not include radiation monitor setpoints, UML's SR for setpoints helps ensure that trip setpoints are

consistent with intended settings (UML's typical setpoints are discussed earlier in this SER section) and that the monitors are able to perform their intended alarm functions and, therefore, the TS 4.6(3) requirement to verify trip setpoints is reasonable and appropriate. The NRC staff also finds that TS 4.6(3) appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 by requiring periodic calibration of radiation monitors, and that the annual surveillance interval in TS 4.6(3) is consistent with the guidance in NUREG-1537, Appendix 14.1, Section 4.7.1, and ANSI/ANS-15.1-2007, Section 4.7.1, item (2). Therefore, based on the above and also based on its 10 CFR 50.36(c)(3) findings for TSs on SRs in SER Section 6.7, the NRC staff concludes that TS 4.6(3) is acceptable.

As discussed in SAR Sections 11.1.4 and 11.1.6, in addition to using the UMLRR's fixed radiation detectors for monitoring radiation dose rates and reactor building and effluent air, UML also conducts routine radiation and contamination surveys at the UMLRR using other portable (e.g., handheld survey instruments such as Geiger-Mueller detectors) and/or analytical (e.g., gamma spectroscopy systems) instruments. UML stated that it performs these surveys to identify and limit the spread of contamination and ensure that radiation levels are ALARA. Work areas and equipment associated with radioactive material use or storage are surveyed with instruments appropriate to detect the applicable type(s) of radiation. Wipe tests are used for removable contamination surveys.

As discussed in SER Section 4.1.5, UML has a personnel monitoring program to help ensure that individual occupational worker doses are within 10 CFR Part 20 limits and ALARA. As discussed in SER Section 4.1.7, UML also conducts monitoring to measure environmental doses outside the reactor building.

The NRC staff reviewed the above information regarding UMLRR radiation monitoring and surveying. The NRC staff also reviewed the UMLRR inspection reports for the years 2006 through 2019 (Ref. 11) and found no significant issues related to radiation monitoring and surveying at the UMLRR. Based on the above, and the results of the NRC staff's inspections, the NRC staff finds that the types and placement of radiation monitors used, and the radiation surveys conducted, at the UMLRR are appropriate for the types and intensities of radiation likely to be encountered at the UMLRR. The NRC staff also finds that the surveillances performed on radiation monitors will help ensure that the monitors function correctly and accurately. The NRC staff finds that UML's proposed changes to its radiation monitoring system, discussed above, help ensure that the UMLRR has individual alarms (audible in the control room) for radiation monitors to immediately alert reactor operators to any potential radiation hazard and that the alarm requirement of TS 3.6.1(1) can be met. The NRC staff also finds that the proposed changes add redundancy by adding additional control room readouts for the TS 3.6.1(1)-required stack monitors. Furthermore, the NRC staff finds that the proposed changes eliminate the ARM CDAS, including the ARMS computer (which receives data for processing via the ARM CDAS controller) and HMI, as a single point of failure for the radiation monitoring system, because the TS-required radiation monitors will have individual control-room-audible alarms that are driven by ratemeters independently of the ARM CDAS, and with the exception of one TS 3.6.1-required monitor (the third floor CAM; see also SER Section 2.5.3), will also have control room readouts that are independent of the ARM CDAS. The NRC staff finds that there is reasonable assurance that expected types of radiation in significant intensities will be detected and that the radiation monitoring and surveying at the UMLRR will help ensure compliance with 10 CFR 20.1501(a) and 10 CFR 20.1501(c). Therefore, the NRC staff concludes that the radiation monitoring and surveying at the UMLRR, including the proposed changes to the radiation monitoring system, are acceptable.

In addition to the above findings and conclusion with respect to radiation monitoring and surveying at the UMLRR, in SER Section 3.7.2, the NRC staff makes additional findings specific to the ARMS instrumentation and control systems and concludes that these portions of the UMLRR instrumentation and control systems are acceptable.

4.1.5 Radiation Exposure Control and Dosimetry

The regulation, 10 CFR 20.1902, requires posting of radiation areas, high radiation areas, and very high radiation areas, as defined in 10 CFR 20.1003. The regulations in 10 CFR Part 20, Subpart G (i.e., 10 CFR 20.1601 and 10 CFR 20.1602), require control of access to high and very high radiation areas. The regulation, 10 CFR 20.1502, requires monitoring of doses to individual workers who are likely to receive an annual external radiation dose of more than 10 percent of the limits in 10 CFR 20.1201(a), or to receive an intake in excess of 10 percent of the applicable annual limits on intake (ALIs) in 10 CFR Part 20, Appendix B, Table 1, Columns 1 and 2.

Radiation exposure control and dosimetry are discussed in SAR Sections 4.4, 7.4.3.2, 7.7.8, 7.7.9, 10.2, and 11.1.5.

As discussed in SAR Section 4.4 and SER Section 2.4, the UMLRR pool water and concrete biological shield provide shielding from the reactor core (and Co-60 sources), limiting the dose rate in the reactor room. As discussed in SER Section 4.1.1.1, the dose from Ar-41 in the reactor room is limited by the facility ventilation system, which directs Ar-41 from experimental facilities directly out through the stack (so it is not released to the reactor room air), and also helps reduce the concentration of Ar-41 from other sources (e.g., argon dissolved in the reactor coolant) that is released to the reactor room. The N-16 dose in the reactor room is reduced by the use of the primary coolant system holdup tank when the reactor is operated in forced flow mode above 100 kWt. When the reactor is operated at 100 kWt or below in natural convection mode, the level of N-16 detected at the surface of the pool is negligible.

SAR Section 11.1.5.1 states that, during reactor operation at 1 MWt, the radiation levels above the pool under the reactor bridge are less than 10 mrem per hour. The radiation levels in most other non-experimental areas of the reactor building are less than 1 mrem per hour. Dose rates near ongoing experiments (e.g., certain beam tube experiments) may be greater than 1 mrem per hour at times; however, in such cases, shielding is used as required to reduce dose rates to acceptable levels. As discussed in SER Sections 4.1.1.1 and 4.1.1.2, dose rates in the pump room are also elevated during reactor operation, but shielding and access controls are used to minimize any resulting doses.

As discussed in SAR Sections 11.1.5.1 and 11.1.5.4, access to the reactor building is controlled, and unescorted access is only given to trained and approved individuals. Radiation doses to any visitors and other non-staff personnel who are not directly involved with the operation and/or use of the reactor are carefully controlled. Areas within the reactor building are posted to indicate radiation hazards. Areas are also controlled per 10 CFR Part 20, Subpart G, to help ensure personnel safety. The use of controls such as locked doors (both manual and automatic), flashing lights, and bells are also utilized within the reactor building where needed to help alert workers to potential radiation fields, such as in irradiation rooms or facilities. SAR Section 10.2 discusses various experimental facilities at the UMLRR, including irradiation facilities such as beam ports, the gamma cave, and the medical embedment (or “ELDRS”). SAR Sections 10.2, 7.7.8, and 7.7.9 also discuss specific radiation exposure controls associated with experimental facilities. Experimental facility exposure controls include, for

example, an interlock system which pneumatically locks the gamma cave when radiation sources (e.g., the Co-60 sources or irradiated fuel elements used for gamma irradiations) are in use, and which is in addition to other manual locks; light trees that alert facility staff to the status of the gamma cave and sources; manually and magnetically locked double doors to the medical embedment; and a control device which causes a reactor scram if personnel enter the beam port bunker area when the reactor is operating.

The NRC staff notes that additional discussion of exposure controls specifically related to the use of Co-60 sources with the gamma cave is in an NRC safety evaluation, dated February 5, 1998 (Ref. 70), which concluded that UML's gamma cave and Co-60 configuration for controlling access to high and very high radiation areas adequately addresses the requirements of 10 CFR 20.1601 and 10 CFR 20.1602 (i.e., 10 CFR Part 20, Subpart G).

As discussed in SER Section 4.1.4, UML has a radiation monitoring system to monitor dose rates, concentrations or airborne radioactive material, and radioactive effluents at the UMLRR and also conducts surveys of radiation and contamination levels using portable equipment. Renewed TS 3.6.1(2), which is discussed and found acceptable in SER Section 4.1.4, specifically requires radiation monitors on the reactor experimental level near in-use experimental facilities, helping to ensure that UMLRR staff are aware of any activities in an experimental facility that could result in elevated dose levels and can take appropriate action. SAR Section 10.2 discusses additional radiation monitors that are located in and around experimental facilities.

As discussed in SER Section 4.1.6, UML uses contamination control procedures to help control radiation exposures by limiting the spread of radioactive contamination. SAR Section 11.1.5.5 discusses protective clothing, face masks, and self-contained compressed air units that are available if needed to protect workers from contamination or airborne radioactivity.

UML stated in SAR Section 11.1.5.6 that based on historical data, it is unlikely that occupational doses at the UMLRR could exceed 10 percent of the 10 CFR Part 20 occupational dose limits. However, UML still provides external dosimetry badges (to measure deep dose and extremity dose) to individuals performing work at the UMLRR. UML uses a National Voluntary Laboratory Accreditation Program (NVLAP)-accredited vendor for its external dosimetry services. SAR Section 11.1.5.7 states that routine bioassays to evaluate internal dose will be performed on individuals who could receive a radioactive material intake in excess of 10 percent of ALI values, and bioassays may also be performed following abnormal events (e.g., an accidental injection or ingestion of radioactive material).

Renewed TS 6.8.3(3), which is discussed and found acceptable and SER Section 6.6.8, requires that radiation exposure records for all personnel monitored be retained for the life of the facility. The NRC staff reviewed the occupational dose records for 2009 through 2013 shown in SAR Section 11.1.5, Table 11-7, and also reviewed the occupational (whole-body and extremity) doses reported for the years 2008 through 2019 in UMLRR annual operating reports (Ref. 10), and noted that the doses are all well below 10 CFR Part 20 limits as well as the ALARA goals (10 percent of 10 CFR Part 20 limits) specified in the annual reports (in many cases, there was no measurable dose). The NRC staff finds that the reported occupational doses help demonstrate the effectiveness of UML's procedures and design features to control radiation exposure at the UMLRR. The NRC staff also reviewed the UMLRR inspection reports for the years 2006 through 2019 (Ref. 11) and found no significant issues related to radiation exposure control and dosimetry.

Renewed TS 3.2.4, “Radiological Protection Scrams,” would state:

Applicability:

This specification applies to reactor scrams associated with radiological protection.

Objective:

Radiological protection scrams are incorporated in the scram circuit to protect personnel, the public, and the environment from possible radiation exposure.

Specification:

The reactor shall not be operated unless the following radiological protection scrams described in Table 3.2.4-1 are operable.

Table 3.2.4-1
Radiological Protection System Scrams

	Scram	Function	Minimum Required
1.	Thermal Column Door Open	Scram if door limit switch open	1
2.	Beam port Chamber Door Open	Scram if door limit switch open	1
3.	First Floor Airlock Integrity	Scram if both doors unsealed	1
4.	Third Floor Airlock Integrity	Scram if both doors unsealed	1
5.	Truck Door Seal	Scram if door unsealed	1

TS 3.2.4, Table 3.2.4-1, items 1 and 2, would require that reactor scrams occur if the thermal column or beam port chamber (bunker) doors are opened during reactor operation. These doors have limit switches that generate these scrams. As discussed in SAR Section 7.4.3.2, although these scrams are part of the reactor protection system scram circuit, they have no function in reactor protection; they are solely incorporated to protect reactor staff and the public from possible radiation exposures. As discussed in SAR Section 10.2.9, the thermal column door, which moves on rails perpendicular to the thermal column shield face, serves as an additional shield to protect operating personnel against gamma radiation. As discussed above in this SER section and in SAR Section 7.7.9, the beam port bunker has a shallow door at its entrance for access control (the bunker has a labyrinthine design to prevent streaming though this door). Although SAR Section 7.7.9 states that the beam port chamber door scram only occurs when the reactor is operating above 1 kWt, in its letter dated September 30, 2020 (Ref. 98), UML clarified that the scram occurs during any reactor operation, consistent with the TS 3.2.4, item 2, requirement. The NRC staff finds that requiring that a reactor scram occur when either of these doors is opened will help ensure that UMLRR staff do not receive excessive radiation doses from accidental exposure to thermal column or entry into the beam port bunker during reactor operation. Therefore, based on the above and also based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.2.4, Table 3.2.4-1, items 1 and 2, are acceptable.

TS 3.2.4, Table 3.2.4-1, items 3, 4, and 5, would require that reactor scrams occur if both doors are unsealed on either of the first or third floor reactor building airlocks, or if the reactor building truck door is unsealed. SAR Section 6.2.1 states that access to the reactor building is provided by the two airlocks and the truck door. The airlock doors and the truck door have inflatable seals that are designed to minimize any air leakage between the reactor building and the outside. The airlocks are designed such that only one door is normally opened at a time, and personnel can enter and exit the reactor building while the reactor is operating without excessive air moving between the reactor building and the outside. The NRC staff finds that TS 3.2.4, Table 3.2.4-1, items 3, 4, and 5, help ensure that confinement integrity is maintained when the reactor is operating by limiting any air leakage from the building through the airlocks and truck door and that any contaminated air leaving the reactor building is directed out of the building via a controlled and monitored pathway (i.e., the ventilation system and stack). The NRC staff also finds that TS 3.2.4, Table 3.2.4-1, items 3, 4, and 5, help ensure that the reactor building can be maintained at a negative differential pressure during reactor operation, which reduces any other building leakage. Therefore, based on the above, and also based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.2.4, Table 3.2.4-1, items 3, 4, and 5, are acceptable.

The NRC staff reviewed the above information related to radiation exposure control and dosimetry. The NRC staff finds that UML described the radiation exposure controls, including design features and procedures, employed at the UMLRR. The NRC staff finds that these controls will help reduce occupational doses by shielding workers from radiation fields; preventing access to and/or alerting workers to areas with radiation fields; and limiting concentrations of Ar-41 in the reactor building. The NRC staff also finds that the use of dosimetry at the UMLRR is appropriate given the types and levels of expected radiation exposure. The NRC staff finds that the radiation control and dosimetry at the UMLRR will help ensure compliance with 10 CFR 20.1502; 10 CFR Part 20, Subpart G; and 10 CFR 20.1902. The NRC staff finds that historical dose data, and the results of UMLRR inspection reports, demonstrate the effectiveness of the radiation exposure controls and the dosimetry program at the UMLRR. Therefore, the NRC staff concludes that the radiation exposure control and dosimetry at the UMLRR are acceptable.

4.1.6 Contamination Control

Contamination control at the UMLRR is described in SAR Sections 11.1.4, 11.1.5.5 , and 11.1.6.

As discussed in SAR Section 11.1.4.1, UML performs surveys to identify and help limit the spread of radioactive contamination at the UMLRR. Work areas and equipment associated with radioactive material use or storage are surveyed for removable contamination (e.g., using wipe testing). As discussed in SAR Section 11.1.6, UML performs periodic routine contamination surveys at UML laboratories that use radioactive material, including the UMLRR. The frequency of these surveys depends on the quantity and frequency of radioactive material use. Contamination surveys would also be performed following any major radioactive material spill or release.

SAR Section 11.1.6 also states that UML posts, and maintains controlled access to, contamination zones for the purpose of contamination control. As discussed in SAR Section 11.1.5.5, UML has various protective clothing available for individuals who must work in contaminated areas. As discussed in SER Section 4.1.3, UMLRR workers receive training to allow them to recognize and protect themselves from sources of radiation, including radioactive contamination.

The NRC staff reviewed the above information related to contamination control. The NRC staff also reviewed the UMLRR inspection reports for the years 2006 through 2019 (Ref. 11) and found no significant issues related to contamination control, which helps demonstrate that adequate controls exist to prevent the spread of radiological contamination within the UMLRR facility. Based on its review of UML's practices to control contamination, and UML's history of satisfactory contamination control, the NRC staff concludes that the contamination control at the UMLRR is acceptable.

4.1.7 Environmental Monitoring

Environmental monitoring at the UMLRR is described in SER Section 11.1.7, as supplemented by UML's response to RAI-11.5 (Ref. 43).

As discussed in SAR Section 11.1.7 and the response to RAI-11.5.b, the environment outside the reactor building is monitored by passive optically stimulated luminescence dosimeters. These dosimeters are located within the first and third floor reactor building airlocks, as well as numerous locations throughout the attached Pinanski building (in the basement, on the first, second, and third floors, and on the roof). The dosimeters are processed by a NVLAP accredited vendor and the UML Radiation Safety Office analyzes the results to ensure that the reported doses are below 10 CFR Part 20 limits and to monitor for trends that would indicate unusual or elevated exposures. UML stated that it determined that the numbers and placement of environmental dosimeters is sufficient based upon historical data accumulated and analyzed from other dosimetry locations that were part of a previous comprehensive background study of areas around the UML campus.

SAR Table 11-9 lists total annual environmental monitoring dose results for 2009 through 2013 for dosimeters located in the first and third floor airlocks and two locations within the Pinanski building. For each year and location, the measured doses were below 10 mrem.

The NRC staff notes that the SAR does not identify any environmental sampling of soil, vegetation, or water performed outside the UMLRR. However, as discussed in SER Sections 2.3 and 4.1.1.2, reactor coolant is regularly monitored for radioactivity and any liquid effluents from the UMLRR are also monitored (before they are released from the liquid waste holding tanks in the Pinanski building) to ensure compliance with release criteria in 10 CFR Part 20. As also discussed in SER Section 2.3, leaked coolant water from past pool leaks collected in the reactor building sump, and water that collects in this sump is pumped to the liquid waste holding tanks. As discussed in SER Sections 4.1.1.1 and 4.1.4, air effluents are also monitored to ensure 10 CFR Part 20 compliance.

Renewed TS 3.6.1, "Radiation Monitoring," would state, in part:

> Applicability:
>
> This specification applies to the availability of radiation monitoring equipment which must be operable during reactor operation.
>
> Objective:
>
> To ensure that radiation monitoring equipment is available for evaluation of radiation conditions in restricted and unrestricted areas.

Specifications:

…

(4) There shall be an environmental monitoring program that shall include the placement of dosimeters, or other devices at points outside the reactor building.

TS 3.6.1(4) would require that UML conduct an environmental monitoring program using radiation measuring devices outside the reactor building. Although TS 3.6.1(4) does not require a specific number, type, or location for these devices (other than the requirement that they be outside the reactor building), as discussed above, UML stated that it determined that the actual number and placement of measuring devices is appropriate based on historical data. The NRC staff finds that TS 3.6.1(4) helps ensure that UML will continue to implement an appropriate environmental monitoring program. The NRC staff also finds that by requiring environmental monitoring, TS 3.6.1(4) appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007. Therefore, based on the above and also based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.6.1(4) is acceptable.

Renewed TS 4.6, "Radiation Monitoring Equipment," would state, in part:

Applicability:

This specification applies to the surveillance requirements for the area radiation monitoring equipment and systems for monitoring airborne radioactivity.

Objective:

To ensure radiation monitoring equipment limiting conditions for operation are met.

Specifications:

…

(4) Environmental monitor measurements shall be checked semi-annually.

TS 4.6(4) would require that UML check measurements from the radiation monitoring devices associated with its TS 3.6.1(4)-required environmental monitoring program at least semi-annually. In its TS 4.6 basis, UML stated that a semi-annual interval is adequate to ensure radiation doses in unrestricted areas are maintained within 10 CFR Part 20 annual limits. During the 2020-2021 audit, as documented in the NRC staff's audit report dated December 17, 2020 (Ref. 86), UML stated that it typically reads environmental monitors quarterly, but the semi-annual interval in TS 4.6(4) allows for flexibility. The NRC staff finds that TS 4.6(4) helps ensure that UML will check environmental monitor readings at intervals that are sufficient to evaluate trends and verify that annual doses in unrestricted areas outside the reactor building are within the annual public dose limit in 10 CFR Part 20. The NRC staff also finds that by requiring environmental monitors to be checked periodically (semi-annually), TS 4.6(4) appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007, including the specific recommended surveillance interval for environmental monitoring in ANSI/ANS-15.1-2007, Section 4.7.2, item (1). Therefore, based on the above and also based

on its 10 CFR 50.36(c)(3) findings for TSs on SRs in SER Section 6.7, the NRC staff concludes that TS 4.6(4) is acceptable.

Renewed TS 6.8.3(2), which is discussed and found acceptable in SER Section 6.6.8, requires that records of offsite environmental monitoring required by TSs be retained for the life of the facility. Renewed TS 6.7.1(6), which is discussed and found acceptable in SER Section 6.6.7, requires that annual operating reports include a summarized result of environmental surveys performed outside the facility.

The NRC staff reviewed the information above related to environmental monitoring. The NRC staff also reviewed the UMLRR inspection reports for the years 2006 through 2019 (Ref. 11) and found no significant issues identified related to the environmental monitoring program at the UMLRR. The NRC staff finds that, based on historical experience and the types of effluents that typically are routinely released from the UMLRR (i.e., Ar-41 and liquid radioactive waste, which are both monitored to ensure compliance with 10 CFR Part 20 limits), UML's environmental monitoring is sufficient to help ensure that any significant radiological impact on the environment from the UMLRR would be detected. Therefore, the NRC staff concludes that the environmental monitoring at the UMLRR is acceptable.

4.2 Radioactive Waste Management

4.2.1 Radioactive Waste Management Program

UML's radioactive waste management program is discussed in SAR Sections 11.1.1.2, 11.1.1.3, and 11.2, as supplemented by UML's responses to RAI-11.3 (Ref. 43) and RAI-11.6 (Ref. 71). As discussed in SER Sections 4.1.1 and 4.1.2, the UML RSP, for which the UML DRS/RSO has delegated management authority, oversees the management and safety of all radiation sources at the UMLRR, including radioactive waste. The UML RSP includes review, audit, and assessment provisions as discussed in SER Section 4.1.2.

In SAR Section 11.2.1, UML stated that, in accordance with the regulations in 10 CFR 20.2001, solid and liquid radioactive waste generated at the UMLRR is disposed of either by transfer to a person properly licensed to receive such waste via a licensing waste vendor or broker; release into sanitary sewerage in conformance with regulations; or decay in storage. (Gaseous radioactive waste is released as airborne effluents, as discussed in SER Sections 4.1.1.1 and 4.2.3.)

As discussed in SER Section 4.1.1.3, solid radioactive waste with a half-life of less than or equal to 120 days may be held for decay in storage at the UMLRR until the radioactivity of the material is statistically indistinguishable from background, and then released. Solid waste with a half-life of greater than 120 days (long-lived waste) is transferred to an authorized recipient (i.e., a disposal facility licensed to receive such waste via a licensed waste vendor or broker) for disposal, although quantities of this waste may be temporarily stored at the UMLRR until a sufficient quantity of the waste has been generated to warrant a transfer. Some long-lived solid waste may be first transferred to another UML license (and moved outside of the UMLRR licensed boundary for temporary storage), and then transferred to an authorized recipient for disposal from the other license.

As discussed in SER Section 4.1.1.2, liquid radioactive waste produced at the UMLRR that is intended for eventual sanitary sewer disposal is transferred to liquid waste hold-up tanks before being released to the sewer. These tanks are in the basement of the Pinanski building

connected to the reactor confinement building. When liquid waste is transferred to the hold-up tanks, it is considered to be transferred to UML's broad scope material license issued by the Commonwealth of Massachusetts (License No. 60-0049). Before any UMLRR liquid radioactive waste is transferred from the reactor sump (in the reactor building) to the hold-up tanks in the Pinanski building, UML cycles and analyzes it for the presence of radioactivity to ensure that the transfer would not cause the possession limits of the broad scope material license to be exceeded. Additionally, before the hold-up tanks in the Pinanski building are discharged to the sanitary sewer, they are also cycled, analyzed, and inspected to ensure that the radioactive material concentrations are within the limits for release to the sewer and that the radioactive material in the effluents in soluble in accordance with 10 CFR 20.2003. Certain liquid radioactive waste, such as liquid scintillation waste, may be transferred to an authorized recipient for disposal (and may be first transferred to another UML radioactive materials license before being transferred for disposal), similar to how long-lived solid waste is disposed of, rather than being disposed of by release to sanitary sewerage.

As discussed in UML's responses to RAI-11.3 and RAI-11.6, transfers of solid and liquid waste to UML's broad scope license are internal to UML, and there is no shipping of radioactive materials involved. The NRC staff notes that, in accordance with the regulations in 10 CFR 20.2108, 10 CFR 30.51, and 10 CFR 30.52, byproduct material licensees must maintain records for all radioactive waste disposed of under the byproduct license, and these records are subject to inspection by the NRC or agreement state, as appropriate. Byproduct material licenses, including UML's broad scope license, require that all waste handling and disposal conducted under the byproduct license be in accordance with NRC regulations.

Renewed TS 6.7.1(5), which is discussed and found acceptable in SER Section 6.6.7, requires that the UMLRR annual operating reports include a summary of radioactive effluents (gaseous and liquid) released under Facility Operating License No. R-125 for the UMLRR to environs beyond the effective control of UML. Based on the above discussion, the NRC staff notes that UML does not typically release liquid waste as effluent directly from the reactor license. However, if it did, TS 6.7.1(5) would require it to report such releases.

TS 6.7.1(5) does not specifically require that UMLRR annual operating reports include any summary of solid or liquid radioactive waste disposed of by transfer to an authorized recipient (either directly or by first transferring the waste to a UML license other than the reactor license). However, the NRC staff notes that the regulation, 10 CFR 20.2108, requires UML to maintain records of any radioactive waste disposal conducted under Facility Operating License No. R-125 for the UMLRR.

The NRC staff reviewed the above information related to UML's radioactive waste management program and practices. The NRC staff also reviewed the UMLRR annual reports covering operation from 2008 through 2019 (Ref. 10) and the UMLRR inspection reports for the years 2006 through 2019 (Ref. 11) and found that UMLRR radioactive material releases complied with TS and regulatory limits and that there were no significant issues related to UML's radioactive waste management program or radioactive waste handling or releases. Based on the descriptions of UML's RSP and radioactive waste management, as well as the historical effectiveness of UML's radioactive waste management program and procedures, the NRC staff finds that there is reasonable assurance that radioactive wastes will not escape UML's control and will not pose an undue risk of radiation exposure to UMLRR staff, the environment, or the public. Therefore, the NRC staff concludes that UML's radioactive waste management program, as implemented at the UMLRR, is acceptable.

4.2.2 Radioactive Waste Controls

UML's controls to minimize radioactive waste production and help avoid inadvertent exposure from, or release of, radioactive waste are discussed in SAR Section 11.2.

SAR Section 11.2.2 discusses radioactive waste minimization guidelines that UML incorporates into its programs for using radioactive materials (including at the UMLRR). UML stated that these guidelines include allowing decay-in-storage of shorter-lived radioisotopes; the substitution of short-lived for long-lived radioisotopes where possible to achieve waste minimization through decay-in-storage; limiting the use of consumables when using long-lived radioisotopes; segregating contaminated versus non-contaminated laboratory waste; and using appropriate decontamination techniques to reduce material that needs to be disposed of as radioactive waste.

As discussed in SAR Section 11.2.1, UML uses procedures and controls to avoid unnecessary exposures from or releases of radioactive waste. These include, for example, using appropriate packaging and labelling of waste containers; inspecting to ensure the integrity of waste containers; performing radiation and contamination surveys of waste containers; posting areas containing radioactive waste, as required; maintaining inventories of waste in storage; and limiting disposal of liquid radioactive wastes to sinks specifically designated for that purpose.

In SAR Sections 11.2.3.1 and 11.2.3.2, UML discussed types of solid and liquid radioactive wastes typically produced at the UMLRR. SAR Section 11.2.3.1 discusses activities of typical solid wastes and states that they are relatively small (e.g., µCi or millicurie levels). As discussed in SER Section 4.1.1.2, activities of liquid wastes have also historically been small, as reported in UMLRR annual reports.

The NRC staff reviewed the above information related to radioactive waste controls. The NRC staff finds that UML monitors and/or assesses radioactive wastes for radioactive material content; controls waste streams to help prevent uncontrolled exposures or releases and protect UMLRR staff, the public, and the environment; and makes appropriate efforts to minimize the amount of waste it produces. The NRC staff also reviewed the UMLRR inspection reports for the years 2006 through 2019 (Ref. 11) and found no significant issues related to radioactive waste controls at the UMLRR. Therefore, the NRC staff concludes that the radioactive waste controls at the UMLRR are acceptable.

4.2.3 Release of Radioactive Waste

SAR Sections 11.1.1.1, 11.1.1.2, and 11.2 discuss the release of radioactive waste from the UMLRR.

As discussed in SER Section 4.1.1.1, Ar-41 is the only radioactive airborne effluent routinely released from the UMLRR in any significant quantity, and the NRC staff finds that UML has shown that UMLRR Ar-41 releases are in quantities and concentrations that will not cause 10 CFR Part 20 dose limits to be exceeded. Renewed TS 3.6.2(2), which is discussed and found acceptable in SER Section 4.1.1.1, requires that maximum concentrations of UMLRR-produced Ar-41 at the location of the maximum exposed individual outside the UMLRR restricted area not exceed 10 CFR Part 20, Appendix B, limits. As discussed in SER Section 4.1.4, the UMLRR radiation monitoring system helps measure and identify gaseous effluents from the UMLRR, which are typically released through the UMLRR stack, helping provide verification that regulatory and TS effluent limits will not be exceeded.

As discussed in SER Section 4.1.1.2, UML primarily releases UMLRR liquid radioactive effluents by release to sanitary sewerage; prior to the release to sewerage, the effluents are first transferred to the hold-up tanks in the Pinanski building, where they become subject to the requirements of UML's broad scope radioactive material license. UML analyzes liquid effluents before they are discharged from the reactor building sump to the hold-up tanks to ensure that they may be possessed under the broad scope license and analyzes them again before they are discharged to the sewer to ensure that they comply with 10 CFR Part 20, Appendix B, concentration limits, and the 10 CFR 20.2003 solubility requirement. Renewed TS 3.6.2(1), which is discussed and found acceptable in SER Section 4.1.1.2, requires that liquid effluents meet 10 CFR Part 20 concentration and solubility requirements.

As discussed in SER Section 4.2.1, UMLRR solid waste, and some UMLRR liquid waste such as liquid scintillation waste that is long-lived and not held for decay-in-storage, is transferred to an authorized recipient (i.e., a disposal facility licensed to receive such waste via a licensed waste vendor or broker) for disposal. It may be disposed of from the UMLRR Facility Operating License No. R-125 or first transferred to UML's broad scope radioactive material license and then disposed of from that license.

As discussed in SER Sections 4.1.1.1 and 4.1.1.2, quantities of gaseous and liquid radioactive effluents from the UMLRR have historically been small, as reported in UMLRR annual reports. As discussed in SER Section 4.2.2, activities of solid wastes are also typically small.

The NRC staff reviewed the above information related to the release of radioactive waste and also reviewed the UMLRR inspection reports for the years 2006 through 2019 (Ref. 11), which did not identify any significant issues related to the release of radioactive waste at the UMLRR. The NRC staff finds that UML discussed the types of radioactive material releases expected from the UMLRR, the methods of the releases, and how UML assesses any consequences of the releases and ensures that they are in accordance with 10 CFR Part 20. Based on the information in SAR Sections 11.1.1.1, 11.1.1.2, and 11.2 and summarized in SER Sections 4.1.1.1, 4.1.1.2, 4.2.1, and 4.2.2 and above, the NRC staff finds that there is reasonable assurance that releases of effluents from the UMLRR will not exceed regulatory limits or pose unacceptable radiation risks to the public or environment. Therefore, the NRC staff concludes that the UMLRR releases of radioactive waste are acceptable.

4.3 Radiation Protection Program and Waste Management Conclusions

Based on its review of the information presented in the UMLRR SAR (as supplemented by responses to RAIs), its observations of the licensee's operations, its review of UMLRR annual operating reports, and the results of the NRC inspection program, the NRC staff concludes the following:

- The UMLRR RSP meets the requirements of 10 CFR 20.1101, is acceptably implemented, and provides reasonable assurance that the facility staff, the public, and the environment are protected from unacceptable radiation exposures. The RSP is acceptably staffed and equipped. The radiation protection staff has adequate lines of authority and communication to carry out the program.

- The UMLRR ALARA Program meets the requirements of 10 CFR 20.1101(b). A review of historical radiation doses and current controls for radioactive material at the UMLRR

provides reasonable assurance that radiation doses to the environment, the public, and facility personnel will be ALARA.

- The radiation monitoring and surveying and dosimetry programs at the UMLRR help ensure compliance with 10 CFR 20.1501 and 10 CFR 20.1502. The results of radiation surveys carried out at the UMLRR, doses to the persons issued dosimetry, and the results of the environmental monitoring program help confirm that the implementation of the radiation protection and ALARA programs is effective.

- UML adequately identified and described potential radiation sources. UML also sufficiently controls access to, and exposure from, radiation sources.

- The systems provided for the control of radioactive effluents are acceptable to ensure that releases of radioactive materials from the UMLRR are within the limits of NRC regulations and are ALARA.

- The radioactive waste management program provides reasonable assurance that radioactive waste produced at the UMLRR will be controlled and handled in accordance with applicable regulations, and the release of waste will not pose an unacceptable radiation risk to the environment and the public.

The NRC staff reviewed the UMLRR radiation protection program and radioactive waste management program as described in the SAR, as supplemented. The NRC staff finds that UML implements adequate and sufficient measures to minimize radiation exposure to facility staff and the public. Therefore, the NRC staff concludes that there is reasonable assurance that the UMLRR radiation protection and radioactive waste management programs will provide acceptable protection to UMLRR staff, the public, and the environment.

5. ACCIDENT ANALYSES

The University of Massachusetts Lowell (UML) Research Reactor (UMLRR) safety analysis report (SAR), as supplemented, provides a series of accident analyses to demonstrate that the health and safety of workers and the public are protected during analyzed reactor transients and other hypothetical accident scenarios. The accident analyses are located in SAR Chapter 13. UML summarized its accident-initiating events and scenarios in SAR Section 13.1 and described its analysis of the events and determination of consequences in SAR Section 13.2.

The accident analyses provide the basis for the UMLRR technical specifications (TSs) that are described and evaluated in this safety evaluation report (SER). The accident analyses also help ensure that no credible accident could lead to unacceptable radiological consequences to UMLRR facility staff, members of the public, or the environment.

UML's accident analyses include an analysis of the consequences of a maximum hypothetical accident (MHA), which is the postulated worst-case fuel failure scenario for the UMLRR and that would lead to the maximum potential radiation hazard to UMLRR personnel and members of the public from a fission product release. The results of the MHA are used to evaluate the ability of the licensee to respond to and mitigate the consequences of a postulated radioactive material release.

NUREG-1537 recommends that licensees consider the applicability of each of the following accident scenarios:

- MHA
- Insertion of excess reactivity
- Loss of coolant accident (LOCA)
- Loss of coolant flow
- Mishandling or malfunction of fuel
- Experiment malfunction
- Loss of normal electrical power
- External events
- Mishandling or malfunction of equipment

5.1 Maximum Hypothetical Accident

UML's MHA analysis is described in SAR Sections 13.1.1 and 13.2.1, as supplemented by the responses to request for additional information (RAI)-6.1 (Ref. 23) and RAI-13.5 (Ref. 43). UML stated that, although its analyses for other non-hypothetical events (e.g., reactivity transients and LOCAs) show no possibility for gross failure of fuel, its MHA analysis assumes that the cladding is completely stripped from one side of one fuel plate, resulting in a fission product release.

Accident Scenario

The MHA scenario assumes a non-mechanistic failure such that that the fuel plate with the highest power is damaged under water (i.e., in the reactor pool) to the extent that it loses the surface of its cladding on one side of the hot fuel plate and releases the fission fragment inventory from that surface into the reactor pool. The scenario assumes continuous full power reactor operation in forced convection mode prior to the failure, for a period of time sufficient to

reach saturation levels for fission product inventory. The licensee stated that, in postulating the hypothetical MHA, it did not consider the cause of the failure, the likelihood of the event, or the mechanism that would produce such a failure.

The amount of the plate activity that is available to be released depends on the burnup, power density, temperature, and the surface area of the fuel that is exposed. The fuel temperature, including during full power operation, is very low; therefore, diffusion from the fuel matrix is negligible. Consequently, the only radionuclides that would be available for release from the denuded surface would be due to the kinetic energy associated with fission fragment recoil.

UML assumed that the fuel failure within the pool results in the instantaneous dispersal, and uniform mixing, of the released fission products throughout the pool. The gaseous fission products (iodines and noble gases) in the pool are released to, and become uniformly mixed in, the reactor confinement air TS 5.1 minimum free volume of 9,486 cubic meters (3.35×10^5 cubic feet) (also instantaneously). Any particulate fission products released from the fuel remain dissolved in the pool. UML assumed that the elevated radiation readings generated by a fuel plate failure would be detected by multiple combinations of fixed radiation monitors. Renewed TS 3.6.1(1), which is discussed and found acceptable in SER Section 4.1.4, imposes requirements for radiation monitors in the reactor building. The high radiation alarms would automatically trigger reactor confinement ventilation isolation (or confinement isolation would be manually triggered by the operator, as discussed in SER Sections 4.1.4 and 6.3.4) and, as discussed in SAR Chapter 6 and UML's response to RAI-6.1, would also normally trigger automatic activation of the UMLRR emergency exhaust system (or it would be manually activated by the operator). UML assumed that the reactor would also be scrammed, either due to the operator's declaration of an actual local radiation emergency alarm (LREA) or general radiation emergency alarm (GREA) (see SER Section 4.1.4), or the operator pressing the control room manual scram button.

As discussed in SAR Chapter 6, as supplemented by UML's response to RAI-6.1, the emergency exhaust system operates independently of the normal ventilation system. The emergency exhaust system routes air from the confinement, through high efficiency particulate air (HEPA) and charcoal filters, and out through the stack, at a lower flow rate and via a separate flow path than the normal ventilation system (see SAR Figure 6-2). The emergency exhaust system is designed to generally allow a negative differential pressure to continue to be maintained inside the reactor confinement.

After the normal ventilation system is shut down, the reactor operators would evacuate the confinement building. As discussed in SAR Section 7.7.5, the evacuation alarm horns are manually activated by the reactor operators. As discussed in SER Section 4.1.4, reactor operators can also make announcements to warn personnel in the reactor building as needed. Although evacuation drills have demonstrated that personnel within the reactor building can be evacuated within 5 minutes, it is conservatively assumed that it takes 10 minutes to evacuate the building.

UML described four scenarios for evaluation of doses from its MHA—Scenario A, Scenario B, Scenario C, and Scenario D.

Scenario A considers the dose to UMLRR staff in the reactor building from inhalation of and submersion in the radionuclides released to the reactor building air.

Scenario B considers the dose to members of the public located outside, but near, the reactor building (i.e., near the reactor building truck door or Pinanski building airlock doors, or at another location near the exterior building wall but away from the doors). Scenario B assumes that normal ventilation shuts down, but that the emergency exhaust system fails to activate and the receptors are exposed to external radiation from the radioactive material held up in the air inside the reactor confinement building.

Scenario C considers the dose to members of the public located near the reactor building truck door, as well as to members of the public located at the nearest residence. Scenario C assumes that normal ventilation shuts down, but that the emergency exhaust system fails to activate, and additionally assumes that the negative differential pressure inside the reactor building is lost and radioactive material leaks out of the building truck door. In Scenario C, receptors receive dose from both the external radiation from the radioactive material held up in the air inside the reactor confinement building and the inhalation of and submersion in radioactive material leaked through the truck door (i.e., at ground level). In response to RAI-6.1, UML stated that in the unlikely event of a substantial overpressure of the reactor building (greater than approximately positive 0.5 inch water column (0.12 kilopascals)) relative to outside pressure, the emergency exhaust system does not continue to operate to relieve pressure (to attempt to return the building to negative differential pressure) because the charcoal filter in the emergency exhaust pathway is not intended for such a high volume, high pressure release.

Scenario D also considers to the dose to members of the public located near the reactor building truck door and stack (UML determined this location, approximately 10 meters from the truck door or base of the stack, to be the maximum public dose location), as well as to members of the public located at the nearest residence. Scenario D assumes that the ventilation system operates as designed (normal ventilation shuts down and the confinement building is isolated; the emergency exhaust system activates, maintaining the negative differential pressure in the reactor building; and the intake fan provides dilution air that is mixed with the air that the emergency exhaust system removes from the reactor building). In Scenario D, receptors receive dose from both the external radiation from the radioactive material in the air inside the reactor confinement building and the inhalation of and submersion in the radioactive material released from the 100 foot (30.5 meter) stack via the emergency exhaust system.

For evaluation of doses to the public, Scenario D, which assumes accident-mitigating functions generally work as designed and required by TSs (e.g., TS 3.5 requirements for ventilation equipment), represents the most probable progression of the MHA event; Scenarios B and C represent calculations that introduce additional conservatism by assuming the failure of certain functions, or an overpressure event, as discussed above.

The U.S. Nuclear Regulatory Commission (NRC, the Commission) staff reviewed the information discussed above related to UML's MHA scenarios. The NRC staff finds that the MHA Scenarios A, B, C, and D are consistent with, or more conservative than, the design and operation of the UMLRR within the bounds of the TSs; are reasonably consistent with MHAs for similar research reactor facilities; and would reasonably bound any credible fission product release event at the UMLRR. For Scenarios B, C, and D, which are postulated to evaluate public dose, the NRC staff finds that UML's Scenario D represents the most probable progression of the MHA event; Scenarios B and C represent unlikely scenarios in which TS-required equipment fails, or there is a unlikely substantial overpressure of the reactor building. Based on its review of the above information, the NRC staff finds that UML's selection of MHA scenarios is acceptable.

Nuclide Inventory

UML determined the saturated radionuclide inventory in the highest power fuel plate using the fuel plate power, and fission product yields for uranium-235 (U-235). UML stated that it assumed a peak plate power of 6.94 kilowatts-thermal (kWt) for this calculation. This peak power is based on an 18-fuel element core with a total radial peaking factor of 2. UML performed its inventory calculation for select iodine and noble gas (krypton and xenon) radionuclides.

The NRC staff performed a calculation of the saturated iodine and noble gas radionuclide inventory in the hot fuel plate to determine if it could confirm UML's calculation. The NRC staff used similar assumptions as UML and used cumulative fission product yields for thermal U-235 fission from the Korea Atomic Energy Research Institute Nuclear Data Center (Ref. 56).

The UML- and NRC-calculated inventories, in curies (Ci), are listed in SER Table 5-1 below.

Table 5-1 Estimates of the MHA Nuclide Inventory

Nuclides	UML Calculations (Ci)	NRC Staff Calculations (Ci)
I-131	168	167
I-132	250	250
I-133	390	390
I-134	454	450
I-135	366	366
Kr-83m	33.6	31.3
Kr-85	16.0	15.9
Kr-85m	75.1	73.3
Kr-87	149	146
Kr-88	205	208
Kr-89	232	268
Kr-90	283	293
Xe-131m	1.69	1.84
Xe-133	390	390
Xe-133m	9.31	11.3
Xe-135	380	380
Xe-135m	105	70.4
Xe-137	353	355
Xe-138	320	370
Xe-139	293	299

The NRC staff reviewed UML's calculated hot plate fission product inventory calculations and compared the results of the calculations to the NRC staff's results. The NRC staff finds that the UML- and NRC-calculated values are reasonably similar, given the uncertainties in fission product yield values and the differences in values from different sources. The NRC staff also finds that UML assumed a peak plate power that is higher (more conservative) that the peak plate power associated with the UMLRR limiting core configuration (LCC). Based on the information discussed in SER Section 2.5.1, the peak element power in the LCC is 69.602 kWt. Dividing this by 16 fuel plates and multiplying by the intra-element radial peaking factor for the LCC hot element (1.364), gives a peak plate power of 5.93 kWt for the LCC; therefore, UML's use of 6.94 kWt in its MHA inventory calculation is conservative. Based on the above

information, including the results of the NRC staff's confirmatory inventory calculation, the NRC staff finds that UML's MHA inventory calculations are acceptable.

Release Fractions

UML assumed that 100 percent of the fraction of noble gases and iodines in the failed fuel plate that are within the fission fragment recoil distance from the denuded surface of the plate are released to the reactor pool. UML used a recoil distance of 1.37×10^{-3} centimeters from NUREG/CR-2079, "Analysis of Credible Accidents for Argonaut Reactors" (Ref. 61). Using this recoil distance and the dimensions of the fuel plate, UML calculated a fission product release fraction of 0.027 for the fuel plate.

UML further assumed, in its MHA analyses, that 100 percent of the noble gases released to the pool are instantaneously released to the reactor building air (i.e., no noble gases remain dissolved in the pool). For the iodines, UML assumed a release fraction of 0.005 for the release from the pool to the reactor building air. UML stated that it based this assumption on the NRC regulatory guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Ref. 62), Appendix B, assumption that if water height above the core is 23 feet (7.0 meters) or greater, then 99.5 percent of the total iodine released by a light water power reactor fuel rod is retained by the pool water. UMLRR TS 2.2.1(4) requires the UMLRR to have a limiting safety system setting (LSSS) for pool water height above the core centerline of 24.25 feet (7.4 meters) or greater; the normal water height is approximately 25 feet (7.6 meters).

UML's release fractions, discussed above, are summarized in SER Table 5-2 below.

Table 5-2 MHA Release Fractions

Group	Release Fraction from Hot Plate to Pool	Release Fraction from Pool to Confinement Air	Total Release Fraction from Hot Plate to Confinement Air
Noble gases	0.027	1.0	0.027
Iodines	0.027	0.005	0.000135

The NRC staff reviewed UML's MHA hot plate release fraction. The NRC staff notes that, although the UMLRR is not an Argonaut reactor, both the UMLRR and Argonaut reactors use aluminum matrix fuels. The NRC staff finds that, due to the similarities in the fuels, the NUREG/CR-2079 recoil range assumed by UML is appropriate for aluminum matrix fuel such as the silicide or aluminide fuel used in the UMLRR and that the use of this recoil range is also consistent with established industry and NRC-accepted practice for similar materials test reactor (MTR)-type research reactors. The NRC staff also performed a calculation to independently verify UML's calculation of the fuel plate release fraction (0.027) using the recoil range and the fuel plate dimensions and obtained a similar result.

