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RESEARCH REACTOR CENTER

November 7, 2016

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Mail Station P1-37
Washington, DC 20555-0001

Reference: Docket 50-186
University of Missouri-Columbia Research Reactor
Amended Facility Operating License No. R-103

Enclosed you will find the University of Missouri-Columbia Research Reactor's responses to the U.S. Nuclear Regulatory Commission's (NRC) request for additional information, dated September 7, 2016, regarding our renewal request for Amended Facility Operating License No. R-103, which was submitted to the NRC on August 31, 2006, as supplemented.

If you have any questions, please contact Bruce Meffert, the facility Reactor Manager, at (573) 882-5118 or MeffertB@missouri.edu.

Sincerely,

Michael A. F..., Acting Director, for Ralph Butler

Ralph A. Butler, P.E.
Director

RAB/jlm

Enclosures

A020
NRR

1513 Research Park Drive Columbia, MO 65211 Phone: 573-882-4211 Fax: 573-882-6360 Web: http://web.missouri.edu/~murrwww
Fighting Cancer with Tomorrow's Technology

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Attention: Document Control Desk
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REFERENCE: Docket No. 50-186
University of Missouri-Columbia Research Reactor
Amended Facility Operating License No. R-103

SUBJECT: Written communication as specified by 10 CFR 50.4(b)(1) regarding responses to the "University of Missouri at Columbia - Request for Additional Information Regarding the Proposed Technical Specifications for the Renewal of Facility Operating License No. R-103 for the University of Missouri at Columbia Research Reactor (TAC No. ME1580)," dated September 7, 2016

On August 31, 2006, the University of Missouri-Columbia Research Reactor (MURR) submitted a request to the U.S. Nuclear Regulatory Commission (NRC) to renew Amended Facility Operating License No. R-103.

By letter dated May 6, 2010, the NRC requested additional information and clarification regarding the renewal request in the form of nineteen (19) Complex Questions. By letter dated September 3, 2010, MURR responded to seven (7) of those Complex Questions.

By letter dated June 1, 2010, the NRC requested additional information and clarification regarding the renewal request in the form of one hundred and sixty-seven (167) 45-Day Response Questions. By letter dated July 16, 2010, MURR responded to forty-seven (47) of those 45-Day Response Questions.

On July 14, 2010, via electronic mail (email), MURR requested additional time to respond to the remaining one hundred and twenty (120) 45-Day Response Questions. By letter dated August 4, 2010, the NRC granted the request. By letter dated August 31, 2010, MURR responded to fifty-three (53) of the 45-Day Response Questions.

On September 1, 2010, via email, MURR requested additional time to respond to the remaining twelve (12) Complex Questions. By letter dated September 27, 2010, the NRC granted the request.

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On September 29, 2010, via email, MURR requested additional time to respond to the remaining sixty-seven (67) 45-Day Response Questions. On September 30, 2010, MURR responded to sixteen (16) of the remaining 45-Day Questions. By letter dated October 13, 2010, the NRC granted the extension request.

By letter dated October 29, 2010, MURR responded to sixteen (16) of the remaining 45-Day Response Questions and two (2) of the remaining Complex Questions.

By letter dated November 30, 2010, MURR responded to twelve (12) of the remaining 45-Day Response Questions.

On December 1, 2010, via email, MURR requested additional time to respond to the remaining 45-Day Response and Complex Questions. By letter dated December 13, 2010, the NRC granted the extension request.

On January 14, 2011, via email, MURR requested additional time to respond to the remaining 45-Day Response and Complex Questions. By letter dated February 1, 2011, the NRC granted the extension request.

By letter dated March 11, 2011, MURR responded to twenty-one (21) of the remaining 45-Day Response Questions.

On May 27, 2011, via email, MURR requested additional time to respond to the remaining 45-Day Response and Complex Questions. By letter dated July 5, 2011, the NRC granted the request.

By letter dated September 8, 2011, MURR responded to six (6) of the remaining 45-Day Response and Complex Questions.

On September 30, 2011, via email, MURR requested additional time to respond to the remaining the remaining 45-Day Response and Complex Questions. By letter dated November 10, 2011, the NRC granted the request.

By letter dated January 6, 2012, MURR responded to four (4) of the remaining 45-Day Response and Complex Questions. Also submitted was an updated version of the MURR Technical Specifications.

On January 23, 2012, via email, MURR requested additional time to respond to the remaining the remaining 45-Day Response and Complex Questions. By letter dated January 26, 2012, the NRC granted the request.

On April 12, 2012, via email, MURR requested additional time to respond to the remaining the remaining 45-Day Response and Complex Questions.

By letter dated June 28, 2012, MURR responded to the remaining six (6) 45-Day Response and Complex Questions. With that set of responses, all 45-Day Response and Complex Questions had been addressed.

On December 20, 2012, the NRC requested a copy of the current Physical Security Plan (PSP) and Operator Requalification Program.

By letter dated January 4, 2013, MURR provided the NRC a copy of the current PSP and Operator Requalification Program.

By letter dated February 11, 2013, the NRC requested updated financial information in the form of four (4) questions because the information provided by the September 14, 2009 response had become outdated.

By letter dated March 12, 2013, MURR responded to the four (4) questions.

By letter dated December 3, 2014, the NRC requested additional information in the form of two (2) questions regarding significant changes to the MURR facility since submittal of the licensing renewal application in August 2006.

By letter dated January 28, 2015, MURR responded to the two (2) questions.

By letter dated April 17, 2015, the NRC requested additional information in the form of ten (10) questions.

On May 29, 2015, via email, MURR requested additional time to respond to the ten (10) questions.

By letter dated June 18, 2015, the NRC requested additional information in the form of two (2) questions.

By letter dated July 31, 2015, MURR responded to the two (2) questions from the June 18, 2015 request.

On September 14, 2015, via telephone, the NRC requested a copy of the Emergency Plan (EP).

By letter dated September 14, 2015, the NRC requested additional information in the form of sixteen (16) questions regarding the PSP.

By letter dated September 15, 2015, MURR provided the NRC a copy of the current EP.

By letter dated October 1, 2015, MURR responded to the ten (10) questions from the April 17, 2015 request.

By letter dated October 28, 2005, the NRC requested additional information regarding the proposed Technical Specifications.

By letter dated December 2, 2015, MURR responded to the fifteen (15) questions from the September 14, 2015 request regarding the PSP.

By letter dated December 17, 2015, the NRC requested additional information in the form of thirteen (13) questions regarding follow-up information from MURR's October 1, 2015 responses to the NRC's April 17, 2015 request for additional information.

By letter dated February 8, 2016, MURR responded to the thirteen (13) questions from the December 17, 2015 request.

By letter dated February 8, 2016, the NRC requested updated financial information in the form of four (4) questions because the information provided by the March 12, 2013 response had become outdated.

By letter dated March 23, 2016, the NRC requested additional information in the form of twenty-one (21) questions regarding follow-up information from MURR's February 8, 2016 responses to the NRC's April 17, 2015 request for additional information.

By letter dated April 8, 2016, MURR responded to the four (4) questions from the February 8, 2016 request.

By letter dated April 15, 2016, MURR responded to the twenty-one (21) questions from the March 23, 2016 request.

By letter dated