The NRC staff also reviewed UML's release fractions from the pool to the confinement air. The NRC staff finds that the release fraction of 1.0 for noble gas release to confinement is conservative because it assumes no noble gases remain dissolved in the pool.

The NRC staff notes the RG 1.183, Appendix B, iodine release fraction of 0.005 is for a release from a power reactor spent fuel pool to the air above the pool for a fuel handling accident involving power reactor fuel, which has a gap between the fuel and cladding. The RG 1.183 scenario that uses this release fraction involves an instantaneous release of the entire gap activity to the pool, followed by the instantaneous release of 0.5 percent (0.005 release fraction) of the iodines to the air above the pool. The RG 1.183 release fraction assumes that the chemical form of iodine released from the fuel to the pool is 95 percent cesium iodide (CsI), 4.85 percent elemental iodine (I_2), and 0.15 percent organic iodine. It is assumed that, although the very soluble CsI completely dissociates to cesium and iodide ions in the pool water, the pH of the water is low enough that some iodide ions can be converted to, and evolve from the pool as, volatile I_2. The decontamination factor (due to the pool water scrubbing iodine from bubbles as they rise the 23 feet (7.0 meters) to the pool surface) is assumed to be 500 for I_2, which comprises 99.85 percent of the iodines, and 1 (i.e., no decontamination) for the organic iodine, which comprises 0.15 percent of the iodines, giving an approximate overall decontamination factor of 200 (i.e., 0.005 iodine release fraction).

The NRC staff notes that the operating temperature of fuel at a research reactor such as the UMLRR is much lower than the operating temperature of power reactor fuel. The NRC staff also notes that UMLRR fuel has a much lower fission product inventory than a typical power reactor core, resulting in less gamma radiation. The NRC staff finds that these UMLRR conditions could help reduce the likelihood of organic iodine (which is more likely than I_2 to be released from the pool) formation in the fuel.

The NRC staff previously found (Ref. 63) that if the conductivity of open-pool research reactor pool water is controlled such that it remains below 5 micromhos per centimeter, the pool water pH will remain between 5.6 and 5.8. UMLRR TS 3.3(1) requires that the pool water conductivity shall be less than 5 micromhos per centimeter when averaged over a month, and UMLRR TS 3.3(2) requires that the pH of the pool water shall be between 5.0 and 7.5 when averaged over a month. In comparison, for power reactor spent fuel pools, an assumption of a pH of about 5 is generally appropriate (Ref. 64). Therefore, the NRC staff finds that the pH of UMLRR pool water is typically similar to or greater than the assumed pH for power reactor spent fuel pool water. The NRC staff notes that higher pH is generally associated with greater retention of iodine in the water, because iodine is more likely to exist as iodide ions and less likely to be in the volatile I_2 form that may be evolved from the pool.

The NRC staff notes that while the power reactor fuel considered in RG 1.183, Appendix B, has a gap between the uranium fuel and cladding, an MTR-type research reactor fuel (such as UMLRR fuel) matrix is clad in aluminum without any gap. The NRC staff finds that this difference either would likely not significantly affect the iodine released from the pool, or would reduce the iodine released from the pool, because fission product gas bubbles released from UMLRR fuel could be smaller (given that they are not released from a gap), and fission products in smaller gas bubbles would be more readily scrubbed by or dissolved in the coolant.

The NRC staff considered the UMLRR fuel dimensions and the TS 2.2.1(4)-required 24.25 foot (7.4 meter) LSSS for minimum water height above the core centerline, and finds that the tops of the fuel plates in the UMLRR core must have approximately 23 feet (7.0 meters) or more of pool water above them during reactor operation.

Based on the above information regarding the differences between UMLRR fuel in the UMLRR pool and power reactor fuel in a spent fuel pool, the NRC staff finds the use of the 0.005 release fraction in RG 1.183, Appendix B, for iodine released from the UMLRR pool to the reactor

building air to be reasonable and conservative. Although RG 1.183, Appendix B, assumes the release from the pool to be instantaneous, the NRC staff notes that some of the release from the UMLRR pool would likely occur over a period of time following an initial release. However, the NRC staff finds that assuming that a 0.5 percent release occurs instantaneously is reasonably bounding, because of the UMLRR conditions discussed above and because the iodine concentration in the pool would be reduced over time by radioactive decay and the primary coolant cleanup system, reducing the iodine available for subsequent release. The NRC staff also finds that an 0.005 release fraction for iodine released from a research reactor pool to the air above it is similar to assumptions used in analyses at similar research reactor facilities. Additionally, the NRC staff notes that UML used other conservative assumptions (e.g., no iodine plate-out in the reactor building or ventilation system) in its calculations of occupational and public doses from iodine. As discussed in SER Section 6.3.5, it is not clear whether UML credited the emergency exhaust system charcoal filter in its analyses, but the effect of crediting this filter would be small (UML did not credit any HEPA filters).

Based on the above, the NRC staff finds that UML's MHA release fractions for noble gases and iodines from the hot plate to the pool, and from the pool to confinement air, are acceptable.

Atmospheric Dispersion

To determine the radionuclide air concentrations used to calculate doses at public receptor locations outside the reactor building from radioactive materials released from the truck door or stack, UML calculated atmospheric dispersion parameters (chi over Q (X/Q) values) using the ARCON96 computer code, which calculates X/Q values for 95th percentile weather conditions. UML's MHA calculations used wind rose data for Hanscom Air Force Base for 2013 and the facility building and stack heights. Also, because UML lacked historical Pasquill atmospheric stability class data for the UMLRR site, the UML MHA atmospheric dispersion analysis included calculations for three different Pasquill atmospheric stability classes (A, D, and F), assuming that each class occurs 100 percent of the time, in order to determine the class that would result in the highest dose for each receptor location. These calculations and assumptions are similar to those discussed and found acceptable related to argon-41 (Ar-41) in SER Section 4.1.1.1. UML stated that its MHA model considered ARCON96-calculated averaged X/Q values representing 0 to 2 hour, 2 to 8 hour, 8 to 24 hour, and 1 to 30-day time intervals (i.e., averaging intervals).

The NRC staff notes that, because UML performed 95th percentile X/Q calculations using an assumption that a worst-case stability class occurs for 100 percent of the year, the X/Q values determined by these calculations are greater than and thus more conservative than 95th percentile values.

For the NRC staff's MHA public dose calculations, which the NRC staff performed to determine if it could confirm the results of UML's calculations, and which are discussed later in this section, the NRC staff used a conservative generic X/Q of 0.01 seconds per square meter, equal to the value assumed for near-field receptors outside a reactor building during accident conditions in NUREG/CR-2387, "Credible Accident Analyses for TRIGA and TRIGA-Fueled Reactors" (Ref. 55), along with an assumption that the wind blows in the direction of the receptor 100 percent of the time.

The NRC staff reviewed UML's atmospheric dispersion calculations for its MHA. The NRC staff finds that the ARCON96 code is adequate for estimating doses to near-field receptors outside a research reactor building and that UML's assumptions for its MHA atmospheric dispersion

calculations, as discussed in the SAR, as supplemented, appear to be reasonable. The NRC staff finds that UML's use of a range of different ARCON96 averaging intervals, as discussed above, is generally consistent with UML's public dose calculations for radioactive material releases from the reactor building that assume material is released from the building over 30 days (Scenario C) or 18 hours (Scenario D), as discussed later in this section. Based on the above information, as well as on the results of the NRC staff's confirmatory calculations using a conservative generic X/Q, discussed below, the NRC staff finds that UML's MHA atmospheric dispersion calculations are reasonable and conservative and, therefore, acceptable.

Dose Conversion Factors

UML stated that, for its occupational and public MHA dose calculations, for most radionuclides, UML used dose conversion factors (DCFs) from Federal Guidance Report No. 11 (FGR 11), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (Ref. 57), and Federal Guidance Report No. 12 (FGR 12), "External Exposure to Radionuclides in Air, Water, and Soil" (Ref. 58). UML stated that, for four radionuclides, krypton-89 (Kr-89), krypton-90 (Kr-90), xenon-137 (Xe-137), and xenon-139 (Xe-139), it used DCFs from other sources, including NRC RG 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I" (Ref. 59), or DOE-EH-0700, "External Dose-Rate Conversion Factors for Calculation of Dose to the Public" (Ref. 60).

The NRC staff notes that FGR 11, FGR 12, RG 1.109, and DOE-EH-0700 do not include DCFs for Xe-139. However, the NRC staff also notes that acceptable DCFs may be obtained from other sources, by using bounding values (i.e., generic values, or values for similar radionuclides that emit radiation of similar types and energies), or by calculation. The NRC staff compared the calculated Xe-139 whole-body deep dose equivalent (DDE) in Table 3 of UML's response to RAI-13.5 to UML's calculated whole-body DDEs for Kr-90 in Table 3 of UML's response to RAI-13.5 (UML labelled these doses as whole-body submersion total effective dose equivalents (TEDEs) in the table) and to the released noble gas activities in Table 2 of UML's response to RAI-13.5. Based on these comparisons, the NRC staff determined that, given UML's Xe-139 dose result and the energies and types of radiation emitted by Xe-139 compared to Kr-90, UML appears to have used a reasonable Xe-139 whole-body submersion DCF in its calculations. The NRC staff also determined that UML's Xe-139 DCF appears to be reasonably consistent with parameters used for calculating Xe-139 dose at similar facilities.

The NRC staff reviewed UML's sources for its DCFs, as discussed above. The NRC staff finds that the use of DCFs from these sources is reasonable and consistent with established industry and NRC-accepted practice and, therefore, acceptable.

Occupational Dose Calculations

For its MHA occupational dose calculations (MHA Scenario A), UML calculated:

- the committed dose equivalent (CDE) to the thyroid (organ dose to the thyroid from radioactive material inhalation and uptake by the thyroid);
- the DDE to the thyroid (organ dose to the thyroid from submersion in radioactive material);
- the total organ dose to the thyroid;

- the committed effective dose equivalent (CEDE) (internal whole-body dose from inhalation of radioactive material and uptake by the thyroid);
- the whole-body DDE (external whole-body dose from submersion in radioactive material); and
- the TEDE (total whole-body dose from internal and external sources).

UML stated that, for its occupational dose calculations, it assumed that a receptor is located in the reactor room for 10 minutes following the release of the radioactive iodines and noble gases to the reactor building, and it assumed that the reactor building is isolated (i.e., no leakage or exhaust of radioactive air during the 10 minutes), which is conservative because the emergency exhaust system (required by renewed TS 3.5(3), which is discussed and found acceptable in SER Section 6.3.5) would normally be removing radioactive air from the building. In SAR Sections 13.1.1 and 13.1.2, UML stated that radioactive decay is not considered for its occupational dose calculations. In its response to RAI-13.5, UML revised its occupational dose calculation by adding the consideration of three additional short-lived noble gas radioisotopes, Kr-90, Xe-137, and Xe-139, but did not appear to specify whether radioactive decay was considered for the additional noble gases. The results of UML's occupational dose calculations are shown below in SER Table 5-3.

The NRC staff performed a calculation to determine if it could confirm UML's calculations of the UMLRR MHA occupational doses. The NRC staff generally used assumptions similar to those used by UML, but the NRC staff did not consider radioactive decay for all radioisotopes, and the NRC staff used the NRC-staff calculated hot plate inventory from SER Table 5-1. The NRC staff also used: DCFs from DOE-EH-0700 for Kr-89, Kr-90, and Xe-137; DCFs from FGR 12 for other noble gases (except Xe-139) and for iodine submersion; and DCFs from FGR 11 for iodine inhalation. For Xe-139, the NRC staff used the Kr-90 DCFs to be representative of Xe-139 DCFs, because the NRC staff finds that this would provide a reasonably conservative estimate of the Xe-139 dose based on the types and energies of radiation emitted by Kr-90 and Xe-139. The results of the NRC staff's occupational dose calculations are also shown below in SER Table 5-3.

Table 5-3 UML- and NRC-Calculated MHA Occupational Doses

Dose Parameters	UML Calculation (mrem)	NRC Confirmatory Calculation (mrem)	10 CFR 20.1201 Dose Limit (mrem)
Thyroid CDE	747	752	50,000
Thyroid DDE	600	771	50,000
Total Thyroid Organ Dose	1,347	1,523	50,000
CEDE	23.7	23.9	5,000
Whole-Body DDE	517	738	5,000
TEDE	540	762	5,000

The NRC staff reviewed UML's occupational MHA dose calculation and results. Based on its review of Table 3 of UML's response to RAI-13.5, and a comparison to the results of the NRC staff's calculation, the NRC staff notes that UML appears to have considered radioactive decay in its evaluation of doses from the short-lived Kr-90, Xe-137, and Xe-139. The NRC staff recognizes that although radioactive decay would significantly reduce the dose from these noble gases over the 10 minute period, consideration of radioactive decay without consideration of

daughter products may not provide a bounding result because it ignores the dose contribution from the daughter products of these noble gases. However, the NRC staff finds that the additional daughter product dose would be relatively small and that the exclusion of this dose contribution is acceptable given the other conservative assumptions in the calculation, such as the 10 minute stay time, the use of a semi-infinite hemispherical cloud model, and the instantaneous release of fission products to the reactor room air. (Although SAR Section 13.2.1.2 states that facility personnel are exposed to a finite, hemispherical cloud of fission products in the reactor building, the NRC staff notes that the submersion DCFs which are used in UML's and the NRC staff's calculations are based on the conservative assumption of a semi-infinite hemispherical cloud.)

During its review of Table 3 of UML's response to RAI-13.5, the NRC staff also noted that UML did not list a thyroid submersion dose (DDE to the thyroid, but listed as "Thyroid Submersion CDE" in the table) for Xe-139. However, the NRC staff notes that the value for total DDE to the thyroid (600 millirem (mrem)) in Table 3 of UML's response to RAI-13.5 appears to include a Xe-139 contribution, because the total thyroid DDE value is greater than the sum of the contributions for other radionuclides listed in Table 3 of UML's response to RAI-13.5.

The NRC staff compared the results of UML's calculations to the results of the NRC staff's calculations, as shown in SER Table 5-3. The NRC staff finds that the results are reasonably similar and that the differences are likely primarily due to UML's consideration of radioactive decay for three short-lived noble gases compared to the NRC staff's conservatively ignoring all radioactive decay.

Based on the above, the NRC staff finds that UML's MHA occupational dose calculation assumptions and methodology are generally reasonable and conservative. The NRC staff also finds that the results of UML's calculations and the NRC staff's calculations confirming UML's results are reasonably similar and are well below the occupational dose limits in 10 CFR 20.1201. Therefore, the NRC staff finds that UML's MHA occupational dose calculations are acceptable.

Public Dose Calculations

As discussed above, for its MHA public dose calculations, UML considered the following three scenarios: Scenario B, in which all radioactive material remains in the reactor building and the public dose outside the building is due to external gamma radiation passing through the building doors or walls; Scenario C, in which the public dose is from radioactive material leaking out the truck door as well as from external radiation from material in the building; and Scenario D, in which the public dose is from radioactive material released through the stack as well as from external radiation from material in the building.

For its public dose calculations, UML calculated the TEDE as the sum of the external dose from radioactive material in the building, the external dose from submersion in material released from the building, and the internal dose from material released from the building and inhaled.

For calculating the external dose outside the reactor building from radioactive material inside the building, UML used a Monte Carlo N-Particle Transport (MCNP) model that it constructed for the building, similar to the methodology that it used for its Ar-41 calculations, as discussed and found acceptable in SER Section 4.1.1.1. UML stated that its MCNP models are conservative in that they ignore many concrete structures in the building (including floors) that would serve as additional shielding.

For Scenarios B and C, UML assumed that a member of the public would occupy a receptor location continuously for 30 days following the occurrence of the MHA. For Scenario D, UML assumed that the member of the public would occupy a receptor location continuously for 18 hours, which is enough time for the emergency exhaust system to exhaust all radioactive material from the building and for the entire plume of released material to pass the receptor.

For Scenario C, UML assumed that the building air containing radioactive material leaks through the truck door at a rate of 1 percent per day. UML stated that it considers this to be a conservative estimate based on engineering judgment and that it represents a leak rate 200 times greater than would be expected to occur for a 1 inch gap around the truck door and a pressure differential of about 7 kilopascals (28 inches of water) representing an extreme weather condition. UML stated that, in an overpressure scenario, leakage would preferentially be directed through the building stack to cause an elevated release, but that some leakage could also occur through the truck door and the 1 percent assumption is intended to bound any such leakage. For Scenario D, UML assumed that the radioactive air in the building is exhausted at the emergency exhaust system flow rate of 320 cubic feet per minute, which causes one complete air change in approximately 18 hours (the NRC staff notes that although the rate of air removal from the reactor building is smaller when the emergency exhaust system is operating compared to normal ventilation operation, as discussed in SER Section 6.3, the total flow rate from the stack does not change significantly because the intake fan normally continues to operate to dilute the air removed from the building by the emergency exhaust system). For the purposes of the calculations of external dose outside the reactor building from radioactive air inside the building, for the 30-day-duration Scenario C, UML assumed that radioactivity in the building decreases as the building leaks. For the shorter-duration Scenario D, UML did not appear to specify whether it assumed radioactivity in the building decreases with building leakage.

UML indicated that it did not consider three noble gas radioisotopes, Kr-90, Xe-137, and Xe-139, in its public dose calculations, because they are short-lived (less than 4-minute half-life). UML also assumed that the dose from the daughter products of these radioisotopes would not be significant, because the daughter products are either stable or would plate out or be captured by the emergency exhaust system filters. (For Scenario D, in which building air is exhausted via the emergency exhaust system and through filters, UML did not otherwise credit the ability of the HEPA filter, or roughing filter that the air also flows through, to reduce the radionuclide concentrations in the exhaust.)

In response to RAI-13.5.e and RAI-13.5.l, UML stated that radioactive decay is not considered for Scenarios C and D, in which radioactive material is released from the reactor building. However, the NRC staff notes that, based on its review of the MHA results shown in Tables 4, 5, and 6 of UML's response to RAI-13.5, radioactive decay appears to have been considered for the Scenario B, C, and D calculations of external dose outside the reactor building from radioactive air inside the building.

The NRC staff performed a calculation to determine if it could confirm UML's estimates of the public dose from radioactive material released from the reactor building. The NRC staff generally used assumptions similar to those that UML used for its public dose calculations, except that the NRC staff used the NRC-staff calculated hot plate inventory from SER Table 5-1, used a very conservative generic X/Q of 0.01 seconds per square meter from (Ref. 55) with an assumption that the wind blows in the direction of the receptor during 100 percent of the release period, and did not consider the dose contribution from four short-lived (less than 4

minute half-life) noble gas radioisotopes (Kr-89, Kr-90, Xe-137, and Xe-139) (whereas UML did not consider the dose contribution from three of these radioisotopes, Kr-90, Xe-137, and Xe-139). The NRC staff finds that these noble gases would mostly decay away before they could be released from the building and that the particulate daughter products of these noble gases would also not be released in significant quantities. The NRC staff also ignored radioactive decay in its calculation and assumed that a receptor would remain in place for the entire time it takes the released plume to pass. Because the NRC staff assumed a bounding, generic X/Q and no radioactive decay (making its calculation independent of any exact receptor location, release point, and release duration), the NRC staff considers its calculation to bound any realistic release scenario from either the truck door (i.e., Scenario C), the stack (i.e., Scenario D), or any other point.

The results of UML's Scenario B calculations are summarized in SER Table 5-4. The results of UML's Scenario C and D calculations, along with relevant NRC staff confirmatory calculations, are shown in SER Table 5-5 and Table 5-6, respectively.

UML noted that, as shown in SER Table 5-4, the dose to an individual positioned 0.16 meters away from the outer surface of the truck door (i.e., effectively at the outer surface of the door) continuously for 30 days would be 123 mrem, above the 100 mrem public dose limit in 10 CFR 20.1301. However, UML stated that this is a highly unrealistic scenario and noted that for individuals at more realistic distances from the door (or for a more realistic stay time) the dose would be much lower. UML stated that, in the highly unlikely event of a scenario that would cause such elevated dose rates outside the truck door, the area around the truck door would be posted and controlled in accordance with 10 CFR Part 20 requirements. (The NRC staff notes that, although the area outside the truck door is outside the area under the UMLRR license, the area is still part of the UML campus and is, therefore, under Commonwealth of Massachusetts control and could be controlled by UML as needed.)

Table 5-4 UML-Calculated MHA Public Doses (Scenario B)

Receptor Location (30 day stay time assumed)	UML Calculations (mrem)	10 CFR 20.1301 Dose Limit (mrem)
Outside first floor airlock outer security door	0.15	100
Outside third floor airlock outer security door	0.88	100
Outside confinement building wall	0.47	100
Outside truck door (0.16 meters away)	123	100
Outside truck door (2.5 meters away)	36.0	100
Outside truck door (9.5 meters away)	4.5	100
Outside truck door (19.5 meters away)	1.6	100

Table 5-5 UML- and NRC-Calculated MHA Public Doses (Scenario C)

Receptor Location (30 day stay time assumed)	UML Calculations (mrem)			NRC Calculations (mrem)			10 CFR 20.1301 Limit (mrem)
	Building Shine Dose	Door Leakage Dose	Total[a]	Door Leakage Dose			
				External	Internal	Total[b]	
Outside truck door (10 meters away)	4.5	31.7[c]	36.2	56.3	3.8	60.1	100
Nearest residence	~0	0.13[c]	0.13	-			100

[a] Total TEDE including contributions from building shine and radioactive air leakage from door
[b] TEDE from radioactive air leakage from door only
[c] Assuming Pasquill Stability Class D or F (worst-case) occurs 100 percent of the time

Table 5-6 UML- and NRC-Calculated MHA Public Doses (Scenario D)

Receptor Location (18 hour stay time assumed)	UML Calculations (mrem)			NRC Calculations (mrem)			10 CFR 20.1301 Limit (mrem)
	Building Shine Dose	Stack Release Dose	Total[a]	Stack Release Dose			
				External	Internal	Total[b]	
Outside truck door (10 meters away)[c]	0.3	6.3[d]	6.7	56.3	3.8	60.1	100
Nearest residence	~0	0.4[e]	0.4	-			100

[a] Total TEDE including contributions from building shine and radioactive air leakage from door
[b] TEDE from radioactive air leakage from door only
[c] UML-determined maximum dose location
[d] Assuming Pasquill Stability Class F (worst-case) occurs 100 percent of the time
[e] Assuming Pasquill Stability Class A (worst-case) occurs 100 percent of the time

The NRC staff reviewed UML's public MHA dose calculations and results. As discussed above, UML appeared to consider radioactive decay for its Scenario B, C, and D calculations of external shine dose outside the reactor building from radioactive air inside the building. The NRC staff notes that, if decay were considered but the daughter products of the decayed radioisotopes were not considered, then UML could underestimate the total external gamma dose rate from radioactive material in the building. However, the NRC staff finds that, given other conservatisms in UML's calculations of external shine dose from material in the building (for example, the continuous occupation by receptors very close to the building), UML's calculations provide a reasonable estimate of dose. Additionally, the NRC staff finds that the daughter product dose contribution (and total shine dose) would be most significant for Scenarios B and C, and less significant for Scenario D, because Scenario D is much shorter duration. As discussed above, Scenario D, which assumes that accident-mitigating functions (as required by TSs) generally work as designed, represents the most probable progression of

the MHA event; Scenarios B and C represent calculations that introduce additional conservatism by assuming the failure of certain functions, or a substantial overpressure event that limits the ability of the emergency exhaust system to maintain negative pressure.

The NRC staff notes that, although the 100 mrem public dose limit in 10 CFR 20.1301 is an annual limit, UML calculated its Scenario B and C doses for a 30-day period. However, the NRC staff finds that the 30-day period is reasonable, given that (1) due to leakage and radioactive decay, most of the dose would realistically occur during the first 30 days of a scenario in which radioactive material were held up in the building and (2) UML conservatively assumed that receptors continually occupy the locations near the reactor building (which are not typically continually occupied) for the entire 30 days.

The NRC reviewed UML's Scenario C assumption that building air containing radioactive material is assumed to leak through the truck door at a rate of 1 percent per day. SAR Section 6.2.8 states that historical leak tests have demonstrated that the total (i.e., from all reactor building penetrations) leakage of the reactor building is less than 10 percent per day at a 2 pounds per square inch gauge (14 kilopascals or 55 inches of water) overpressure. The NRC staff notes that, given UML's re-designation of its reactor building from a containment to a confinement in conjunction with the UMLRR license renewal (see UML's response to RAI-6.1 and SER Sections 1.8 and 6.3), surveillances related to reactor building leak rates will no longer be required to be performed. However, because any prolonged, realistic building overpressure scenario would likely be much less severe than 2 pounds per square inch gauge (14 kilopascals or 55 inches of water), or the 7 kilopascals (28 inches of water) discussed above, and because most leakage would likely not occur from the truck door during an overpressure scenario, the NRC staff finds that UML's 1 percent per day value for leakage from the truck door is a reasonable approximation for its Scenario C calculation. The NRC staff additionally notes that its confirmatory calculations, which are independent of leak rate (and thus would bound more rapid releases from the truck door) and release point (and thus would bound alternate scenarios in which reactor building air leaked outside from multiple points simultaneously), as discussed above, demonstrate that public doses from releases remain below 10 CFR Part 20 limits.

Although, as stated above, UML did not appear to specify whether activity in the building was assumed to decrease with building leakage for Scenario D, the NRC staff finds that this assumption would be reasonable and justifiable, if used.

Although Scenario D is a release from the UMLRR stack, UML determined that the maximum dose location is near the reactor building, approximately 10 meters (32.8 feet) from the truck door or base of the stack. The NRC staff notes that this is likely due to the calculation methodology used by the ARCON96 code, which considers releases from stacks to be ground releases for some fraction of the time if the stacks are less than 2.5 times the height of adjacent buildings, in order to account for potential plume downwash from building wakes. As discussed above, the NRC staff finds that UML's use of ARCON96 for its MHA atmospheric dispersion calculations is acceptable.

The NRC staff compared the results of UML's calculations to the results of the NRC staff's confirmatory calculations, as well as to the dose limits in 10 CFR Part 20. The NRC staff notes that its calculated doses for truck door leakage (Scenario C) are greater than UML's calculated doses and that the NRC staff's calculated doses for stack release dose (Scenario D) are much greater than UML's calculated doses. The NRC staff finds that the differences for Scenario C are likely due primarily to the NRC staff's assumption that the receptor remains in place for the entire time it takes the plume to pass, and the differences for Scenario D are likely due primarily

to the NRC staff's use of a very conservative generic X/Q. The NRC staff finds that, except for UML's Scenario B calculation for a receptor adjacent to the truck door, the results of all of UML's and the NRC's staff calculations are below the 100 mrem public dose limit in 10 CFR 20.1301 and, therefore, acceptable. The NRC staff finds that the results of UML's Scenario B calculation for a receptor adjacent to the truck door exceed 100 mrem, but the NRC staff considers this calculation to be a conservative sensitivity study because it is based on unrealistic assumptions, as discussed above; realistic doses for any receptor adjacent to the truck door would be very unlikely to exceed the 100 mrem public dose limit.

The NRC staff finds that UML's MHA public dose calculation assumptions and methodology are generally reasonable and conservative and also finds that the results of UML's public dose calculations and the NRC staff's confirmatory calculations demonstrate that the MHA public dose would be within 10 CFR Part 20 limits. Therefore, the NRC staff finds that UML's MHA public dose calculations are acceptable.

Conclusion

The NRC staff reviewed UML's MHA occupational and public dose calculations and finds that the methodologies and assumptions used by the licensee are generally reasonable, conservative, and consistent with established industry practices. As discussed above, the NRC staff also performed independent confirmatory calculations of the MHA occupational and public dose calculations. Based on its review of UML's dose calculations, and the results of the NRC staff's confirmatory calculations, the NRC staff finds that the MHA results demonstrate that the MHA occupational dose and public dose would be below the occupational dose limits in 10 CFR 20.1201 and the public dose limit in 10 CFR 20.1301, respectively. Therefore, the NRC staff concludes that the UMLRR MHA is acceptable.

5.2 Insertion of Excess Reactivity

UML's insertion of excess reactivity (reactivity transient) analyses are discussed primarily in SAR Sections 13.1.2 and 13.2.2, as supplemented by UML's response to RAI-13.1 (Ref. 23). Additional information relating to UML's methodology, assumptions, and inputs for its analyses is also discussed in SAR Sections 4.5.1.1, 4.5.4, 4.5.5, 4.5.6, 4.5.7, and 4.6.

As discussed in SER Section 2.5, for thermal-hydraulic analyses evaluating the consequences of reactivity transients in the reactor core, UML used Program for the Analysis of Reactor Transients (PARET), which is an analysis code that simulates the behavior associated with reactivity-induced transients within the system.

UML used PARET to evaluate two broad types of reactivity transients: step insertions of reactivity and ramp insertions of reactivity. Step insertions could occur due to the failure of a fixed experiment or other device, due to a rapid cold-water insertion event, or due to the potential rapid ejection of an experimental bayonet from one of the radiation baskets in the UMLRR core. Ramp insertions could occur due to an event such as the inadvertent withdrawal of a control blade, either from a malfunction or from operator error. For its analyses, UML assumed that potential step insertions would occur instantaneously and that potential ramp insertions would occur at a fixed rate of reactivity insertion per unit time. As discussed below, UML evaluated several generic step and ramp insertion scenarios for forced flow and natural convection operation and also performed a separate, specific analysis of cold water insertion events due to the primary pump being turned on while the UMLRR is operating in natural convection mode.

UML stated that PARET does not utilize hot channel factors, similar to those used for UML's NATCON and PLTEMP steady-state thermal hydraulic analyses, to account for calculation uncertainties. Instead, UML stated that it accounted for uncertainties by using conservative input conditions for its PARET analyses.

UML's generic step insertion analyses assumed that 0.6 percent delta k over k (% Δk/k) of reactivity (greater than the TS 3.7.1 maximum total movable experiment worth or single secured experiment worth of 0.5% Δk/k) is instantaneously inserted into the core. UML's generic ramp insertion analyses assumed that 0.07% Δk/k per second of reactivity (greater than the TS 3.2.2(1) maximum reactivity insertion rate by the regulating rod and most reactive control blade simultaneously of 0.05% Δk/k per second) is inserted into the core.

For its generic step and ramp insertion PARET analyses, UML used inputs and/or assumptions including the following:

- kinetics parameters and reactivity feedback coefficients which are discussed and found acceptable in SER Section 2.5.2;
- total radial peaking factor of 2.1 and the VENTURE-calculated axial profile for the fuel element in core location B5 of the operational core configuration (OCC) (see SER Section 2.5.1);
- initial reactor power of 0.01 watt-thermal or 1.25 megawatt-thermal (MWt), for forced flow scenarios, and 0.01 watt-thermal or 0.125 MWt, for natural convection scenarios;
- for forced flow scenarios, a primary flow rate of 1,370 gallons per minute (gpm), which is below the LSSS value of 1,400 gpm, and which corresponds to a coolant mass flux of 799.1 kilograms per second per square meter for a 21 silicide fuel element core (calculated using UML's PLTEMP-calculated aluminide element flow distributions and rates, as discussed in SER Section 2.6);
- for natural convection scenarios, an initial coolant mass flux up through the core of 0.001 kilograms per second per square meter (because PARET will not execute properly with an initial flow rate set to exactly zero);
- core coolant inlet temperature of 110 degrees Fahrenheit (°F) (43.3 degrees Celsius (°C)), which is above the LSSS value of 108 °F (42.2 °C) for pool inlet temperature (in forced flow mode) or bulk pool temperature (in natural convection mode);
- short period scram required by TSs is inoperable;
- reactor scrams when a high-power scram set point of 1.25 MWt (for forced flow scenarios) or 0.125 MWt (for natural convection scenarios) is reached, which is above the respective LSSS values of 1.15 MWt and 0.115 MWt;
- instrument delay (time between when the high-power scram setpoint is reached and when the control blades begin to drop) of 210 milliseconds (msec), which is greater than the measured value of approximately 185 msec;
- the most reactive control blade is stuck fully out and does not drop with the other three control blades when the reactor scrams;
- control blades drop from a critical height of 15 inches;
- control blade insertion speed of 0.533 meters per second (21.0 inches per second), calculated assuming that the control blades move from fully withdrawn (25 inches out) to 4 inches out (approximately 84 percent inserted; the last approximately 4 inches of free-fall control blade travel are dampened by a dashpot to prevent mechanical damage to the blades) in 1 second; and

- control blade worth curves determined based on actual measurements demonstrating that, from a blade height of 15 inches to fully inserted, the three least reactive control blades that scram have a combined worth of about $7.00 or 5.5% Δk/k (the total worth of these three blades, based on these measurements and the data in SAR Table 4-3, was about 8% Δk/k).

Renewed TS 3.2.1(2), which is discussed and found acceptable in SER Section 2.2.2, requires that the time from initiation of a scram signal and movement of each control blade from the fully withdrawn position to 80 percent of the fully inserted position shall be less than 1 second. As discussed in SAR Section 4.5.5 and SER Section 2.2.2, the 1 second time in the TS includes both instrument delay time and the time that it takes the control blades to move from fully withdrawn to 80 percent inserted. However, UML stated that, for its PARET analyses, it conservatively considered its assumed 210 msec instrument delay time to be separate from its assumed 1 second time needed for the control blades to move from fully withdrawn to 84 percent inserted (i.e., there would be a 210 msec delay before the blades would begin to fall and then the blades would fall at a rate based on it taking 1 second to go from fully withdrawn to 84 percent inserted).

As discussed in UML's response to RAI-13.4 (Ref. 23), even if one or both of the trip actuator amplifiers (TAAs) designed for initiating an electronic scram (i.e., fast scram) failed (each of the two TAAs powers two of the four control blade drive electromagnets), the slower relay scram would not be affected. As discussed in SAR Section 7.4.3, the instrument delay time that UML used is conservatively based on a relay scram. However, because the UMLRR power and period channels can also normally provide an electronic scram, which has a much shorter instrument delay time (approximately 5 msec), the actual instrument delay time for a scram would likely be much shorter.

The NRC staff reviewed UML's methodology, inputs, and assumptions for its generic step and ramp reactivity insertion analyses, as discussed above. The NRC staff finds that the use of PARET for reactivity transient calculations is reasonable and consistent with research reactor industry practice.

The NRC staff finds that the total radial peaking factor and axial profile used by UML are appropriate and are conservative because they are consistent with the hot fuel plate of the UMLRR LCC discussed and found acceptable in SER Section 2.5.1. The NRC staff finds that UML analyzed step reactivity insertions that bound the maximum total movable experiment worth or single secured experiment worth of TS 3.7.1, and ramp reactivity insertions that bound the maximum reactivity insertion rate of TS 3.2.2(1). The NRC staff finds that events such as the rapid insertion of a fuel element into the core or the rapid ejection or removal of a control blade, when the reactor is critical or subcritical by less than the worth of the fuel element or control blade could potentially cause a supercritical rapid reactivity insertion greater than the 0.6% Δk/k assumed by UML, but that such an event is not credible given the design of the reactor and control blades and standard industry practice for operating research reactors such as the UMLRR. The NRC staff also finds that UML assumed values for power level scram setpoint, temperature, and coolant flow that are more conservative (i.e., higher, higher, and lower, respectively) than TS-required LSSS settings. The NRC staff finds that UML's calculated coolant mass flux for forced flow operation is reasonable given the LSSS flow rate and the dimensions and layout of the UMLRR core and fuel and finds that UML's coolant mass flux for natural convection operation is reasonable because it approximates the mass flux to be effectively zero. The NRC staff additionally finds that UML used starting power levels (at 0.01 watt-thermal and at greater than LSSS values) that reasonably bound the range of possible

allowed reactor operation. The NRC staff finds that UML's assumption that the single TS-required short period scram channel is inoperable, meaning that the reactor will scram on high power (which, for the analyses with low starting power, results in an additional time delay until the reactor scrams), is consistent with the single initiating malfunction criterion, and is consistent with the practice for reactivity transient analyses for similar research reactors. The NRC staff notes that because there are two TS-required power scram channels for the UMLRR, the single failure of one of those channels would not affect the analyses. The NRC staff finds that UML's assumption that the most reactive control blade is stuck fully withdrawn and does not scram, even though TS 3.2.1(1) requires all four control blades to be operable, adds additional conservatism to the analyses.

The NRC staff notes that UML assumed that the control blades would be 84 percent inserted 1 second after they began to fall from fully withdrawn, but TS 3.2.1 only requires the blades to be 80 percent inserted after 1 second. However, the NRC staff finds that UML's consideration of a conservative instrument delay time as separate from the 1 second blade drop time helps compensate for the small overestimation of control blade drop speed resulting from UML's assumption that the blades fall to 84 percent inserted in 1 second.

Additionally, the NRC staff notes that, although UML assumed that the blades would fall at a constant 0.533 meters per second (21.0 inches per second) for the entire range from 15 inches withdrawn to fully inserted, the actual blade speed could vary (for example, the blades could fall at a slower rate when they first start to fall, or over the last approximately 4 inches because of the dampening by the dashpot). Also, the NRC staff notes that the initial critical blade height from which the control blades fall could be greater or less than 15 inches. If the initial height from which the blades fall is greater than 15 inches, the blades could take longer to be fully inserted, and the differential worth of the blades could be smaller when the blades first start to fall (although the worth of the blades from critical height to fully inserted could also be greater). If the initial height is less than 15 inches, the worth of the blades from critical height to fully inserted could be less (but it would need to be at least 1.0% $\Delta k/k$ per the shutdown margin TS 3.1.1(2)). The NRC staff finds that UML's separation of the conservative instrument delay time also helps compensate for any variance in the initial critical height from which the control blades fall, and for the blades falling more slowly when they first begin to drop and more rapidly as they fall further. The NRC staff additionally finds that, based on the control blade worth data that UML assumed for its PARET analyses and described in SAR Table 4-6, the reactivity worth of the last 4 inches of the control blades is small (less than 7 percent of the total of approximately $7.00 of negative reactivity inserted by the blades as they fall from 15 inches to fully inserted), and the movement of the blades at a slower rate over the last 4 inches would not significantly affect the results of the analyses.

As illustrated by UML's results below, reactor power decreases quickly and sharply as soon as the blades begin to insert, and fuel and cladding temperatures start to decrease very shortly thereafter. The NRC staff therefore finds that the timing of the beginning of blade insertion is much more significant than the exact negative reactivity insertion rate or amount for the purposes of estimating peak reactor power and fuel temperatures.

Based on the above and also on the reactivity transient analysis results below that demonstrate a large margin of safety, the NRC staff finds that UML's methodology, inputs, and assumptions for its generic step and ramp reactivity insertion analyses are generally reasonable and conservative and, therefore, acceptable.

Step Insertions of Reactivity

UML performed analyses of step insertions of reactivity for four scenarios: forced flow mode operation with initial reactor power of 0.01 watt-thermal; forced flow mode operation with initial reactor power of 1.25 MWt; natural convection mode operation with initial reactor power of 0.01 watt-thermal; and natural convection mode operation with initial reactor power of 0.125 MWt. As discussed above, these scenarios all assumed a step insertion of 0.6% Δk/k. The results of these analyses are graphically illustrated in SER Figure 5-1, Figure 5-2, Figure 5-3, and Figure 5-4, which are reproduced from UML's response to RAI-13.1, and show plots of power level and temperature (of the coolant, clad, and fuel) versus time for each of the four scenarios.

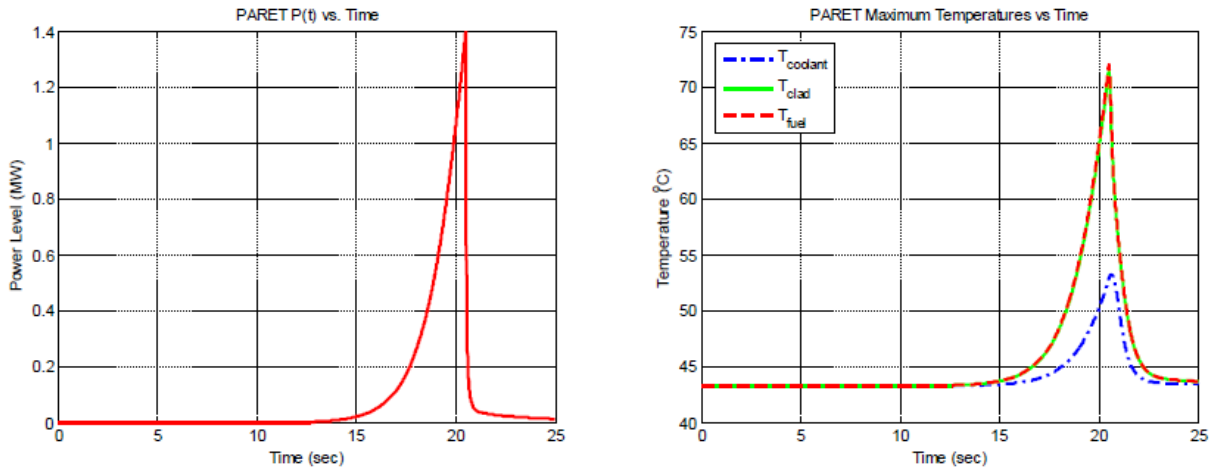

Figure 5-1 Step Insertion of 0.6% Δk/k for Forced Flow and Initial Power of 0.01 Watt-thermal

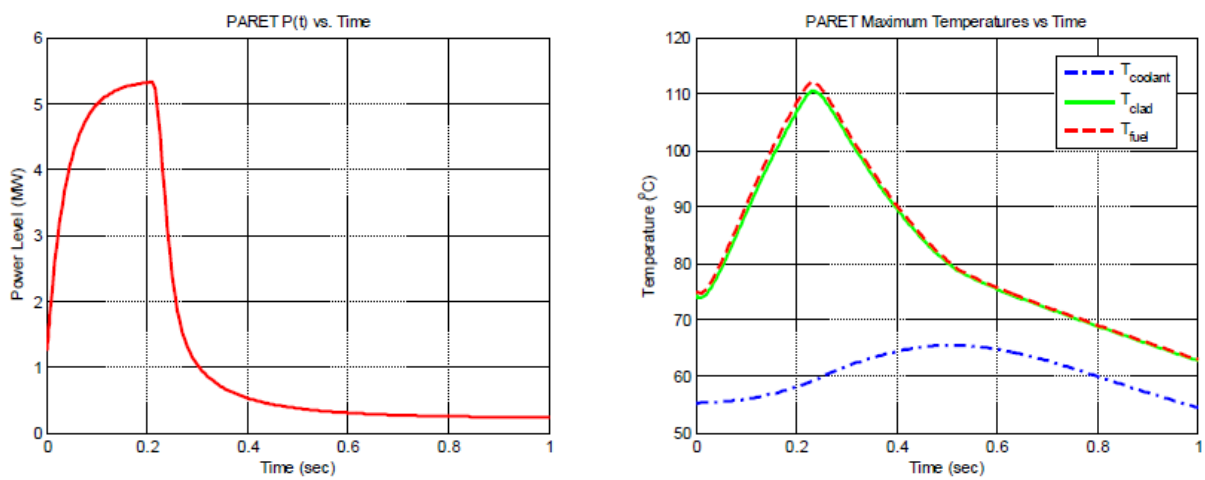

Figure 5-2 Step Insertion of 0.6% Δk/k for Forced Flow and Initial Power of 1.25 MWt

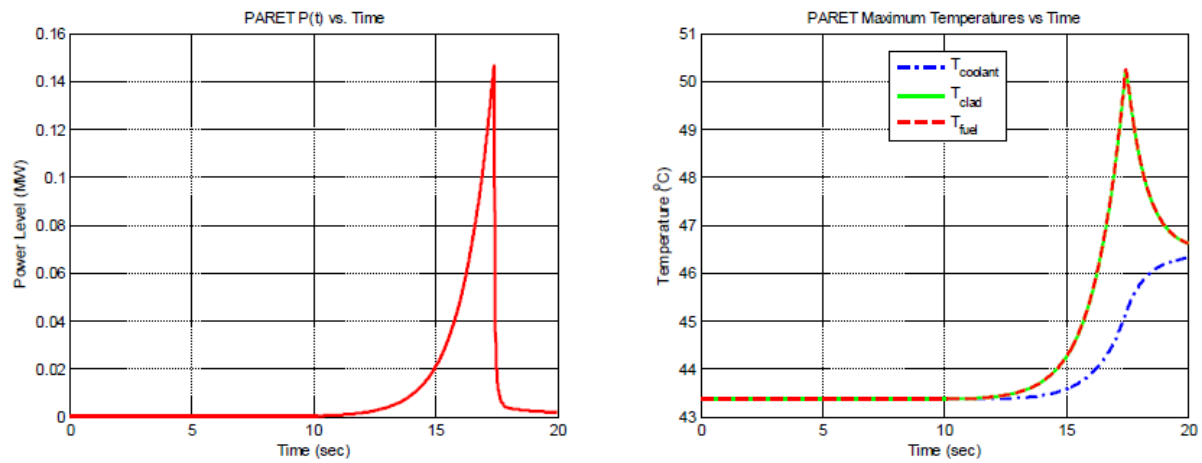

Figure 5-3 Step Insertion of 0.6% Δk/k for Natural Convection and Initial Power of 0.01 Watt-thermal

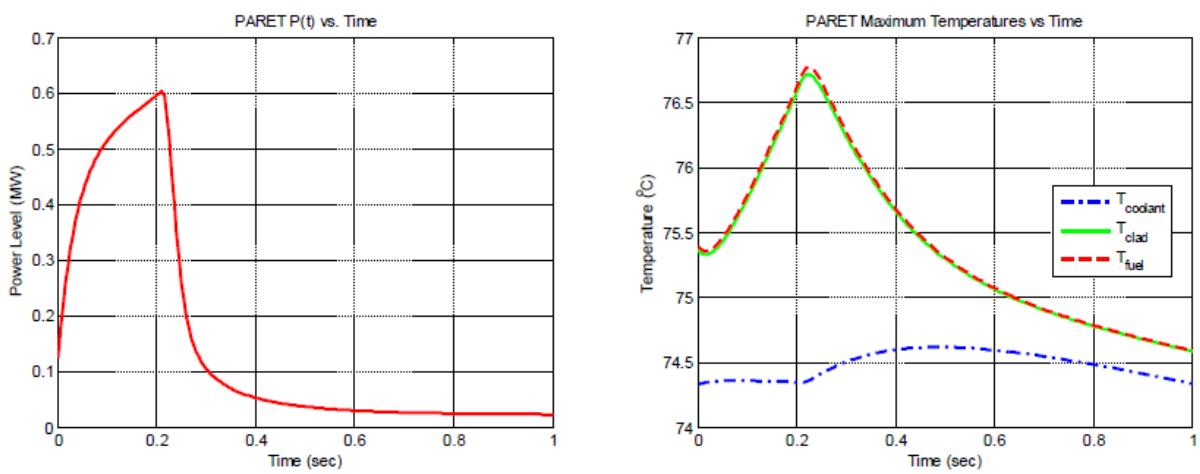

Figure 5-4 Step Insertion of 0.6% Δk/k for Natural Convection and Initial Power of 0.125 MWt

The figures show that for all four scenarios, the maximum fuel cladding temperature remains below the minimum onset of nucleate boiling (ONB) threshold of 118 °C (244 °F) and well below the TS 2.1 safety limit (SL) of 530 °C (986 °F). UML stated that its analyses show that ONB does not occur for any of its step insertion scenarios. In the worst-case step insertion scenario, for forced flow operation and initial reactor power of 1.25 MWt (SER Figure 5-2), the maximum cladding temperature reached is approximately 111 °C (232 °F) and the maximum reactor power level reached is approximately 5.3 MWt.

The NRC staff reviewed the results of UML's step reactivity insertion analyses, discussed above. The NRC staff finds that the results of the analyses demonstrate that credible rapid reactivity insertion accident scenarios (e.g., experiment failures) at the UMLRR would result in maximum fuel cladding temperatures that are well below the SL and temperatures at which fuel failure could occur. Therefore, the NRC staff finds that the results of UML's step reactivity insertion analyses are acceptable.

Ramp Insertions of Reactivity

UML performed analyses of ramp insertions of reactivity for four scenarios: forced flow mode operation with initial reactor power of 0.01 watt-thermal; forced flow mode operation with initial reactor power of 1.25 MWt; natural convection mode operation with initial reactor power of 0.01 watt-thermal; and natural convection mode operation with initial reactor power of 0.125 MWt. As discussed above, these scenarios all assumed a ramp insertion of 0.07% Δk/k per second. The results of these analyses are graphically illustrated in SER Figure 5-5, Figure 5-6, Figure 5-7, and Figure 5-8, which show plots of power level and temperature (of the coolant, clad, and fuel) versus time for each of the four scenarios (figures are from UML's response to RAI-13.1).

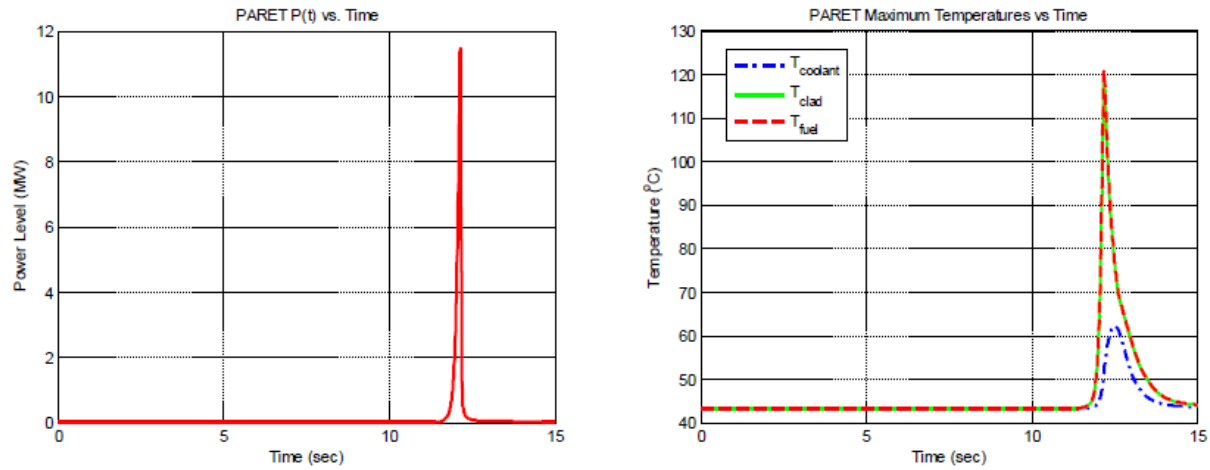

Figure 5-5 Ramp Insertion of 0.07% Δk/k per second for Forced Flow and Initial Power of 0.01 Watt-thermal

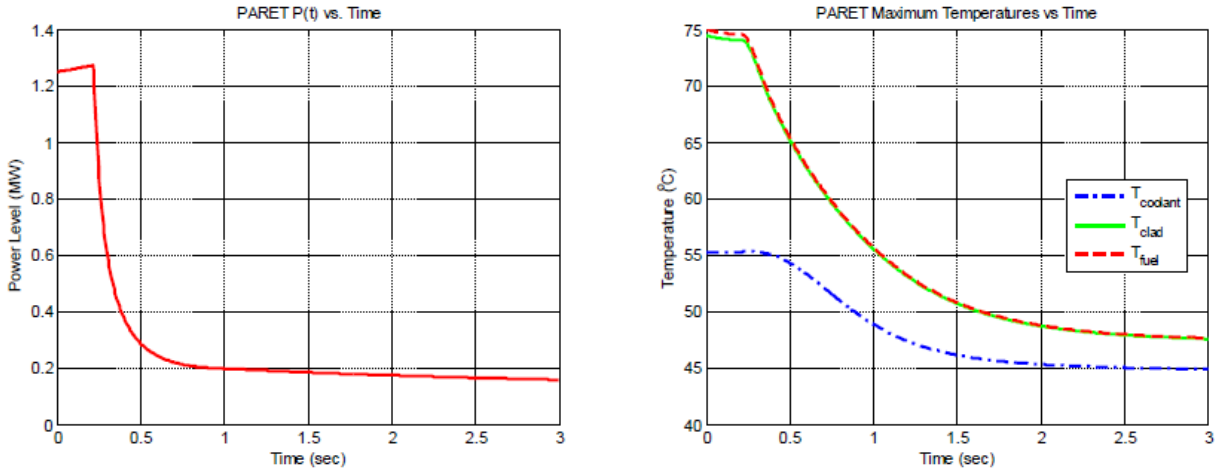

Figure 5-6 Ramp Insertion of 0.07% Δk/k per second for Forced Flow and Initial Power of 1.25 MWt

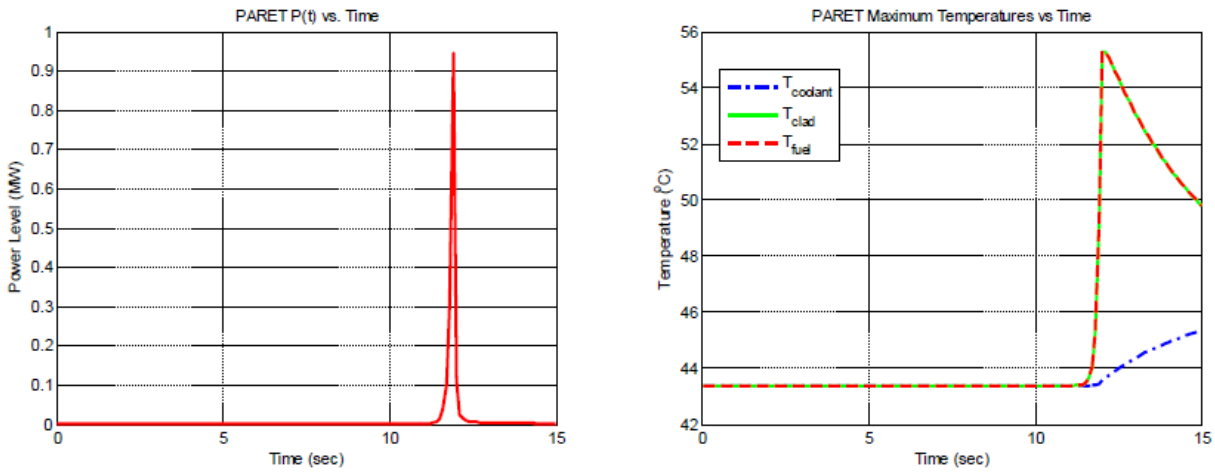

Figure 5-7 Ramp Insertion of 0.07% Δk/k per second for Natural Convection and Initial Power of 0.01 Watt-thermal

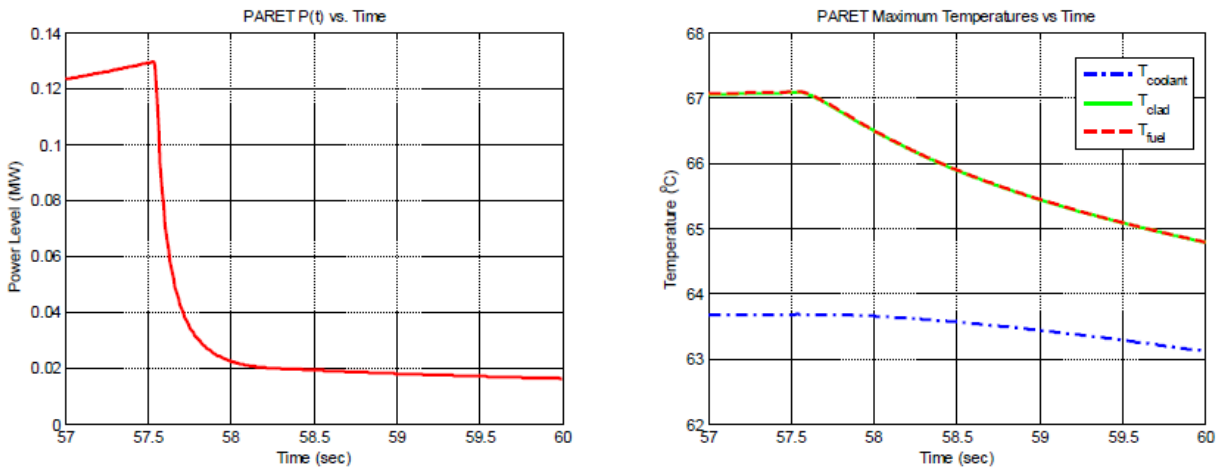

Figure 5-8 Ramp Insertion of 0.07% Δk/k per second for Natural Convection and Initial Power of 0.125 MWt

The figures show that for all ramp insertion scenarios except for the forced flow mode operation with initial reactor power of 0.01 watt-thermal scenario, which is the worst-case scenario and which is illustrated in SER Figure 5-5, the maximum fuel cladding temperature remains well below the TS 2.1 SL and the minimum ONB threshold of 118 °C (244 °F). In the worst-case ramp insertion scenario, the maximum cladding temperature reached is approximately 120 °C (248 °F) and the maximum reactor power level reached is approximately 11.5 MWt. However, UML stated that its analyses for all four ramp insertion scenarios showed that ONB does not actually occur. Additionally, the figures for all four scenarios show that the maximum cladding temperature remains well below the TS 2.1 SL of 530 °C (986 °F).

The NRC staff reviewed the results of UML's ramp reactivity insertion analyses, discussed above. The NRC staff finds that the results of the analyses demonstrate that credible slow reactivity insertion accident scenarios (e.g., control blade drive malfunction or operator error) at the UMLRR would result in maximum fuel cladding temperatures that are well below the SL and temperatures at which fuel failure could occur. Although UML's analysis for a ramp insertion starting from low power and in forced flow mode operation shows that the minimum ONB threshold could be exceeded for that scenario, the NRC staff notes that that scenario analyzed by UML is highly unlikely because it assumes a reactivity insertion rate (and other parameters) that exceed TS limits. Additionally, UML determined that ONB would not actually occur for this scenario. (Although 118 °C (244 °F) is the minimum ONB threshold, the NRC staff notes that ONB may not occur until a slightly higher temperature, depending on heat flux and flow conditions. Also, any boiling may be limited because the scenario is very rapid and heat transfer conditions could be essentially adiabatic.) Based on the above, the NRC staff finds that the results of UML's ramp reactivity insertion analyses are acceptable.

NRC Staff Confirmatory Calculation of Ramp Insertion of Reactivity

Because it was UML's bounding reactivity insertion scenario, the NRC staff performed a calculation to determine if it could confirm the results of UML's analysis of a ramp insertion of reactivity in forced flow operation mode, starting from low power (0.01 watt-thermal),. For its analysis, the NRC staff used the non-LOCA TRAC/RELAP Advanced Computational Engine (TRACE) model discussed in SER Section 2.6 and used inputs and/or assumptions including the following:

- reactivity insertion rate of 0.05% Δk/k per second (equal to the TS 3.2.2(1) maximum reactivity insertion rate);
- kinetics parameters and reactivity feedback coefficients which are discussed and found acceptable in SER Section 2.5.2;
- power peaking factors consistent with the hot plate of the LCC;
- coolant inlet temperature of 108 °F (43.3 °C), equal to the LSSS value;
- primary flow rate of 1,400 gpm, equal to the LSSS value;
- the short period scram required by TSs is inoperable;
- reactor scrams when a high-power scram set point of 1.15 MWt, equal to the LSSS value, is reached;
- instrument delay time of 200 msec before the control blades begin to fall;
- all four control blades are inserted within 1 second after they begin to fall; and
- a total of approximately 10.57% Δk/k (assumed to be the worth of the four control blades from a high critical height to fully inserted) of negative reactivity is inserted at a uniform rate over the 1 second insertion time (the total worth of all four blades, based on the data in SAR Table 4-3, was 11.55% Δk/k).

The results of the NRC staff's confirmatory calculation are shown graphically in SER Figure 5-9, which shows plots of reactor power and fuel cladding temperature versus time. For the calculation with results illustrated in Figure 5-9, the maximum cladding temperature is approximately 135 °C (275 °F) and the maximum reactor power is approximately 7.8 MWt.

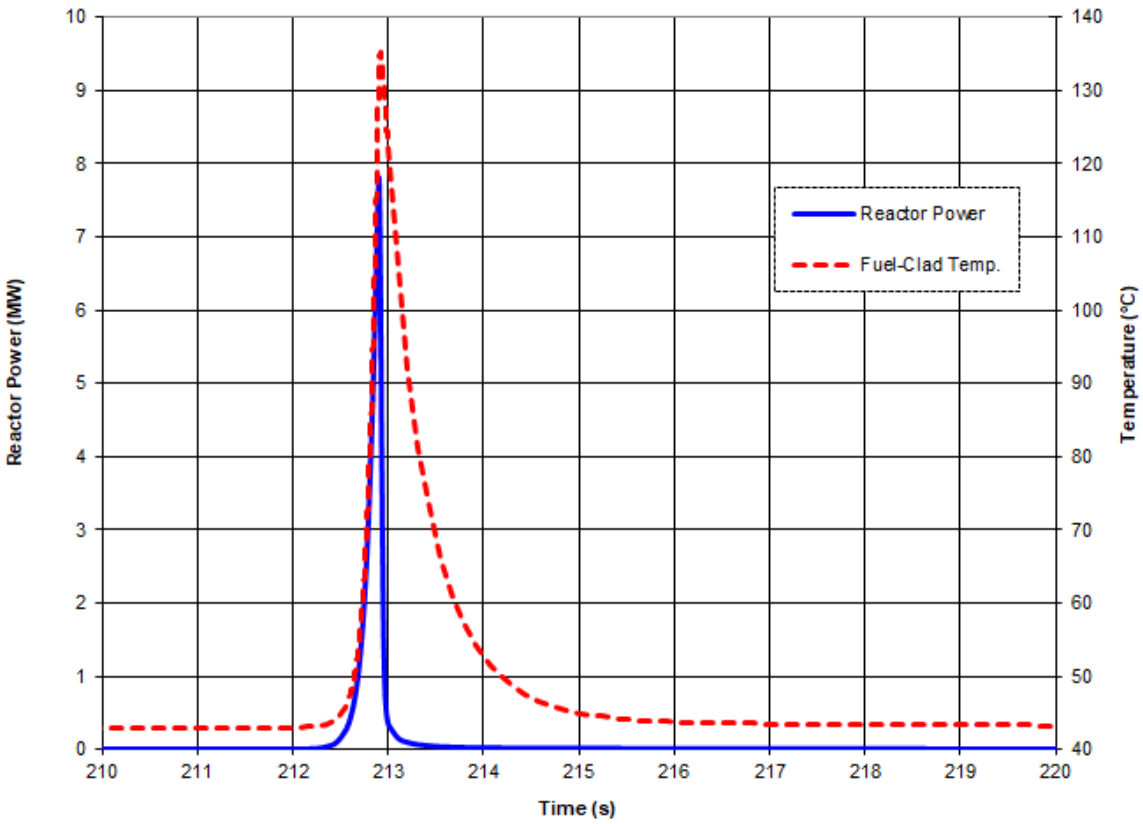

Figure 5-9 NRC Staff Confirmatory Calculation of Ramp Insertion of 0.05% Δk/k per second for Forced Flow and Initial Power of 0.01 Watt-thermal

In addition, the NRC staff performed a second calculation that is similar to its calculation discussed above and illustrated in Figure 5-9, except that the NRC staff conservatively ignored all reactivity feedback effects. For its second calculation, the maximum cladding temperature reached is approximately 142 °C (288 °F) and the maximum reactor power is approximately 13.2 MWt.

The NRC staff notes that, for both NRC staff calculations (with and without reactivity feedback), the maximum cladding temperature exceeds the minimum ONB threshold of 118 °C (244 °F). The NRC staff also determined that ONB conditions would be reached. However, the results of the NRC staff's calculations, even when reactivity feedback is ignored, are well below the TS 2.1 SL of 530 °C (986 °F). Both NRC staff calculations additionally demonstrated that although nucleate boiling could very briefly occur, departure from nucleate boiling conditions (i.e., departure from nucleate boiling ratio > 2.0) would not be reached, indicating that bulk boiling could not occur. The NRC staff notes that, although its calculation used some inputs that were less conservative than those used by UML for its similar calculation of a ramp reactivity insertion during forced flow operation starting from low power, the NRC staff's calculated maximum cladding temperature (approximately 135 °C (275 °F)) was greater than UML's calculated maximum cladding temperature (approximately 120 °C (248 °F)). However, the NRC staff finds that this variation is likely due to other differences in the methodologies and assumptions used

for the calculations, including TRACE's use of explicit, conservative hot channel factors to account for uncertainties.

Cold Water Insertion Events

As discussed above, UML also performed a separate analysis of cold-water insertion events due to the primary pump being turned on while the UMLRR is operating in natural convection mode. As discussed in SAR Section 4.5.4 and SER Section 2.5.2, the UMLRR has negative temperature reactivity feedback coefficients. Therefore, a rapid decrease in core temperature, such as could occur during a sudden insertion of cold water into the core (e.g., turning on the primary pump during natural convection operation would cause rapid replacement of warm water in the core with cooler water from above the core), would lead to an increase in reactivity.

UML constructed a PARET model to analyze three pump-on scenarios: initial reactor power of 0.1 MWt and core inlet temperature of 43.3 °C (110 °F); initial reactor power of 0.125 MWt and core inlet temperature of 43.3 °C (110 °F); and initial reactor power of 0.1 MWt and core inlet temperature of 30 °C (86 °F). The third scenario represents nominal conditions. UML's PARET model for these scenarios incorporated the results of experiments, discussed in SAR Section 4.5.6, demonstrating that the primary pump approaches full flow approximately 2 seconds after being turned on. The reactor scrams at 0.125 MWt for all three scenarios. The results of UML's pump-on analyses are illustrated in SER Figure 5-10, Figure 5-11, and Figure 5-12 (which are reproduced from SAR Figure 13-16).

The results of UML's pump-on analyses illustrate that, for all scenarios evaluated, the maximum fuel cladding temperatures remain below approximately 91 °C (196 °F). The highest cladding temperatures were those seen at the beginning of each scenario. UML stated that, for its pump-on analyses, the initial peak temperatures are slightly higher than expected because the PARET model used for the pump-on analyses underestimates the realistic initial natural convection flow rate in the hot channel of the fuel by assuming one average flow rate for the entire core. UML also stated that its pump-on analyses demonstrate that a pump-on event during steady-state natural convection operation is not a limiting transient scenario for the UMLRR.

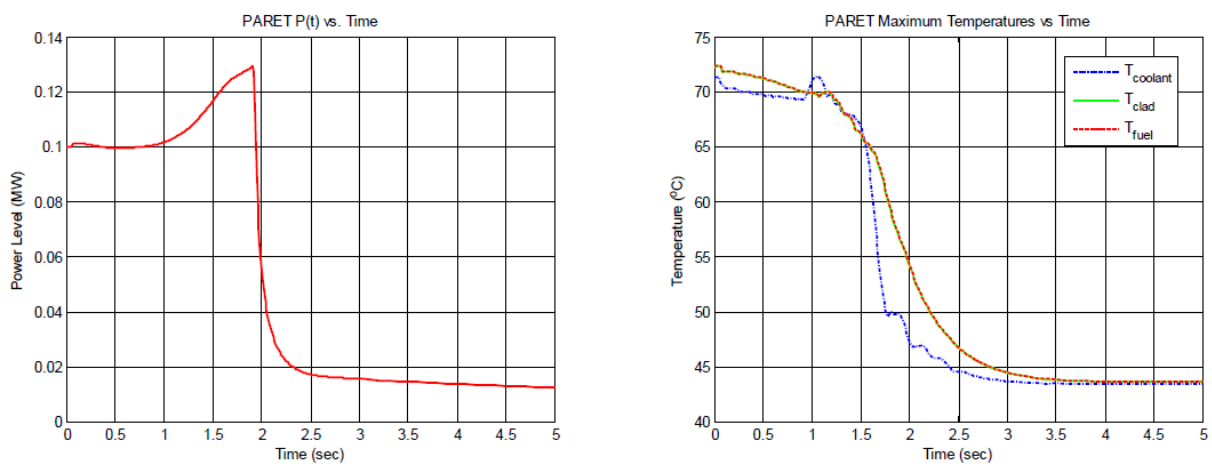

Figure 5-10 Pump-On Scenario for Natural Convection with Initial Power of 0.1 MWt and Core Inlet Temperature of 43.3 °C

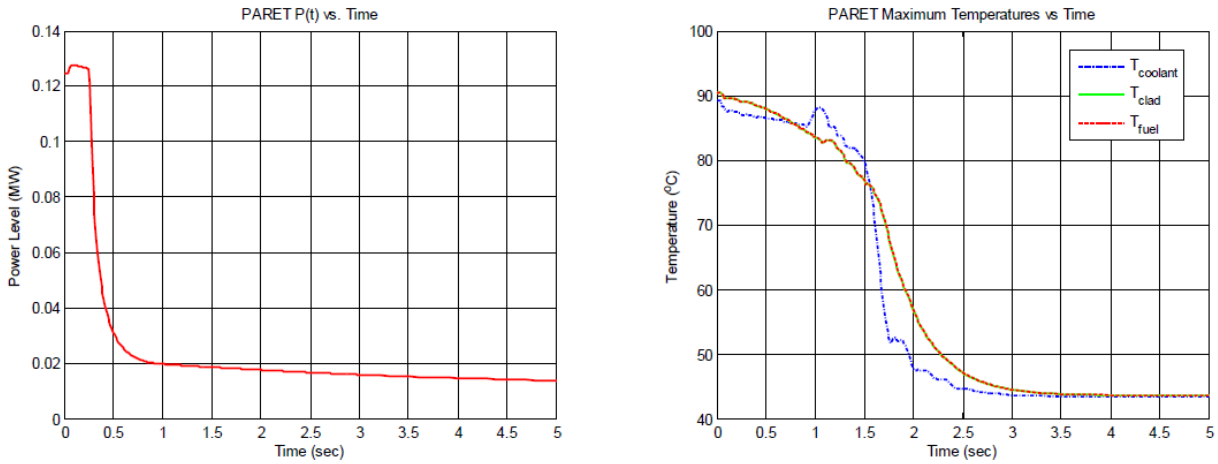

Figure 5-11 Pump-On Scenario for Natural Convection with Initial Power of 0.125 MWt and Core Inlet Temperature of 43.3 °C

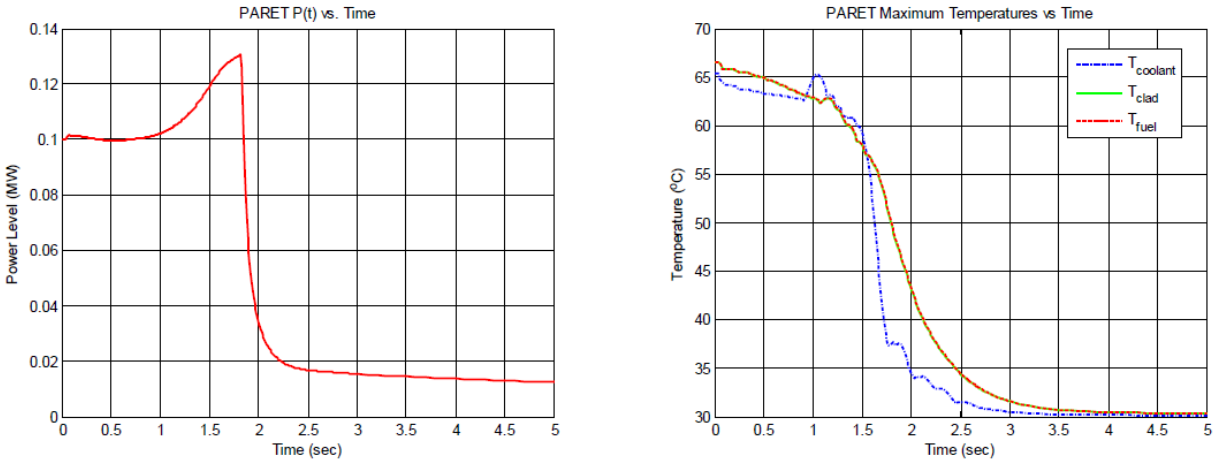

Figure 5-12 Pump-On Scenario for Natural Convection with Initial Power of 0.1 MWt and Core Inlet Temperature of 30 °C

Also, UML discussed a second cold water insertion scenario caused by turning on the secondary cooling system during forced flow operation on a cold winter day when the sump temperature is low, but provided experimental data for an initial sump temperature of approximately 55 °F (13 °C) to demonstrate that any core temperature and reactivity change caused by this scenario would be relatively small and slow, especially for the cross-stall mode (where coolant that leaves the heat exchangers is fed into the pool section not occupied by the reactor, instead of being fed directly into the core inlet channel (or downcomer plenum) on the suspension frame, as discussed in SAR Section 5.2 and UML's response to RAI-14.3.33 (Ref. 71)) in which the reactor is normally operated for forced flow operation. Even if the reactor were operated in the downcomer mode (in which coolant is supplied from the heat exchangers directly to the inlet plenum above the core via the core inlet channel, as also discussed in SAR Section 5.2 and UML's response to RAI-14.3.33 (Ref. 71)), any reactivity change from a sudden decrease in primary coolant inlet temperature would be much smaller and slower than an instantaneous insertion of 0.6% Δk/k.

The NRC staff reviewed UML's pump-on analyses and finds that UML evaluated scenarios that reasonably bound possible pump-on reactivity insertion scenarios at the UMLRR, and that the assumptions for UML's analyses, as discussed in the SAR, are generally reasonable and conservative because they are based on nominal or bounding conditions, as appropriate. The NRC staff finds that, for the scenarios that UML evaluated, the SL for the fuel would not be exceeded and the maximum cladding temperatures would remain well below temperatures at which fuel failure could occur.

The NRC staff noted that UML did not consider pump-on scenarios starting from low power (e.g., 0.01 watt-thermal), with the single failure assumption that the short period scram does not function, similar to other analyses above. However, the NRC staff finds that, because scenarios involving a surge of cold primary coolant are more similar to step than ramp insertion events, pump-on scenarios starting from higher power would be reasonably bounding in terms of maximum cladding temperatures reached. Additionally, the NRC staff finds that having the primary pump operating, but the high power scram setting at the natural convection mode setpoint as required by TSs for natural convection operation, would help limit the maximum fuel temperature that could be reached.

As discussed above, UML stated that the PARET model used for the pump-on analyses underestimates the realistic initial natural convection flow rate in the hot channel of the fuel. The NRC staff notes that the initial natural convection flow rate is also likely underestimated for UML's natural convection step and ramp reactivity insertion analyses discussed above, given UML's assumed initial coolant mass flux up through the core of 0.001 kilograms per second per square meter for those analyses. The different initial peak fuel temperatures (i.e., fuel temperatures at the beginning of the transients) for the step, ramp, and pump-on 0.125 MWt natural convection analyses could be due to differences in the degree of convergence of the assumed initial natural convection flow rate to a realistic steady-state value prior to the beginning of the transient. However, the NRC staff finds that underestimation of the initial flow rate is likely a conservative assumption because a lower natural convection flow rate corresponds to a higher initial fuel temperature.

The NRC staff also reviewed UML's discussion of potential reactivity transients caused by turning on the secondary coolant system during forced flow operation and finds that such reactivity transients would be relatively small and slow and, therefore, not significant.

The NRC staff finds that UML's analyses help demonstrate that reactivity insertion events related to cold water insertion would be bounded by other analyzed reactivity insertion events, discussed earlier in this SER section, that result in higher peak cladding temperatures and, therefore, the NRC staff finds that UML's cold water insertion event analyses are acceptable.

Conclusion

The NRC staff reviewed UML's analyses of reactivity insertion scenarios. The NRC staff finds that UML discussed possible methods by which excess reactivity could be accidently inserted (in a rapid or a slow manner) into the UMLRR. The NRC staff finds that UML provided generic analyses demonstrating that possible reactivity insertions within the bounds of the UMLRR TSs could not result in damage to the UMLRR fuel and also provided specific analyses helping to show that cold water insertion events would be bounded by the generic analyses. The NRC staff finds that the methodologies and assumptions used by UML for its reactivity insertion analyses are generally reasonable, conservative, and consistent with established industry practices. As discussed above, the NRC staff also performed an independent confirmatory

calculation of UML's bounding reactivity insertion scenario and the NRC staff's calculation also demonstrated that fuel failure would not occur. Based on the above, the NRC staff concludes that UML's insertion of excess reactivity analyses are acceptable.

5.3 Loss of Coolant

UML's LOCA analyses include analyses to demonstrate that a loss of coolant from the pool could not result in fuel reaching temperatures that could result in fuel failure, as well as analyses of the potential radiological consequences of LOCAs (due to the external gamma (photon) radiation from the unshielded core and cobalt-60 (Co-60) irradiation source). UML's discussion of LOCAs and LOCA fuel integrity analyses is provided in SAR Section 13.4.2, as supplemented by UML's responses to RAI-13.2 (Ref. 23) and RAI-13.6 (Ref. 43) and UML's letters dated February 1, 2018 (Ref. 44), September 30, 2020 (Ref. 98). UML's LOCA radiological analyses are provided in SAR Sections 13.4.2.1 through 13.4.2.11, as supplemented by UML's response to RAI-13.7 (Ref. 43).

UML stated that a LOCA at the UMLRR is a highly unlikely event due to the design of the facility. The UMLRR pool is designed with thick concrete walls and an aluminum liner to resist the most severe earthquake that might reasonably be expected near the facility.

The UMLRR pool penetrations are summarized in SER Table 2-2 in SER Section 2.3. UML stated that there is no penetration of the reactor pool wall below the top of the core that is open to pool water. The primary piping connection penetrations, which are approximately 13 feet below the pool surface, are connected to 10-inch aluminum lines that can feed water to and from the core during forced cooling operation as discussed in SAR Section 5.2. In order to prevent water from being siphoned out of the pool such that the core could become uncovered following a possible primary piping break outside the pool, anti-siphon lines are connected to the discharge and return lines of the primary coolant loop. These lines would prevent water from being siphoned below the level of the primary piping penetrations.

There are several experimental facilities built into the concrete shielding surrounding the pool liner and/or penetrating the pool liner at the level of the core (e.g., the medical embedment and gamma cave). UML stated that these experimental facilities are provided with multiple barriers to protect against the possibility of pool water leakage. For example, as discussed in SER Section 2.3, the medical embedment is bolted closed, and the pool liner at the location of the gamma cave is reinforced by a strengthened aluminum plate.

There are also three in-service beam ports (two 6-inch diameter and one 8-inch diameter) that extend through the concrete pool wall and aluminum liner, and beyond the liner into the pool, at the core centerline level. The in-service beam ports are welded shut at the pool end. (As discussed in SAR Sections 10.2.1 and 10.2.2, for the three additional beam ports that have been removed from use, the inner section that formerly extended into the pool has been removed, and aluminum plates have been bolted over the associated penetrations in the pool liner on the inside of the pool.) Each beam port also has a heavy lead shutter within the pool wall and a bolted shield plug at the outer shield wall, which also help preclude accidental draining through the ports. UML stated that due to radiological considerations, even after an extended shutdown period, the lead shutter and shield plug are not simultaneously opened when the reactor is in the stall end of the pool. Additionally, renewed TS 3.8, which is discussed and found acceptable in SER Section 6.3.8, states that the lead shutter and shield plug may only be simultaneously opened when the reactor is in the bulk pool and also prohibits reactor operation when both the lead shutter and shield plug are opened (regardless of the reactor

location in the pool). As discussed in SER Section 2.3, the reactor is currently only operated above 0.1 MWt when it is located in the stall pool, because only the stall pool primary coolant system connections (needed for forced convection operation) are in service and used (though bulk pool operation above 0.1 MWt using the bulk pool primary coolant connections is not prohibited by TSs).

UML stated that it is highly unlikely that the portions of the beam ports that extend into the reactor pool could be severely damaged (i.e., sheared off) by a falling object while the reactor is in the stall pool, due both to the restricted space, and the protection afforded by the reactor bridge overhead. Additionally, UML stated that per procedure, regardless of the position of the reactor in the pool, UML does not move or handle heavy loads (e.g., using the overhead crane) over the stall pool.

UML also stated that the pool divider gate, which allows the stall and bulk sides of the reactor pool to be sealed off from each other, provides additional protection in the event of a severe leak in one side of the pool. If a severe leak were to occur, UML could prevent the core from becoming uncovered by moving the reactor to the unaffected side of the pool, and closing the gate. UML stated that this operation could be conducted within a few minutes. Because the reactor is required to be in the bulk pool when a beam port lead shutter and shield plug are both opened, as discussed above, the reactor would not need to be moved and only the gate would need to be closed (if not already closed) if a leak through a beam port occurred in this situation. UML further stated that if a leak through a beam port occurred while the lead shutter was open, the shutter could also be closed to help mitigate the leak, given the high density of the lead and the ability of the shutter to displace any water that may be in the shutter assembly.

The NRC staff notes that UML may be unable to close the gate or shutter without offsite power because these actions typically require use of the overhead crane, as discussed in SER Section 6.3.8; however, the NRC staff considers a severe LOCA, coincident with a loss of offsite power, an extremely unlikely scenario. As also discussed in SER Section 6.3.8, the NRC staff notes that although there is a possibility that UML would be unable to immediately close the shutter at the onset of a LOCA due to water pressure in the beam tube, UML should eventually be able to close the shutter as the pool drains and the water pressure drops. The NRC staff further expects that even if the shutter cannot be closed at all (and nothing else is done to mitigate the LOCA), once the pool completely drained to the level of a bottom of a beam tube, UML (depending on radiation levels around the beam tube) may be able to replace the beam tube plug, which would allow the pool to begin to be refilled and the fuel to be re-covered with water, mitigating any temperature rise in the exposed fuel, providing shielding of radiation, and terminating the LOCA event. However, the NRC staff notes that, as discussed later in this SER section, the results of the NRC staff's LOCA confirmatory calculation indicate that no fuel failure would occur even if UML were unable to take any action to mitigate a worst-case LOCA through a beam tube.

UML stated that the UMLRR facility has a pool level sensor that initiates an automatic pool fill system that operates at a flow rate of up to 5 gpm through the cleanup demineralizer. Additionally, a fire hose available within the reactor building could be used to provide up to 300 gpm to the reactor pool in an emergency situation, to help ensure that the core remains covered with water.

Although, as discussed above, UML described how the UMLRR is designed and operated in a manner that would make any event that could uncover the core highly unlikely, UML provided LOCA scenario analyses, which are described below.

UML LOCA Fuel Integrity Analysis

In SAR Section 13.2.4, UML stated that based on historical studies of other plate-type fueled reactors (Ref. 93 and Ref. 94), it is reasonable to conclude that irradiated UMLRR fuel could be adequately cooled by natural air convection if fuel become completely uncovered with water. However, because the complete uncovering of the UMLRR core immediately following reactor operation is an unrealistic scenario based on the design of the UMLRR (including the robust pool construction and the location and design of the pool penetrations, as discussed above), UML analyzed a scenario in which the core becomes only partially uncovered following some period of time after a reactor scram, due to a postulated beam tube break that allows water to drain through the beam tube.

To demonstrate that a UMLRR LOCA could not cause the UMLRR fuel to reach temperatures that could result in fuel failure, UML submitted (as Appendix B to its response to RAI-13.6) an analysis of a LOCA at the 2 MWt Rhode Island Nuclear Science Center (RINSC) research reactor, which UML stated is similar to the UMLRR, as discussed below. UML also provided a discussion and analysis adapting the RINSC analysis to the UMLRR to demonstrate that a UMLRR LOCA occurring due to a beam tube break would not produce unacceptable consequences.

The RINSC analysis submitted by UML demonstrated that, for a LOCA at the RINSC reactor, if the decay power fraction drops to 0.827 percent (i.e., if the decay heat produced by the fuel has decreased to less than 0.827 percent of the heat produced during full-power reactor operation) before the level of water in the pool drops to the elevation of the bottom of an 8 inch beam tube (which is 4 inches below core centerline, because, like the UMLRR, the RINSC beam ports are centered at core centerline), then the fuel cladding temperature will not exceed 530 °C (986 °F) and fuel failure will not occur. This calculation is based on a heat transfer model that, once a portion of the fuel is uncovered, considers heat generated in the fuel to be conducted down the fuel into the water covering the bottom portion of the fuel, and to boil this water away, producing steam that removes additional heat from the exposed portion of the fuel by convection. The water level would continue to very slowly drop after it reaches the bottom of an 8 inch beam tube due to the water boiling away, but the RINSC analysis determined that the continued drop in decay heat due to further decay over time compensates for this additional loss of water and that the water level drop due to boiling can be neglected for the purposes of determining peak fuel cladding temperature. The RINSC heat transfer model assumes that heat generation in the hot plate is uniform in the axial direction; the RINSC analysis demonstrated that this assumption is bounding because it causes a greater fraction of the heat to be generated in the portion of the fuel above the water line than would be expected for a more realistic axial power profile. In the RINSC analysis, the highest temperature in the fuel is at the top of the hot plate, which is furthest from the remaining coolant. Due to the localized boiling, the temperature of the RINSC fuel plate at the surface of the coolant remains only slightly above the saturation temperature of the coolant.

The RINSC analysis provided by UML also includes a drain time model, which is based on an equation derived from Bernoulli's equation, and that is used to calculate the time required for the RINSC pool to drain from a certain level above a drainage hole to another level above the hole for a drainage hole of a certain size (i.e., cross-sectional area). The RINSC analysis determined, based on extrapolation from data in American National Standards Institute/American Nuclear Society (ANSI/ANS)-5.1-2005 (Ref. 65), that assuming the reactor had been operating for an infinite period of time prior to a low pool level scram, it would take

16,232 seconds following the reactor scram for the decay power fraction to drop to 0.827 percent. This time is used with the drain time model equation to solve for the maximum size hole(s) through which a RINSC LOCA could occur such that the pool would drain slowly enough that the decay power fraction can sufficiently decrease. The RINSC analysis conservatively assumes that the drain time is the time to drain to the top of the RINSC grid box, which is well above the top of the RINSC fuel matrix, instead of the time to drain to the bottom of an 8 inch beam port, below the core centerline. Additionally, although for the RINSC reactor forced convection flow through the core from the primary coolant system would continue following a low pool level scram unless the primary pump was manually turned off, the RINSC analysis conservatively assumes that the primary pumps shut off at the time of the low pool level scram.

UML stated that the RINSC LOCA analysis is relevant to the UMLRR because the RINSC and UMLRR facilities have similar physical characteristics. Both are open pool reactors using similar MTR plate-type fuel, and both have a similar reactor core structure and beam ports (although the UMLRR has fewer beam ports in use because three have been removed from use, as discussed above). In terms of physical core differences, the most significant is that the RINSC reactor has a (nominally) 14 element, beryllium-reflected core, while the UMLRR has a (nominally) 21 element, graphite- and water-reflected core. UML stated that the RINSC fuel also has more plates per element than UMLRR silicide or aluminide fuel, and smaller fuel channel width (0.2620 centimeters versus 0.2953 centimeters or 0.2709 centimeters for UMLRR silicide or aluminide fuel, respectively). UML stated that the widths of the UMLRR and RINSC fuel matrix are similar (however, the NRC staff notes that the widths stated in UML's response to RAI-13.2(a), (b), and (d) appear to contain a misplaced decimal point, compared to the values provided in SAR Table 4-1 and in the RINSC LOCA analysis provided by UML).

The NRC staff notes that although the matrix thickness of the UMLRR silicide fuel is less than that of UMLRR aluminide fuel as discussed in SER Section 2.2.1, the matrix thickness of the limiting UMLRR silicide fuel and the RINSC fuel are similar. The NRC staff also notes that the UMLRR silicide and aluminide fuel matrix height is slightly greater than the RINSC fuel matrix height, but finds that this adds conservatism in application of the RINSC model to the UMLRR, because a slightly larger fraction of the UMLRR fuel matrix would be below water when the pool water level reaches the bottom of the 8 inch beam tube.

The NRC staff notes that the RINSC fuel is silicide fuel. Although the UMLRR can contain aluminide fuel in addition to silicide fuel, the UMLRR silicide fuel is limiting from a peak plate power standpoint, as discussed in SER Section 2.5.1. The NRC staff finds that the fact that the highest decay power generation (and greatest risk of fuel element overheating) is in the UMLRR silicide fuel supports UML's application of the RINSC LOCA model to the UMLRR.

UML stated that there are some differences in the thermal conductivity of materials used in the UMLRR and RINSC reflectors and flux traps, which could help conduct additional heat from the fuel down into the coolant during a LOCA. While the thermal conductivity of both graphite (used in the UMLRR reflector) and beryllium (used in the RINSC reflector, including the central flux trap) are high, the thermal conductivity of beryllium is higher. The thermal conductivity of aluminum (used in the UMLRR flux trap, which is not considered part of the UMLRR reflector) is even higher than the thermal conductivity of beryllium. However, the RINSC heat transfer model ignores heat transfer to the reflector, including the flux trap, and the NRC staff finds that this adds conservatism in the application of the RINSC model to either the UMLRR or the RINSC.

The NRC staff also notes that, as discussed in NUREG-1313 (Ref. 52), the thermal conductivities of a silicide fuel matrix (used in UMLRR and RINSC fuel) and an aluminide fuel matrix (used in UMLRR fuel) are similar.

UML stated that given the similar general physical characteristics of the RINSC and UMLRR cores and fuel, the larger convection cells in the UMLRR fuel (given the larger channel widths), and the lower peak plate power and average plate power in the UMLRR core, it is reasonable to assume that the heat transfer characteristics of the UMLRR core would be similar or more conservative than those in the RINSC core, and it is therefore reasonable to apply the general RINSC heat transfer model for the UMLRR.

UML calculated that based on the RINSC hot plate power (for 2 MWt full-power operation) of 9.653 kWt and a UMLRR hot plate power of 6.25 kWt, which is greater (more conservative) than the 5.93 kWt peak plate power associated with the UMLRR LCC (see SER Section 5.1), the UMLRR peak plate power is approximately 65 percent of the RINSC peak plate power. UML divided the RINSC maximum decay power fraction, 0.827 percent as discussed above, by the ratio of UMLRR to RINSC peak plate power to determine a UMLRR maximum decay power fraction (to prevent fuel cladding failure when the water in the pool has fallen to the level of the bottom of an 8 inch beam tube) of 1.28 percent. UML used this decay power fraction with the data in ANSI/ANS-5.1-2005 to extrapolate that, assuming the UMLRR had been operating at full power for an infinite period of time prior to a low pool level scram, it would take 3,947 seconds following the reactor scram for the decay power fraction to drop to 1.28 percent.

UML used its UMLRR decay time, 3,947 seconds, with the RINSC drain time model equation to solve for the maximum area hole through which a UMLRR LOCA could occur such that the pool would drain slowly enough that the decay power fraction can sufficiently decrease. For this equation, UML assumed a pool surface area of 350 square feet as an estimate of average pool surface area that conservatively accounts for the fact that the pool surface area would decrease as the pool level dropped during a LOCA. The surface area would decrease because the actual pool surface area is 400 square feet (200 square feet each for the stall and bulk portions of the pool), but the walls of the stall pool taper down (see SER Figure 1-1) such that the area of the stall pool is only approximately 110 square feet at the bottom. In using the equation, UML also assumed that the initial pool level is 30.25 feet (24.25 feet above core centerline), equivalent to the LSSS for pool level, and that the pool drains to 6 feet (the level of the core centerline and the center of the beam tubes). UML calculated a maximum drainage area of 0.11 square feet, which is equivalent to a round hole with a diameter of approximately 4.5 inches.

The NRC staff notes that, as discussed above, the RINSC drain time model equation is used to calculate the time required for the pool to drain from a certain level above a drainage hole to another level above the hole for a drainage hole of a certain size. The NRC staff notes that, in applying the drain time model equation for the UMLRR, UML took the initial and final pool levels to be the levels above the bottom of the pool, rather than the levels above the pool drainage point (i.e., a beam tube). The NRC staff finds that UML's use of pool levels above the bottom of the pool is conservative, because it overestimates the water level above the drainage point during the entire draining period, causing the rate at which water flows out of the pool through the draining point to also be overestimated.

UML stated that there are no other pipes (e.g., conduit, fill, or drainage pipes), other than the beam tube penetrations themselves, associated with the beam tubes that exceed 4 inches in diameter. While the beam tubes exceed the 4.5 inch maximum drain diameter calculated by UML, UML stated that given the combination of controls and design features at the UMLRR

(such as design features that make the shearing of a beam tube highly unlikely, the TS prohibition on having the lead shutter and shield plug of a beam port simultaneously opened, procedural controls on moving or handling heavy loads over the stall pool, and the ability to close the pool divider gate to mitigate a LOCA), a scenario in which the pool could be drained to the level of the fuel through a 6 or 8 inch beam port, and fuel cladding failure could possibly occur, is not credible. UML also noted that its analysis of a LOCA event uses very conservative assumptions, including the assumption that the reactor had been operated for an infinite period of time prior to the event (the reactor actually has been operated much less frequently even though there is no restriction on operating frequency in the reactor license).

The NRC staff reviewed UML's LOCA fuel integrity analysis described above. The NRC staff notes that the RINSC analysis provided by UML is similar to the RINSC LOCA analysis that was previously reviewed and approved by the NRC staff for the renewal of Facility Operating License No. R-95 for the RINSC research reactor, as documented in the NRC staff's SER related to the RINSC license renewal, dated January 5, 2017 (Ref. 66). Based on the discussion in UMLRR SER Section 2.2.1, the NRC staff finds that 530 °C (986 °F) is a reasonable assumed fuel temperature limit to prevent cladding failure of either the RINSC or the similar UMLRR aluminum-clad MTR fuel; 530 °C (986 °F) is the UMLRR SL defined in UMLRR TS 2.1, which is discussed and found acceptable in SER Section 2.2.1, and the NRC staff notes that the RINSC reactor has the same SL. Given the similarities between the RINSC reactor and the UMLRR, particularly the similarities in the fuel, core design and layout, and beam tube location and design, the NRC staff finds that UML's application of the RINSC heat transfer model to the UMLRR is reasonable. Additionally, because of the conservatism built into the RINSC model, and because of minor differences in the UMLRR design such as the larger fuel plate channel widths, the NRC staff finds that the use of this model for the UMLRR is conservative. The NRC staff also finds that UML's adjustment of the decay power fraction to account for the lower peak plate power in the UMLRR is appropriate. The NRC staff finds that the RINSC drain time model is sufficiently general such that it may reasonably be applied to the UMLRR, and the NRC staff further finds that the way UML used the drain time model for the UMLRR, including its pool surface area assumptions, initial and final pool levels, and reactor operating history assumptions, are reasonable and conservative. The NRC staff finds that UML's analysis conservatively ignores the ability of the primary coolant system to continue to provide some level of core cooling after the onset of a LOCA. The NRC staff finds that UML's analysis helps demonstrate that any LOCA resulting from a 4.5 inch or smaller diameter hole at the core centerline could not cause the UMLRR fuel to reach temperatures at which fuel cladding failure could occur. The NRC staff additionally finds that, because of the controls and facility design features that would help prevent or mitigate any severe LOCA that could result from a greater than 4.5 inch diameter hole at or below the core centerline (such as through a beam tube) and possibly result in fuel failure, such a severe unmitigated LOCA is an extremely unlikely scenario for the UMLRR. Based on the above, and also based on the results of the NRC staff's calculations of UMLRR LOCA fuel cladding temperatures, which are discussed below, the NRC staff finds that a UMLRR LOCA that could result in fuel failure is not a credible accident and that UML's LOCA fuel integrity analysis is acceptable.

NRC Staff Confirmatory Calculations of LOCA Fuel Integrity

The NRC staff performed a calculation of the maximum fuel cladding temperatures that could be reached following a LOCA at the UMLRR to determine if it could confirm UML's determination that a credible LOCA would not result in fuel failure. Although, as discussed above, an unmitigated UMLRR LOCA from a greater than 4.5 inch diameter hole is an extremely unlikely scenario, the NRC staff conservatively considered a scenario in which the reactor pool is able to

drain unmitigated through an 8 inch diameter (equivalent to 0.03243 square meters by area) beam tube.

For its analysis, the NRC staff used TRACE. As discussed in SER Section 2.6, for the LOCA calculations, the NRC staff used a TRACE model (illustrated in SER Figure 5-13 below) that is modified from the model (illustrated in SER Figure 2-15 in SER Section 2.6) used for all confirmatory steady-state thermal-hydraulic, reactivity transient, and loss of flow analyses discussed in this SER. The LOCA TRACE model adds a pool pipe component (numbered 150 in SER Figure 5-13) and a break component (numbered 20 in SER Figure 5-13) for a variably-sized drainage hole. For its LOCA analysis, the NRC staff set the break component drainage hole size to 0.03243 square meters and also set the break to be located approximately 4 inches below the core centerline (to be representative of the level of the bottom of an 8 inch beam tube, which is the lowest level to which the pool could drain following any 8-inch beam tube break). The break component allows TRACE to calculate the pool drain time. The NRC staff's LOCA fuel integrity analyses assume power peaking factors consistent with the hot plate of the LCC and use explicit hot channel factors, similar to the NRC staff's other TRACE confirmatory analyses for the UMLRR.

Similar to UML's analysis, the NRC staff assumed that the reactor scrams on low pool level when the reactor pool is at a height of 24.25 feet above the core centerline, consistent with the pool level LSSS. Also similar to UML's analysis, the NRC staff assumed that the reactor has been operating for an infinite period of time prior to the scram, such that fission product inventory and decay heat would be maximized.

As discussed in SER Section 2.3, the 31-foot-deep reactor pool holds approximately 76,000 gallons of water. However, because only the portion of the water above the drainage point (assumed to be at the lowest point of an 8-inch beam tube) would drain, because the pool would already be approximately 0.75 foot below its full level when the reactor scrams, and considering that the walls of the stall portion of the pool taper down, the NRC staff used 49,600 gallons as a conservative estimate of the pool water available to drain out of the core for its analysis. Although drainage from the pool stops once the level has dropped to the drainage point, the NRC staff considered that pool level would continue to slowly drop due to water boiling off near the submerged portion of the fuel.

In response to RAI-13.6.c, UML stated that the bottom of the UMLRR core box is hydraulically connected to the reactor pool because there is an approximately 0.75-inch diameter through-hole located at the bottom center of the lower (outlet) core coolant plenum. The NRC staff also notes that, given the design and construction of the core box, it is not likely to be leak tight. For its analysis, the NRC staff therefore conservatively assumed that the core box is hydraulically connected to the pool via a 0.5-inch diameter hole. This hydraulic connection allows water to circulate between the core box and the pool once the pool water level has dropped below the top of the core box, leading to improved cooling of the core.

The results of the NRC staff's LOCA fuel integrity calculations are illustrated in SER Figure 5-14 and Figure 5-15 below. Figure 5-14 shows the peak cladding temperature versus time following the reactor scram. Figure 5-15 shows the mass flow rate of coolant out of the pool versus time following the reactor scram.

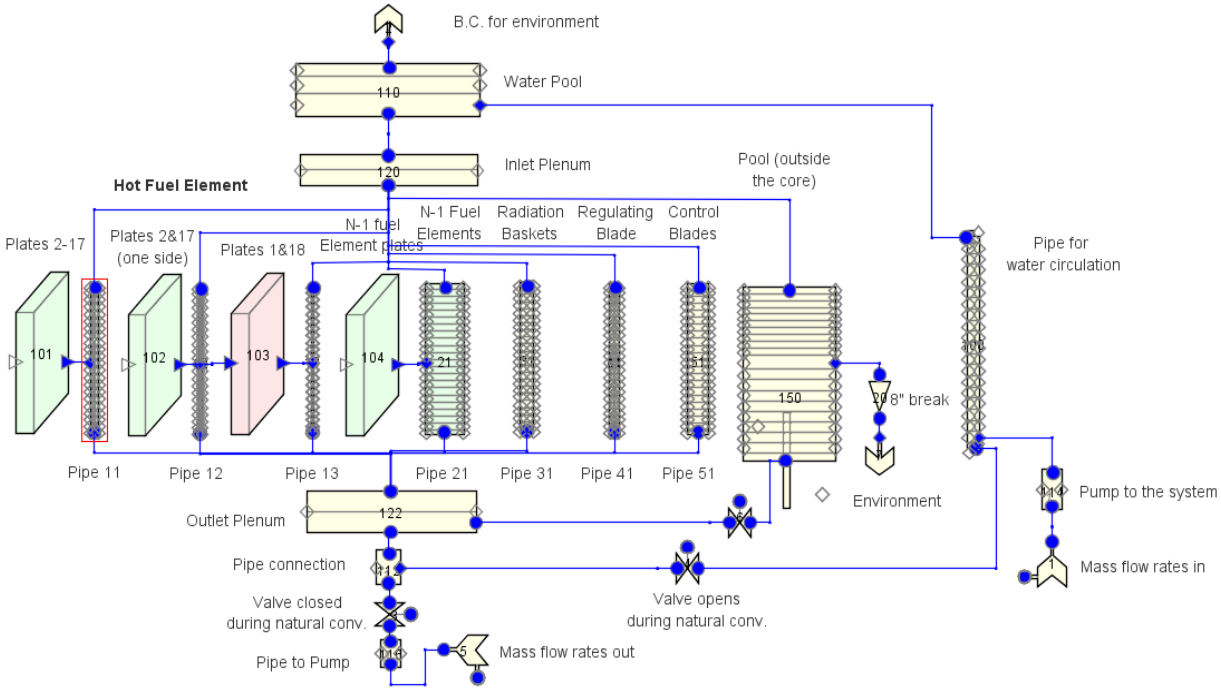

Figure 5-13 LOCA TRACE Model for NRC Confirmatory Calculations

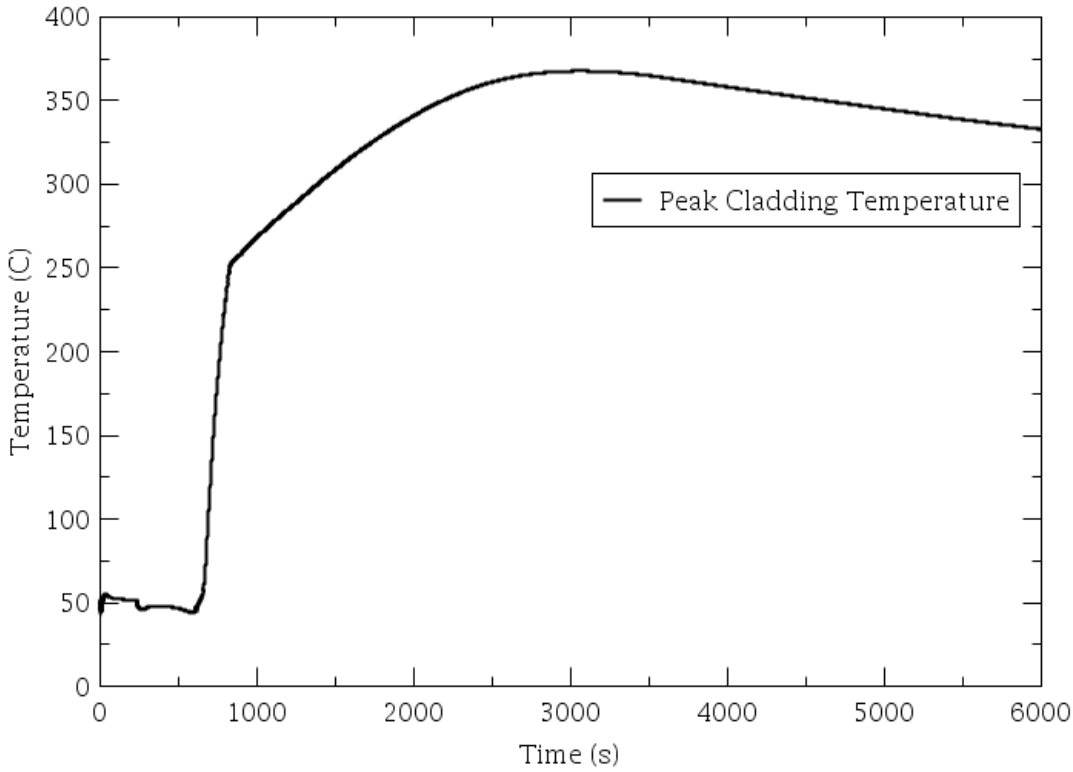

Figure 5-14 Peak Fuel Cladding Temperature versus Time for NRC Staff LOCA Fuel Integrity Confirmatory Analysis

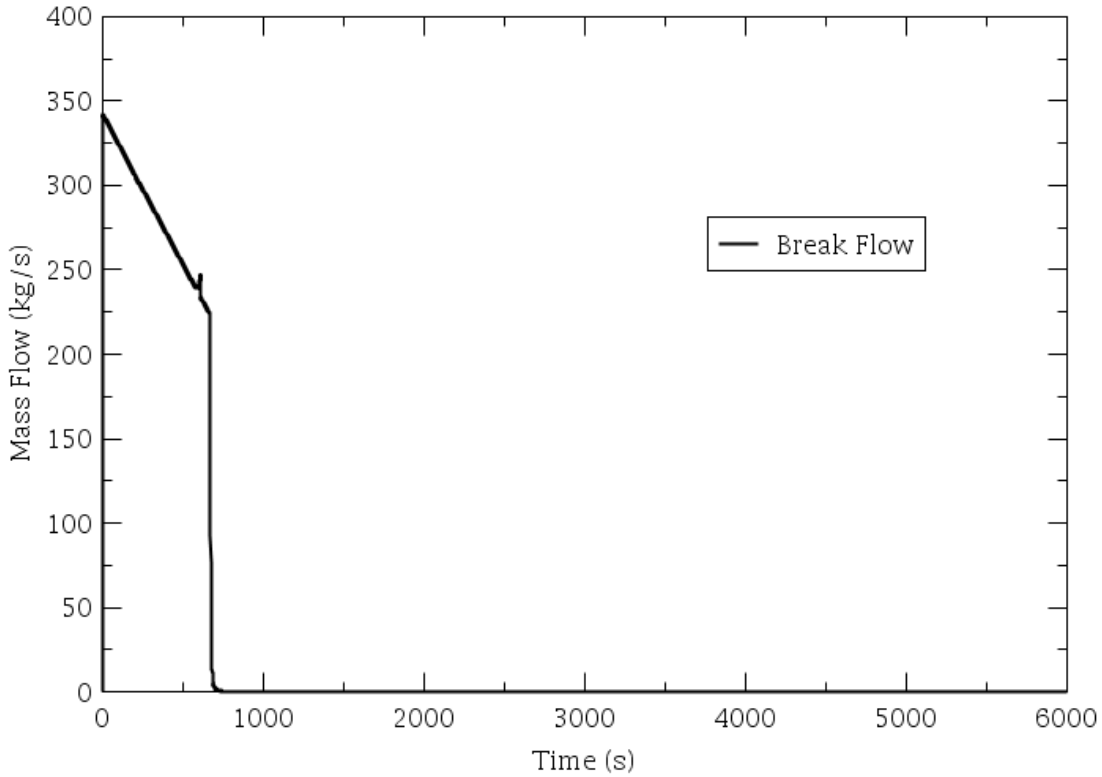

Figure 5-15 Coolant Leakage Rate from the Pool versus Time for NRC Staff LOCA Fuel Integrity Confirmatory Analysis

The NRC staff's analysis determined that, for an 8-inch diameter break, the maximum fuel cladding temperature is approximately 367.7 °C (693.9 °F), which is well below the SL and temperatures at which fuel cladding failure could occur. The pool drains much more quickly than in UML's analysis, given the larger break size. The analysis shows that the cladding temperature remains relatively low and constant while the pool is draining, then increases to a maximum after a portion of the fuel becomes exposed. For the 8-inch diameter break, the maximum temperature is reached approximately 3,038 seconds after the scram. (The NRC staff notes that, because the RINSC and UML analyses discussed above are based on a steady-state solution, they are independent of, and do not calculate, the time at which the maximum temperature occurs.)

The NRC staff acknowledges that, although the pool drains more quickly in its analysis than in UML's analysis, the maximum fuel cladding temperature is much lower. However, the NRC staff notes that the differences in the maximum fuel cladding temperatures in UML's and the NRC staff's analyses likely stem from the use of significantly different methodologies and assumptions. As discussed above, the NRC staff finds that the RINSC analysis, as adapted by UML for the UMLRR, includes many conservatisms.

The NRC staff notes that the time between when the core becomes uncovered and reaches its maximum temperature is significant (approximately 40 minutes), based on the NRC staff's confirmatory analysis. Therefore, in an unlikely scenario in which the pool could completely and rapidly drain to the level of a beam tube without mitigation, UML would likely have time to perform mitigating actions after the pool completely drained (e.g., actions to close the beam tube plug and/or shutter and to begin refilling the pool) before the maximum fuel temperature were reached.

In summary, the results of the NRC staff confirmatory calculations of LOCA fuel integrity confirmed the NRC staff's findings above that a UMLRR LOCA that could result in fuel failure is not a credible accident and that UML's fuel integrity analysis of a LOCA is acceptable.

UML LOCA Radiological Consequence Analysis

As discussed above, UML also provided analyses of the potential radiological consequences of a LOCA. The radiological hazard could occur due to the external gamma (photon) radiation from the unshielded core following the LOCA. UML additionally provided an analysis of the radiological hazard that could result from the Co-60 irradiation source in the pool losing its pool water shielding.

UML stated that, in the unlikely event that the pool completely drains, there would be the potential for both direct and indirect doses from the radiation sources in the pool. A direct dose could occur for individuals in the reactor building who are directly exposed to the primary photons emerging from the top of the open and empty pool. The indirect dose is attributed mainly to primary photons that emerge from the open pool and scatter (i.e., due to Compton scattering interactions) off the reactor building dome. The primary source terms for direct/indirect dose rate are the Co-60 source and the reactor core. UML considered these as unshielded point sources 6 feet above the bottom center of the reactor pool. The pool is approximated as a cylinder, with a radius chosen to provide a top surface area for the cylinder that is similar to top surface area of the actual (rectangular) pool, thus providing a comparable irradiation of the reactor dome by primary photons that emerge from the pool surface.

For determining the photon emission from the Co-60 source, UML assumed that the Co-60 source term is 100,000 Ci, which is equal to the maximum Co-60 irradiation source possession limit in the UMLRR license (see SER Section 1.10), and that two 1.25 mega electron-volt (MeV) photons are emitted per disintegration. The Co-60 source photon emission rate is assumed to be independent of time, given that the half-life of Co-60 is approximately 5 years. For determining photon emission from fission products in the reactor core, UML used an approach described in Shultis and Faw (Ref. 67) to estimate the emission rate for photons from each of six energy groups (with average energies of 0.5, 1.5, 2.5, 3.5, 4.5, and 6.25 MeV), as a function of time. UML assumed that the reactor had either been operated at full power for six hours (to represent typical reactor operations) or an infinite period of time (to represent bounding conditions), prior to a reactor scram caused by the LOCA.

UML calculated the dose rates directly above the reactor pool (at the level of the top of the pool wall, 27 feet from the sources, from direct radiation), at the edge of the reactor pool (from indirect/scattered radiation), and outside the reactor building truck door (from indirect/scattered radiation). UML calculated dose rates from the core for six elapsed times following the scram: 27 minutes (UML stated that it determined that this is the approximate time required for a rupture in an 8-inch beam tube to cause the reactor pool to drain to the level of the core

centerline), 1 hour, 1 day, 7 days, 30 days, and 90 days. The Co-60 irradiation source dose rate is assumed to be independent of time.

For calculating direct dose rates from the photons emitted by the sources, UML considered attenuation (and buildup) in the intervening air. For calculating indirect dose rates, UML did not consider attenuation of the primary photons in the air between the sources and the reactor dome, attenuation of primary or scattered photons in the reactor dome, or attenuation of scattered photons in air between the dome and the truck door (for the truck door exterior dose calculation). UML considered the attenuation (and buildup) of scattered photons in the air between the dome and the reactor pool edge (for the pool edge dose calculation) and in the truck door (for the truck door exterior dose calculation).

For its calculation of indirect dose rates at the truck door exterior for the reactor core, UML considered the annihilation photons generated by interactions (specifically, pair production) of primary reactor core photons in the dome, in addition to the photons generated by Compton scattering. UML estimated that the additional dose rate from annihilation photons in this case is small, adding less than about 6 percent to the total dose rate. For other indirect dose rate calculations (for the Co-60 source and/or at the edge of the reactor pool), UML did not consider the annihilation photon contribution because it determined that it would not be significant, given the low probability of pair production interactions for Co-60 photons and the relatively small fraction of annihilation photons in the radiation at the pool edge (outside the truck door, annihilation photons make up a larger fraction of the photon radiation, given that they more readily penetrate the 0.375-inch thick steel (assumed to be iron) truck door).

For its calculations of indirect dose rates due to Compton scattered or annihilation photons, UML numerically integrated expressions for the fluence rate of photons from the dome over the irradiated surface of the reactor dome, in order to consider the different possible photon paths and scattering angles between the sources and a receptor location. UML's expressions for fluence rate of Compton scattered photons from the dome included the Klein-Nishina differential cross section formula photon scattering.

UML did not consider any photon scattering or other interactions in structures within the reactor building (e.g., floors and the walls of the reactor building or reactor pool enclosure), including "double-scatter" events (i.e., additional scattering of photons which already scattered off of the dome). However, to account for the possibility of increased indirect doses due to these additional scattering events, UML conservatively assumed that the reactor dome is constructed of iron with a uniform thickness of 1 inch, rather than the actual thickness of approximately 0.625 inch in the area of the dome where most photons would be incident, which results in increased photon scattering from the dome.

The NRC staff noted that, as discussed in SAR Section 3.5.1, there is a 2-inch-thick layer of fiberglass insulation attached to the interior of the iron reactor dome. However, the NRC staff finds that any additional photon scattering or other interactions in the fiberglass would be small relative to those in the iron and that UML's conservatively increasing the thickness of the iron in the dome would help account for any additional scattering in the fiberglass.

The results of UML's LOCA dose rate calculations are shown in SER Table 5-7, Table 5-8, Table 5-9, and Table 5-10 (along with the results of the NRC staff's calculations, which are discussed below, done to determine if the NRC staff could confirm the results of UML's LOCA dose rate calculations). SER Table 5-7 shows the direct dose rates above the reactor pool from the uncovered reactor core. SER Table 5-8 shows the indirect dose rates at the edge of the

reactor pool from the uncovered reactor core. SER Table 5-9 shows the indirect dose rates at the truck door exterior from the uncovered reactor core. SER Table 5-10 shows the direct and indirect dose rates from the uncovered Co-60 irradiation source.

Table 5-7 Direct Dose Rates above the Reactor Pool from the Core

Time after scram	Direct dose rates in mrem/hr following 6 hours of reactor operation		Direct dose rates in mrem/hr following infinite reactor operation	
	UML calculation	NRC calculation	UML calculation	NRC calculation
27 minutes	7.14×10^{6}	9.62×10^{6}	1.26×10^{7}	2.33×10^{7}
1 hour	4.62×10^{6}	6.40×10^{6}	9.95×10^{6}	1.99×10^{7}
1 day	2.21×10^{5}	4.59×10^{5}	3.89×10^{6}	1.05×10^{7}
7 days	2.58×10^{4}	4.98×10^{4}	2.27×10^{6}	7.13×10^{6}
30 days	4.70×10^{3}	8.83×10^{3}	1.26×10^{6}	5.33×10^{6}
90 days	1.15×10^{3}	2.37×10^{3}	7.42×10^{5}	4.28×10^{6}

Table 5-8 Indirect Dose Rates at the Edge of the Pool from the Core

Time after scram	Indirect dose rates in mrem/hr following 6 hours of reactor operation		Indirect dose rates in mrem/hr following infinite reactor operation	
	UML calculation	NRC calculation	UML calculation	NRC calculation
27 minutes	4.34×10^{4}	1.66×10^{3}	8.65×10^{4}	4.03×10^{3}
1 hour	2.81×10^{4}	1.11×10^{3}	7.06×10^{4}	3.43×10^{3}
1 day	1.85×10^{3}	7.94×10^{1}	3.23×10^{4}	1.82×10^{3}
7 days	1.97×10^{2}	8.62×10^{0}	1.85×10^{4}	1.23×10^{3}
30 days	3.16×10^{1}	1.53×10^{0}	1.15×10^{4}	9.21×10^{2}
90 days	1.09×10^{1}	4.10×10^{-1}	7.36×10^{3}	7.40×10^{2}

Table 5-9 Indirect Dose Rates at the Truck Door Exterior from the Core

Time after scram	Indirect dose rates in mrem/hr following 6 hours of reactor operation		Indirect dose rates in mrem/hr following infinite reactor operation	
	UML calculation	NRC calculation	UML calculation	NRC calculation
27 minutes	2.58×10^{3}	1.59×10^{2}	4.89×10^{3}	3.84×10^{2}
1 hour	1.67×10^{3}	1.06×10^{2}	3.94×10^{3}	3.28×10^{2}
1 day	9.71×10^{1}	7.58×10^{0}	1.70×10^{3}	1.74×10^{2}
7 days	1.07×10^{1}	8.23×10^{-1}	9.79×10^{2}	1.18×10^{2}
30 days	1.79×10^{0}	1.46×10^{-1}	5.86×10^{2}	8.79×10^{1}
90 days	5.51×10^{-1}	3.91×10^{-2}	3.67×10^{2}	7.06×10^{1}

Table 5-10 Direct and Indirect Dose Rates from the Co-60 Irradiator

Location	Dose rates in mrem/hr	
	UML calculation	NRC calculation
Direct 10-m above the reactor	1.89×10^{6}	1.80×10^{6}
Indirect at the edge of the pool	5.31×10^{3}	2.11×10^{2}
Indirect at the truck door exterior	4.62×10^{2}	4.62×10^{1}

UML noted that its calculations indicate that the direct dose rates above the reactor pool (from the core or the Co-60 irradiation source) would be very high. However, UML stated that, in practice, personnel would be evacuated from the reactor building during any LOCA event before the core could drain to the level of the core or the Co-60 irradiation source, meaning that personnel should not actually be exposed to these dose rates.

UML also noted that, although its calculations show significant dose rates at the truck door exterior, most the reactor building is thick concrete with an exterior steel lining, which provides much more shielding than the truck door. UML stated that most of the scattered or annihilation photons that are obliquely incident on the containment wall would be attenuated to negligible levels, and based on the geometry of the UMLRR, at any distance beyond about 24 feet from the truck door, all scattered or annihilation photons from the reactor building dome are intercepted at an oblique angle by the wall of the containment vessel. Therefore, UML stated that, at this distance, the dose rate would be negligible (i.e., much less than 2 mrem per hour).

The NRC staff reviewed UML's LOCA radiological consequence analysis methodology and assumptions. The NRC staff finds that UML's methodology for the analysis approximates important radiation interactions in the reactor building and is, therefore, reasonable. The NRC staff also finds that UML's assumptions for its analysis are conservative and are consistent with the design and geometry of the reactor pool and building and the maximum radioactive material inventory that may be present in the core and the Co-60 irradiation source. The NRC staff notes that specific conservative assumptions in UML's calculation include ignoring self-shielding within the core and the Co-60 irradiation source (given that the sources are assumed to be point

sources); ignoring shielding from the water that would likely still cover the bottom portion of the core (given that, even for a severe LOCA involving drainage through a beam tube, the pool could only drain to the level of the bottom of the beam tube, which is above the bottom of the core); ignoring the shielding provided by the core support structure and piping and the interior structure of the reactor building (i.e., floors and walls); and, as discussed above, ignoring attenuation of incident and scattered photons in the dome, for calculation of indirect dose rates. The NRC staff notes that, although 6 hours is a typical operation duration for the UMLRR, the core dose rate calculations following only 6 hours of operation could underestimate the dose from longer-lived fission products remaining in the core from previous operation. However, the NRC staff finds that UML's calculation of dose rate following infinite reactor operation bounds any possible UMLRR operational history. Infinite reactor operation assumes that all fission products are at saturated (maximum) inventories, even though, in reality, many of the fission products that are important contributors to external dose rate from reactor fuel, such as cesium-137, would not reach saturation even after decades of continuous reactor operation. The NRC staff notes that UML's estimate of 27 minutes for the drain time through an 8-inch beam tube is longer than both the NRC staff's approximately 11-minute estimate in its LOCA fuel integrity confirmatory calculations (see SER Figure 5-15) and the NRC staff's approximately 20-minute estimate based on scaling UML's calculation performed for a 4.5 inch drainage hole (see SER Section 6.3.8). However, given that an unmitigated LOCA through an 8-inch diameter hole is extremely unlikely, as discussed above, and that it would likely take longer than 27 minutes for the pool to drain to the core level in any realistic LOCA scenario, the NRC staff finds that 27 minutes is a reasonable estimate of minimum drain (and core decay) time for the purposes of estimating the radiological consequences of a realistic LOCA scenario. The NRC staff also recognizes that mitigating actions could be taken to help prevent any LOCA from causing the pool to drain to the core level, as discussed above.

The NRC staff also reviewed UML's LOCA radiological consequence analysis results. The NRC staff finds that, given the very high dose rates from direct radiation above the pool following a complete LOCA, the reactor bridge would be inaccessible. However, given that an instantaneous LOCA is not a credible accident, if personnel were on the reactor bridge during the initiation of a LOCA, there would be sufficient time to evacuate the personnel from the bridge. The dose rates in other areas within the reactor building, including the edge of the pool, would also be significant (because of the additional distance and/or shielding between the sources and most reactor building areas other than the pool edge, the NRC staff finds that the dose rates at the pool edge would reasonably bound dose rates in most other areas of the building). However, given that in any realistic LOCA scenario, the pool would likely take significant time to drain (when the pool is partially drained, dose rates would still be elevated, but would be well below the values listed in the tables above), and that it would only take 10 minutes or less to evacuate the reactor building (as shown by UML evacuation drills that demonstrate personnel within the reactor building can be evacuated within 5 minutes; see SER Section 5.1), the NRC staff finds that any dose to occupational workers prior to their evacuation would be low, and well below the 5,000 mrem occupational dose limit in 10 CFR 20.1201. If personnel need to temporarily remain in (or re-enter) the reactor building following the onset of the LOCA to perform mitigating actions, as discussed above and in SER Section 6.3.8, the NRC staff finds that these actions could be performed within a few minutes and that reactor staff completing mitigating actions could locate themselves to minimize any direct radiation from the fuel or the Co-60 irradiation source in the pool and, therefore, would primarily only be exposed to scattered radiation (if any). However, based on the NRC staff's confirmatory LOCA fuel integrity analysis discussed above, even if the pool drained very rapidly (i.e., in as little as 11 minutes through a sheared 8 inch beam tube) and UML personnel had to evacuate the reactor building before any mitigating actions could be completed, the NRC staff finds that a SL would

not be reached and that no core fuel failure would occur. Additionally, even if a complete LOCA were to occur, the NRC staff finds that the dose rates would be low enough that personnel could likely occupy most areas within the reactor building for brief periods of time, if necessary, to perform any mitigating actions that were still appropriate (e.g., actions needed to refill the pool) and/or actions needed for recovery operations. Therefore, the NRC staff finds that occupational doses due to shine from the reactor core and the Co-60 irradiation source during a LOCA could reasonably be maintained below the 5,000 mrem occupational dose limit in 10 CFR 20.1201.

The NRC staff notes that UML's calculations show that the dose rate at the truck door exterior could be as high as 4,890 mrem per hour from the core, plus an additional 462 mrem per hour from the Co-60 irradiation source, for a LOCA in which the pool completely drained within 27 minutes. However, these calculations are based on multiple conservative assumptions, as discussed above. Additionally, the NRC staff notes that, as discussed in SAR Section 6.2.1 and UML's response to RAI-6.1 (Ref. 23), there is an additional roll-up door outside the truck door that creates a small vestibule space between the truck door and roll-up door, and there is also a chain link fence and a vehicle crash beam about 2.5 meters from the truck door (see Figure 6.1.1 in UML's response to RAI-6.1). Therefore, although it is outside the UMLRR licensed boundary, UML's response to RAI-6.1 states that the area within about 2.5 meters of the truck door would have limited public accessibility (i.e., would not normally be occupied by members of the public). As discussed in SAR Section 13.2.1.3, beyond the fence and crash beam is a parking lot area, which is accessed, though not continually occupied, by members of the public. Given the additional distance to the parking lot area from the truck door and the geometry of the reactor building and truck door, the NRC staff finds that dose rates in areas that would realistically be accessed by members of the public for any significant amount of time would be significantly less than the dose rates at the truck door exterior. As stated above, UML estimated that, beyond about 24 feet from the truck door, the dose rate would be much less than 2 mrem per hour; the NRC staff finds that this estimate is reasonable given that most scattered radiation reaching those areas would have passed through a significant thickness of concrete. Although some areas beyond 24 feet from the truck door, such as nearby buildings, are normally continually occupied by members of the public, the maximum possible LOCA dose rate (even for a short duration) in those areas would therefore be small.

Further, the NRC staff notes that, because the UMLRR is located on the UML campus, much of the area beyond the UMLRR licensed boundary, including nearby parking lots and buildings, is under Commonwealth of Massachusetts control. Therefore, the NRC staff finds that UML could control access to this area, as needed, should elevated dose rates exist, and should likely have time to do so, given the time that it would take for the pool to drain for any realistic LOCA scenario. Although the pool could theoretically drain to expose the core and the Co-60 irradiation source in as little as 11 minutes, this worst-case scenario is extremely unlikely, as discussed earlier in this SER Section. The NRC staff finds that such a rapid drainage scenario (where the pool could potentially drain before any individuals could be removed from the area outside the truck door), combined with a worst-case UMLRR operating history that maximizes dose rate from the core, and an individual being in a location outside the truck door in a geometry and for a sufficient period of time to allow the individual to receive an excessive dose greater than 100 mrem, represents an even more unlikely scenario. Based on the information above, the NRC staff finds that there is reasonable assurance that any public doses due to shine from the reactor core and the Co-60 irradiation source during a LOCA will be maintained below the 100 mrem dose limit for members of the public in 10 CFR 20.1301.

The NRC staff notes that other radioactive material in the reactor pool, particularly irradiated fuel, which is stored in racks along the pool wall as discussed in SER Section 2.7, could also

contribute to elevated dose rates in and outside the reactor building in the event of a LOCA. As discussed in SAR Section 3.5.9, fuel stored in these racks is shielded by at least 14 feet of pool water. Although fuel in the racks could be at a higher location in the pool than the core or the Co-60 irradiation source, the NRC staff finds that any LOCA that could cause irradiated fuel in the storage racks to become uncovered would still be unlikely because the fuel is still below any of the pool penetrations except for the experimental facilities, discussed above, located at the core centerline (the primary piping connections are approximately 13 feet below the pool surface, as discussed above, which is above any fuel stored in the pool). Additionally, in its letter dated September 30, 2020 (Ref. 98), UML stated that for fuel being stored in the pool, it typically places higher-activity fuel (i.e., fuel that has been more recently irradiated and/or has higher burnup, which generates more radiation and decay heat) in the lower racks, which are deeper in the pool and at approximately the same height as the core (see SER Figure 1-1 and SAR Figure 9-3); the NRC staff notes that this would help reduce any radiation from fuel stored in the pool that became exposed. Even if stored irradiated fuel did become uncovered, the NRC staff finds that the radiological consequences would likely be bounded by UML's estimated consequences of the core and/or the Co-60 irradiation source becoming uncovered, based on the radioactive decay that would have occurred for the stored fuel and given the conservatism in the core and the Co-60 irradiation source dose rate calculations.

Based on the above discussion, as well as on the results of the NRC staff's confirmatory calculation of LOCA radiological consequences (which are shown in SER Table 5-7, Table 5-8, Table 5-9, and Table 5-10 above, and are discussed below), the NRC staff finds that the methodology, assumptions, and results of UML's LOCA radiological consequence analysis are acceptable. Also, the NRC staff finds that even in the unlikely event of a complete LOCA at the UMLRR, doses to occupational workers and members of the public could reasonably be maintained below 10 CFR Part 20 limits.

NRC Staff Confirmatory Calculation of LOCA Radiological Consequences

The NRC staff performed calculations to determine if it could confirm the results of UML's LOCA radiological consequence analysis. For its calculations, the NRC staff used methodologies and assumptions generally similar to those used by UML, including using the Klein-Nishina formula for differential cross section of Compton scattering. However, for its indirect dose rate calculations, instead of numerically integrating expressions for the fluence rate of photons from the dome over the irradiated surface of the reactor dome in order to consider the different possible photon paths and scattering angles between the sources and a receptor location, the NRC staff performed simplified calculations that only considered one discrete path (and fixed scattering angle) between each source and receptor location. For its indirect dose rate calculations, the NRC staff assumed that a wide beam of photons, roughly equivalent in area to the pool surface, and with a fluence rate based on the distance from the point source (core or the Co-60 irradiation source), is incident on the dome. Additionally, based on the methodology in Stephenson (Ref. 68, see pages 193 through 199), the NRC staff assumed that the reactor building dome is a thick slab of iron, in order to maximize photon backscattering from the dome. For core dose rate calculations for direct dose rate and indirect dose rate at the truck door exterior, instead of considering a range of energies for photons emitted from the reactor core, the NRC staff assumed that only 1 MeV photons (one photon per beta decay in the core) are emitted from the core; this is a conservative assumption, since most of the decay photons from the core will have energies less than 1 MeV, and would generally result in lower doses. For core dose rate calculations for indirect dose rate at the edge of the pool, however, the NRC staff assumed that only 0.5 MeV photons (one photon per beta decay in the core) are emitted from the core, because it determined that, for this case, the higher scattering cross section

(probability) of the lower energy photons in the dome compensates for the lower energy of the scattered photons, and the 0.5 MeV photons result in slightly higher dose rates. For the Co-60 irradiation source, the NRC staff assumed, similar to UML, that two 1.25 MeV photons are emitted per decay. The NRC staff's calculations did not consider air attenuation, but considered attenuation in the dome (in accordance with the Stephenson methodology) and attenuation (and buildup) in the truck door. The NRC staff did not consider annihilation photons in its calculations.

The results of the NRC staff's confirmatory calculations are shown in SER Table 5-7, Table 5-8, Table 5-9, and Table 5-10 above, along with the results of UML's analysis. The NRC staff notes that the differences in the direct dose rates are generally small (NRC staff calculations are larger by a factor of approximately two or less) for dose rates following a short duration of reactor operation and/or short decay times. However, there are significant differences in direct dose rates following infinite reactor operation and longer decay (NRC staff calculations are larger by up to a factor of approximately six), which are likely primarily due to the NRC staff's simplified core photon emission assumptions. For the indirect dose rates, the NRC staff's calculated dose rates are significantly smaller than UML's values (the differences range from a factor of approximately five to 27). The significant differences in indirect dose rates are also likely due to the NRC staff calculations' use of the various different assumptions and methodologies discussed in the paragraph above. The NRC staff notes that, although the differences in some of the NRC staff's and UML's calculations are significant, there is a high degree of uncertainty inherent in modeling the dose rates, particularly dose rates from scattered photons. The NRC staff finds that, given that its calculated dose rates for locations at which individuals could potentially be present following a LOCA are less than those calculated by UML (as discussed above, individuals would not be on the reactor bridge following a LOCA), the confirmatory calculations support the NRC staff's above finding that LOCA doses could reasonably be maintained below 10 CFR Part 20 limits.

Conclusion

The NRC staff reviewed UML's LOCA fuel integrity and radiological consequence analyses. The NRC staff finds that UML discussed possible methods by which a LOCA could occur and how the UMLRR is designed and operated in a manner that would make any LOCA that could cause fuel or the Co-60 irradiation source to become uncovered unlikely. The NRC staff finds that UML provided discussion and analyses that, as supported by the NRC staff's confirmatory analysis, demonstrate that no credible LOCA could cause the fuel to reach temperatures at which fuel failure could occur. The NRC staff finds that UML has also analyzed the possible radiological consequences that could result from the loss of core and the Co-60 irradiation source shielding due to a LOCA (if a LOCA were able to cause pool drainage to the level of the core or the Co-60 irradiation source), and the NRC staff finds that LOCA doses could reasonably be maintained below 10 CFR Part 20 limits. The NRC staff finds that the methodologies and assumptions used by UML for its LOCA analyses are generally reasonable and conservative. As discussed above, the NRC staff also performed independent confirmatory calculations of UML's fuel integrity and radiological consequence analyses and the results of the NRC staff's calculations generally showed a larger margin of safety or lower dose rates than UML's results, except as discussed above. Based on the above, the NRC staff concludes that UML's LOCA analyses are acceptable.

5.4 Loss of Coolant Flow

Potential loss of coolant flow events at the UMLRR are discussed primarily in SAR Sections 13.1.3 and 13.2.3, as supplemented by UML's response to RAI-13.8 (Ref. 43). Additional information relating to UML's methodology, assumptions, and inputs for its analyses is also discussed in SAR Section 4.5.6.

UML stated that during normal forced flow operation, the primary pump provides a flow rate of approximately 1,700 gpm down through the core. If, for any reason, forced flow is lost, then the downward flow ceases (after the forced flow from the pump coasts down), and upward flow is initiated by the difference in density of the water in the heated fuel channels and the water in the surrounding pool. As discussed in SAR Section 5.2 and UML's response to RAI-14.3.33 (Ref. 71), a riser gate on the outlet channel (or riser plenum) opens to facilitate natural convection (the downcomer gate would also open if not already open, i.e., if the reactor had been operating in downcomer mode). Eventually, equilibrium natural convection flow is reached when the buoyancy forces balance the friction forces in the fuel channels.

UML stated that a loss of flow event could occur from a pump failure, a loss of electrical power, or operator error, all of which are credible events. Therefore, UML evaluated sudden pump-off scenarios, which could result from these events.

UML Loss of Coolant Flow Analyses

As discussed in SER Section 2.5, for its loss of flow analyses, UML used PARET, similar to its reactivity transient analyses, since PARET can be used to evaluate both reactivity- and flow-induced transients. UML used its LCC peaking factors as inputs to its PARET calculations. For its PARET loss of flow analyses, UML used a modeled pump-off flow rate profile (coast-down curve), which is based on measured data as discussed in SAR Section 4.5.6, and which assumes that pump flow rate decreases exponentially after the pump shuts off, reaching exactly zero after 10 seconds. The analyses considered the flow direction reversal transient period and the natural convection flow that becomes established after the pump shuts off.

UML analyzed three cases for its loss of flow analyses:

- case 1, in which the reactor is initially operating at 1.25 MWt with a pool inlet temperature of 43.3 °C (109.9 °F) (both beyond LSSS conditions), and an initial primary flow rate of 1,700 gpm (nominal value);
- case 2, which is the same as case 1 except that the initial primary flow rate is 1,370 gpm (beyond LSSS conditions); and
- case 3, in which the reactor is initially at nominal conditions, i.e., operating at 1 MWt with a pool inlet temperature of 30.0 °C (86.0 °F) and a primary flow rate of 1,700 gpm.

In each case, the reactor is allowed to scram when a low flow scram setpoint of 1,370 gpm (more conservative than the LSSS scram setpoint of 1,400 gpm) is reached. For case 2, the reactor scrams almost immediately (the instrument delay time is accounted for) after the pump shuts off, because the flow rate was exactly equal to the scram setpoint before the pump shut off. For cases 1 and 3, it takes about 1.5 seconds for the flow rate to coast down to the scram setpoint and for the scram to occur.

SER Figure 5-16, Figure 5-17, and Figure 5-18 (which are reproduced from SAR Section 13.2.3) illustrate the results of UML's loss of flow analyses for cases 1, 2, and 3, respectively.

The figures show that the maximum fuel cladding temperature reached is about 75 °C (167 °F) for case 2 immediately prior to the scram, which is well below the SL, ONB, and temperatures at which any fuel failure could occur. UML noted that, for all three cases, the fuel temperatures quickly drop when the reactor scrams (for cases 1 and 3, there is a notable small increase in temperature before the reactor scrams from the decrease in flow rate prior to the scram). The temperature then rises back up as the pump flow rate continues to decrease, but falls again as natural convection flow is established.

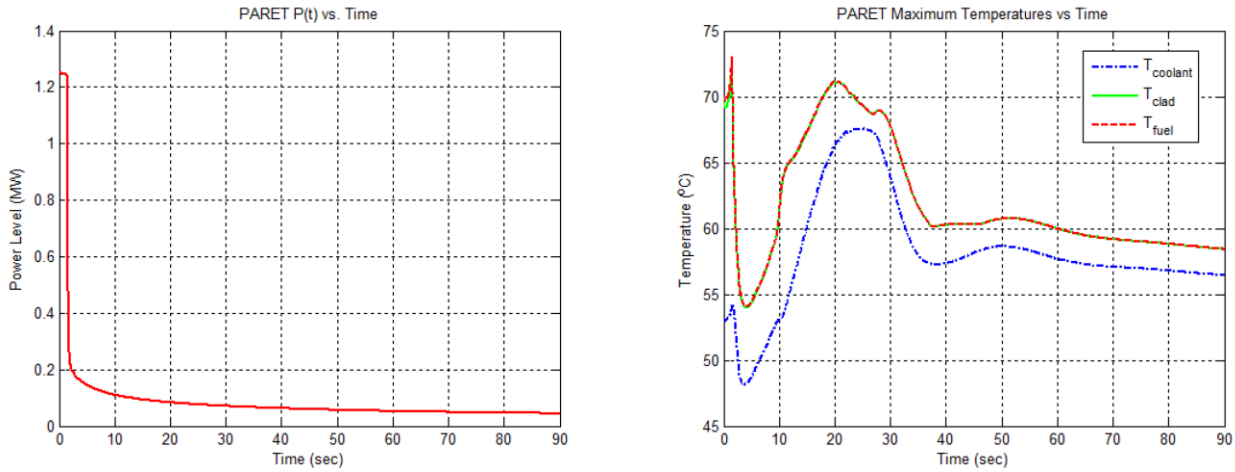

Figure 5-16 Loss of Flow Case 1 (Reactor Initially Operating at 1.25 MWt with Inlet Temperature 43.3 °C and Flow Rate 1,700 gpm)

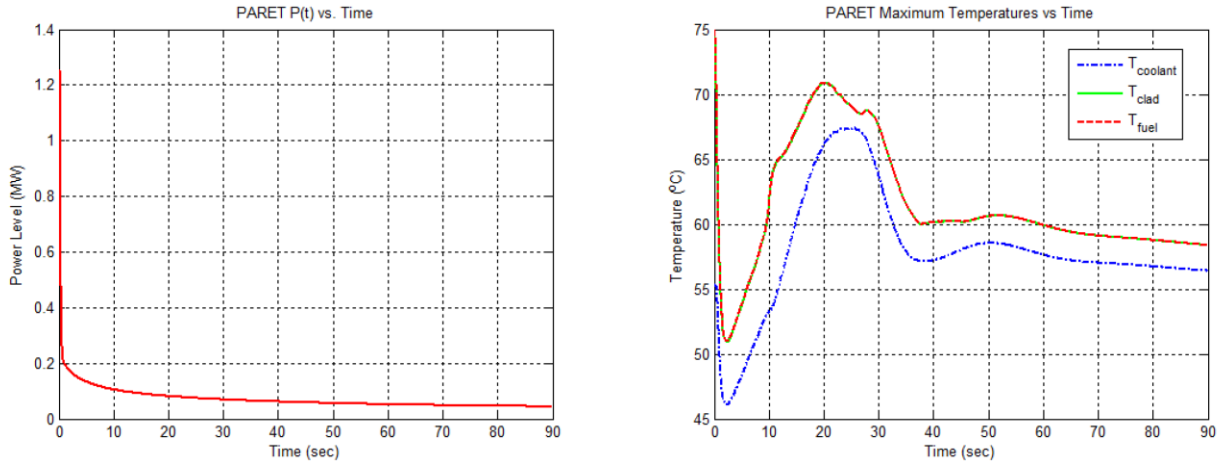

Figure 5-17 Loss of Flow Case 2 (Reactor Initially Operating at 1.25 MWt with Inlet Temperature 43.3 °C and Flow Rate 1,370 gpm)

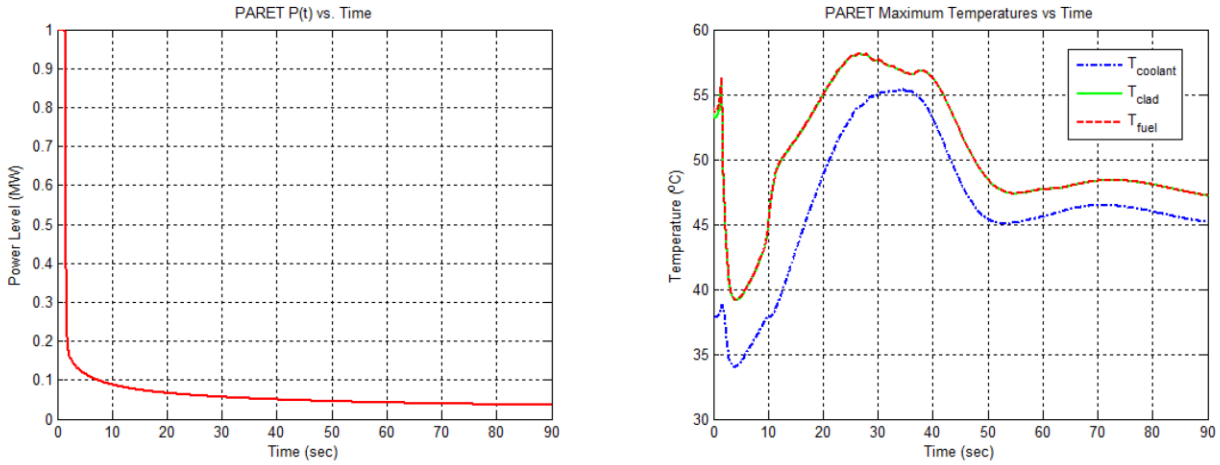

Figure 5-18 Loss of Flow Case 3 (Reactor Initially Operating at 1 MWt with Inlet Temperature 30.0 °C and Flow Rate 1,700 gpm)

The NRC staff reviewed UML's loss of flow analyses. The NRC staff finds that UML used a methodology and assumptions that are reasonable, conservative, and consistent with accepted practice and UMLRR design and operation. The NRC staff finds that UML's analyses show that loss of flow from a pump-off event at the UMLRR, from any initiating cause (e.g., pump failure, loss of electrical power, or operator error), would not cause fuel to reach temperatures that could result in fuel failure. The NRC staff notes that UML's analyses assume that natural convection flow is eventually established through the fuel channels. However, based on experience with similar facilities, the NRC staff finds that even if normal natural convection flow were somehow inhibited (for example, in the unlikely event that the riser gate failed to open), the fuel cladding would still likely remain well below the SL temperature once the reactor is shutdown given the large amount of pool water providing a heat sink and the decay heat decreasing with time. The NRC staff also finds that, even if the fuel temperature reached the saturation temperature of the coolant, localized boiling would help keep the fuel temperature within a few degrees of the saturation temperature and well below the fuel SL temperature, as long as the core remained covered with water. Based on the above, as well as the results of the NRC staff's confirmatory calculations of a loss of flow, which are discussed below, the NRC staff finds that UML's analyses of a loss of coolant flow from a pump-off scenario are acceptable.

NRC Staff Loss of Flow Confirmatory Analysis

The NRC staff performed an analysis to determine if it could confirm the results of UML's calculations of a pump-off loss of coolant flow scenario at the UMLRR. For its analysis, the NRC staff used the non-LOCA TRACE model discussed in SER Section 2.6, with LCC peaking factors and explicit hot channel factors. Similar to UML's analyses, the NRC staff's analysis assumed that the pump shuts off, the reactor scrams when the low flow setpoint is reached, pump flow coasts down, flow direction in the core reverses, and equilibrium natural convection flow is established. The NRC staff assumed that initially, the reactor power is 1.0 MWt (equal to the license condition 2.C.1 and TS 3.1.2 power level limits), the primary flow rate is 1,400 gpm (equal to the LSSS), and the pool inlet temperature is 43 °C (109 °F) (beyond the LSSS). The NRC staff also assumed that the flow rate scram is set to the LSSS value of 1,400 gpm; therefore, the reactor scrams almost instantly after the pump shuts off (following the instrument delay).

The NRC staff calculated a maximum fuel cladding temperature of approximately 57 °C (135 °F), reached approximately 20 seconds after the pump shuts off. This is similar to UML's result for case 3, illustrated in SER Figure 5-18 above, which similarly assumed an initial reactor power of 1.0 MWt. Although the NRC staff's calculation did not consider a worst-case (i.e., LSSS) initial power level, the NRC staff finds that its calculation result illustrates the reasonableness of UML's results, and supports UML's determination that pump-off loss of flow scenarios would not cause the fuel to reach temperatures that could cause damage. The NRC staff also finds that the differences in its result and UML's results are reasonable given the differences in methodologies and assumptions used.

Partial Loss of Flow Resulting from Potential Coolant Channel Blockage

In its response to RAI-13.8, UML discussed the possibility of a partial loss of coolant flow from a different initiator, specifically, the blockage of a coolant channel by a foreign object or debris. UML stated that a combination of administrative controls and UMLRR design features minimize the potential for foreign objects or debris to cause a coolant channel blockage. UML has procedures to check the reactor bridge and core area for loose or unnecessary items and to remove items that do not need to be present in these areas. The concrete pool wall is designed such that it extends above the level of the pool, helping to prevent any items from accidentally falling or being kicked into the pool. If UML reactor staff notice any object or debris falling into the pool while working above the pool on the reactor bridge, the reactor staff would visually track the object or debris to its final resting place in the pool; operational experience has shown that items falling into the pool generally drop to the pool floor where they may be recovered. Should reactor staff on the bridge notice objects or debris fall to the top of the core, the staff could scram the reactor using a remote scram button on the bridge. Even if an object or debris fell to the top of the core or was drawn into the primary flow pathway unnoticed, the design of the fuel element end box, which includes the lifting handle (or bale) for the element, would help inhibit the object or debris from reaching the fuel plates where it cause potentially cause a blockage.

The NRC staff reviewed the above information. The NRC staff finds that administrative controls and design features at the UMLRR would help minimize the possibility of a partial loss of flow resulting from foreign objects or debris blocking a coolant channel. The NRC staff also finds that, even if an object or debris were drawn into a coolant channel, it could result in power oscillations that would help give reactor operators an indication of a possible problem and allow the issue to be addressed before damage could occur. The NRC staff further finds that, in the unlikely event that any fuel damage occurred, it would likely be limited to a small portion of the fuel (e.g., part of the fuel plates comprising one coolant channel), and would likely be bounded by the MHA, which assumes cladding is completely lost from one side of the highest power fuel plate in the core, following extended full power reactor operation with maximum fission product inventory available for release, as discussed in SER Section 5.1. Therefore, the NRC staff finds that a partial loss of flow resulting from a coolant channel blockage that could cause fuel failure at the UMLRR is highly unlikely and, even if it did occur, it would likely be bounded by the MHA.

Conclusion

The NRC staff reviewed UML's discussion and analyses of loss of flow at the UMLRR. The NRC staff finds that UML discussed possible methods by which coolant flow through some (due to a blockage) or all (due to a pump-off event) could be interrupted while the UMLRR is operating. The NRC staff finds that UML demonstrated that a pump-off loss of flow event could

not cause the fuel to reach temperatures that would cause fuel failure. The NRC staff performed a confirmatory calculation that supports the reasonableness of the results of UML's analysis. The NRC staff also finds that, because of the administrative controls and design features at the UMLRR, a partial loss of flow resulting from a coolant channel blockage that could cause fuel failure would be a highly unlikely event and, in any case, likely bounded by the MHA. Therefore, the NRC staff concludes that UML's loss of flow discussion and analyses are acceptable.

5.5 Mishandling or Malfunction of Fuel

SAR Sections 13.1.1, 13.1.5, and 13.2.5 discuss the possibility of a mishandling or malfunction of fuel at the UMLRR.

UML stated that, when not in the reactor core, fuel elements are stored in the fuel storage racks located underwater along the reactor pool walls. As required by renewed TS 5.4(1), which is discussed and found acceptable in SER Section 2.7, the fuel elements in the storage racks are stored in a criticality-safe configuration. As also discussed in SER Section 2.7, UML performed analyses demonstrating that the k-effective of fuel in the UMLRR storage racks could not exceed 0.90, even when configurations that are of an even more optimal-geometry (i.e., racks and individual fuel elements spaced more closely together, such as could potentially occur during an accidental re-arrangement of the stored fuel) than actual configurations are considered.

UML stated that fuel elements are only handled one at a time so that they are maintained in a criticality-safe configuration (the NRC staff notes that this would also help prevent fuel handling accidents that could damage multiple elements). UML stated that it uses a specially designed fuel handling tool that latches onto the element to prevent the inadvertent drop of an element during movement, and that an element has never been damaged during movement using this tool. If any damage occurred or were suspected to occur, an evaluation (likely including inspection) of the element would be performed before it would be utilized in the reactor. UML stated that any fuel handling accident that could cause fuel clad damage significant enough to lead to a fission product release would be highly unlikely and, even if it did occur, the release would be small (and much less than that associated with the UMLRR MHA) because the damage would likely be limited to a deep scratch or puncture of a fuel plate.

UML stated that there have been no instances of fuel malfunction at the UMLRR (with either the previous highly enriched uranium or the current low-enriched uranium fuel). UML stated that its fuel elements must meet strict quality control standards and that UMLRR operation does not impose significant thermal stresses on the fuel; the temperature change of the fuel from cold shutdown to a hot operating condition is typically no more than 18 °C (32 °F). As discussed in SER Sections 5.2, 5.3, 5.4, even the worst-case credible reactivity transient, loss of coolant, or loss of flow scenarios could not cause the fuel to reach temperatures (i.e., the SL) at which damage could occur. UML also stated that the water chemistry of the pool is maintained to minimize any possibility of fuel corrosion. As required by renewed TS 4.1(8), which is discussed and found acceptable in SER Section 2.2.1, the fuel is periodically inspected for any deterioration, corrosion, or other physical changes that could lead to loss of cladding integrity. UML stated that the UMLRR radiation monitoring systems would also provide an early indication of any fuel malfunction causing a fission product release, allowing the damaged element(s) to be identified.

The NRC staff reviewed the above information regarding a mishandling or malfunction of fuel at the UMLRR. The NRC staff finds that UML discussed events that could potentially damage fuel while it is in the core, in storage, or being handled. The NRC staff finds that damage to a fuel element from mishandling or malfunction of the fuel would be unlikely, given UML's methods for handling fuel and the UMLRR operating conditions and historical experience. The NRC staff finds that damage to fuel elements from other potential events, such as objects being dropped onto the fuel, would also be unlikely given the design and location of the fuel storage racks and core box. The core box is beneath the core suspension structure, which would help protect the fuel in the core box from falling objects. If any fuel damage did occur, it would likely be minor, and any associated fission product release to the reactor building would be small and less severe than for the MHA. As discussed in SER Section 5.1, the UMLRR MHA assumes that the cladding is completely stripped from one side of one fuel plate; the NRC staff finds that this is consistent with bounding fuel damage scenarios assumed for similar research reactors. The NRC staff also finds that an inadvertent criticality of stored fuel is unlikely, based on the fuel storage configurations being designed for criticality safety. Therefore, the NRC staff concludes that UML's analysis of mishandling or malfunction of fuel events is acceptable.

5.6 Experiment Malfunction

SAR Sections 13.1.6 and 13.2.6, as supplemented by UML's response to RAI-13.3 (Ref. 23), discuss experiment malfunctions at the UMLRR. SAR Chapter 10 also provides additional detail on UMLRR experimental facilities, and SAR Section 10.3, as supplemented by UML's responses to RAIs 14.6.16, 14.6.17, 14.6.18, and 14.6.19 (Ref. 71), specifically discusses UMLRR experiment review.

UML stated that neutron irradiation experiments conducted at the UMLRR are divided into 2 general types: ex-core experiments (e.g., experiments utilizing the beam tubes, thermal column, or fast neutron irradiator, which are decoupled from the reactor such that they cannot affect reactor neutronics or thermal-hydraulics) and in-core experiments (e.g., experiments utilizing the center flux trap, radiation baskets, or pneumatic tube system, which are in or near the core and, therefore, could affect the reactor performance). Experiments at the UMLRR could also utilize other gamma and/or neutron irradiation facilities adjacent to the bulk side of the pool (e.g., the gamma cave, hot cell, and medical embedment). Gamma irradiations in the UMLRR pool could utilize either the Co-60 irradiation source or irradiated fuel elements in the fuel storage racks.

UML stated that improperly controlled experiments could potentially result in damage to the reactor, unnecessary radiation exposure, and/or inadvertent release of radioactive material. Therefore, UML has TS and procedural requirements addressing the review and approval of experiments at the UMLRR. All new experiments (or classes of experiments), and substantive changes to experiments, must be reviewed by the Reactor Safety Subcommittee (RSSC). The RSSC reviews assess such considerations as criticality, reactivity, heat generation, off-gassing, possible chemical reactions, and shielding. Renewed TS 6.5, which is discussed and found acceptable in SER Section 6.6.5, imposes requirements for the RSSC review and Reactor Supervisor approval that are conducted for all new experiments (or classes of experiments) and substantive changes to previously approved experiments (i.e., changes that cause an experiment to be outside the scope of an existing experiment review and approval). All proposed experiments, including routine experiments (i.e., experiments that have already been reviewed by the RSSC and approved by the Reactor Supervisor), are initially evaluated by the experimenter and another staff member approved by the RSSC. This initial evaluation is

reviewed by the Reactor Supervisor and radiation safety officer to determine that the experiment complies with the UMLRR license and TSs and 10 CFR Part 20.

Renewed TSs 6.4 and 6.5, which are discussed and found acceptable in SER Sections 6.6.4 and 6.6.5, respectively, require the use of written procedures for the conduct of irradiations and experiments.

SER Section 4.1.5 discusses radiation exposure control at the UMLRR, including controls to ensure that personnel do not receive excessive radiation doses from the use of UMLRR irradiation facilities.

Renewed TS 3.7, which is discussed and found acceptable in SER Section 6.3.7, includes several restrictions and requirements to minimize hazards of experiments installed in the reactor and associated experimental facilities. Specifically, TS 3.7.1 limits the individual and total reactivity worth of movable and secured in-core experiments (including pneumatic rabbits, which are considered to be movable experiments), to help ensure that reactivity changes or transients caused by the normal operation (for example, movement of a moveable experiment) or failure of an experiment would be less than the reactivity changes analyzed as discussed in SER Section 5.2, and could not damage the reactor. TSs 3.7.2(1), 3.7.2(2), and 3.7.2(3) impose requirements to minimize interferences between the reactor and experiments, help ensure that any corrosive materials used in experiments cannot damage the reactor, and help ensure that experiment failures cannot cause doses in excess of 10 CFR Part 20 limits. TS 3.7.2(4) limits the quantity of explosive materials in any experiment and requires that any experiment containing explosive materials be in a properly pressure-tested container. TS 3.7.2(5) limits the radioactive material content of any fueled experiment to help ensure that the consequences of the failure of any failed experiment would be bounded by the MHA discussed in SER Section 5.1.

The NRC staff reviewed the above information. The NRC staff finds that UML has proper controls established for experiments to minimize the likelihood and consequences of any experiment malfunction. The NRC staff also finds that the performance of experiments within the requirements of the UMLRR TSs provides reasonable assurance that experimental malfunctions would not damage the reactor and that the potential consequences of experiment malfunctions would be bounded by the analyses discussed in SER Section 5.2 for malfunctions involving reactivity transients and the dose limits in 10 CFR Part 20 for malfunctions that could result in radiation exposure to UMLRR staff or the public. Therefore, the NRC staff concludes that UML's discussion and analysis of experiment malfunctions is acceptable.

5.7 Loss of Normal Electrical Power

A loss of normal electrical power at the UMLRR is discussed in SAR Sections 6.2.3, 8.2, 13.1.7, and 13.2.7, as supplemented by UML's response to RAI-6.1 (Ref. 23) and its letter dated September 30, 2020 (Ref. 98).

UML stated that because of its passive safety features, the UMLRR design is fail-safe under a condition where electrical power is lost. When normal offsite power is lost, the reactor will scram and normal reactor building ventilation will be isolated. As discussed in SER Section 2.2.2, the four reactor control blades are attached to their drives by electromagnets, and when power is lost, the magnets de-energize and the blades insert due to gravity. The UMLRR is designed for the removal of decay heat by natural convection. As demonstrated by UML's loss of coolant flow analyses, which are discussed in SER Section 5.4, no forced flow cooling is

required after a reactor scram to prevent the fuel from reaching temperatures (i.e., the SL) at which fuel failure could occur.

UML stated that the UMLRR is equipped with an emergency generator that can provide backup power to certain systems, including control room instrumentation, area radiation monitors, reactor building radiation alarms, reactor building public address system, emergency exhaust fan, ventilation valve control, and the compressor that supplies air for the airlock doors. Additionally, an uninterruptible power supply will continue powering the control room instrumentation during the switch to emergency power and can also continue to supply power to the instrumentation for approximately 30 minutes if the emergency generator fails to start.

However, UML stated that the emergency generator and uninterruptible power supply are not required for safety purposes. No power is required for the reactor to scram or to cool down by natural convection. If control room instrumentation is not functioning, the reactor operator can visually verify that the control blades have inserted into the core using battery operated lights, and this is sufficient verification that the reactor has shut down. Emergency lights with backup batteries are located throughout the reactor building and will turn on after loss of normal power, even if emergency power is also lost or unavailable, allowing personnel to exit the building safely. Compressed air line pressure and backup compressed air tanks within the two personnel airlocks will provide sufficient air to allow the building to be evacuated while maintaining building confinement.

UML stated that normal building ventilation system isolation is achieved by de-energizing solenoids in the various building air supplies. Therefore, power is not needed to isolate normal ventilation. Although normal reactor ventilation shuts down on a loss of normal electrical power, as discussed in SAR Section 6.2.3 and UML's response to RAI-6.1, the emergency exhaust system, which is powered by the emergency generator, would normally begin to operate after the loss of power and isolation of normal ventilation. As discussed in SER Section 5.1, although a scenario in which the emergency exhaust system operates as normal (i.e., MHA Scenario D) represents the most probable progression of an event involving a release of radioactive material into the reactor building air, UML also analyzed scenarios (MHA Scenarios B and C) in which the emergency exhaust system fails to operate, and demonstrated that doses to members of the public would still not exceed 10 CFR Part 20 limits.

The NRC staff notes that, following a scenario where offsite power is lost (but the emergency exhaust system is still able to operate on backup power), the emergency exhaust system would not normally operate exactly as assumed in MHA Scenario D. This is because the emergency exhaust fan and the intake fan, which are on the emergency power distribution switchboard as illustrated in SAR Figures 8-2 and 8-3, would shut off for a brief period of time (while the emergency generator automatically starts) following a loss of offsite power, until they normally restart once the emergency generator is online and providing electrical power (see SAR Section 6.2.3 and UML's letter dated September 30, 2020 (Ref. 98)). (As discussed in SAR Section 6.2.3, the intake fan normally operates along with the emergency exhaust system to give additional dilution and momentum to potentially contaminated air exiting the stack, as assumed in MHA Scenario D.) However, given the TS 3.5(2) negative reactor building pressure requirement, it is unlikely that any significant leakage from the reactor building would occur during the brief period after the building is isolated and before the emergency generator starts. Additionally, a significant airborne radioactive release in the reactor building (i.e., a release similar to that assumed for the UMLRR MHA), coincident with a loss of offsite power, is an extremely unlikely event, and even if it did occur and the emergency exhaust system and/or intake fan completely failed to function (due to failure of the emergency generator to start or

another cause), it would not cause 10 CFR Part 20 dose limits to be exceeded, as demonstrated by UML's analysis of its MHA Scenarios B and C, and the NRC staff's MHA public dose calculation, which bounds any realistic UMLRR release scenario as discussed in SER Section 5.1.

UML stated that information on radioactive effluents is not necessary once the reactor is shut down and the building ventilation is isolated following a loss of power. The NRC staff notes that the emergency exhaust system would normally cause some air to continue to be released from the reactor building via the stack following a loss of power. However, given that the reactor is shut down, and the air release rate from the building is slow compared to the air release rate during normal reactor and ventilation system operation, the NRC staff expects that any radioactive effluents released through the stack would not typically be significant following normal reactor operation.

As discussed in SAR Section 11.1.4.2, UML has a range of portable radiation survey instruments available for use at the UMLRR. The NRC staff notes that these instruments could be used to help provide an indication of radiation levels in the event that electrical power for the fixed area monitors was unavailable.

As discussed in SER Sections 5.3 and 6.3.8, UML may use the reactor building overhead crane to help mitigate a LOCA (by closing the pool divider gate and/or beam tube shutter), and the overhead crane requires offsite electrical power (and is also not powered by the emergency generator). However, a severe LOCA coincident with a loss of offsite power is an extremely unlikely event. Additionally, even if an unmitigated, worst-case LOCA occurred through an 8-inch beam tube, the SL would not be exceeded, as demonstrated by the NRC staff's LOCA confirmatory calculation discussed in SER Section 5.3.

The NRC staff reviewed the above information. The NRC staff finds that the UMLRR is designed to safely and automatically shut down on a loss of power and that no power is required to ensure that decay heat can be adequately removed from the reactor. The NRC staff also finds that the UMLRR has backup power capabilities that help provide redundant power for certain reactor systems, but that backup power capabilities are not necessary to ensure reactor safety once the reactor is shutdown. Therefore, the NRC staff concludes that UML's discussion of a loss of normal electrical power is acceptable.

5.8 External Events

Possible accidents caused by external events are primarily discussed in SAR Sections 13.1.8 and 13.2.8. Additionally, SAR Chapter 2 discusses the site characteristics of the UMLRR, including the potential for external events such as weather and seismic events, and the potential for events at nearby offsite facilities to affect the UMLRR, and SAR Chapter 3 discusses the design of UMLRR structures, systems, and components (including the reactor building and pool) to withstand weather, flooding, and seismic events.

UML stated that damage to the reactor core from external events, including lightning, floods, and meteorological disturbances, is not credible since the core is contained within a deep tank of water made of thick reinforced concrete, which is in turn located within the thick concrete and steel reactor building. The building and pool effectively isolate the reactor from flooding and meteorological disturbances, including tornadoes and hurricanes; moreover, the UMLRR is in a low flood risk zone, and tornadoes and hurricanes are infrequent in the area, as discussed in SAR Sections 2.3.1, 2.4, and 3.2. SAR Section 3.2.2 states that the shape of the reactor

building dome causes rain to be shed at once and snow to slide off as soon as a small amount accumulates, limiting snow and ice loading on the dome.

The reactor building and deep pool also provide effective isolation against seismic events. SAR Sections 2.5.5 and 3.5.1 state that the reactor building foundation, the pool, the reinforced concrete parts of the reactor building, and the steel shell of the reactor building have been designed to withstand a gravitational acceleration corresponding to Modified Mercalli Scale VII, as recommended by the Uniform Building Code for structures in the Lowell area, and that this exceeds the expected felt intensity of earthquakes in the area. Renewed TS 3.2.3, which is discussed and found acceptable in SER Section 2.5.3, requires that the reactor protection system include a scram for a seismic event. The control blades are positioned in the core by shrouds, which help ensure that the blades would still be able to insert into the core by gravity during a seismic event.

As discussed in SER Section 5.3, a LOCA at the UMLRR, including a LOCA caused by an earthquake, is highly unlikely because the UMLRR pool is designed with thick concrete walls and an aluminum liner to resist the most severe earthquake that might reasonably be expected near the facility. Other features of the UMLRR, including the location of reactor pool penetrations and the anti-siphon values in the primary piping, would also help prevent or mitigate a potential LOCA caused by an earthquake.

Even if a LOCA occurred due to earthquake-related damage (e.g., damage to a beam tube), the analyses discussed in SER Section 5.3 demonstrate that no credible LOCA at the UMLRR could cause the fuel to reach temperatures (i.e., the SL) at which fuel failure could occur. The NRC staff notes that an earthquake could cause water to splash from the pool, but given the depth of the pool, this could not cause the pool to drop to a level that would uncover fuel.

UML stated that the reactor building and pool also effectively isolate the reactor from potential explosions or toxic releases resulting from an offsite event. For an explosion to affect the reactor, it would have to penetrate the building wall, propagate across the space between the building wall and the pool wall, penetrate the thick pool wall, and then still have sufficient effect to damage the core box containing the fuel. For an external toxic release, UML would be able to shut down the ventilation system such that outside air would not be drawn into the building.

The NRC staff notes that because UML is re-designating its reactor building from a containment to a confinement (see SER Section 1.8), UML is no longer required to perform leak tests to verify the leak-tightness of the building; however, the NRC staff finds that because air intake into the building can still be secured, and given the robust design of the building, it would still provide appropriate protection against an external toxic release.

The NRC staff reviewed the above information. The NRC staff finds that the UMLRR is designed to adequately withstand external events, including weather and seismic events, and offsite events at nearby facilities, and that any damage that could occur due to an external event would reasonably be bounded by the other accidents discussed in SAR Chapter 13, as supplemented. The NRC staff finds that the design of the UMLRR would also protect the Co-60 irradiation source, because it is also located in the reactor pool and protected by the reactor building and pool walls similar to the reactor core. Therefore, the NRC staff concludes that UML's discussion of external events is acceptable.

5.9 Mishandling or Malfunction of Equipment

Mishandling or malfunction of equipment events are discussed in SAR Sections 13.1.9 and 13.2.9. UML identified operator errors and the malfunction of safety-related instruments or equipment as falling into this category of event.

UML stated that credible operator errors at the UMLRR include: passive inattentiveness leading to a power drift when operating in manual control mode; inadvertent continuous withdrawal of a control blade or the regulating rod while the reactor is supercritical; and inadvertent shutoff of the primary coolant pump while the reactor is in forced convection mode. UML stated that a power drift would eventually cause a reactor scram. As discussed in SER Section 5.2, UML analyzed reactivity insertion events, including ramp reactivity insertions from a continuous inadvertent rod withdrawal, and showed that the consequences of these events would be acceptable. As discussed in SER Section 5.4, UML analyzed a loss of flow during full power forced convection operation, and showed that the consequences of this event would also be acceptable. UML stated that the reactor protection system helps ensure that an operational error by the operator, or even the loss of the operator, would not result in a situation creating an accident beyond those analyzed in SAR Chapter 13, as supplemented.

UML stated that the UMLRR is designed such that safety-related instrument or equipment failures generally result in a reactor shutdown.

The NRC staff reviewed the above information. The NRC staff finds that the reactor protection system helps ensure that any credible operator error at the UMLRR would result in the reactor shutting down and would not cause any accident that could exceed those analyzed in SAR Chapter 13, as supplemented. The NRC staff also finds that the reactor protection system includes sufficient redundancy, and is designed in a fail-safe manner to ensure that the reactor is shut down if safety-related instruments or equipment are lost (see SER Chapter 3), which also helps prevent any accident that could exceed those analyzed in SAR Chapter 13, as supplemented. Therefore, the NRC staff concludes that UML's discussion of mishandling or malfunction of equipment events is acceptable.

5.10 Accident Analysis Conclusions

The NRC staff has reviewed the accident analyses presented in the SAR, as supplemented, and finds that UML has considered a sufficient range of accident categories and analyzed limiting scenarios for each category to bound all credible accidents for the UMLRR. Based on its review, the NRC staff concludes the following:

- UML considered the expected consequences of a sufficiently broad spectrum of postulated credible accidents and an MHA, emphasizing those that could lead to a loss of integrity of fuel element cladding and a release of fission products.

- UML analyzed the most significant credible accidents and the MHA and determined that, under conservative assumptions, the most significant credible accidents and the realistic doses from the MHA will not result in occupational radiation exposure of the UMLRR staff or exposure to a member of the public in excess of the applicable 10 CFR Part 20 limits.

- UML generally employed appropriate methods in performing the accident and consequence analyses.

- UML's analyses demonstrated that a credible LOCA would not result in unacceptable fuel element temperatures.

- External events that would lead to fuel failure are unlikely.

- The accident analyses confirm the acceptability of the UMLRR's licensed power of 1.0 MWt, including with respect to the UMLRR's response to anticipated transients and accidents.

- The accident analyses confirm the general 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 the UMLRR. The NRC staff finds that 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 members of the public and the occupational workers at the UMLRR meet 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 UMLRR design features and administrative restrictions required by 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 cause significant radiological risk and that the continued operation of the UMLRR would not pose an undue risk to the facility staff, members of the public, or the environment.

6. TECHNICAL SPECIFICATIONS

In this section of the safety evaluation report (SER), the U.S. Nuclear Regulatory Commission (NRC, the Commission) staff provides its evaluation of the University of Massachusetts Lowell's (UML's) proposed technical specifications (TSs) for the UML Research Reactor (UMLRR). The proposed TSs define specific features, characteristics, organizational and reporting requirements, and conditions required for the safe operation of the UMLRR. The TSs would be 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 NUREG-1537 (Ref. 14), Chapter 14, and American National Standards Institute/American Nuclear Society (ANSI/ANS)-15.1-2007 (Ref. 18). The NRC staff specifically evaluated the content of the TSs to determine whether they meet the requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36. The NRC staff also relied on other chapters, as applicable, of NUREG-1537 to perform its review.

6.1 UMLRR Proposed Renewed TSs Section 1, "Introduction"

6.1.1 Scope and Application

Renewed TS 1.1, "Scope," would state:

> This document constitutes the technical specifications for The University of Massachusetts Lowell Research Reactor under facility license No. R-125. The technical specifications include definitions, safety limits, limiting safety system settings, limiting conditions for operation, surveillance requirements, design features, and administrative controls in accordance with 10CFR 50.36. Also included are the bases for the technical specifications. The bases, which provide the technical support for the individual technical specifications, are for information purposes only. They are not part of the technical specifications, and they do not constitute limitations or requirements to which the licensee must adhere.

Renewed TS 1.2, "Application," would state:

> 1.2.1 Purpose
>
> The technical specifications represent the agreement between the licensee and the U.S. Nuclear Regulatory Commission (NRC) on administrative controls, operational parameters, and equipment requirements, for safe reactor operation and for dealing with abnormal situations. They are typically derived from the safety analysis report (SAR). These specifications represent a comprehensive envelope for safe operation. The operational parameters and equipment requirements directly related to preserving this safe envelope are included.
>
> 1.2.2 Format
>
> The format of this document is in general accordance with ANSI/ANS-15.1-2007.

The NRC staff reviewed TSs 1.1 and 1.2 and finds that the information in TSs 1.1 and 1.2 related to the scope, purpose, and format of the TSs is consistent with the overall UMLRR TSs, with the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007, and with 10 CFR 50.36. The NRC staff finds that the UMLRR TSs include a safety limit (SL), limiting safety

system settings (LSSSs), limiting conditions for operation (LCOs), surveillance requirements (SRs), design features, and administrative controls, consistent with the requirements of 10 CFR 50.36(c). The NRC staff finds that the SL, LSSSs, LCOs, SRs, and design features in the TSs include applicability and objective statements consistent with the guidance in ANSI/ANS-15.1-2007 and NUREG-1537, Appendix 14.1. The NRC staff also finds that the bases for the SL, LSSSs, LCOs, SRs, and design features in the TSs summarize the rationale for those TSs, but that the bases are not part of the TSs, as required by 10 CFR 50.36(a)(1). Therefore, the NRC staff concludes that TSs 1.1 and 1.2 are acceptable.

6.1.2 Definitions

UML proposed the following TS definitions to provide for the uniform interpretation of terms and phrases used in the UMLRR TSs and to be generally consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. UML's TS definitions include minor modifications to some definitions in the guidance and some additional facility-specific definitions.

Renewed TS 1.3, "Definitions, would state:

> ADMINISTRATIVE CONTROLS – Those organizational and procedural requirements established by the NRC and/or the facility management.
>
> 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.
>
> 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 which the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip and shall be deemed to include a channel test.
>
> CHANNEL CHECK – A channel check is a qualitative verification of acceptable performance by observation of channel behavior, or by comparison of the channel with other independent channels or systems measuring the same parameter.
>
> CHANNEL TEST – A channel test is the introduction of a signal into the channel for verification that it is operable.
>
> CONFINEMENT – Confinement is an enclosure of the reactor building that is designed to limit the release of effluents between the enclosure and its external environment through controlled or defined pathways (see also Reactor Building).
>
> CONTROL BLADE – See Rod, Control.
>
> CORE CONFIGURATION – The core configuration includes the number, type, or arrangement of fuel elements, graphite reflector elements, control blades, regulating rod, irradiation baskets, source holders, lead void boxes, and grid plugs occupying the core grid.
>
> EXCESS REACTIVITY – Excess reactivity is that amount of reactivity that would exist if all control blades and the regulating rod were moved to the fully withdrawn position from the point where the reactor is exactly critical (k_{eff} = 1) at reference core conditions.

EXPERIMENT – Any operation, hardware, or target (excluding devices such as detectors, foils, etc.) that is designed to investigate non-routine reactor characteristics or that is intended for irradiation within the pool, on or in a beam port or irradiation facility. Hardware rigidly secured to a core or shield structure so as to be a part of their design to carry out experiments is not normally considered an experiment.

LICENSE – The written authorization, by the NRC, for an individual or the organization to carry out the duties and responsibilities associated with a personnel position, material, or facility requiring licensing.

LICENSEE – An individual or organization holding a license.

MEASURED VALUE – The measured value is the value of a parameter as it appears on the output for a channel.

MOVABLE EXPERIMENT – A movable experiment is one where it is intended that all or part of the experiment may be moved in or near the core or into and out of the reactor while the reactor is operating.

OPERABLE – Operable means a component or system is capable of performing its intended function.

OPERATING – Operating means a component or system is performing its intended function.

OPERATIONS MODE – Operations mode refers to the method by which the reactor core is cooled, either natural convection mode or forced convection mode of operation.

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 specific limit.

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.

REACTOR BUILDING – The reactor building is the enclosure housing the research reactor (see also Confinement).

REACTOR OPERATING – The reactor is operating whenever it is not secured or shut down.

REACTOR OPERATOR – An individual who is licensed by the NRC to manipulate the controls of the reactor.

REACTOR SAFETY SYSTEM – Reactor safety systems are those systems, including their associated input channels, that are designed to initiate automatic reactor protection or to provide information for initiation of manual protective action. The reactor safety system is also referred to as the reactor protection system.

REACTOR SECURED – The reactor is secured when:

(1) *Either* there is insufficient moderator available in the reactor to attain criticality or there is insufficient fissile material present in the reactor to attain criticality under optimum available conditions of moderation and reflection;

(2) *Or* the following conditions exist:

(a) All four control blades are fully inserted;

(b) The master key switch is in the off position and the key is removed from the lock;

(c) No work is in progress involving core fuel, core structure, installed control rods, or control rod drives unless they are physically decoupled from the control rods;

(d) No experiments are being moved or serviced that have, on movement, a reactivity worth exceeding the maximum value allowed for a single experiment (0.5% $\Delta k/k$).

REACTOR SHUTDOWN – The reactor is shut down if it is subcritical by at least one dollar (0.78% $\Delta k/k$) in the reference core condition with the reactivity worth of all installed experiments included.

REFERENCE CORE CONDITION – The condition of the core when it is at ambient temperature (cold) and the reactivity worth of xenon is negligible (<0.2% $\Delta k/k$).

RESEARCH REACTOR – A research reactor is defined as a device designed to support a self-sustaining neutron chain reaction for research, developmental, educational, training, and experimental purposes and that may have provisions for the production of radioisotopes. The terms research reactor and reactor may be used interchangeably.

RESEARCH REACTOR FACILITY – Includes those areas described in TS 5.1.2 within which the licensee directs authorized activities associated with the reactor. The terms research reactor facility, reactor facility, and facility may be used interchangeably.

ROD, CONTROL – A control rod is a device fabricated from neutron-absorbing material that is used to establish neutron flux changes and to compensate for routine reactivity losses. A control rod can be coupled to its drive unit allowing it to perform a safety function when the coupling is disengaged. The terms control rod and control blade may be used interchangeably.

ROD, REGULATING – The regulating rod is a low worth control device, used primarily to maintain an intended power level and does not have scram capability. Its position may be varied manually or by a servo-controller.

SCRAM TIME – Scram time is the elapsed time between the initiation of a scram signal and a specified movement of a control or safety device.

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.

SENIOR REACTOR OPERATOR – An individual who is licensed to direct the activities of reactor operators. Such an individual is also a reactor operator.

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.

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 safety systems starting from any permissible operating condition and with the most reactive control blade and regulating rod fully withdrawn and that the reactor will remain subcritical without further operator action.

SITE – The UMLRR site includes the reactor confinement building and the attached academic building (Pinanski Hall).

SURVEILLANCE TIME INTERVALS – The maximum allowable intervals listed as follows are to provide operational flexibility only. Established frequencies shall be maintained over the long term.

- 5 Year (interval not to exceed 6 years)
- Biennial (interval not to exceed 30 months)
- Annual (interval not to exceed 15 months)
- Semiannual (interval not to exceed 7-1/2 months)
- Quarterly (interval not to exceed 4 months)
- Monthly (interval not to exceed 6 weeks)
- Weekly (interval not to exceed 10 days)
- Daily (shall be done during the same working day)
- Prior to the first reactor start-up of the day

TRUE VALUE – The true value is the actual value of a parameter.

UNSCHEDULED SHUTDOWN – An unscheduled shutdown is defined as any unplanned shutdown of the reactor caused by actuation of the reactor safety 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.

The NRC staff reviewed the above TS definitions (including the defined surveillance intervals) and finds: (1) that they are either appropriate facility-specific definitions or are standard definitions used in research reactor TSs and (2) that they are consistent with applicable guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Therefore, the NRC staff concludes that the TS definitions are acceptable.

6.2 UMLRR Proposed Renewed TSs Section 2.0, "Safety Limits and Limiting Safety Settings"

Renewed TS 2.1, "Safety Limit," is evaluated and found acceptable in Section 2.2.1 of this SER.

Renewed TS 2.2, "Limiting Safety System Settings," including renewed TS 2.2.1, "Forced Convection Mode," and renewed TS 2.2.2, "Natural Convection Mode," is evaluated and found acceptable in Section 2.6 of this SER.

6.3 UMLRR Proposed Renewed TSs Section 3.0, "Limiting Conditions For Operation"

6.3.1 Proposed Renewed TS 3.1, "Reactor Core Parameters

Renewed TS 3.1.1, "Reactivity and Core Configurations," would state, in part:

> Applicability:
>
> These specifications apply to the reactivity condition of the reactor, core configuration, and experiments.
>
> Objective:
>
> To ensure that the reactor can be safety operated and shutdown and maintained in a safe shutdown condition at all times such that the Safety Limit will not be exceeded.
>
> Specifications:
>
> When the reactor is operating, the following conditions shall exist:
>
> …
>
> (6) The reactor shall not be operated whenever the reactor core is in the same end of the reactor pool as any portion of the cobalt-60 source.

Renewed TSs 3.1.1(1) and 3.1.1(2) are discussed and found acceptable in SER Section 2.5.3.

Renewed TSs 3.1.1(3) and 3.1.1(4) are discussed and found acceptable in SER Section 2.2.

Renewed TS 3.1.1(5) is discussed and found acceptable in SER Section 2.2.1.

Renewed TS 3.1.1(6) would require that the reactor not be operated when it and any portion of the cobalt-60 (Co-60) irradiation source are in the same end (bulk section or stall section) of the pool. In SAR Section 10.3, UML stated that TSs related to experiments and the Co-60 irradiation source are intended, in part, to prevent direct interference with reactor operations. In its basis for TS 3.1.1(6), UML stated that the specification prevents the Co-60 irradiation source from causing signal interference with the power measuring detectors, particularly at lower power levels. The NRC staff finds that TS 3.1.1(6) helps minimize the possibility of interference (related to instruments or physical activities or procedures) between the Co-60 irradiation source and the reactor. Therefore, based on the above and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.1.1(6) is acceptable.

Renewed TS 3.1.2, “Maximum Power Level,” is discussed and found acceptable in SER Section 2.5.1.

6.3.2 Proposed Renewed TS 3.2, “Reactor Control and Safety Systems”

Renewed TS 3.2.1, “Control Blades” is discussed and found acceptable in SER Section 2.2.2.

Renewed TS 3.2.2, “Maximum Reactivity Insertion Rate and Regulating Rod Worth” is discussed and found acceptable in SER Section 2.2.2.

Renewed TS 3.2.3, “Reactor Protection System Scrams,” is discussed and found acceptable in SER Section 2.5.3.

Renewed TS 3.2.4, “Radiological Protection Scrams,” is discussed and found acceptable in SER Section 4.1.5.

Renewed TS 3.2.5, “Minimum Channels Needed for Reactor Operation,” is discussed and found acceptable in SER Section 2.5.3.

Renewed TS 3.2.6, “Reactor Control System Interlocks,” is discussed and found acceptable in SER Sections 2.2.2 and 2.5.3.

6.3.3 Proposed Renewed TS 3.3, “Reactor Coolant Systems”

Renewed TS 3.3, “Reactor Coolant Systems,” is discussed and found acceptable in SER Section 2.3.

6.3.4 Proposed Renewed TS 3.4, “Confinement”

As discussed in SER Section 1.8, in conjunction with its license renewal request, UML requested approval to change the designation of the UMLRR reactor building from a containment to a confinement and to make associated changes to the operation of its ventilation system, and also requested associated TS changes. Renewed TSs 3.4 and 3.5, which are discussed and found acceptable below and in SER Section 6.3.5, respectively, include specifications related to UMLRR confinement (re-designated from containment) and ventilation. Renewed TSs 4.4 and 4.5 include SRs related to the TS 3.4 and 3.5 specifications and are discussed and found acceptable in SER Section 6.4. Additionally, UML’s TS definition for confinement is discussed and found acceptable in SER Section 6.1.2.

UML discussed its request to designate its reactor building as a confinement in its response to request for additional information (RAI)-6.1 (Ref. 23) and its letter dated September 30, 2020 (Ref. 98). Additionally, in its response to RAI-6.2 (Ref. 71), UML provided clarification that it is requesting NRC review of this change in conjunction with license renewal (rather than implementing the change under 10 CFR 50.59).

In its response to RAI-6.1, UML stated that, although its reactor building was constructed as a containment building (with the ability to be sealed to support a defined negative or positive differential pressure), it is appropriate for the building to be re-designated as a confinement, given that the UMLRR is licensed to operate at only 1 megawatt-thermal (MWt) (considerably lower than other research reactors that have a containment), and given that the SAR analyses which UML states demonstrate that the ability to seal the building and/or maintain a negative

pressure in the building are not necessary to maintain potential accident doses within 10 CFR Part 20 limits (however, the building will still normally be kept at a negative pressure as required by TS 3.5(2) following the re-designation). UML stated that, in conjunction with re-designating its reactor building as a confinement, it would also modify the conditions under which its emergency exhaust system would operate.

As discussed in SAR Section 6.2.3, the normal ventilation system shuts off and associated intake and exhaust values close to isolate the reactor building in response to any of the following: alarming of certain combinations of radiation monitors (see SAR Sections 7.7.5, 7.7.6, and 7.7.7 and SER Section 4.1.4); manual action by the reactor operator; and a loss of offsite electrical power. This response is referred to as a general reaction in the ventilation system (GRVS). Following the re-designation of the reactor building as a confinement, the normal ventilation system would continue to isolate the building following any of these events. As discussed in SER Section 5.1, the emergency exhaust system operates independently of the normal ventilation system. The emergency exhaust system routes air from the confinement, through high efficiency particulate air (HEPA) and charcoal filters, and out through the stack, at a lower flow rate and via a separate flow path than the normal ventilation system (see SAR Figure 6-2). The emergency exhaust system is designed to generally allow a negative differential pressure to continue to be maintained inside the reactor building and to relieve any positive overpressure.

As discussed in SER Section 5.7, the UMLRR backup electrical power system can provide power to ventilation valve control (such that the emergency exhaust valve can open to allow the emergency exhaust system to operate) and the emergency exhaust system fan, such that the emergency exhaust system can operate after a loss of offsite power. Given that a loss of offsite power results in a GRVS and that, with the reactor building re-designated as a confinement, the emergency exhaust system would activate following a GRVS (see discussion below), the emergency exhaust system would normally operate during a loss of offsite power.

Although, as discussed in SAR Section 6.2.3 and UML's letter dated September 30, 2020 (Ref. 98), the reactor building is isolated (intake and exhaust values closed, and exhaust fan off) following a GRVS, the intake fan normally remains on (except when there is a loss of offsite electrical power, in which case the intake fan shuts off but normally restarts once the emergency generator starts), and a valve opens (to its fail-safe position) which directs the flow from this fan up this stack. This allows for additional dilution of any contaminated air from the emergency exhaust system leaving the stack, if the emergency exhaust system is operating. Because the nominal intake flow rate is about 14,500 cubic feet per minute (cfm), and this is mixed with the emergency exhaust system flow of about 320 cfm when the emergency exhaust system is operating, the total exhaust flow from the stack when the emergency exhaust system is operating is similar to the total stack exhaust rate of about 15,000 cfm for normal ventilation system operation.

SAR Section 6.2.4 (which UML revised by information provided in its response to RAI-6.1 requesting to re-designate its reactor building as a confinement) discusses the previous operation of the emergency exhaust system. Previously, with the reactor building designated as a containment, the emergency exhaust system could be activated either manually by the reactor operator or automatically based on measurement of pressure differentials. Normally, the emergency exhaust system would operate in automatic mode. In automatic mode, the emergency exhaust system would activate if the differential pressure of the inside of the reactor building relative to outside reached or exceeded approximately positive 0.25 inches of water. Automatic mode operation of the system would continue until either: (1) the differential pressure

dropped to approximately negative 0.25 inches of water or (2) the differential pressure rose to or exceeded approximately positive 0.50 inches of water. A GRVS would not automatically activate the emergency exhaust system. If a GRVS occurred and the containment were isolated, the emergency exhaust system would only activate as discussed above to relieve overpressure in the sealed building.

UML's response to RAI-6.1 and its letter dated September 30, 2020 (Ref. 98), discuss the conditions under which the emergency exhaust system would operate following the re-designation of the reactor building as a confinement. With the reactor building designated as a confinement, the emergency exhaust system would be activated either manually by the reactor operator, automatically upon a GRVS, or automatically based on measurement of pressure differentials. For operation based on pressure differentials, the system would activate if the differential pressure of the inside of the reactor building relative to outside reached or exceeded approximately negative 0.10 inches of water. Automatic mode operation of the system would continue until either: (1) the GRVS is manually cleared by the operator; (2) the differential pressure dropped to approximately negative 0.25 inches of water; or (3) the differential pressure rose to or exceeded approximately positive 0.50 inches of water.

As discussed in UML's response to RAI-6.1 and in SER Section 5.1, in the unlikely event of a substantial overpressure of the reactor building (greater than approximately positive 0.50 inches of water (0.12 kilopascals)) relative to outside pressure, the emergency exhaust system does not continue to operate to relieve pressure (to attempt to return the building to negative differential pressure) because the charcoal filter in the emergency exhaust pathway is not intended for such a high volume, high pressure release. The emergency exhaust system would restart if the overpressure dropped back down to less than approximately positive 0.50 inches of water.

UML's analysis of the UMLRR maximum hypothetical accident (MHA), which is the bounding fission product release accident for the UMLRR, is discussed and found acceptable in SER Section 5.1. As discussed in SER Section 5.1, UML considered three scenarios for evaluation of MHA public doses. In Scenario D, which represents the most probable progression of the MHA event, accident-mitigating functions, including the emergency exhaust system (which comes on after radiation monitors detect the fission product release into the reactor building, resulting in a GRVS), generally work as designed. Scenarios B and C represent calculations that introduce additional conservatism by assuming that the emergency exhaust system does not operate (this could occur either due to an equipment failure or an overpressure scenario). Scenarios B and C consider that members of the public could be exposed to radiation due to direct external radiation from fission products held up in the reactor building or due to ground-level leakage of fission products from the building. UML's analyses demonstrated that, for any of these scenarios, the realistic maximum public doses from external radiation plus leakage would be below the 10 CFR Part 20 limits. Additionally, as also discussed in SER Section 5.1, the NRC staff performed confirmatory calculations of public doses from reactor building ground leakage, which are independent of leak rate (meaning that they would bound releases that are even more rapid than UML considered) and release point (meaning that they would bound alternate scenarios in which reactor building air leaked from locations other than the truck door considered by UML). The NRC staff's calculations also demonstrated that public doses from releases would remain below the 10 CFR Part 20 limits.

The NRC staff reviewed the above information regarding UML's proposed re-designation of its reactor building as a confinement, and associated changes (i.e., changes in the operation of the emergency exhaust system). The NRC staff finds that the results of the UMLRR MHA, which

are discussed and found acceptable in SER Section 5.1, demonstrate that it is not necessary for the reactor building to function as a containment (i.e., be able to be sealed) for the results of the MHA to be acceptable. The NRC staff also finds that configuring the emergency exhaust system to operate such that it normally automatically turns on when a GRVS occurs is appropriate given the re-designation of the building, because the emergency exhaust system will work to maintain negative building pressure and minimize leakage through pathways other than the stack (which provides a controlled, monitored, diluted flow path) following building isolation (if the building is treated as a confinement and its leakage rates are no longer measured, it cannot be considered to be airtight). However, as discussed above and in SER Section 5.1, even if the emergency exhaust system (which is required by TS 3.5(3)) fails to function as designed, public and occupational MHA doses will still be below 10 CFR Part 20 limits. The NRC staff finds that although the reactor building would no longer be designated as a containment, it still maintains robust design elements of a containment building, such as thick concrete, that continue to provide benefits such as shielding and protection of the reactor from external events. The NRC staff also reviewed renewed TSs 3.4, 3.5, 4.4, and 4.5 related to confinement and ventilation; these TSs are discussed and found acceptable below and in SER Sections 6.3.5 and 6.4. Based on the above, the NRC staff concludes that UML's proposed re-designation of its reactor building as a confinement, and associated changes in emergency exhaust system operation, are reasonable and appropriate, and that potential accident doses will remain within acceptable limits, as demonstrated by the results of the MHA, and, therefore, the changes are acceptable.

Renewed TS 3.4.1, "Operations Requiring Confinement," would state:

> Applicability:
>
> This specification applies to the reactor building.
>
> Objective:
>
> To restrict the release of airborne radioactive material into the environment in the event of an accident.
>
> Specifications:
>
> The operations requiring confinement shall be the following:
>
> (1) The reactor is operating.
>
> (2) Movement of irradiated fuel is being performed, except when the fuel is in a properly sealed and approved shipping container.
>
> (3) Core or control blade work that could cause a change in reactivity of more than 0.5% Δk/k.
>
> (4) The handling of radioactive material with the potential for significant airborne release.
>
> (5) Movement of experiments that could cause a change of total worth of more than 0.5%Δ k/k.

Renewed TS 3.4.1 would require that reactor building confinement be maintained when the reactor is operating; when irradiated fuel is being moved (unless it is in a properly sealed, NRC-licensed shipping container); when radioactive material with a potential for significant airborne release (e.g., fueled or non-fueled reactor experiments that contain significant activation and/or fission products and that are not appropriately contained to prevent the possibility of release) is being moved or handled; when core or control blade work that could cause a reactivity change of more than 0.5 percent delta k over k (% Δk/k) is being conducted; or during movement of experiments that could cause a reactivity change of more than 0.5% Δk/k. In its response to RAI-6.1 (Ref. 23), UML stated that designating the UMLRR reactor building as a confinement is adequate and appropriate to control any accidental radioactive material releases from the building, given the reactor operations and accident analysis (MHA). The NRC staff finds that by specifying the conditions or activities when confinement must be maintained (and, as discussed below and in SER Section 6.3.5, the TS 3.4.2 conditions needed to achieve confinement must be met and the ventilation equipment required by TS 3.5 must be operating or operable, as appropriate), TS 3.4.1 helps ensure that equipment required to minimize radioactive releases from the reactor building will be able to perform as intended during times when a potential accident condition could occur, and that the assumptions in the MHA are valid. The NRC staff finds that the conditions that require confinement are appropriate because they cover the likely range of conditions when a release of radioactive material into the reactor building could occur. The NRC staff finds that by requiring irradiated fuel movement (outside of a shipping container) and handling of radioactive material (with the potential for significant airborne release) to occur under confinement, TS 3.4.1 also helps ensure that UML will not conduct these activities under the reactor license outside of the reactor building. The NRC staff finds that by specifying appropriate conditions when confinement is required that are similar to or more limiting than the recommended conditions in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007, TS 3.4.1 appropriately implements the guidance in these documents (the NRC staff notes that the 0.5% Δk/k experiment movement limit is more conservative than the one dollar recommended in ANSI/ANS-15.1-2007, Section 3.4.1, but is equal to the UMLRR TS 3.7.1 single secured experiment, and total of absolute values of all movable experiments, reactivity limits). Therefore, based on the above and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.4.1 is acceptable.

Renewed TS 3.4.2, “Conditions Needed to Achieve Confinement,” would state:

> Applicability:
>
> This specification applies to the reactor building equipment required to achieve a confinement configuration.
>
> Objective:
>
> To prevent the release of reactor building exhaust air through other than defined pathways.
>
> Specifications:
>
> For any of the conditions in specification 3.4.1, the following equipment requirements shall be met:
>
> (1) At least one door in each of the personnel air locks is sealed and the truck door is sealed.

(2) All ventilation isolation valves, and the bypass valve are either operable or in the fail-safe position.

Renewed TS 3.4.2 would require that, in order to achieve confinement as required during the conditions in TS 3.4.1, (1) at least one door in each of the personnel air locks (i.e., the first and third floor personnel airlocks) must be sealed and the truck door must be sealed and (2) all ventilation isolation valves (i.e., valves that close when the reactor confinement building ventilation is isolated) and the bypass valve (i.e., the valve designated valve F in SAR Section 6.2, which can open to direct the air from the intake fan up the stack and increase dilution of air leaving the stack) are either operable (i.e., capable of closing, or opening in the case of valve F, when the building is isolated) or in the fail-safe position (i.e., closed for the isolation valves, open for the bypass valve). As discussed in UML's response to RAI-14.3.22 (Ref. 71), TS 3.4.2 includes requirements for conditions needed to achieve confinement, but TS 3.5 (which is discussed and found acceptable in SER Section 6.3.5) separately imposes requirements for equipment needed to achieve and maintain confinement. Both TSs 3.4.2 and 3.5 require that the conditions and equipment in the TSs be met or operating/operable, as applicable, during conditions or activities where confinement is required per TS 3.4.1. The NRC staff finds that TS 3.4.2, in conjunction with TS 3.5, helps ensure that the status of reactor conditions and equipment when confinement is required (i.e., when there is the potential for a significant airborne release of radioactive material into the reactor building) is appropriate to mitigate and control any release from the reactor building to the environment and that the assumptions in the MHA are valid. The NRC staff finds that TS 3.4.2(1) helps minimize any leakage of potentially contaminated air from the airlocks or truck door. The NRC staff finds that TS 3.4.2(2) helps ensure that the reactor building can be isolated as discussed in the SAR, as supplemented (when a GRVS occurs due to automatic or manual action, as discussed above), and assumed in the MHA analyses. The NRC staff also finds that TS 3.4.2(2) helps ensure that, when a GRVS occurs, the bypass valve is open to allow air from the intake fan to dilute exhaust air before it exits the stack, as discussed in the SAR and assumed in the MHA analysis. The ventilation isolation values are discussed in SAR Section 6.2 and illustrated in SAR Figure 6-2 and include valves designated A, B, C, E, G, and H. The bypass valve is also discussed in SAR Section 6.2 and illustrated in SAR Figure 6-2, and is designated valve F. The NRC staff notes that although TS 3.4.2(2) allows valves to be inoperable if they are in their fail-safe position (conservative for accident scenarios), UML must still ensure that the reactor building is properly ventilated during normal operations such that 10 CFR Part 20 occupational dose limits are met and doses are maintained ALARA, and that other ventilation TSs, such as the TS 3.5(2) requirement to maintain a negative reactor building pressure, are met. The NRC staff also finds that, by specifying minimum equipment and conditions needed to achieve confinement, TS 3.4.2 appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007. Therefore, based on the above and also based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.4.2 is acceptable.

6.3.5 Proposed Renewed TS 3.5, "Ventilation System"

Renewed TS 3.5, "Ventilation System," would state:

Applicability:

This specification applies to the normal and emergency exhaust ventilation equipment.

Objective:

To maintain a controlled pathway for reactor building exhaust air and minimize exposures from a release of airborne radioactive materials.

Specifications:

For any of the operations specified in 3.4.1:

(1) The main intake fan shall be operating. If a malfunction of the main intake fan occurs, the operations may continue only if the main intake fan is restored to the operating condition within 15 minutes of discovery.

(2) Building pressure shall be maintained at or more negative than 0.1-inch water column.

(3) The emergency exhaust system shall be operable.

(4) The emergency exhaust system charcoal filter shall have an efficiency of 95% or greater.

Renewed TS 3.5(1) would require that during conditions when confinement is required, the main intake fan shall be operating. Renewed TS 3.5(1) additionally would allow that, if the main intake fan malfunctions during operations when confinement is required, such operations may continue only if the main take fan is returned to its operating condition with 15 minutes of discovery of the fan malfunction. As discussed in SER Section 6.3.4 and in UML's basis for TS 3.5, the main intake fan normally provides fresh air to the confinement building, and under a condition where the reactor building normal ventilation is isolated, the fan provides dilution air up the stack (when the building is isolated, a bypass valve, which is designated valve F in SAR Section 6.2, and which is required by TS 3.4.2(2) to be operable or in its fail-safe (open) position when confinement is required, opens (if not already open) to its fail-safe position to direct the air from the fan up the stack). As discussed in SER Section 5.1, UML's MHA Scenario D, in which the ventilation system operates as designed, assumes that when a GRVS occurs following the release of radioactive material into the reactor building air, the emergency exhaust system activates and the main intake fan flow is redirected to provide dilution air to the emergency exhaust system flow before it leaves the stack. In its letter dated September 30, 2020 (Ref. 98), UML stated that the TS 3.5(1) allowance for operations to continue following an intake fan malfunction, if the fan can be restored to operation within 15 minutes of discovery of the malfunction, provides an action statement to avoid default violation of TS 3.5(1) should fan failure occur during operations that require confinement.

The NRC staff finds that, by requiring that the intake fan be operating during any conditions when confinement is required, TS 3.5(1) helps ensure that dilution air is mixed with potentially contaminated exhaust system air before it is released from the stack during a potential accident condition. The NRC staff finds that TS 3.5(1) helps ensure that the additional dilution air helps reduce the concentration of any radioactive material in the air. The NRC staff also finds that the dilution air increases the exhaust velocity of the air from the stack, which enhances atmospheric dispersion. As discussed in SER Section 5.1, UML used ARCON96 for its MHA public dose calculations, and the NRC staff notes that stack exhaust velocity is an ARCON96 input parameter; therefore, the NRC staff finds that TS 3.5(1) helps ensure the validity of UML's methodology and assumptions for its MHA Scenario D calculations of the most likely

progression of events following a radioactive material release within the reactor building (MHA Scenarios A, B, and C assume that the emergency exhaust system fails to function at all and radioactive material is held up and the reactor building and/or leaked out the truck door; therefore, stack exhaust velocity is not relevant for those scenarios). As also discussed in SER Section 5.1, the nominal intake fan flow rate is about 14,500 cfm. The NRC staff finds that the intake fan's operation at this nominal approximate flow rate is part of the intake fan performing its intended function during a potential accident scenario. As discussed in SER Section 5.7, the intake fan would normally operate to provide emergency exhaust system dilution following a loss of offsite power (although it would shut down for a brief period of time until the emergency generator starts). It is unlikely that a significant release to the reactor building would occur coincident with a loss of offsite power, but even if it did occur and the intake fan failed to continue operating, it would not cause 10 CFR Part 20 dose limits to be exceeded, as demonstrated by the NRC staff's confirmatory MHA public dose calculation.

The NRC staff further finds that the TS 3.5(1) allowance for continuance of operations is reasonable, because the fan is normally required to be operating during operations for which confinement is required; because 10 CFR Part 20 dose limits would not be exceeded even in the highly unlikely event that an MHA scenario occurred during a brief period while the fan were inoperable; and because the 15-minute time period is short compared to the 8-hour surveillance interval for verifying intake fan operability in renewed TS 4.4(1), which is discussed and found acceptable in SER Section 6.4.4. The NRC staff finds that in addition to helping ensure dilution during an accident, TS 3.5(1), in conjunction with TS 3.5(2), also helps ensure adequate normal ventilation of the reactor building to remove routine effluents such as argon-41 (Ar-41) and minimize doses to reactor staff. The NRC staff also finds that by specifying minimum equipment/conditions needed during any UMLRR activities when confinement is required, TS 3.5(1) appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007. Therefore, based on the above and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.5(1) is acceptable.

TS 3.5(2) would require that during conditions when confinement is required, the reactor building pressure must be maintained at or more negative than 0.1-inch water column. As discussed in SAR Section 6.2.3, during normal ventilation system operation, the intake blower brings air into the reactor building at a nominal rate of 14,500 cfm, while the exhaust blower removes air from the building at a nominal rate of 15,000 cfm. The difference in flow rates maintains a negative reactor building pressure, which is intended to cause the building to have in-leakage rather than out-leakage so that all building exhaust is through the stack (and is an elevated, controlled, monitored release). As discussed in SAR Section 6.2 and SER Section 4.1.1.1, there are experimental facility exhaust blowers, which can provide additional air removal from the reactor building, separate from the main (15,000 cfm) exhaust blower. As discussed in SER Section 6.3.4, when normal ventilation is isolated, the emergency exhaust system normally removes air from the reactor building at 320 cfm, but no air is blown into the reactor building (the intake fan air is instead directed to dilute the exhaust air), so a negative reactor building pressure can still be maintained. The NRC staff finds that TS 3.5(2) helps ensure that, during normal operating conditions or when the reactor building ventilation is isolated, the reactor building will be at a negative pressure with respect to the outside to ensure that any radioactive material releases are through the stack. The NRC staff finds that TS 3.5(2) helps ensure the validity of UML's methodology and assumptions for its MHA Scenario D calculations of the most likely progression of events following a radioactive material release within the reactor building (however, as discussed in SER Section 5.1, UML also considered MHA Scenario C in which negative differential pressure is unable to be maintained and radioactive material is released through the truck door, but showed that doses would still be within 10 CFR Part 20 limits). The

NRC staff finds that TS 3.5(2), in conjunction with TS 3.5(1), also helps ensure adequate normal ventilation of the reactor building to remove routine effluents such as Ar-41 and minimize doses to the reactor staff. The NRC staff also finds that by specifying minimum equipment/conditions needed during any UMLRR activities when confinement is required, TS 3.5(2) appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007. Therefore, based on the above and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.5(2) is acceptable.

TS 3.5(3) would require that during conditions when confinement is required, the emergency exhaust system shall be operable. SER Sections 5.1, 5.7, and 6.3.4 discuss the operation of the emergency exhaust system. As discussed in SER Section 5.1, UML's MHA Scenario D calculations of the most likely progression of events following a radioactive material release within the reactor building assume that the reactor building ventilation system, including the emergency exhaust system, operates normally. However, UML also considered MHA Scenarios B and C, in which the emergency exhaust system fails to operated even though it is required by TS 3.5(3), and determined that public doses would still be within the 10 CFR Part 20 limit (UML's MHA Scenario A, which evaluated doses to workers in the reactor building, also conservatively assumed that there is no ventilation from the reactor building by the emergency exhaust system or otherwise). The NRC staff finds that although UML's MHA calculations show that worker and public doses would be within 10 CFR Part 20 limits even if the emergency exhaust system failed to operate, TS 3.5(3) helps ensure that the assumptions of UML's most likely and realistic MHA Scenario D are valid and that worker and public doses are minimized in the event a radioactive material release in the reactor building. The NRC staff finds that TS 3.5(3) helps ensure that any contaminated air releases from the building following an accident are through a controlled, monitored, and filtered (as discussed in SAR Section 6.2, the emergency exhaust system passes air through both charcoal and HEPA filters; renewed TS 3.5(4), which is discussed and found acceptable below, separately imposes a specific requirement on the efficiency of the charcoal filter) pathway and that a negative building pressure can continue to be maintained. The NRC staff finds that TS 3.5(3) also helps ensure that contaminated air is removed from the reactor building to reduce exposure to any reactor staff who may continue to be present in the building or have to re-enter the building following an accident. The NRC staff finds that the emergency exhaust system's operation at its nominal flow rate of approximately 320 cfm and the operability of ventilation system valve D, which opens to allow emergency exhaust air to exit the reactor building (see SAR Figure 6-2), are part of the system performing its intended function. The NRC staff also finds that by specifying minimum equipment/conditions needed during any UMLRR activities when confinement is required, TS 3.5(3) appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007. Therefore, based on the above and also based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.5(3) is acceptable.

TS 3.5(4) would require that during conditions when confinement is required, the charcoal filter in the emergency exhaust system shall have an efficiency of at least 95 percent. As discussed in SAR Section 6.2, the emergency exhaust system passes air through both charcoal and HEPA filters. SAR Section 6.2.5 states that the minimum 95 percent efficiency of the charcoal filter in the emergency exhaust system is for removal of elemental iodine. In SAR Section 13.2.1.2, UML stated that it does not take credit for either the charcoal or HEPA filters in the emergency exhaust system in its MHA analyses. However, in its response to RAI-14.4.19 (Ref. 71), UML stated that the charcoal filter is considered in the MHA analysis. Based on the results of its MHA Scenario D confirmatory calculations discussed in SER Section 5.1, the NRC staff notes that most of the public total effective dose equivalent (TEDE) outside the reactor building is due

to noble gas radionuclides; less than 10 percent of the TEDE is due to iodine radionuclides, even when the charcoal filter is not credited (the NRC staff did not consider the charcoal filter in its analysis). The NRC staff also notes that the 95 percent efficiency of the filter is for elemental iodine, but some of the iodine reaching the filter in the MHA would likely be organic iodine, which charcoal filters are generally less likely to remove. Therefore, the NRC staff finds that whether the charcoal filter is credited does not make a significant difference in the MHA results. However, the NRC staff finds that the required charcoal filter will remove some of any radioactive iodine that its released through the emergency exhaust system following a potential accident and, therefore, TS 3.5(4) helps reduce the iodine released from the building and any dose to the public. The NRC staff also finds that by specifying minimum equipment/conditions needed during any UMLRR activities when confinement is required, TS 3.5(4) appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007. Therefore, based on the above and also based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.5(4) is acceptable.

In its responses to RAI-6.1 (Ref. 23) and RAI-14.3.20 (Ref. 71), UML stated that in conjunction with its renewal request, it was providing revised TSs to reflect its proposed re-designation of the UMLRR reactor building from a containment to confinement. UML indicated that, therefore, current TSs related to building leak rate testing and verification that the building still meets containment requirements following modifications of the building or its penetrations should not be included in the renewed TSs. In its response to RAI-14.3.22 (Ref. 71), UML also stated that a TS for the reactor building vacuum relief valve (which, as described in SAR Section 6.2.1, is a passive design feature of the reactor building that allows air to pass into the building to prevent the building from reaching a vacuum of more than 0.5 inches of water with respect to the outside, helping protect the reactor building from under-pressure), is no longer necessary because the relief valve is not a component needed to achieve and maintain confinement (the NRC staff notes that the UMLRR containment TSs prior to this renewal included an LCO for the vacuum relief valve, but no surveillance of the valve's operability).

The NRC staff finds that TSs related to building leak rate and containment integrity are not necessary for the reactor building to serve the purpose of a confinement, which is to limit the release of effluents from the building, as stated in the UMLRR TS definition of "confinement," but not necessarily to be airtight. The vacuum relief valve remains a part of the existing building as described in the renewal SAR, as supplemented (UML proposed to change the conditions under which the emergency exhaust system operates, as discussed in SER Section 6.3.4, but did not propose any physical changes to the reactor building in conjunction with its proposed re-designation as a confinement). The NRC staff finds that the vacuum relief valve continues to be an appropriate design feature to protect the reactor building, given that the building was originally designed to be a containment. However, the NRC staff finds that a specific TS for the valve is not necessary because, even if the valve failed to open and a building under-pressure could cause damage to or failure of reactor building seals (e.g., the truck door seal), the building is not intended to be airtight, and any resulting additional building leakage would likely continue to be in-leakage. The NRC staff also finds that because the reactor building is being re-designated as a confinement, TS requirements related to building leakage rates, containment integrity, and the vacuum relief valve are not necessary for the renewed TSs to appropriately reflect the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007. Therefore, based on the above, the NRC staff concludes that TSs related to building leak rate, containment integrity, and the vacuum relief valve are not required for the UMLRR. As discussed in SER Section 6.7, the NRC staff finds that UML has LCO TSs that satisfy the provisions of 10 CFR 50.36(c)(2).

6.3.6 Proposed Renewed TS 3.6, "Radiation Monitoring Systems and Effluents"

Renewed TS 3.6.1, "Radiation Monitoring," specifications (1) through (3), are discussed and found acceptable in SER Section 4.1.4.

Renewed TS 3.6.1, "Radiation Monitoring," specification (4), is discussed and found acceptable in SER Section 4.1.7.

Renewed TS 3.6.2, "Effluents," specification (1), is discussed and found acceptable in SER Section 4.1.1.2.

Renewed TS 3.6.2, "Effluents," specification (2), is discussed and found acceptable in SER Section 4.1.1.1.

6.3.7 Proposed Renewed TS 3.7, "Experiments"

The applicability and objective of renewed TS 3.7, "Experiments" (which are followed by the specifications for TSs 3.7.1 and 3.7.2), would state:

> Applicability:
>
> This specification applies to experiments to be installed in the reactor and associated experimental facilities.
>
> Objective:
>
> To prevent damage to the reactor or excessive release of radioactive materials in the event of experiment failure.

Renewed TS 3.7.1, "Reactivity Limits," would state:

> Specifications:
>
> (1) The absolute reactivity worth of any single movable experiment shall not exceed 0.25%Δk/k.
>
> (2) The sum of the absolute values of reactivity worths of all movable experiments shall not exceed 0.5%Δk/k.
>
> (3) The absolute reactivity worth of any single secured experiment shall not exceed 0.5%Δk/k.
>
> (4) The sum of the absolute values of the reactivity worths of all secured experiments shall not exceed 2.5%Δk/k.
>
> (5) The sum of the absolute values of the reactivity worths of all experiments shall not be greater than 2.5%Δk/k.

Renewed TSs 3.7.1(1) through 3.7.1(5) would impose requirements on the reactivity worth of movable and secured experiments, as defined in the TS definitions (see SER Section 6.1.2). Renewed TSs 3.7.1(1) and 3.7.1(3) would require that the absolute worth (i.e., the absolute

value of the positive or negative reactivity) of individual movable or secured experiments not exceed 0.25% Δk/k or 0.5% Δk/k, respectively. Renewed TSs 3.7.1(2) and 3.7.1(4) would require that the sum of absolute worths of all movable or all secured experiments not exceed 0.5% Δk/k or 2.5% Δk/k, respectively. Renewed TS 3.7.1(5) would require that the total sum of the absolute worths of all movable and secured experiments not exceed 2.5% Δk/k. As discussed in SER Section 5.2, UML analyzed reactivity insertion accidents involving the rapid (step) insertion of 0.6% Δk/k of reactivity and demonstrated that maximum fuel cladding temperatures following any such event would remain well below the SL. In its basis for TS 3.7.1, UML stated that the total maximum reactivity of 2.5% Δk/k in TSs 3.7.1(4) and 3.7.1(5) is a reasonable upper limit on the worth of all experiments which is compatible with the allowable excess reactivity and the shutdown margin (SDM) and is consistent with the functional mission of the reactor.

The NRC staff finds that TSs 3.7.1(1) and 3.7.1(3) require that the absolute reactivity worth of any individual experiment be less than the reactivity assumed in UML's analyzed step reactivity insertions, helping ensure that no failure of or operator error involving (e.g., unintentional insertion or removal of an experiment during operation) any individual experiment could cause a reactivity transient that could result in the SL being exceeded. The NRC staff finds that TS 3.7.1(2) helps ensure that even if all movable experiments were moved at the same time during reactor operation, or in the unlikely event that multiple movable experiments failed simultaneously, any resulting reactivity change would be bounded by UML's analyzed step reactivity insertions and the SL would not be exceeded. The NRC staff finds that the TS 3.7.1(1) limit of 0.25% Δk/k (equivalent to approximately 32 cents for the UMLRR) for moveable experiments is conservative compared to the NUREG-1537, Appendix 14.1, Section 3.8.1, recommended limit of $1.00 for movable experiments, and helps ensure that no movable experiment could cause an inadvertent prompt criticality. The NRC staff finds that because TS 3.1.1(2) (see SER Section 2.5.3) requires that SDM be determined with all installed experiments in their most reactive state, the movement of a movable experiment should not increase core reactivity such that the SDM TS 3.1.1(2) could be violated; also, as discussed in SER Section 2.5.3, UML is able to maintain a large margin (much greater than the maximum 0.25% Δk/k for any movable experiment) to its minimum SDM. The NRC staff finds that the TS 3.7.1(3) limit of 0.5% Δk/k (equivalent to approximately 64 cents for the UMLRR) for secured experiments is consistent with the NUREG-1537, Appendix 14.1, Section 3.8.1, recommendation that reactivity limits for secured experiments be approximately twice those for movable experiments. The NRC staff notes that the TS 3.7.1(5) limit of 2.5% Δk/k (equivalent to approximately $3.21 for the UMLRR) for all experiments exceeds the NUREG-1537, Appendix 14.1, Section 3.8.1, recommendation that the limit for all experiments not be more than twice the limit on individual secured experiments. However, the NRC staff finds that given the conservatism in the limits for individual movable and secured experiments, and because the TS 3.7.1(5) limit is less than $4.00 (twice $2.00, which would be the individual secured experiment limit if, as recommended by NUREG-1537, the individual secured experiment limit was twice the maximum NUREG-1537 recommended limit of $1.00 for individual movable experiments), the TS 3.7.1(5) limit is reasonable and consistent with the intent of the guidance in NUREG-1537. Although the TS 3.7.1(5) limit of 2.5% Δk/k is greater than the 0.6% Δk/k rapid reactivity insertion accident that UML analyzed, the NRC staff finds that a simultaneous failure of all movable and secured experiments such that 2.5% Δk/k of reactivity would be rapidly inserted is not a credible event. The NRC staff finds that TS 3.7.1(4) is bounded by TS 3.7.1(5) because TS 3.7.1(5) places a limit on all experiments, but TS 3.7.1(4) places the same limit on secured experiments only, which are less likely to cause core reactivity changes because they cannot be moved. The NRC staff finds that the requirement in TSs 3.7.1(1) through 3.7.1(5) to consider the absolute values and sums of absolute values of experiment worths is conservative

and consistent with guidance in NUREG-1537, Appendix 14.1, Section 3.8.1, and ANSI/ANS-15.1-2007. Therefore, based on the above and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.7.1 is acceptable.

Renewed TS 3.7.2, "Design and Materials," would state:

> Specifications:
>
> (1) Experiments shall be designed such that a credible failure of the experiment shall not result in releases or exposures in excess of 10 CFR Part 20 limits.
>
> (2) Experiments shall be designed such that a failure of an experiment shall not contribute to the failure of another experiment, core components, or principal physical barriers to uncontrolled release of radioactivity.
>
> (3) All materials to be irradiated shall be either corrosion resistant or encapsulated within corrosion resistant containers to prevent interaction with reactor components, pool water, or Co-60 sources. Corrosive materials shall be doubly encapsulated. Should a failure of the encapsulation occur that could damage the reactor or Co-60 sources, the potentially damaged components shall be inspected.
>
> (4) 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. In addition, the irradiation container for this material shall be designed and tested for a pressure exceeding two times the maximum expected pressure from detonation.
>
> (5) Each fueled experiment shall be limited such that the total inventory of iodine-131 through iodine-135 in the experiment is not greater than 100 mCi.

TS 3.7.2(1) would require that experiments be designed such that if an experiment fails, any resulting radiation exposure to reactor staff or the public will not exceed 10 CFR Part 20 dose limits. TS 3.7.2(2) would require that experiments be designed such that if an experiment fails, the failure of the experiment shall not contribute to the failure of another experiment, core components (e.g., core structural components or control blades), or principal physical barriers to uncontrolled release of radioactivity (e.g., fuel cladding). In its bases for TSs 3.7.2(1) and 3.7.2(2), UML stated that these TSs are intended to prevent experiments from having any unacceptable radiological consequences. The NRC staff finds that TS 3.7.2(1) helps ensure that UML designs experiments (e.g., by limiting radioactive material inventory in an experiment) such that no experiment failure could cause an unacceptable radiation dose in excess of 10 CFR Part 20 limits and that TS 3.7.2(2) helps ensure that UML designs experiments such that their failure will not cause other experiment or reactor failures that could result in additional radiological consequences. The NRC staff finds that TSs 3.7.2(1) and 3.7.2(2) are consistent with the guidance in ANSI/ANS-15.1-2007, Section 3.8.3, for TSs related to failures and malfunctions of experiments, and appropriately implement the overall guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 that TSs should provide limitations for experiments. Therefore, based on the above and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.7.2(1) and TS 3.7.2(2) are acceptable.

TS 3.7.2(3) would require that materials irradiated in experiments be corrosion resistant, or if they are corrosive (i.e., not corrosion resistant), the materials must be double encapsulated in corrosion resistant containers. Additionally, TS 3.7.2(3) requires that if a failure of corrosive material encapsulation occurs such that there is the potential for damage to reactor components (e.g., fuel, control blades, or core structural components) or the Co-60 irradiation source, the potentially damaged components or sources shall be inspected for damage. In its basis for TS 3.7.2(3), UML stated that this TS is intended to provide assurance that experiments will not cause unintended chemical reactions that could damage structures or components and potentially cause unacceptable radiological consequences. In its response to RAI-14.3.30 (Ref. 71), UML stated that although TS 3.7.2(3) would require inspection of a potentially damaged item, it would not necessarily require the removal (e.g., from the pool) of an item such as irradiated fuel or a Co-60 source since radiological conditions may not permit the removal. The NRC staff finds that TS 3.7.2(2) helps ensure that the reactor and the Co-60 irradiation source are protected from any damage caused by experiments containing corrosive material, and if there is the potential for corrosion damage, components will be appropriately inspected. The NRC staff finds that TS 3.7.2(3) is consistent with guidance in NUREG-1537, Appendix 14.1, Section 3.8.2, regarding corrosive materials, and appropriately implements the overall guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 that TSs should provide limitations for experiments. Therefore, based on the above and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.7.2(3) is acceptable.

TS 3.7.2(4) would require that if an experiment contains explosive materials, and/or if there is the potential for explosive materials to be generated within an experiment, the quantity of explosive materials contained and/or generated shall not exceed 25 milligrams of trinitrotoluene (TNT)-equivalent. Additionally, TS 3.7.2(4) would require that any experiment that contains explosive material and/or has the potential for explosive material generation be in an irradiation container that is designed and tested for a pressure exceeding two times the maximum expected pressure from potential detonation. In its basis for TS 3.7.2(4), UML stated that the TS provides assurance that explosive materials in an experiment will not cause encapsulation failure or damage structures or components and potentially cause unacceptable radiological consequences. In its response to RAI-14.3.31 (Ref. 71), UML stated that its TS 3.7.2(4) includes an allowance to have explosive materials in experiments to give it flexibility for reactor usage for education and research, and that the TS is similar to those of similar research reactors. The NRC staff notes that UMLRR License Amendment No. 5, issued January 15, 1982 (Ref. 74), which originally authorized UML to use the Co-60 sources as a pool irradiation source in the UMLRR pool, also revised the TSs to prohibit irradiation of any quantity of explosive material in the reactor pool or experimental facilities to prevent damage to the Co-60 sources (or reactor); however, UMLRR License Amendment No. 12, issued July 31, 1997 (Ref. 37), subsequently revised the TSs to allow irradiation of less than 25 milligrams of explosive materials (including TNT), provided that testing and analysis showed that detonation of the explosive would not damage the explosive containers, reactor, or Co-60 sources. The NRC staff finds that TS 3.7.2(4) helps ensure that the reactor and Co-60 sources are protected from any potential for damage from explosive material in experiments. The NRC staff also finds that TS 3.7.2(4) helps ensure that UML designs and tests containers for experiments containing explosives to verify that they can withstand any potential detonation of the explosives. As discussed below, other TSs require UML to review any specific new experiment or class of experiments, including experiments containing explosive material as allowed by TS 3.7.2(4), to ensure that they would be designed to limit the possibility of damage to the reactor and/or Co-60 sources, or any radiological release. The NRC staff additionally finds that TS 3.7.2(4) is consistent with guidance in NUREG-1537, Appendix 14.1, Section 3.8.2, which recommends a maximum TS limit of 25 milligrams of TNT in experimental facilities, and recommends that TSs

require prior testing or analyses of explosive material encapsulations to ensure no reactor damage in the event of detonation. The NRC staff also finds that TS 3.7.2(4) appropriately implements the overall guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 that TSs should provide limitations for experiments. Therefore, based on the above and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.7.2(4) is acceptable.

Renewed TS 3.7.2(5) would require that any fueled experiment at the UMLRR be limited such that the experiment will not contain more than 100 millicuries of the radioisotopes iodine-131 (I-131) through iodine-135 (I-135) (i.e., from fission product generation). In its response to RAI-13.9 (Ref. 71), UML stated that the 100 millicurie limit is approximately half of the activity of I-131 through I-135 used in the MHA analysis in the SAR, as supplemented, and that the occupational and public dose consequences of the MHA are well below 10 CFR Part 20 limits. As discussed in SER Section 5.1, the NRC staff reviewed the UMLRR MHA and finds that the doses are well within 10 CFR Part 20 limits, and that the MHA is acceptable. The NRC staff reviewed the MHA analysis and verified that 100 millicuries of I-131 through I-135 is slightly less than half of the total I-131 through I-135 released from the reactor pool to the reactor building air for the MHA (approximately 220 millicuries, after applying the release fractions that UML used as discussed in SER Section 5.1). UML determined the MHA values of I-131 through I-135 based on saturated inventories; the NRC staff finds that this is conservative for the MHA or fueled experiments because it maximizes the fraction of the total iodine inventory that could be I-131, and that I-131 has the highest inhalation dose conversion factor of the five iodine radioisotopes. The NRC staff finds that UML's use of a limit of 100 millicuries, based on approximately half the released radioiodine in the MHA, for the total inventory of a fueled experiment is conservative because it effectively assumes that 100 percent of the iodine, krypton, and xenon fission products in the fueled experiment could be released to the reactor building air (generation of a given saturated inventory of iodine will also result in saturated inventories of most krypton and xenon fission products, and the MHA considered krypton and xenon as well as iodine). As discussed in SER Section 5.1, the MHA did not consider the release of particulate fission products (i.e., fission products other than iodine, krypton, and xenon) from the pool because the MHA fuel failure is assumed to occur in the pool water and most of the particulate fission products released (if any) would remain dissolved in the pool water. The NRC staff notes that given that a fueled experiment may be irradiated outside the pool (e.g., in a beam tube), a failure could occur in air, and that there is some potential for a greater particulate release from a fueled experiment failure. However, because (1) the maximum iodine, krypton, and xenon that could be released from a failed fueled experiment (even assuming a 100-percent release fraction) is less than half of the release assumed in the MHA; (2) the particulate fission products are less volatile and generally less likely to be released; (3) many particulate fission products (e.g., strontium-90, which has a 28.8 year half-life) would not be likely to approach a significant inventory in the realistic duration of typical fueled experiments limited to only 100 millicuries of iodine (the NRC staff estimated that a fueled experiment with a 100 millicurie saturated iodine inventory would require approximately 4.8 years of continuous irradiation to reach a 5 millicurie inventory of strontium-90); and (4) the significant conservatisms assumed in the MHA analysis (see SER Section 5.1), the NRC staff finds that the realistic consequences of a failure of a fueled experiment conducted in accordance with TS 3.7.2(5) would likely be bounded by the MHA. Therefore, the NRC staff finds that TS 3.7.2(5) helps ensure that fueled experiments at the UMLRR are bounded by the MHA and could not result in an unacceptable radiation dose in excess of 10 CFR Part 20 limits. TS 3.7.2(1), discussed above, separately requires that experiments, including fueled experiments, be designed such that any credible failure of an experiment not result in dose consequences exceeding 10 CFR Part 20 limits. As discussed below, other TSs require UML to

review any specific new experiment or class of experiments, including fueled experiments, to ensure that the experiment would be designed to limit the possibility of failures that could result in any radiological release (e.g., by using controls such as double encapsulation, if needed), and that the experiment would be appropriately bounded by SAR analyses and could not cause unacceptable radiological consequences if it did fail. The NRC staff also finds that by effectively limiting fissionable materials in experiments by limiting the fission product inventory, TS 3.7.2(5) appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007. Therefore, based on the above and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.7.2(5) is acceptable.

UML did not provide specific surveillance TSs for TSs 3.7.1 or 3.7.2. However, renewed TS 6.2.3(1), item c., which is discussed and found acceptable in SER Section 6.6.2, and renewed TS 6.5, which is discussed and found acceptable in SER Section 6.6.5, require that experiments be reviewed (including verification that the experiments meet TS 3.7 and other applicable TSs) and approved prior to initiation. UML's experiment review procedures are discussed in SAR Sections 10.3 and 12.2.3, as supplemented by UML's responses to RAI-14.6.16 through RAI-14.6.19 (Ref 71). The NRC staff finds that the TSs requiring experimental review and approval are sufficient to help ensure that UML verifies its compliance with TSs 3.7.1 and 3.7.2 and that UML's not providing separate, specific surveillance TSs for TSs 3.7.1 and 3.7.2 is reasonable and consistent with the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007, which states that reviews to determine compliance of experiments with TSs are typically not part of surveillance TSs. Also, as discussed in SER Section 6.7, the NRC staff finds that the UMLRR has surveillance TSs that satisfy the provisions of 10 CFR 50.36(c)(3). Therefore, the NRC staff concludes that separate surveillance TSs are not required for TSs 3.7.1 and 3.7.2.

Regarding the TS 3.7.2 limits on experiment design and materials, the NRC staff notes that consistent with guidance in NUREG-1537, Appendix 14.1, Sections 3.8.2 and 3.8.3, and ANSI/ANS-15.1-2007, Section 3.8, and consistent with TSs for other similar, recently renewed research reactor licenses, TS 3.7.2 is generally less prescriptive compared to the experiment TS limits prior to this license renewal (i.e., TS 3.6 in the current UMLRR TSs). For example, TS 3.7.2 does not prohibit any specific material (e.g., cryogenic liquids or highly water reactive materials) from being used in experiments, does not impose any specific thermal-hydraulic limits on experiments, and does not require any specific encapsulation of experiments other than experiments with corrosive or explosive material. However, as discussed above and in SER Sections 6.6.2 and 6.6.5, the TSs require UML to evaluate all new experiments or classes of experiments to help ensure that experiments are designed and planned such that the routine operation or credible failure of any experiment will not unacceptably interfere with other facility operation, cause any thermal-hydraulic limit to be exceeded, or cause unacceptable radiological consequences that may not be bounded by the MHA or could create the potential for 10 CFR Part 20 dose limits to be exceeded.

6.3.8 Proposed Renewed TS 3.8, "Beam Port Operations"

Renewed TS 3.8, "Beam Port Operations," would state:

<u>Applicability</u>:

This specification applies to restrictions associated with operation of the beam ports.

Objective:

To minimize the possibility and effect of a loss of coolant accident.

Specifications:

(1) The reactor shall not be operated with both a beam port lead shutter in the up (open) position and the corresponding beam-port shield plug removed.

(2) The shield plug may be substituted or modified so long as the overall open diameter shall not exceed an area equivalent to 4 inches in diameter.

(3) When a beam port lead shutter is in the up position while the corresponding shield plug is also removed, the reactor shall be positioned in the bulk pool.

(4) When the pool divider gate is in position to separate the bulk pool and the stall pool, and the reactor is in the stall pool, the beam port shutters shall be in the down (closed) position.

TS 3.8(1) would require that during reactor operation, a beam port lead shutter shall not be open at the same time that the shield plug for that beam port is removed. As discussed in SER Section 5.3, each beam port has a heavy lead shutter within the pool wall, and a bolted shield plug at the outer shield wall, which would help preclude accidental draining through the ports in the unlikely event that the portion of the beam tube that extends into the reactor pool were accidentally sheared off. As also discussed in SER Section 5.3, UML performed a loss-of-coolant accident (LOCA) analysis in which it calculated that in order to avoid exceeding the SL following a LOCA, the reactor core needs to remain covered with water for at least 3,947 seconds following reactor operation (as well as during operation) to allow for adequate reduction in decay heat. Additionally, UML calculated that during a LOCA, as long as the area through which the pool drains is equivalent to a round hole of diameter 4.5 inches or less, it will take at least 3,947 seconds for the pool to drain enough that any part of the fuel becomes uncovered. The NRC staff finds that by requiring that for each beam port, either the shutter be closed or the bolted plug be in place when the reactor is operating, TS 3.8(1) helps ensure that in the unlikely event a beam tube is sheared during reactor operation, either the shutter or plug will help prevent any possibility of a LOCA where the pool is able to be drained through a hole or holes larger than the maximum diameter round hole determined in UML's analysis. Therefore, the NRC staff finds that TS 3.8(1) helps ensure that a LOCA during reactor operation could not cause the SL to be exceeded.

The NRC staff notes that although UML calculated that the reactor needs to remain covered with water for at least 3,947 seconds following reactor operation, and TS 3.8(1) restricts beam port configurations to help ensure that the pool cannot drain in less than 3,947 seconds, TS 3.8(1) is not applicable when the reactor is not operating. If a beam tube were completely sheared off shortly after reactor operation when a beam port's shutter and plug were both open, the reactor could, potentially, become uncovered in less than 3,947 seconds following reactor operation. However, in its letter dated September 30, 2020 (Ref. 98), UML stated that per TS 3.8, in order for UML to open a beam tube plug while the corresponding beam tube shutter is also open, the reactor must not only be shut down but must also be moved to the bulk pool. If the shutdown reactor were in the stall pool and a beam tube were damaged, any drainage would be limited to a 4-inch diameter round opening (similar to the 4.5 inches used in UML's LOCA analysis) per the TS 3.8(2) limit on the size of any opening in a beam tube plug. If the

shutdown reactor were in the bulk pool, a beam tube damage scenario could potentially cause drainage through an 8-inch diameter round opening if both a beam tube plug and the corresponding shutter were open. However, UML stated that although its LOCA analysis only assumes that drainage occurs through a 4.5 inch diameter opening, the assumptions in its analysis are extremely conservative, including that the reactor has been operated at full power for an infinite period of time and that pool drainage starts immediately upon reactor shutdown. In actuality, even without a TS requiring UML to wait a certain amount of time after shutdown to open a beam port plug and shutter, it would take UML some time to move the reactor to the bulk pool (as required by TS 3.8(3)) and actually perform the beam tube reconfiguration, and this time would allow for some decay following shutdown. Additionally, given the design of, and operational practices at, the UMLRR, the complete shearing of a beam tube that would allow the reactor pool to be drained through an 8-inch diameter opening is an extremely unlikely scenario. UML stated that, per procedure, it does not move or handle heavy loads (e.g., using the overhead crane) over the stall pool (in which the beam tubes are located), regardless of the position of the reactor in the pool. Furthermore, even if such a scenario did occur, UML would be able to mitigate the scenario. UML could close the pool divider gate to isolate the reactor in the bulk pool (UML stated that this could be accomplished very quickly, within a few minutes and well before the pool could drain to the level of the reactor even with an 8-inch break, because the reactor would already be in the bulk pool), preventing the reactor from becoming uncovered. The pool divider gate would be closed using the overhead crane, and this could be accomplished by personnel located to the side of the pool such that any potential exposure to direct radiation from fuel or the Co-60 sources in the pool (the radiation level above the pool could increase as the pool level decreases) would be minimized. UML could also close the beam tube shutter (also typically using the overhead crane, and similarly minimizing any exposure of personnel to direct radiation from the pool) to effectively stop the pool drainage because the heavy lead shutter would block the beam tube.

The NRC staff reviewed UML's justification for TS 3.8(1) not being applicable when the reactor is not operating. The NRC staff notes that if TS 3.8(1) is not applicable during the 3,947 seconds following reactor operation, a potential LOCA scenario could exceed the bounds of UML's analysis discussed in SER Section 5.3. By scaling the area of the 4.5 inch diameter opening assumed in UML's analysis to the area of a potential 8 inch diameter opening, the NRC staff estimated that the pool could potentially be drained in approximately 20 minutes, less than the 3,947 seconds (approximately 66 minutes) that UML calculated the core needed to remain covered with water to prevent the SL from being exceeded. However, the NRC staff finds that a scenario involving severe damage to a beam tube that could cause significant pool drainage (faster than assumed in UML's LOCA analysis) through the beam tube is unlikely. Although the beam tubes are less protected when the reactor is in the bulk pool (given the vertical protection provided by the bridge when the reactor is in the stall pool), the NRC staff expects UML to exercise appropriate caution when conducting any activities over the pool in which objects could fall into the pool that could damage the core, beam tubes, or other UMLRR equipment (regardless of the position of the reactor or the beam tube configuration), consistent with standard industry practice. As discussed above, UML stated that its procedures prohibit movement or handling of heavy loads over the portion of the pool where the beam tubes are located, regardless of whether the beam tubes have overhead protection from the bridge, to prevent a dropped load from damaging a beam tube. Furthermore, even if damage did occur, it would not likely be a complete shearing of the tube and the drainage rate would be less than that associated with a complete shearing. Additionally, the NRC staff finds that in any realistic LOCA, the UMLRR operational history would be less limiting than that assumed in UML's LOCA analysis (in which pool drainage starts immediately following an infinitely-long period of full-

power reactor operation). UML's LOCA analysis also uses other conservative assumptions as discussed in SER Section 5.3.

As discussed above, UML stated that it would have at least two actions it could take to mitigate the pool drainage: isolating the reactor in the bulk pool using the divider gate and closing the beam tube shutter. As discussed in SAR Section 4.3, the divider gate is put in place using the overhead crane and two individuals, and as discussed above, UML stated that it could accomplish the closure of the reactor in the bulk pool within a few minutes, and that this action would not require facility staff to be located directly above the reactor pool (where they could potentially be exposed to direct radiation from the fuel or Co-60 sources in the pool, which could increase as the pool drained and shielding is reduced). A photo of the pool divider gate and reactor pool is provided in UML's response to RAI-13.2 (Ref. 23). The NRC notes that, as indicated in SAR Figure 8-3, the crane is not on emergency power; therefore, UML would be unable to use the crane to close the divider gate to mitigate a LOCA if offsite power were lost. Closing the beam tube shutter could similarly be accomplished while minimizing any potential UMLRR staff exposure to direct radiation from the fuel or Co-60 sources, but would also typically be done using the overhead crane. However, a severe LOCA occurring shortly following reactor shutdown, coincident with a loss of offsite power, is an extremely unlikely scenario. The NRC staff also notes that there is a possibility that UML may be unable to immediately close the shutter following the onset of a severe LOCA through a beam tube, depending on the water pressure in the tube; however, the NRC staff expects that UML should be able to close the shutter before the water level in the pool reaches the core, because the water pressure will decrease as the pool starts to drain. Additionally, as discussed in SER Section 5.3, the NRC staff notes that in a very unlikely scenario where the pool drained very rapidly, UMLRR staff could potentially need to evacuate the reactor building before mitigating actions could be completed. However, the NRC staff finds that, if necessary, UMLRR staff would likely be able to occupy (or re-occupy) the building for brief periods of time following the drainage to perform appropriate mitigating actions (e.g., actions needed to refill the pool) and/or actions needed for recovery operations, although as discussed below, the NRC staff finds that, even for a postulated worst-case beam tube LOCA without any mitigating actions, the SL would not be exceeded. Based on the above, the NRC staff finds that although one or both options may not be possible in every circumstance, UML has at least two different options for mitigating a severe LOCA (other options it may have include the use of the fire hose discussed in SER Section 5.3).

As discussed in SER Section 5.3, the NRC staff performed a calculation of a bounding LOCA scenario assuming that the pool divider gate is open and that the pool is able to drain unmitigated through an 8 inch diameter round hole. The NRC staff's calculation determined that even in this scenario, which involves a much more severe LOCA (shorter drainage time) than UML's analysis, the maximum core fuel temperature remains well below the SL. Therefore, based on the results of the NRC staff's calculation, the SL would not be exceeded for a LOCA resulting from drainage through a beam tube, regardless of whether TS 3.8(1) is applicable following reactor operation (i.e., whether or not UML is prohibited from opening a beam port while the corresponding beam port shutter is also open, once the reactor is shut down and moved to the bulk pool).

Regarding LOCA radiological consequences, the NRC staff notes that the initial direct and scattered radiation dose rate from the exposed core would be somewhat greater if the core becomes uncovered faster (i.e., if the pool is drained through an 8 inch beam tube rather than a 4 inch tube), but the drainage time (based on UML's estimate of 27 minutes for an 8-inch break) should still be long compared to the time needed for personnel to be moved off the reactor

bridge (above the pool), take mitigating actions, and/or evacuate from the reactor building. Mitigating actions such as closing the pool divider gate could likely be completed well before the water level dropped to the level of the core (or Co-60 sources, which are stored at approximately the same level as the core), even for an 8-inch break. However, even if the water dropped to a low level and much of the pool water shielding were lost, reactor staff completing mitigating actions could locate themselves to minimize any direct radiation from the core, Co-60 sources, or fuel in storage racks in the pool, and would primarily only be exposed to any scattered radiation. For radiological consequences due to Co-60 sources or fuel in storage racks becoming exposed following a LOCA, the time when the Co-60 sources or fuel in storage racks becomes exposed after reactor shutdown or the onset of the LOCA would not significantly affect the dose rate from the Co-60 sources or fuel in storage racks because they contain predominately longer-lived radionuclides compared to the fuel in the recently-shutdown core. Although UML would likely be able to perform mitigating actions during a LOCA based on UML's estimated drain time, the NRC staff expects that if the pool were able to drain more rapidly (e.g., the approximately 11 minutes estimated by the NRC staff's confirmatory calculation) such that UML may not be able to complete the actions before the pool drained, UML could still remove any personnel from the reactor bridge and evacuate them from the reactor building before they received an excessive dose. Therefore, the NRC staff finds that whether TS 3.8(1) is applicable during the 3,947 seconds following reactor operation would not be likely to significantly affect the radiological consequences of a LOCA.

Because, as discussed above, (1) a scenario involving severe damage to a beam tube that could cause significant pool drainage is unlikely; (2) UML's LOCA analysis is based on conservative assumptions including that the LOCA drainage begins immediately following infinite full-power reactor operation; (3) UML has options for mitigating a severe LOCA in progress; (4) the NRC staff's LOCA confirmatory calculation shows that even if the pool did drain through an 8 inch beam tube starting immediately following infinite full-power operation, core fuel damage would not occur; and, (5) even if the TS 3.8(1) beam port configuration restriction were applicable following reactor operation (i.e., when the reactor is not operating), the radiological consequences of a LOCA would not be likely to be significantly affected, the NRC staff finds that it is not necessary for the TS 3.8(1) configuration restriction to apply following reactor operation. The NRC staff also finds that allowing UML to have a beam port plug open while that beam port's shutter is also open (provided that the reactor is shutdown, and also moved to the bulk pool as required by TS 3.8(3)) provides appropriate operational flexibility for conducting reactor experiments. As also discussed above, the NRC staff also finds that TS 3.8(1), which is applicable during reactor operation, helps ensure that a LOCA during reactor operation could not cause the SL to be exceeded. Based on the above and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.8(1) is acceptable.

TS 3.8(2) would permit a beam port shield plug to be substituted or modified to be open to the reactor building as long as the total open cross-sectional area of the plug does not exceed the area of a round opening 4 inches in diameter. In its response to RAI-14.3.18 (Ref. 71), UML stated that allowing openings in beam port plugs provides flexibility for usage of the beam ports for education and research. UML stated that an open area equal to a 4-inch diameter round hole is less than the maximum 4.5-inch diameter round hole that would provide the minimum drain time (3,947 seconds) determined in UML's LOCA analysis. The NRC staff notes that some pool water could still drain through a plug with an opening, while the shutter was open, in the unlikely event a beam tube were sheared; however, as discussed in SER Section 5.3, if a leak through a beam port (including a leak through a plug opening) were occurring while the lead shutter were open, the shutter could be closed to mitigate the leak (among other possible

mitigating actions), significantly reducing the flow and providing further margin to UML's calculated maximum 4.5 inch diameter. Additionally, as also discussed in SER Section 5.3, the NRC staff performed an independent calculation of a UMLRR LOCA, which showed that even if a pool leak was able to occur through an 8 inch diameter hole (e.g., if a beam port plug were completely removed while the shutter was open and the beam tube sheared), the fuel temperature would still not exceed the SL. The NRC staff finds that TS 3.8(2) would allow UML flexibility in the use of beam port plugs with openings to allow for experiments (while still meeting TS 3.8(1) if the beam port shutter is open while such a plug is in use), while helping to continue to prevent any possibility of a LOCA where the pool is able to be drained through a hole or holes larger than the maximum diameter round hole determined in UML's analysis. Therefore, based on the above and also based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.8(2) is acceptable.

TS 3.8(3) would require that the reactor be positioned in the bulk pool if, for any beam port, both the shutter is open and the shield plug is removed. TS 3.8(3) would be applicable for any facility condition, but because TS 3.8(1) prohibits having both the shutter open and shield plug removed during reactor operation, UML could only open a beam port shutter and remove a shield plug when the reactor is shutdown. In its responses to RAI-13.6.d (Ref. 43) and RAI-14.3.18 (Ref. 71), UML stated that TS 3.8(3) requires that the reactor be in the bulk pool when a beam tube shutter and plug are both open because the response to a LOCA event could include placing the pool divider gate in place to prevent the bulk pool from draining, if the reactor were in the bulk pool and the drainage was through a beam tube (as illustrated in SER Figure 1-1 in SER Section 1.3, the beam tubes all extend into the stall pool). If the reactor is already in the bulk pool when a beam port lead shutter and shield plug are both opened, the reactor would not need to be moved and only the gate would need to be closed (if not already closed) if a leak through a beam port occurred in this situation.

Although TS 3.8(1) separately mitigates LOCA risk by prohibiting having both a beam port plug removed and a shutter open while the reactor is operating (i.e., when potential LOCA consequences are greatest), the NRC staff finds that TS 3.8(3) helps ensure that UML can quickly and easily mitigate the potential consequences of a LOCA when the reactor is shutdown, even though potential LOCA consequences are less significant at that time (as discussed in SER Section 5.3, UML determined that the core only needs to remain covered with water for 3,947 seconds following reactor shutdown to prevent the SL from being exceeded, and the NRC staff's confirmatory calculation determined that the core only needs to remain covered for an even shorter time, but there could be radiological consequences from the radiation emitted by the exposed core and Co-60 sources if a LOCA occurred at any time). The NRC staff notes that while TS 3.8(3) requires the reactor to be in the bulk pool to allow the pool divider gate to be readily installed to close the reactor in the bulk pool, it does not require that the gate be installed to open a shutter and beam port. The NRC staff finds that having the gate installed ahead of time could provide additional protection of the reactor core in the event of a LOCA through a beam tube (because UML would not need to install the gate after a leak starts), but does not need to be required for all situations where a shutter and beam port are open, because it may not be appropriate in all situations. For example, the NRC staff notes that if a leak occurred in the stall pool while the reactor was in the bulk pool and the gate was closed, the stall pool could drain more rapidly to expose any irradiated fuel that may be in the stall pool storage racks (both the bulk and stall pool may have fuel storage racks as illustrated in SAR Figure 9-3), decreasing the time available for personnel evacuation and/or mitigating actions to prevent radiological consequences of stored irradiated fuel becoming uncovered with water. Also, due to the pressure differential, the NRC staff notes that UML could be unable to open the gate, if necessary, if one side of the pool were leaking and the water levels of the sides were

different while the gate was closed. Therefore, the NRC staff finds that TS 3.8(3) gives UML appropriate flexibility to have the gate open or closed, as suitable for operational flexibility and ensuring safety. Based on the above and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.8(3) is acceptable.

TS 3.8(4) would require that when the reactor is in the stall pool and the pool divider gate is in position to separate the bulk and stall sections of the pool, all beam port shutters shall be in the closed (down) position. TS 3.8.(4) would be applicable for any facility condition, but because TS 3.3(4) would prohibit having the divider gate in position to separate the bulk and stall sections of the pool during reactor operation, UML could only have the reactor in the stall pool and pool divider gate in position to separate the bulk and stall sections of the pool when the reactor is shutdown. As discussed in SER Section 2.3, a beam tube being damaged while the reactor is closed in the stall pool is an unlikely event (and putting the pool divider gate in place while the reactor is in the stall pool is also a very infrequent evolution), but if it did occur, having the shutters closed would mitigate the leak, helping prevent the (shutdown) core from becoming potentially rapidly uncovered if the pool divider gate were in place. The NRC staff finds that by requiring all beam port shutters to be closed when the reactor is isolated in the stall pool, TS 3.8(4) helps prevent a rapid LOCA scenario (due to the possibility for fast drainage of the enclosed stall end of the pool which is the smaller portion of the pool) that could cause a recently-shutdown core to be uncovered quickly and in potentially less time than the 3,947 seconds that UML determined the core needs to remain covered with water following extended full-power reactor operation (see SER Sections 2.3 and 5.3). Therefore, based on the above and based on its 10 CFR 50.36(c)(2) findings for LCOs in SER Section 6.7, the NRC staff concludes that TS 3.8(4) is acceptable.

6.4 UMLRR Proposed Renewed TSs Section 4.0, "Surveillance Requirements"

Renewed TS 4.0, "Surveillance Requirements," would state:

> Applicability:
>
> This specification applies to the surveillance requirements of systems related to reactor safety.
>
> Objective:
>
> To verify the proper operation of systems related to reactor safety.
>
> Specification:
>
> A. Surveillance requirements may be deferred during reactor shutdown (except TS 4.1(8); 4.2.1(2); 4.3(1, 2, 3); 4.3(5); 4.4; 4.5; and 4.6); however, they shall be completed prior to reactor startup unless reactor operation is required for performance of the surveillance. Such surveillance shall be performed as soon as practicable after reactor startup. Scheduled surveillance, which cannot be performed with the reactor operating, may be deferred until a planned reactor shutdown.

B. The appropriate surveillance testing on any Limiting Condition for Operation required equipment shall be conducted after replacement, repair, or modification before the equipment is considered operable and returned to service.

Renewed TS 4.0, item A., would provide for the deferral of surveillance TS requirements (excluding TSs 4.1(8), 4.2.1(2), 4.3(1), 4.3(2), 4.3(3), 4.3(5), 4.4, 4.5, and 4.6) in Section 4 of the UMLRR TSs when the reactor is shutdown. TS 4.0, item A., also would provide for the deferral of surveillance TS requirements until the reactor is shutdown, if certain surveillances cannot be performed while the reactor is operating. In its basis for TS 4.0, UML stated that TS 4.0, item A., ensures that LCO requirements in Section 3 of the UMLRR TSs are met. The NRC staff finds that TS 4.0, item A., provides UML appropriate flexibility to defer surveillances that are unnecessary and/or cannot be performed (e.g., reactivity measurements or thermal power calibrations) when the reactor is not operating (such as during an extended shutdown condition), while helping ensure that such surveillances are performed before reactor operation resumes, or as soon as practicable once the reactor is restarted, as applicable. The NRC staff finds that TS 4.0, item A., helps ensure UMLRR facility safety during periods when the reactor is shut down by requiring that certain appropriate surveillances (e.g., fuel and control rod inspections, pool water chemistry, and surveillances related to confinement, ventilation, and radiation monitoring equipment) continue to occur at normal TS-required intervals during the shutdown. The NRC staff also finds that TS 4.0, item A., provides UML appropriate flexibility for extended reactor operations that may be necessary for experiments, etc., by allowing it to defer any TS-required scheduled surveillances in Section 4 of the UMLRR TSs that cannot be performed when the reactor is operating until a scheduled reactor shutdown. Additionally, the NRC staff finds that by specifying surveillance TSs that may be deferred during reactor shutdown, TS 4.0, item A., appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007. Therefore, based on the above and based on its 10 CFR 50.36(c)(3) findings for TSs on SRs in SER Section 6.7, the NRC staff concludes that TS 4.0, item A., is acceptable.

TS 4.0, item B., would require that when any LCO TS-required equipment is replaced, repaired, or modified, the appropriate corresponding TS-required surveillance for the equipment shall be performed before the equipment is considered operable and returned to service. In its basis for TS 4.0, UML stated that TS 4.0, item B., ensures that LCO TS-required equipment will continue to operate as intended and described in the SAR following any replacement, repair, or modification. The NRC staff finds that TS 4.0, item B., helps ensure that if equipment is replaced, repaired, or modified, the applicable TS-required surveillances are performed before the equipment is considered operable and returned to service (regardless of when the surveillances were last performed or when they are next due) to verify that the equipment continues to perform its intended function. (The NRC staff notes that this special surveillance may change the due date of the next regularly scheduled surveillance of a given type.) The NRC staff also finds that by specifying that surveillances be performed following replacement, repair, or modification of TS-required equipment, TS 4.0, item B., appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007. Therefore, based on the above and based on its 10 CFR 50.36(c)(3) findings for TSs on SRs in SER Section 6.7, the NRC staff concludes that TS 4.0, item B., is acceptable.

6.4.1 Proposed Renewed TS 4.1, "Reactor Core Parameters"

Renewed TS 4.1, "Reactor Core Parameters," would state, in part:

Applicability:

This specification applies to surveillance requirements for the various reactor core parameters.

Objective:

To ensure the reactor core parameters meet the specified limiting conditions for operation.

Specifications:

…

(5) Prior to the first reactor start-up of the day, visual verification shall be made that the reactor is not in the same end of the reactor pool as any portion of the cobalt-60 source.

(6) Prior to the first reactor start-up of the day, a visual verification shall be made confirming the beam ports meet the criteria of TSs 3.8(1) and 3.8(2).

(7) Prior to any beam port configuration change, a visual verification shall be made confirming TS 3.8(3) is met.

….

Renewed TSs 4.1(1) and 4.1(2) are discussed and found acceptable in SER Section 2.5.3.

Renewed TSs 4.1(3) and 4.1(4) are discussed and found acceptable in SER Section 2.2.

TS 4.1(5) would require that prior to the first reactor start-up of the day, UML visually verify that the reactor is not in the same end (i.e., bulk section or stall section) of the pool as any of the Co-60 sources. The NRC staff finds that TS 4.1(5) helps ensure that UML verifies that the location of the reactor and Co-60 sources meets the requirement of TS 3.1.1(6) before the reactor is operated, to help minimize the possibility of interference between the Co-60 sources and the reactor. Therefore, based on the above and based on its 10 CFR 50.36(c)(3) findings for TSs on SRs in SER Section 6.7, the NRC staff concludes that TS 4.1(5) is acceptable.

TS 4.1(6) would require that prior to the first reactor start-up of the day, UML visually verify that the beam ports meet the requirements of TSs 3.8(1) and 3.8(2). The NRC staff finds that TS 4.1(6) helps ensure that UML verifies that, before the reactor is operated, the beam port shutter and beam port shield plug configurations meet the requirements of TSs 3.8(1) and 3.8(2), such that the possibility and potential consequences of any LOCA are minimized. Therefore, based on the above and based on its 10 CFR 50.36(c)(3) findings for TSs on SRs in SER Section 6.7, the NRC staff concludes that TS 4.1(6) is acceptable.

TS 4.1(7) would require that prior to any beam port configuration change, UML visually verify that TS 3.8(3) is met. The NRC staff finds that TS 4.1(7) helps ensure that, before UML makes a beam port configuration change such as opening a beam port shutter or plug, it verifies that the reactor is in the bulk pool (if TS 3.8(3) requires the reactor to be in the bulk pool for the new configuration). The NRC staff finds that by requiring UML to verify that the reactor is in the bulk pool, such that UML could quickly close the reactor in the bulk pool using the pool dam if a pool leak through a stall pool beam port occurred, TS 4.1(7) helps ensure that the possibility and consequences of a potential LOCA are minimized. Therefore, based on the above and also based on its 10 CFR 50.36(c)(3) findings for TSs on SRs in SER Section 6.7, the NRC staff concludes that TS 4.1(7) is acceptable.

Renewed TS 4.1(8) is discussed and found acceptable in SER Section 2.2.1.

6.4.2 Proposed Renewed TS 4.2, "Reactor Control and Safety Systems"

Renewed TS 4.2.1, "Control Blades," and renewed TS 4.2.2, "Rod Reactivity Insertion Rate," are discussed and found acceptable in SER Section 2.2.2.

Renewed TS 4.2.3, "Reactor Protection System Scrams," specifications (1) through (3) and specifications (5) through (8), are discussed and found acceptable in SER Section 2.5.3.

Renewed TS 4.2.3, "Reactor Protection System Scrams," specification (4), is discussed and found acceptable in SER Section 2.5.1.

6.4.3 Proposed Renewed TS 4.3, "Coolant Systems"

Renewed TS 4.3, "Coolant Systems," is discussed and found acceptable in SER Section 2.3.

6.4.4 Proposed Renewed TS 4.4, "Confinement"

Renewed TS 4.4, "Confinement," would state:

Applicability:

This specification applies to the surveillance requirements for the reactor building confinement.

Objective:

To ensure the confinement limiting conditions for operation are met.

Specifications:

(1) Prior to any of the operations specified in 3.4.1 and at no less than 8 hour intervals during, the main intake fan shall be verified as operating.

(2) Prior to any of the operations specified in 3.4.1 and at no less than 8 hour intervals during, the building pressure compared to ambient shall be verified at or more negative than 0.1 inch water column.

(3) The ventilation isolation valves and bypass valve shall be verified as operable or in a fail-safe position semi-annually.

TS 4.4(1) would require that prior to, and at least every 8 hours during, any of the operations specified in TS 3.4.1 (i.e., conditions when confinement is required), UML shall verify that the main intake fan is operating. Renewed TS 3.5(1), which is discussed and found acceptable in SER Section 6.3.5, requires that the main intake fan be operating during any operations specified in TS 3.4.1. The NRC staff finds that TS 4.4(1) helps ensure that before and during operations when the intake fan is required by TS 3.5(1) to be operating, UML verifies that the fan is operating such that TS 3.5(1) is met and reactor operations are consistent with the assumptions of UML's MHA analysis (see SER Sections 5.1 and 6.3.5). The NRC staff notes that, as discussed in SER Section 6.3.5, when the reactor building is isolated, a valve (designated valve F in SAR Figure 6-2) opens to allow the air from the main intake fan to dilute air leaving the stack. Renewed TS 4.4(3), which is discussed and found acceptable below, separately provides for surveillance of this valve. The NRC staff also notes that, as discussed in SER Section 6.3.5, the main intake fan has a nominal flow rate of about 14,500 cfm, and fan operation at approximately this flow rate is part of the fan's performance of its intended dilution function as described in the SAR (however, as discussed in SER Section 5.1, the NRC staff's confirmatory calculations for the MHA show that even if this fan did not provide any dilution air at all, 10 CFR Part 20 dose limits would still not be exceeded). The NRC staff also finds that the surveillance interval in TS 4.4(1) is conservative compared to the NUREG-1537, Appendix 14.1, Section 4.5, recommended quarterly interval for verifying ventilation system blower operability. Therefore, based on the above and based on its 10 CFR 50.36(c)(3) findings for TSs on SRs in SER Section 6.7, the NRC staff concludes that TS 4.4(1) is acceptable.

TS 4.4(2) would require that prior to, and at least every 8 hours during, any of the operations specified in LCO TS 3.4.1 (i.e., conditions when confinement is required), UML shall verify that the reactor building pressure is negative relative to ambient by at least 0.1 inch water column. Renewed TS 3.5(2), which is discussed and found acceptable in SER Section 6.3.5, requires that building pressure be negative by at least 0.1 inch water column during any operations specified in TS 3.4.1. The NRC staff finds that TS 4.4(2) helps ensure that before and during operations when the building pressure is required to be negative, UML verifies that the minimum negative pressure differential specified in TS 3.5(2) is met and that any leakage of air from the reactor building is minimized. The NRC staff notes that renewed TS 3.4.2(1), which is discussed and found acceptable in SER Section 6.3.4, and which requires airlock doors and the truck door to be sealed during operations specified in TS 3.4.1, does not have a specific corresponding surveillance TS; however, because sealing airlock doors and the truck door facilitates UML's maintenance of the reactor building at a negative differential pressure, the NRC staff finds that TS 4.4(2) would provide surveillance for both TS 3.4.2(1) and TS 3.5(2) (the NRC staff notes that the UMLRR also has TS-required scrams that would shut down the reactor if an airlock or truck door gasket deflated during reactor operation). Based on the above and based on its 10 CFR 50.36(c)(3) findings for TSs on SRs in SER Section 6.7, the NRC staff concludes that TS 4.4(2) is acceptable.

TS 4.4(3) would require that the reactor building confinement/ventilation system isolation valves, and the bypass valve, be verified operable (or in their fail-safe positions) semi-annually. The operation of the confinement system is discussed in SAR Section 6.2.3, as supplemented by UML's response to RAI-6.1 (Ref. 23), and in SER Sections 6.3.4 and 6.3.5. When the reactor building is isolated (i.e., when a GRVS occurs, as discussed in SER Section 6.3.4), the confinement system causes isolation values (designated A, B, C, E, G, and H in SAR Figure 6-2) to close, a bypass valve (designated F in SAR Figure 6-2) to open to allow dilution

air from the main intake fan to travel up the stack, and the main exhaust fan and experimental facilities blowers to cease to operate. In its response to RAI-14.4.17 (Ref. 71), UML stated that it considers a semi-annual frequency of confinement system testing to be adequate given the fail-safe design of the ventilation isolation valves and given that semi-annual functional tests of the confinement (formerly containment) isolation initiation system and valves done since UMLRR operation began in 1974 have shown no instance in which the initiation system failed to function or an isolation valve failed to close. UML also stated that frequent testing of the pneumatically pressurized and spring-loaded fail-safe valves creates substantial wear on the valve components and, therefore, excessive testing of the valves is undesirable. Renewed TS 3.4.2(2), which is discussed and found acceptable in SER Section 6.3.4, permits the ventilation isolation valves and bypass valve to be inoperable if they are in the fail-safe position. Therefore, TS 4.4(3) does not require testing of a valve if UML knows that the valve is inoperable; verification and documentation that the valve is in the fail-safe position is sufficient to satisfy TS 4.4(3) for the inoperable valve.

The NRC staff finds that TS 4.4(3) helps ensure that UML periodically verifies that the UMLRR confinement isolation and bypass valves are capable of performing their intended function, including that the confinement initiation system is operable and that the confinement system has the ability to rapidly close appropriate isolation valves and open the main intake fan dilution air valve (valve F) following a GRVS. The NRC staff finds that TS 4.4(3) helps ensure that TS 3.4.2(2), which requires isolation valves A, B, C, E, G, and H to be operable or closed and bypass valve F to be operable or open, is met, such that the ventilation and confinement systems can function as designed during potential accident conditions. The NRC staff notes that the UMLRR TS for containment building isolation system surveillance prior to this renewal included a requirement that the surveillance test verify that isolation valves closed within 2.5 seconds after the initial signal to close. The NRC staff finds that while an appropriate closure time of the valves is important to the valves performing their intended function, a TS specifying a certain closure time is unnecessary because the TS definitions require that operable equipment be fully capable of performing its intended function(s). The NRC staff notes that the guidance in NUREG-1537, Appendix 14.1, Section 4.4.2, and ANSI/ANS-15.1-2007, Section 4.4.2, recommends quarterly testing of confinement system operability, which is more frequent than the semi-annual testing specified in TS 4.4(3); however, the NRC staff finds that UML's interval is reasonable and appropriate given the past successful operation of the system and UML's desire to minimize unnecessary wear on the system. The NRC staff also finds that by requiring functional testing of the confinement isolation system, TS 4.4(3) appropriately implements the intent of the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007. Based on the information above and based on its 10 CFR 50.36(c)(3) findings for TSs on SRs in SER Section 6.7, the NRC staff concludes that TS 4.4(3) is acceptable.

6.4.5 Proposed Renewed TS 4.5, "Ventilation Systems"

Renewed TS 4.5, "Ventilation Systems," would state:

> Applicability:
>
> This specification applies to the surveillance requirements for the confinement building emergency exhaust system.
>
> Objective:
>
> To ensure the emergency exhaust system limiting conditions for operation are met.

Specifications:

(1) An operability check of the emergency exhaust system shall be performed quarterly.

(2) The carbon filter efficiency in the emergency exhaust system shall be tested biennially.

TS 4.5(1) would require that UML verify the operability of the emergency exhaust system quarterly. Renewed TS 3.5(3), which is discussed and found acceptable in SER Section 6.3.5, requires that the emergency exhaust system be operable during any operations specified in TS 3.4.1 (i.e., conditions when confinement is required). SER Sections 5.1, 5.7, 6.3.4, and 6.3.5 discuss the operation of the emergency exhaust system. The NRC staff notes that, as discussed in SER Section 6.3.5, the emergency exhaust system has a nominal flow rate of about 320 cfm and that system operation at approximately this flow rate is part of the system's performance of its intended function as described in the SAR (however, the NRC staff also notes that as discussed in SER Section 5.1, UML's calculations for the MHA show that even if the emergency exhaust system failed to function at all, 10 CFR Part 20 dose limits would still not be exceeded). As discussed in SER Sections 5.1 and 6.3.4, the emergency exhaust system routes air through HEPA and charcoal (carbon) filters, but the NRC staff notes that TS 4.5(1) does not include measurement of specific efficiencies for these filters. Renewed TS 4.5(2), which is discussed and found acceptable below, separately requires biennial efficiency testing of the charcoal filter; the NRC staff notes that although the HEPA filter with a minimum efficiency of 99.9 percent for removal of 0.3 micron particles is part of the emergency exhaust system as described in the SAR, there is no TS to specify the efficiency of the HEPA filter. The NRC staff finds that TS 4.5(1) helps ensure that UML verifies that the UMLRR emergency exhaust system, including its initiating system, is operable as required by TS 3.5(3), such that reactor operations are consistent with the assumptions of the MHA. The NRC staff also finds that TS 4.5(1) appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 because it requires functional testing of the TS-required emergency exhaust system, and because the specific surveillance interval in TS 4.5(1) is consistent with the quarterly interval specified by guidance in NUREG-1537, Appendix 14.1, Section 4.5, and ANSI/ANS-15.1-2007, Section 4.5, item (1). Therefore, based on the above and based on its 10 CFR 50.36(c)(3) findings for TSs on SRs in SER Section 6.7, the NRC staff concludes that TS 4.5(1) is acceptable.

TS 4.5(2) would require that UML test the charcoal (carbon) filter in the emergency exhaust system biennially to verify that its efficiency meets TS 3.5(4), which is discussed and found acceptable in SER Section 6.3.5. In its response to RAI-14.4.19 (Ref. 71), UML stated that TS 4.5(2) only requires efficiency testing of the charcoal filter in the emergency exhaust system, i.e., it does not require efficiency testing of any other facility filters, such as the emergency exhaust system HEPA filter or experimental facility filters, and indicated that these other filters are not credited in any SAR analyses. During the 2020-2021 regulatory audit, as documented in the NRC staff's audit report dated December 17, 2020 (Ref. 86), UML stated that for charcoal filter surveillances, it performs an in-place test of the filter with stable elemental iodine. As discussed in SER Section 6.3.5, it is not clear whether UML credited the emergency exhaust system charcoal filter in its MHA analyses, but the NRC staff finds that the effect of crediting this filter would be small. Additionally, as discussed in SER Section 5.1, the NRC staff performed an MHA confirmatory calculation which demonstrated that if the charcoal filter is not credited, the MHA doses remain below 10 CFR Part 20 limits. The NRC staff finds that although the

dose-mitigating effect of the charcoal filter is small for the MHA, TS 4.5(2) helps ensure that UML verifies that the efficiency of the charcoal filter meets the minimum value required by TS 3.5(4), such that iodine released from the building and any dose to the public following a potential accident are minimized. ANSI/ANS-15.1-2007, Section 4.5, item (2), recommends that TSs require that filter efficiency measurements be performed annual to biennially, or following major maintenance. NUREG-1537, Appendix 14.1, Section 4.5, recommends that TSs require that "[t]he function and efficiency of filters … be tested annually or in accordance with manufacturer's recommendations and acceptance criteria and following repair or maintenance…." The NRC staff also finds that, considering the recommendations in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007, as well as the likely relative importance of the charcoal filters during an accident, and the likely frequency of use of the filters (the emergency exhaust system is not operating during normal routine reactor operation), the surveillance interval required by TS 4.5(2) (in conjunction with the requirements of TS 4.0, item B.) is reasonable and appropriate. Therefore, based on the above and based on its 10 CFR 50.36(c)(3) findings for TSs on SRs in SER Section 6.7, the NRC staff concludes that TS 4.5(2) is acceptable.

6.4.6 Proposed Renewed TS 4.6, "Radiation Monitoring Equipment"

Renewed TS 4.6, "Radiation Monitoring Equipment," specifications (1) through (3), are discussed and found acceptable in SER Section 4.1.4.

Renewed TS 4.6, "Radiation Monitoring Equipment," specification (4), is discussed and found acceptable in SER Section 4.1.7.

6.5 UMLRR Proposed Renewed TSs Section 5.0, "Design features"

6.5.1 Proposed Renewed TS 5.1, "Site and Facility Description"

Renewed TS 5.1, "Site and Facility Description," would state:

Applicability:

These specifications apply to the physical location of the reactor and supporting structures.

Objective:

To specify the bounds of the facility.

Specifications:

(1) The reactor and associated equipment shall be located at 1 University Avenue, Lowell, Massachusetts.

(2) The facility shall be the area under the reactor license. It shall include the reactor building, designed for confinement, and the attached three-story building. The reactor building shall be the minimum restricted area as defined in 10 CFR Part 20. The reactor building shall have a minimum free volume of 335,000 ft^3 that is exhausted through a 100 ft. high stack. The three-story building attached to the reactor building shall include spaces necessary for supporting licensed activities

including radiation protection, emergency preparedness, physical security, and the reactor building ventilation.

TSs 5.1(1) and 5.1(2) would provide a description of the UMLRR facility, including important features of the facility physical design. TS 5.1(1) would specify the location of the UMLRR. TS 5.1(2) would specify the area under the reactor license as the facility, which includes both the reactor building and the attached Pinanski building (see also "research reactor facility" in the TS definitions), and would specify the minimum restricted area as the reactor building. TS 5.1(2) would also specify a minimum reactor building free volume of 335,000 cubic feet, and the height of the UMLRR stack at 100 feet, and would specify that the Pinanksi building attached to the reactor building includes spaces and equipment supporting activities conducted under the reactor license. The NRC staff finds that TSs 5.1(1) and 5.1(2) appropriately describe important design features and characteristics of the UMLRR facility and are consistent with information in the SAR, as supplemented, and the UMLRR TS definitions. The NRC staff finds that designating the reactor building and the attached Pinanski building as the licensed boundary is appropriate because the boundary encompasses not only the area where the primary UMLRR activities such as reactor operation and irradiations occur, but also any areas that support these activities and in which UML may consequently possess and use radioactive materials under the reactor license. (The NRC staff notes that UML is also authorized to possess certain radioactive materials within this boundary under UML's broad scope radioactive material license issued by the Commonwealth of Massachusetts (License No. 60-0049), for example, radioactive material in the liquid waste tanks, as discussed in SER Section 4.1.1.2). The NRC staff finds that TS 5.1(2) describes an appropriate minimum restricted area, while giving UML the flexibility to expand the restricted area to other parts of the licensed area (i.e., Pinanski building areas) if necessary. The NRC staff finds that the reactor building minimum free volume and stack height in TS 5.1(2) are consistent with the assumptions UML used in its Ar-41 and MHA analyses. The NRC staff also finds that TSs 5.1(1) and 5.1(2) appropriately implement the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 because the TSs provide a general description of the UMLRR site and facility. Therefore, based on the above and based on its 10 CFR 50.36(c)(4) findings for TSs on design features in SER Section 6.7, the NRC staff concludes that TSs 5.1(1) and 5.1(2) are acceptable.

6.5.2 Proposed Renewed TS 5.2, "Reactor Coolant System"

Renewed TS 5.2, "Reactor Coolant System," is discussed and found acceptable in SER Section 2.3.

6.5.3 Proposed Renewed TS 5.3, "Reactor Core and Fuel"

Renewed TS 5.3, "Reactor Core and Fuel," specifications (1) through (6), are discussed and found acceptable in SER Section 2.2.

Renewed TS 5.3, "Reactor Core and Fuel," specification (7), is discussed and found acceptable in SER Section 2.5.1.

6.5.4 Proposed Renewed TS 5.4, "Fissionable Material Storage"

Renewed TS 5.4, "Fissionable Material Storage," is discussed and found acceptable in SER Section 2.7.

6.6 UMLRR Proposed Renewed TSs Section 6.0, "Administrative Controls"

6.6.1 Proposed Renewed TS 6.1, "Organization"

Renewed TS 6.1.1, "Structure," would state:

> The organization for the management and operation of the research reactor facility in matters related to the license and these technical specifications shall be as shown in Figure 6-1.

TS 6.1.1 would require that the structure of the UMLRR operating organization be as shown in TS Figure 6-1, which is reproduced below as SER Figure 6-1. As indicated by the TS Figure 6-1 caption, solid lines denote reporting lines, and dashed lines denote communication lines. The NRC staff finds that TS 6.1.1 helps ensure that the UMLRR organizational structure, including the communication and reporting lines, is appropriately defined and properly delineated in the TSs. The NRC staff finds that UML's designation of two individuals as Level 2 authorities is appropriate, because both individuals have management authority over UMLRR operation, and because TS 6.1.1 (in conjunction with TS 6.1.2) clearly designates which Level 2 has the higher authority. The NRC staff also finds that by delineating the UMLRR organizational structure, TS 6.1.1 appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007. Therefore, based on the above and based on its 10 CFR 50.36(c)(5) findings for TSs on administrative controls in SER Section 6.7, the NRC staff concludes that TS 6.1.1, including TS Figure 6-1, is acceptable.

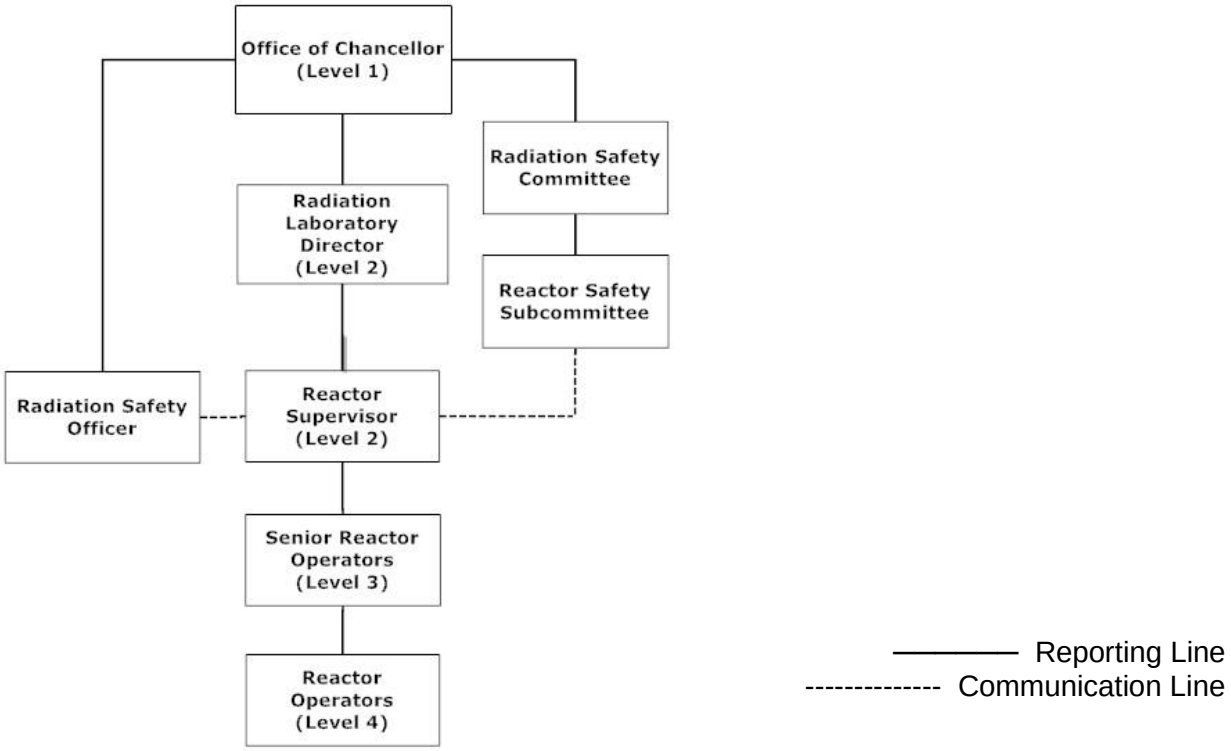

Figure 6-1 UMLRR TS Figure 6-1

Renewed TS 6.1.2, “Responsibility,” would state:

(1) The Chancellor shall designate an individual (Level 1), at a position of associate vice chancellor or higher, to be responsible for the reactor license.

(2) The Reactor Supervisor (Level 2) shall be directly responsible for the safety of all operations at the research reactor facility, and in all matters pertaining to these Technical Specifications.

(3) In all matters pertaining to safe operation of the reactor facility and to these Technical Specifications, the Reactor Supervisor shall report to and be directly responsible to the Director of the Radiation Laboratory (Level 2).

(4) The UML Radiation Safety Officer shall be responsible for radiation protection at the UMLRR and shall advise the Reactor Supervisor on all matters pertaining to radiation protection.

(5) In matters pertaining to radiation safety, the UML Radiation Safety Officer shall report to and be directly responsible to the Level 1 individual in the Office of the Chancellor.

TS 6.1.2 would require that the Level 1 authority indicated in TS Figure 6-1 be an individual at a position of associate vice chancellor or higher, who has overall responsibility for the UMLRR license, and is responsible for ensuring that UMLRR activities are in accordance with NRC regulations. TS 6.1.2 would also require that the Reactor Supervisor (lower Level 2 authority indicated in TS Figure 6-1) report to, and have direct responsibility to, the Director of the Radiation Laboratory (upper Level 2 authority indicated in TS Figure 6-1) for the safety of UMLRR activities and compliance with the UMLRR TSs; per TS Figure 6-1, the Director of the Radiation Laboratory in turn reports to the Level 1. TS 6.1.2 would additionally require that the UML campus radiation safety officer (RSO) (see SER Section 4.1.2) be responsible for radiation protection at the UMLRR, advise the Reactor Supervisor on all matters related to radiation protection, and report to and be directly responsible to the Level 1 in matters pertaining to radiation safety. The NRC staff finds that TS 6.1.2 helps ensure that the responsibilities of key positions in the UMLRR organizational structure, including responsibilities for compliance with the UMLRR license and NRC regulations, are clearly delineated. The NRC staff finds that TS 6.1.2, in conjunction with TS 6.1.1, helps ensure that the Level 1 authority is an upper-level UML official who has ultimate responsibility for the UMLRR license and who can make UML resources available as needed to ensure facility safety. Regarding TS 6.1.2, and other TSs in Section 6.0, “Administrative Controls,” the NRC staff notes that it does not consider any personnel responsibilities, review requirements, procedure requirements, etc., delineated in the TSs to imply applicability for any purpose other than ensuring nuclear and radiological safety and that NRC requirements are met, with respect to activities conducted under NRC Facility Operating License No. R-125 for the UMLRR. The NRC staff also finds that by clearly defining the responsibilities to UML personnel as they relate to the UMLRR, TS 6.1.2 appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007. Therefore, based on the above and based on its 10 CFR 50.36(c)(5) findings for TSs on administrative controls in SER Section 6.7, the NRC staff concludes that TS 6.1.2 is acceptable.

Renewed TS 6.1.3, “Staffing,” would state:

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

a. A reactor operator or senior reactor operator shall be in the control room.

b. A second designated person shall be present at the facility. This individual shall be a senior reactor operator, reactor operator or an individual able to carry out prescribed written instructions.

c. If a senior reactor operator is not at the facility, a senior reactor operator shall be readily available on call. “Readily available on call” shall mean an individual who:

1. has been specifically designated and the designation known to the operator on duty,

2. keeps the operator on duty informed of where he/she may be rapidly contacted and the phone number, and

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

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

a. management personnel,

b. radiation safety personnel, and

c. other operations personnel

(3) The following events shall require the presence of a senior reactor operator at the facility:

a. initial startup and approach to power.

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

c. recovery from an unplanned or unscheduled shutdown or power reduction of 200kW or greater.

TS 6.1.3(1) would require that when the reactor is not secured, there shall be an NRC-licensed reactor operator (RO) or senior reactor operator (SRO) in the control room, and a second person capable of carrying out prescribed written instructions must be present at the facility. Additionally, if an SRO is not present at the facility, an SRO must be readily available on call (specifically designated and the designation known to the operator on duty, rapidly available to the operator by phone or other communication method, and capable of getting to the facility within a reasonable time under normal conditions).

TS 6.1.3(1) would not require any specific facility staffing when gamma irradiation facilities are in use, if the reactor is secured. However, in its letter dated September 30, 2020 (Ref. 98), UML stated that the gamma facilities are operated by individuals who are trained in UMLRR gamma facility operation and have unescorted access to the UMLRR facility. These individuals are present for the setup and termination of gamma irradiations (including when gamma sources are being moved into or out of place for irradiations), although ongoing static irradiations may continue during periods when there is no one present at the UMLRR facility (e.g., overnight). The NRC staff notes that, based on information in Attachments 2 and 3 of the NRC safety evaluation related to the gamma irradiation facilities, dated February 5, 1998 (Ref. 70), moving gamma sources for irradiations typically involves at least two individuals (one to move the source and one to supervise/observe and assist with the source movement and time the irradiation). Per renewed TS 6.4(1), item g., which is discussed and found acceptable in SER Section 6.6.4, UML is required to have and use written procedures (which have been reviewed by the Reactor Safety Subcommittee (RSSC) and approved by the Reactor Supervisor) for operation of the gamma irradiation facilities. By letter dated September 30, 2020 (Ref. 98), UML also stated that its procedures specifically require two individuals to be present when Co-60 sources are being moved. As discussed in SER Section 4.1.5, the UMLRR also has multiple engineered controls to control any personnel exposure from gamma irradiation facilities.

In its letter dated September 30, 2020 (Ref. 98), UML stated that although there would not necessarily be anyone in the control room to monitor the area radiation monitoring system (ARMS) when gamma irradiation facilities are in use (including setup and termination of irradiations), the gamma irradiation facilities have multiple fixed radiation monitors with local alarms that can alert personnel to radiation hazards, including monitors that are both connected to, and independent of, the ARMS (renewed TS 3.6.1(2), which is discussed and found acceptable in SER Section 4.1.4, also specifically requires local area radiation monitors that will alert personnel when a gamma irradiation source is in use). Portable monitors are also used when personnel enter irradiation facilities (per procedures) to supplement fixed monitors.

The NRC staff finds that TS 6.1.3(1) helps ensure appropriate staffing and availability of personnel at the UMLRR and helps ensure that UML complies with the regulation in 10 CFR 50.54(k), which states that “[a]n operator or senior operator licensed pursuant to [10 CFR Part 55] shall be present at the controls at all times during the operation of the facility.” The NRC staff finds that because UML has gamma irradiation facility operators who are trained in the use of those facilities present for the setup and termination of gamma irradiations; because setup and termination of gamma irradiations requires manual actions for which individuals must be physically present (e.g., moving sources into or out of position); because UMLRR radiation monitoring equipment can alert users or any other personnel in the vicinity of a gamma irradiation facility of irradiation facility radiation hazards, regardless of whether there is anyone in the control room; and because use of the gamma irradiation facilities does not directly affect the reactor or change reactivity, no RO or SRO is required for gamma irradiation facility operations and, therefore, it is not necessary for TS 6.1.3(1) to require specific facility staffing when gamma irradiation facilities are in use while the reactor is secured. The NRC staff also finds that because the gamma irradiation facilities and sources are in a static condition during ongoing irradiations, UML does not necessarily need to have anyone present at the UMLRR facility during ongoing gamma irradiations (except for during the setup and termination of the irradiations as discussed above). The NRC staff finds that by defining minimum facility staffing when the reactor is not secured, TS 6.1.3(1) appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007. Therefore, based on the above and based on its 10 CFR 50.36(c)(5) findings for TSs on administrative controls in SER Section 6.7, the NRC staff concludes that TS 6.1.3(1) is acceptable.

TS 6.1.3(2) would require that a contact list with names and phone numbers of management personnel, radiation safety personnel, and other operations personnel be available in the control room for use by the reactor operator. The NRC staff finds that TS 6.1.3(2) helps ensure that the reactor operator can contact appropriate UMLRR personnel if necessary and that TS 6.1.3(2) appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 by requiring posting of important names and contact information. Therefore, based on the above and based on its 10 CFR 50.36(c)(5) findings for TSs on administrative controls in SER Section 6.7, the NRC staff concludes that TS 6.1.3(2) is acceptable.

TS 6.1.3(3) would require that an SRO be present at the UMLRR facility during initial startup and approach to power; all fuel or control blade relocations within the reactor core region; and recovery from an unplanned or unscheduled shutdown or power reduction of 200 kilowatts-thermal or greater. The NRC staff finds that TS 6.1.3(3) helps ensure that an SRO is present at the facility at appropriate times to provide supervision. The NRC staff notes that TS 6.1.3(3) does not specifically require that an SRO be present during relocation of any experiment or the regulating rod, but finds that this is acceptable because TSs 3.2.2(2) and 3.7.1 limit any reactivity change associated with a single experiment or regulating rod relocation to 0.5% $\Delta k/k$ (less than $1.00) and, therefore, the relocation of any experiment or the regulating rod could not result in a significant reactivity change (e.g., a prompt excursion). The NRC staff also finds that TS 6.1.3(3) helps ensure that UML complies with the regulation in 10 CFR 50.54(m)(1), which states that "[a] senior [reactor] operator licensed pursuant to [10 CFR Part 55] 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 start-up and approach to power, recovery from an unplanned or unscheduled shut-down or significant reduction in power, and refueling, or as otherwise prescribed in the facility license." The NRC staff also finds that by defining when a senior reactor operator must be present at the facility, TS 6.1.3(3) appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007. Therefore, based on the above and based on its 10 CFR 50.36(c)(5) findings for TSs on administrative controls in SER Section 6.7, the NRC staff concludes that TS 6.1.3(3) is acceptable.

Renewed TS 6.1.4, "Selection and Training of Personnel," would state:

(1) The Director of the Radiation Laboratory shall be a tenured faculty member in a science or engineering discipline.

(2) The selection, training, and requalification of operations personnel should meet or exceed the requirements (most current revision) of American National Standard, ANSI/ANS-15.4 "Selection and Training of Personnel for Research Reactors." (R2016 or later revision).

TS 6.1.4(2) would recommend that selection, training, and requalification of UMLRR operations personnel meet or exceed the guidance in ANSI/ANS-15.4-2016, "Selection and Training of Personnel for Research Reactors" (Ref. 20), or a more current revision of this guidance. TS 6.1.4(1) would additionally require that the Director of the Radiation Laboratory (Level 2 authority) be a tenured UML faculty member in a science or engineering discipline. The NRC staff finds that TS 6.1.4(2) helps ensure that the selection and training of UMLRR operators and other personnel is accomplished using appropriate guidance. As discussed in SER Section 1.1.1, as part of its renewal review, the NRC staff reviewed the UMLRR operator requalification program (ORP), Revision 3, dated November 2016, and concluded that it is consistent with the guidance in ANSI/ANS-15.4-2016 and the applicable regulations in 10 CFR Part 55, and is

acceptable for use at the UMLRR. The NRC staff finds that the use of "should" to denote a recommendation in TS 6.1.4(2) is appropriate because it gives UML flexibility to apply guidance in ANSI/ANS-15.4-2016 using facility-specific considerations, and to use guidance in future revisions of ANSI/ANS-15.4-2016 (within the constraints of the NRC-approved UMLRR ORP). The NRC staff finds that the TS 6.1.4(1) requirement exceeds the guidance in ANSI/ANS-15.4-2016 because it specifies a specific position, typically requiring significant science and/or engineering training and experience, that the Director of the Radiation Laboratory is required to hold, which is in addition to the general ANSI/ANS-15.4-2016 recommendations for Level 2 qualifications. Additionally, the NRC staff finds that by specifying requirements and recommendations related to selection and training of personnel and referencing ANSI/ANS-15.4-2016, TS 6.1.4 appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007. Therefore, based on the above and based on its 10 CFR 50.36(c)(5) findings for TSs on administrative controls in SER Section 6.7, the NRC staff concludes that TSs 6.1.4(1) and 6.1.4(2) are acceptable.

6.6.2 Proposed Renewed TS 6.2, "Review and Audit"

The introductory text of renewed TS 6.2, "Review and Audit," would state:

> There shall be a Reactor Safety Subcommittee (RSSC) which shall review reactor facility operations to ensure that the facility is operated in a manner consistent with public safety and within the terms of the facility license. The RSSC shall be a subcommittee of the University Radiation Safety Committee which has overall authority in the use of all radiation sources at the University.

Renewed TS 6.2.1, "Composition and Qualifications," would state:

> The RSSC shall be composed of at least five members, one of whom shall be the Radiation Safety Officer and another of whom shall be the Reactor Supervisor. Members of the RSSC shall be knowledgeable in the areas of reactor operation and radiation safety. The membership of the RSSC shall include at least two faculty members from the engineering or science disciplines. Members shall be appointed by the Office of the Chancellor Level 1 designee. The RSSC chairman shall be elected from among the membership and shall be outside the reactor facility operating staff or Level 1.

The introductory text of TS 6.2 would requires UML to have an RSSC (which is a subcommittee of the overall Radiation Safety Committee (RSC) for the entire UML campus), and would require that the RSSC be responsible for oversight functions to ensure that the UMLRR is operated safety and in accordance with NRC Facility Operating License No. R-125. TS 6.2.1 would require that the RSSC include at least five members, including the RSO and Reactor Supervisor, and at least two UML engineering or science faculty members, and that the members be knowledgeable in reactor operation and radiation safety, and appointed by the Level 1 authority. TS 6.2.1 also would require that the RSSC chairman be elected from among the RSSC membership, and not be the Level 1 or be from the UMLRR operating staff. The NRC staff notes that "operating staff," as used in ANSI/ANS-15.1-2007, Section 6.2.2, item (2), is considered to be "the Level 2 facility director or administrator and anyone who reports to that person" and, therefore, the NRC staff considers "reactor facility operating staff" in TS 6.2.1 to include the UMLRR Level 2 individuals (the Radiation Laboratory Director and the Reactor Supervisor, who reports to the Director) and anyone who reports to them, consistent with ANSI/ANS-15.1-2007, Section 6.2.2, item (2). The NRC staff finds that the introductory text of

TS 6.2, and TS 6.2.1, help ensure that UML has a designated committee (the RSSC) for oversight functions for UMLRR operations, that the members of the committee are appropriately qualified, and that the committee is appropriately independent from UMLRR operations in that its chairman is not part of the UMLRR operating organization and is also someone other than the individual with overall responsibility for the UMLRR license. The NRC staff notes that renewed TS 6.2.2 also helps ensure independence of the RSSC, as discussed below. The NRC staff also finds that by establishing an oversight method and establishing requirements for the committee responsible for oversight functions, the introductory text of TS 6.2, and TS 6.2.1, appropriately implement the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007. Therefore, based on the above and based on its 10 CFR 50.36(c)(5) findings for TSs on administrative controls in SER Section 6.7, the NRC staff concludes that the introductory text of TS 6.2 and TS 6.2.1 are acceptable.

Renewed TS 6.2.2, "Charter and Rules," would state:

> The RSSC shall follow the rules specific to it under the charter and rules of the Radiation Safety Committee. Notwithstanding that charter and rules, the RSSC functions shall be conducted as follows:
>
> (1) Meetings shall be held at least once per calendar year and more frequently as circumstances warrant, consistent with effective monitoring of facility activities.
>
> (2) A meeting quorum shall consist of at least one-half of the membership where the operating staff does not constitute a majority.
>
> (3) Meeting minutes shall be distributed to RSSC members within three months of the meeting.

TS 6.2.2 would require that the RSSC follow rules specific to it under the charter and rules of UML's overall RSC, but notwithstanding the overall RSC charter and rules, the RSSC must hold meetings at least every calendar year; require at least one-half its membership be present (where the UMLRR operating staff do not constitute a majority of those present) to achieve a quorum; and distribute meeting minutes to its members within 3 months following each meeting. In addition to the quorum requirement in TS 6.2.2(2), SAR Section 12.2.2 states that a quorum of the RSSC must also include the RSO or designee and the RSSC chairman or designee. The NRC staff notes that it considers "operating staff" as used in TS 6.2.2(2) to include the UMLRR Level 2 individuals and anyone who reports to them, consistent with "operating staff" as used in ANSI/ANS-15.1-2007, Section 6.2.2, item (2). The NRC staff finds that TS 6.2.2 helps ensure that the RSSC functions are appropriately conducted in accordance with an established charter and that the RSSC follows appropriate rules related to meeting frequency, quorum, and timely distribution of meeting minutes. The NRC staff also finds that the TS 6.2.2(2) requirement that UMLRR operating staff not be a majority in order to achieve an RSSC quorum helps ensure the independence of the RSSC. Additionally, the NRC staff finds that by establishing rules for the RSSC and requiring the RSSC to operate in accordance with its charter, TS 6.2.2 appropriately implements the guidance of NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007. Therefore, based on the above and based on its 10 CFR 50.36(c)(5) findings for TSs on administrative controls in SER Section 6.7, the NRC staff concludes that TS 6.2.2 is acceptable.

Renewed TS 6.2.3, “Review Function,” would state:

(1) The RSSC shall review the following:

a. Evaluations performed as required by 10 CFR 50.59.

b. All new procedures and major revisions thereto having safety significance and proposed changes in reactor facility equipment or systems having safety significance.

c. All new experiments or classes of experiments.

d. Proposed changes in the technical specifications or license.

e. Violations of technical specifications or license, and violations of internal procedures having safety significance.

f. Operating abnormalities having safety significance.

g. Reportable occurrences listed in TS 6.6.2.

h. Audit reports.

(2) A written report or minutes of the findings and recommendations of the RSSC shall be submitted to the Level 1 individual in the Office of the Chancellor and to the RSSC members within three months after a review has been completed.

TS 6.2.3 would list items that are required to be reviewed by the RSSC and also would require that written reports or minutes of RSSC reviews be submitted to the RSSC members and the Level 1 within 3 months following completion of a review. The NRC staff finds that TS 6.2.3 helps ensure that the scope of RSSC reviews is appropriate and includes items that could relate to UMLRR facility safety, and that RSSC review findings are reported in a timely manner. The NRC staff also finds that by defining the RSSC’s review function, TS 6.2.3 appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007. Therefore, based on the above and based on its 10 CFR 50.36(c)(5) findings for TSs on administrative controls in SER Section 6.7, the NRC staff concludes that TS 6.2.3 is acceptable.

Renewed TS 6.2.4, “Audit Function,” would state:

(1) Audits of the following functions shall be performed by an individual or group without immediate responsibility for the area being audited.

(2) The scope of the audits shall include, as a minimum, the following:

a. Facility operations for conformance to the technical specifications and license conditions on an annual basis.

b. The requalification program for the operating staff on a biennial basis.

c. Corrective actions associated with deficiencies in the reactor facility equipment, systems, structures, or methods of operation that affect reactor facility safety on an annual basis.

d. The reactor facility emergency plan and implementing procedures on a biennial basis.

(3) Deficiencies uncovered that affect reactor facility safety shall immediately be reported to the Chancellor's Level 1 designee. A written report of the findings of the audit shall be submitted to the Chancellor's Level 1 designee and to all RSSC members within three months after the audit has been completed.

TS 6.2.4 would impose requirements for audits, including that audits be performed by an individual or group without immediate responsibility for the area being audited; that audits be performed (annually or biennially, as specified) of facility operations for conformance with the UMLRR TSs and license, of the ORP, of corrective actions, and of the emergency plan and its implementing procedures; that deficiencies uncovered by audits that affect UMLRR facility safety be immediately reported to the Level 1 authority; and that follow-up written reports of audit findings be submitted to the Level 1 authority and all RSSC members within 3 months after the completion of an audit. In its letter dated September 30, 2020 (Ref. 98), UML stated that although the TSs do not include a requirement for security audits, the NRC-approved physical security plan for the UMLRR (see SER Section 1.1.3) has an annual audit requirement which is performed and documented by the Reactor Supervisor. The NRC staff finds that TS 6.2.4 helps ensure that audits are conducted with appropriate independence, that the scope of audits is appropriate and includes important items relating to UMLRR facility safety, and that audit findings are reported in a timely manner. The NRC staff notes that while audits are an RSSC function, TS 6.2.4 does not require that audits necessarily be performed by an individual or group that is part of the RSSC (i.e., other individuals or groups, including qualified contractors/consultants, could be assigned to conduct an audit performed to help satisfy TS 6.2.4). The NRC staff also finds that by establishing requirements related to audits, TS 6.2.4 appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007. Additionally, the audit intervals specified in TS 6.2.4(2) are consistent with the recommended intervals provided in ANSI/ANS-15.1-2007, Section 6.2.4, as applicable. Therefore, based on the above and based on its 10 CFR 50.36(c)(5) findings for TSs on administrative controls in SER Section 6.7, the NRC staff concludes that TS 6.2.4 is acceptable.

The NRC staff notes that, as discussed in SER Section 4.1.2, the RSC (of which the RSSC is a subcommittee) separately conducts reviews and audits of the UML Radiation Safety Program (RSP) to satisfy the requirements of 10 CFR 20.1101(c) and TS 6.3(1); these reviews and audits are separate from the requirements of TSs 6.2.3 and 6.2.4.

6.6.3 Proposed Renewed TS 6.3, "Radiation Safety"

Renewed TS 6.3, "Radiation Safety," is discussed and found acceptable in SER Section 4.1.2.

6.6.4 Proposed Renewed TS 6.4, "Operating Procedures"

Renewed TS 6.4, "Operating Procedures," would state:

(1) Written procedures shall be reviewed by the RSSC in accordance with TS 6.2.3(1.a) and 6.2.3(1.b) and shall be in effect and followed for the following items. The procedures shall be adequate to ensure the safe operation of the reactor and gamma irradiation facilities, but shall not preclude the use of independent judgment and action should the situation require such.

a. startup, operation, and shutdown of the reactor;

b. fuel loading, unloading, and movement within the reactor;

c. maintenance of major components of systems that could have an effect on reactor facility safety;

d. surveillance checks, calibrations, and inspections required by the technical specifications or those that may have an effect on reactor facility safety;

e. personnel radiation protection, consistent with applicable regulations and guidelines, and TS 6.3(3);

f. administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect reactor facility safety or core reactivity;

g. the conduct of irradiations and experiments in the gamma irradiation facilities;

h. implementation of required plans such as emergency or security plans;

i. use, receipt, and transfer of byproduct material.

(2) The Reactor Supervisor or designee shall approve the procedures for 6.4(1) with the exception of 6.4(1.e). The Radiation Safety Officer or designee shall approve the procedures for 6.4(1.e).

(3) Temporary deviations from procedures required by TS 6.4(1) may be made by a senior reactor operator (Level 3) or member of the radiation safety staff, as applicable. Such deviations shall be documented and reported within 24 hours or the next working day to the Reactor Supervisor or Radiation Safety Officer or designees, as applicable.

TS 6.4(1) would require that UML have and use written procedures that are adequate to ensure the safe conduct of activities at the UMLRR facility. TS 6.4(1) would require that UML have procedures for the items listed in TS 6.4(1), items a. through i., and that the procedures in use have been reviewed by the RSSC as required by separate TS 6.2.3(1), items a. and b. (TS 6.2.3(1), item a., requires that the RSSC review UMLRR 10 CFR 50.59 evaluations, and TS 6.2.3(1), item b., requires, in part, that new procedures and major revisions (having safety

significance) to existing procedures be reviewed by the RSSC.). TS 6.4(2) would require that TS 6.4(1)-required procedures in use have been approved by the Reactor Supervisor, except for personnel radiation protection procedures (TS 6.4(1), item e.), which must have been approved by the RSO. TS 6.4(3) would allow SROs (or members of the health physics staff, as applicable) to make temporary deviations from procedures, but would require that such deviations be documented and reported within 24 hours or the next working day to the Reactor Supervisor or RSO, as applicable.

In its response to RAI-14.6.15 (Ref. 71), UML stated that TS 6.4(1), item g., was added to cover the UMLRR gamma irradiation facilities (the NRC staff notes that this is an additional facility-specific item that is not included in the generic list provided in ANSI/ANS-15.1-2007, Section 6.4). Also in its response to RAI-14.6.15, as updated by its letter January 30, 2021 (Ref. 99), UML stated that although there are UMLRR-specific procedures that relate to personnel radiation safety (e.g., for the UMLRR radiation monitoring system and experimental facilities), and there are radiation safety steps (e.g., surveys) included within several UMLRR-specific operations procedures associated with the reactor and gamma irradiation facilities, there are no specific separate UMLRR procedures for general personnel radiation safety (i.e., personnel radiation protection). UML indicated that these types of procedures, required by TS 6.4(1), item e., for the UMLRR, are generic to the entire UML RSP, which applies to the entire UML campus, including the UMLRR; SAR Section 12.3.2 lists the types of procedures that fall under generic RSP procedures. Therefore, TS 6.4(2) specifies that campus-wide generic RSP personnel radiation protection procedures used at the UMLRR (as required by TS 6.4(1), item e.) are approved by the RSO, instead of being approved by the Reactor Supervisor (however, all TS-required procedures used at the UMLRR are reviewed by the RSSC, as required by TS 6.2.3(1), item b., and TS 6.4(1)).

The NRC staff finds that TSs 6.4(1) and 6.4(2) help ensure that UML has and uses properly reviewed and approved procedures for activities that relate to UMLRR facility safety. The NRC staff also finds that, by requiring that the procedures in use be reviewed and approved, TSs 6.4(1) and 6.4(2) help ensure that any permanent, substantive changes to existing procedures are properly reviewed and approved prior to implementation. (SAR Section 12.3.1 states that changes to UMLRR facility operations procedures, including editorial changes or changes without safety significance, must be reviewed by the RSSC and approved by the Reactor Supervisor or designee. The RSSC review must occur prior to approval and implementation for safety-significant, substantive changes, although editorial changes or changes without safety significance may be reviewed by the RSSC after Reactor Supervisor approval and implementation. SAR Section 12.3.2 states that changes to campus-wide RSP procedures are approved by the RSO or designee. As discussed above, TS 6.2.3(1), item b., which is referenced in TS 6.4(1), requires RSSC review of all new TS-required procedures and major/substantive revisions (having safety significance).)

Additionally, the NRC staff finds that the approval of generic UML radiation protection procedures that are used at the UMLRR by the RSO (instead of the Reactor Supervisor) is reasonable and appropriate, given the subject matter of these procedures and their applicability to the entire UML campus. The NRC staff finds that the TS 6.4(3) allowance for temporary deviations from procedures is reasonable because it helps ensure that appropriate UMLRR staff and/or health physics personnel have flexibility to handle special or unusual circumstances or conditions, while also helping ensure documentation and timely notification of any such deviation. The NRC staff notes that while TS 6.4 does not explicitly state that new procedures, permanent changes to existing procedures, or temporary deviations from existing procedures are subject to 10 CFR 50.59, all new procedures, and all permanent or temporary changes to

existing procedures, are potentially subject to reviews required by 10 CFR 50.59 (in addition to being subject to the review and approval requirements explicitly stated in TSs 6.2.3(1), 6.4(1), and 6.4(2) for permanent changes). Changes to any procedures used at the UMLRR, including the campus-wide RSP procedures, are subject to 10 CFR 50.59 criteria. The NRC staff additionally finds that by establishing requirements for procedures, TSs 6.4(1) and 6.4(2) appropriately implement the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007. Therefore, based on the above and based on its 10 CFR 50.36(c)(5) findings for TSs on administrative controls in SER Section 6.7, the NRC staff concludes that TSs 6.4(1), 6.4(2), and 6.4(3) are acceptable.

6.6.5 Proposed Renewed TS 6.5, "Experiments Review and Approval"

Renewed TS 6.5, "Experiments Review and Approval," would state:

(1) All new experiments or classes of experiments shall be reviewed by the RSSC, subject to the requirements of 10CFR 50.59, and approved in writing by the Reactor Supervisor or designated alternate prior to initiation.

(2) Approved experiments shall be carried out in accordance with established and approved written procedures.

(3) Substantive changes to previously approved experiments shall be made only after review by the RSSC, subject to the requirements of 10 CFR 50.59, and approved in writing by the Reactor Supervisor or designated alternate prior to initiation.

TS 6.5(2) would require that experiments be carried out in accordance with established and approved written procedures. TS 6.5(1) would require that all new experiments or classes of experiments be reviewed by the RSSC and approved by the Reactor Supervisor or designee prior to initiation. TS 6.5(3) would further require that any substantive changes to previously approved experiments or classes of experiments similarly be made only after RSSC review and Reactor Supervisor approval (the review and approval must be done prior to initiation of the changed experiment). Additionally, TSs 6.5(1) and 6.5(3) would specify that new experiments and classes of experiments, and substantive changes to experiments, are subject to the requirements of 10 CFR 50.59. The NRC staff finds that TS 6.5(2), in conjunction with TS 6.4(1), helps ensure that experiments are conducted following appropriate procedures. The NRC staff finds that TSs 6.5(1) and 6.5(3) help ensure that new experiments, and changes to experiments, are subject to an appropriate review and approval process before they are initiated. The NRC staff notes that although minor changes to experiments (i.e., non-substantive changes that do not significantly alter the experiment) are not addressed in TS 6.5, any change to an experiment is potentially subject to the requirements of 10 CFR 50.59; however, the NRC staff notes that research reactor experiment safety evaluations typically cover multiple experiment variations to help prevent any minor changes from causing an experiment to be outside the scope of its review and approval. The NRC staff also finds that by establishing requirements for experiment procedures and review and approval, TS 6.5 appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007. Therefore, based on the above and based on its 10 CFR 50.36(c)(5) findings for TSs on administrative controls in SER Section 6.7, the NRC staff concludes that TS 6.5 is acceptable.

6.6.6 Proposed Renewed TS 6.6, "Required Actions"

Renewed TS 6.6.1, "Action To Be Taken In The Event The Safety Limit Is Exceeded," would state:

(1) The reactor shall be shut down and reactor operation shall not be resumed until authorization is obtained from the NRC.

(2) The safety limit violation shall be promptly reported to the Reactor Supervisor or designee, the Chancellor's Level 1 designee, and the Chairman of the RSSC.

(3) The safety limit violation shall be reported to the NRC in accordance with TS 6.7.2(1).

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

a. The time and date of the violation, reactor status at the time of the violation, and a description of the violation.

b. The applicable circumstances leading to the violation including, when known, the cause and contributing factors.

c. The effect of the violation upon reactor facility components, systems, or structures and on the health and safety of personnel and the public.

d. Corrective action to be taken to prevent recurrence.

(5) The report shall be reviewed by the RSSC and shall be submitted to the NRC in accordance with TS 6.7.2(2).

TS 6.6.1 would require specific actions that must be taken if the UMLRR SL is exceeded. Specifically, TS 6.6.1(1) would require that the reactor be shut down until a restart is authorized by the NRC; TS 6.6.1(2) would require prompt reporting of the violation to the Reactor Supervisor or designee (Level 2), the Level 1 authority in the UML Chancellor's office, and the RSSC chairman; TS 6.6.1(3) would require reporting of the violation to the NRC Headquarters Operations Center no later than the following working day in accordance with renewed TS 6.7.2(1), which is discussed and found acceptable in SER Section 6.6.7; and TSs 6.6.1(4) and 6.6.1(5) would require the preparation, RSSC review, and submission to the NRC within 14 days as required by TS 6.7.2(2), of a SL violation report. The NRC staff finds that TS 6.6.1 helps ensure that UML will take appropriate actions in the event of a SL violation, including reporting, analysis, and documentation of the violation, and also helps ensure that the reactor will not be restarted until it is safe to do so, and it has been determined that recurrence of the violation is unlikely. The NRC staff notes that the content of, and RSSC review of, the SL violation report are subject to the requirements of TS 6.6.2(2), item e., in addition to TSs 6.6.1(4) and 6.6.1(5). The NRC staff also notes that the SL violation report required to be prepared and submitted within 14 days in accordance with TSs 6.6.1(4) and 6.6.1(5) could accompany a request for NRC authorization to restart the reactor referred to in TS 6.6.1(1), or a UML request for an NRC restart authorization submitted more than 14 days after the violation, if necessary, could be accompanied by a separate report. The NRC staff finds that TS 6.6.1 helps ensure that UML meets the requirements in 10 CFR 50.36(c)(1)(i)(A) related to follow-up

actions for a SL violation and will notify the NRC promptly. The NRC staff also finds that by listing specific actions that must be taken if the SL is exceeded, TS 6.6.1 appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007. Therefore, based on the above and based on its 10 CFR 50.36(c)(5) findings for TSs on administrative controls in SER Section 6.7, the NRC staff concludes that TS 6.6.1 is acceptable.

Renewed TS 6.6.2, "Action To Be Taken in the Event of a Reportable Occurrence," would state:

(1) A reportable occurrence shall be any of the following conditions:

a. Release of radioactivity from the reactor facility into unrestricted areas above allowed limits.

b. Operating with any safety system setting less conservative than that stated in Section 2.2 of these specifications.

c. Operating in violation of a limiting condition for operation established in Section 3.0 of these specifications unless prompt remedial action is taken as specified in TS 3.5(1) or TS 3.6.1(3).

d. A reactor safety system component malfunction that 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 shall be required.

e. An uncontrolled or unanticipated change in reactivity in excess of 0.6%$\Delta k/k$. Reactor trips resulting from a known cause are excluded.

f. An abnormal and significant degradation in reactor fuel and/or cladding, coolant boundary, or confinement boundary (excluding minor leaks).

g. An observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy causes or could have caused the existence or development of an unsafe condition in connection with operation of the reactor or gamma irradiation facilities.

(2) In the event of a reportable occurrence, the following actions shall be taken:

a. If involving the reactor, the reactor conditions shall be returned to normal, or the reactor shall be shutdown, to correct the occurrence. If shutdown, the reactor shall not be operated until authorized by the Reactor Supervisor.

b. If involving a gamma irradiation facility, the conditions shall be returned to normal, or gamma facilities operations shall cease, to correct the occurrence. If operations cease, the gamma irradiation facility shall not be operated until authorized by the Reactor Supervisor.

c. The Reactor Supervisor shall be notified as soon as possible.

d. The Nuclear Regulatory Commission shall be notified in accordance with TS 6.7.2(1).

e. A report shall be submitted to the NRC in accordance with TS 6.7.2(2) that includes the time and date of the occurrence, facility status at the time of the occurrence, a description of the occurrence, an evaluation of the cause of the occurrence, a record of the corrective action taken, and recommendations for appropriate action to prevent or reduce the probability of recurrence. This report shall be reviewed by the RSSC no later than its next regularly scheduled meeting.

TS 6.6.2(1) would specify UMLRR conditions (other than SL violations, which must be reported in accordance with TS 6.6.1) that constitute reportable occurrences that must be reported to the NRC, and other actions taken. TS 6.6.2(2) would specify the actions that must be taken in the event of a reportable occurrence. The NRC staff finds that TSs 6.6.2(1) and 6.6.2(2) help ensure that important information regarding UMLRR conditions and events will be provided to the NRC in a timely manner, and that UML will take other appropriate actions following reportable occurrences that will help prevent future occurrences and ensure the safety of subsequent UMLRR operations. The NRC staff finds that TS 6.6.2(1), item a., applies to releases of radioactive material covered under the UMLRR Facility Operating License No. R-125 only. The NRC staff further finds that by specifying what conditions constitute reportable occurrences at the UMLRR facility and what actions must be taken in the event of a reportable occurrence, TSs 6.6.2(1) and 6.6.2(2) appropriately implement the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007, with facility-specific modifications, e.g., the definition of any reactivity change greater than the specific value analyzed in the UMLRR SAR, as supplemented, as a reportable event, and a requirements for reportable events related to a gamma irradiation facility. (The NRC staff notes that although the guidance in ANSI/ANS-15.1-2007 lists recommended reportable events in Section 6.7.2, UML included its list of reportable occurrences in TS 6.6.2(1); this non-substantive difference did not affect the NRC staff's findings.) Therefore, based on the above and based on its 10 CFR 50.36(c)(5) findings for TSs on administrative controls in SER Section 6.7, the NRC staff concludes that TSs 6.6.2(1) and 6.6.2(2) are acceptable.

6.6.7 Proposed Renewed TS 6.7, "Reports"

Renewed TS 6.7.1, "Operating Reports," would state:

An annual or operating report shall be submitted to the NRC Document Control Desk within ninety days following the 30th of June of each year. Its content shall include:

(1) A narrative summary of reactor operating experience including a tabulation showing the energy generated by the reactor (in megawatt days), the number of hours the reactor was critical, and the cumulative total energy output since initial criticality.

(2) The number of emergency shutdowns and inadvertent scrams, including the reasons therefore, and where applicable, corrective actions to preclude recurrence.

(3) Tabulation of major preventative and corrective maintenance operations having safety significance.

(4) A description, including a summary of the safety evaluations of changes in the facility and procedures and of tests and experiments carried out pursuant to 10 CFR 50.59.

(5) A summary of the nature and amount of radioactive effluents released or discharged to environs beyond the effective control of the licensee, as determined at, or before, the point of such release or discharge. The summary shall include to the extent practicable an estimate of individual radionuclides present in the effluent. If the estimated average release after dilution or diffusion is <25% of the 10 CFR 20 Appendix B concentration limits, a statement to this effect is sufficient.

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

(7) A summary of exposures received by facility personnel and visitors where such exposures are >25% of the regulatory limits in 10 CFR 20.

TS 6.7.1 would require that UML submit, within 90 days following June 30 of each year, a report to the NRC providing information related to UMLRR operations during the one-year period leading up to that June 30. TS 6.7.1 would also specify the minimum information that is required to be included in each report. The NRC staff finds that TS 6.7.1 helps ensure that important information regarding facility operations is provided to the NRC at intervals that are appropriate to allow timely NRC review (and to ensure compliance with the regulation in 10 CFR 50.59(d)(2) related to reporting of changes, tests, and experiments). With regard to TS 6.7.1(5), the NRC staff notes that, as discussed in SER Section 4.2.1, UML does not typically release liquid waste as effluent directly from the reactor facility operating license (it first transfers the waste to UML's broad-scope materials license), but if it did, TS 6.7.1(5) would require it to report such releases. The NRC staff also finds that by specifying requirements related to annual operating reports, TS 6.7.1 appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007, including guidance related to the content of annual reports. Therefore, based on the above and based on its 10 CFR 50.36(c)(5) findings for TSs on administrative controls in SER Section 6.7, the NRC staff concludes that TS 6.7.1 is acceptable.

Renewed TS 6.7.2, "Special Reports," would state:

(1) A report shall be made not later than the following working day by telephone and confirmed in writing by fax or similar conveyance to the NRC Headquarters Operations Center, of any of the following:

a. Operation in violation of a safety limit.

b. Any reportable occurrence as defined in TS 6.6.2.

(2) A written report shall be provided as a follow-up to the verbal one within 14 days of the occurrence. This report shall provide the information required by TS 6.6.1(4) and/or 6.6.2(2.e), as applicable. The report shall be submitted to the NRC Document Control Desk.

(3) A written report shall be submitted within 30 days to the NRC Document Control Desk in the event of:

a. A permanent change in the personnel serving as Level 1 or Level 2.

b. Any significant change in the transient or accident analyses as described in the SAR.

TS 6.7.2 would establish requirements for special reports to be submitted to the NRC regarding SL violations, other reportable occurrences, and changes in UMLRR personnel or analyses described in the SAR. The NRC staff finds that TS 6.7.2 helps ensure that important information regarding the facility and its personnel is provided to the NRC in a timely manner. The NRC staff also finds that by delineating reporting requirements for SL violations, other reportable occurrences, and changes in UMLRR personnel or analyses described in the SAR, TS 6.7.2 appropriately implements the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007 (although the guidance in ANSI/ANS-15.1-2007 lists recommended reportable events in Section 6.7.2, UML included its list of reportable occurrences in TS 6.6.2(1); additionally, although ANSI/ANS-15.1-2007, Section 6.7.2, lists releases of radioactivity as events which should require special reports, UML included releases of radioactivity as events that are reportable occurrences as defined in TS 6.6.2(1); these non-substantive differences did not affect the NRC staff's findings.). Therefore, based on the above and also based on its 10 CFR 50.36(c)(5) findings for TSs on administrative controls in SER Section 6.7, the NRC staff concludes that TS 6.7.2 is acceptable.

6.6.8 Proposed Renewed TS 6.8, "Records"

Renewed TS 6.8.1, "Five-Year Record Retention," would state:

The following records shall be retained for five years or for the life of the component involved if less than five years:

(1) Records of normal reactor facility operation. (but not including supporting documents such as checklists, log sheets, etc., which shall be retained for a period of at least one year.)
(2) Records of principal maintenance operations.
(3) Records of reportable occurrences.
(4) Records of surveillance activities that are required by these technical specifications.
(5) Records of reactor facility radiation and contamination surveys.
(6) Records of experiments performed with the reactor.
(7) Records of fuel inventories, receipt, and shipments.
(8) Records of approved changes made in the operating procedures.
(9) Records of meeting minutes and reports of audits required by TS 6.2.4.

Renewed TS 6.8.2, "Six-Year Record Retention," would state:

Records of individual licensed staff members indicating qualifications, experience, training, and requalification shall be retained at all times that an individual is employed or until the operator license is renewed.

Renewed TS 6.8.3, “Records To Be Retained for the Life of the Facility,” would state:

> The following records shall be retained for the life of the facility. 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 personnel monitored.
> (4) Drawings of the reactor facility.
> (5) Reviews and reports pertaining to a violation of a safety limit, limiting safety system setting, or limiting condition for operation.

TSs 6.8.1, 6.8.2, and 6.8.3 would provide requirements for record retention at the UMLRR. TS 6.8.1 would specify records that must be maintained for 5 years (or for the life of the component involved if less than 5 years). TS 6.8.2 would specify records related to licensed reactor operators that must be maintained for the term of each operators’ license (the NRC staff notes that operators’ licenses are issued for 6-year terms, pursuant to 10 CFR 55.55(a)). TS 6.8.3 would specify records that must be maintained until the facility license is terminated. With regard to TS 6.8.1(9), “meeting minutes” are referenced in TSs 6.2.2(3) and 6.2.3(2), and refer to meeting minutes (or reports) of the RSSC (although TS 6.8.1(9) requires that meeting minutes be retained for 5 years, SAR Section 12.6.2 states that UML maintains RSSC meeting minutes for the life of the facility). The NRC staff finds that TSs 6.8.1, 6.8.2, and 6.8.3 help ensure that UML maintains important records and retains them for appropriate time periods. The NRC staff finds that TSs 6.8.1, 6.8.2, and 6.8.3 appropriately implement the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007, related to recordkeeping requirement TSs, that TSs 6.8.1(6) and 6.8.1(8) help ensure UML’s compliance with the 10 CFR 50.59(d) requirement to retain records of changes in procedures and records of tests and experiments for 5 years, and that TS 6.8.3(5) helps ensure UML’s compliance with the regulations in 10 CFR 50.36(c) requiring licensees to retain the record of the results of each review pertaining to a violation of a SL, LSSS, or LCO until the NRC terminates the license for the reactor. The NRC staff notes that 10 CFR 50.59(d) also requires that records of changes in the facility be maintained until the termination of the operating license. Therefore, based on the above and based on its 10 CFR 50.36(c)(5) findings for TSs on administrative controls in SER Section 6.7, the NRC staff concludes that TSs 6.8.1, 6.8.2, and 6.8.3 are acceptable.

6.7 Conclusions on Technical Specifications

The NRC staff evaluated the proposed TSs as part of its review of the application for renewal of Facility Operating License No. R-125 for the UMLRR. The proposed TSs define certain features, characteristics, organizational and reporting requirements, and conditions governing the operation of the UMLRR. The renewed license includes the proposed TSs as Appendix A. The NRC staff reviewed the format and content of the proposed TSs for consistency with the guidance in ANSI/ANS-15.1-2007, and NUREG-1537, Appendix 14.1, and finds that they are appropriately consistent with this guidance. The NRC staff also specifically evaluated the content of the proposed TSs to determine if they meet the requirements in 10 CFR 50.36. Based on its review, the NRC staff concludes that the proposed TSs meet the requirements of the regulations. The NRC staff based this conclusion on the following findings:

- As required by 10 CFR 50.36(a)(1), the licensee, an applicant for a license to operate a utilization facility, provided proposed TSs with its license renewal application and included appropriate summary statements of the bases or reasons for the TSs. The bases are included for reference on applicable TS pages, but are not part of the TSs as required by 10 CFR 50.36(a)(1).

- As required by 10 CFR 50.36(b), each license for a research reactor must include TSs and the TSs must be derived from the analyses and evaluation included in the SAR, and amendments thereto, and the Commission may include such additional specifications as it finds appropriate. The licensee provided TSs derived from analyses in the SAR, as supplemented.

- The UMLRR TSs specify a SL (TS 2.1) on the fuel temperature, which is the important process variable necessary to reasonably protect the integrity of the fuel cladding, which is the primary fission product barrier; and LSSSs (TS 2.2), which are settings for the automatic reactor protection system, for variables having significant safety functions, intended to prevent reaching the SL, that satisfy 10 CFR 50.36(c)(1)(i)(A) and (ii)(A) requirements.

- The TSs contain LCOs (TS Section 3.0), which are the lowest functional capability or performance levels of equipment required for safe operation of the facility, for each item that meets one of more of the criteria specified in 10 CFR 50.36(c)(2)(ii).

- The TSs contain SRs (TS Section 4.0), which relate to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within SLs, and that the LCOs will be met, that satisfy the provisions of 10 CFR 50.36(c)(3).

- The TSs contain design features (TS Section 5.0), which are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety, that satisfy the requirements of 10 CFR 50.36(c)(4).

- The TSs contain administrative controls (TS Section 6.0), which are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner, that satisfy the requirements of 10 CFR 50.36(c)(5). The TSs contain requirements for initial notification, written reports, and records that satisfy 10 CFR 50.36(c)(1), (2), and (7); and for special reports that the NRC staff deemed necessary in accordance with 10 CFR 50.36(c)(8).

- The TSs acceptably implement the recommendations of Section 1.3 of Appendix 14.1 to Part 1 of NUREG-1537, and Section 1.3 of ANSI/ANS-15.1-2007, by using definitions (TS 1.3) that are acceptable and appropriate.

On the basis of its review, the NRC staff finds the TSs to be acceptable and concludes that normal operation of the UMLRR within the limits of the TSs will not result in radiation exposures in excess of the limits specified in 10 CFR Part 20 for members of the general public or for occupational exposures. The NRC staff also concludes that the TSs provide reasonable assurance that the facility will be operated as analyzed in the SAR, as supplemented; that

adherence to the TSs will limit the likelihood of malfunctions and the potential accident scenarios analyzed in the SAR, as supplemented, and discussed in SER Chapter 5; that facility operation will be in accordance with the applicable regulations; and that the conduct of activities by the licensee will not endanger the facility staff or members of the public.

7. 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 staff concludes the following:

- The application for license renewal dated October 20, 2015, as supplemented, complies with the standards and requirements of the Atomic Energy Act (AEA) of 1954, as amended, 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, as well as the provisions of the AEA and the rules and regulations of the Commission.

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

- 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 Commission.

- The applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," have been satisfied.

- The issuance of this license is in accordance with 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," and all applicable requirements have been satisfied, as documented in the Environmental Assessment and Finding of No Significant Impact published in the *Federal Register* on August 4, 2021 (86 FR 41998), which concluded that the renewal of the University of Massachusetts Lowell Research Reactor 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 Commission'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."

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

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Includes as attachments the following documents referenced in the SER. These documents are, in their entirety, proprietary and are withheld from public disclosure per 10 CFR 2.390, and are therefore not publicly available in ADAMS:

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Includes, in part, the following specific documents referenced in the SER:

a. General Atomics, “TRIGA INL Software Configuration Management Plan,” Drawing No. T9S900D970-CMP, Revision A. ADAMS Accession No. ML20017A146 (redacted version).

b. General Atomics, “NMP-1000 Multi-range Linear Module User Manual,” Document T3401000-1UM, Revision B, dated June 2015. ADAMS Accession No. ML20017A149 (redacted version).

c. General Atomics, “Operation and Maintenance Manual NMP-1000 Linear Power Channel,” Document No. E117-1017, Revision 1, 1991. ADAMS Accession No. ML20017A151 (redacted version).

d. General Atomics, “Software Quality Assurance Plan,” Document No. T9S900SQAP, Revision B, dated December 2010. ADAMS Accession No. ML20017A145 (redacted version).

e. General Atomics, “TRIGA INL Channels Software Development Plan,” Drawing No. T9S900D970-SWP-2, Revision A, dated January 2014. ADAMS Accession No. ML20017A147 (redacted version).

104. U.S. Nuclear Regulatory Commission, Federal Register Notice of Environmental Assessment and Finding of No Significant Impact, dated July 30, 2021 (published in *Federal Register* August 4, 2021, 86 FR 41998). ADAMS Package Accession No. ML17115A401.

105. U.S. Nuclear Regulatory Commission, Issuance of License Amendment No. 9 for the University of Lowell Related to Facility Operating License Renewal, dated November 21, 1985. ADAMS Package Accession No. ML20137A862.

106. U.S. Nuclear Regulatory Commission, Issuance of License Amendment No. 11 for the University of Massachusetts Lowell, dated February 20, 1992. ADAMS Package Accession No. ML20092L544.

107. Institute of Electrical and Electronics Engineers, IEEE 7-4.3.2-1993, "IEEE Standard Criteria for Programmable Digital Devices in Safety Systems of Nuclear Power Generating Stations," American Nuclear Society, New York, NY.

108. ASME-NQA-1-2008, 1a-2009, "Quality Assurance Requirements for Nuclear Facility Applications," American Society of Mechanical Engineers, New York, New York.

109. E-mails between L. Bobek, University of Massachusetts Lowell, and E. Helvenston, U.S. Nuclear Regulatory Commission, "Draft UMLRR Renewed License," dated January 27, 2022. ADAMS Package Accession No. ML22027A549.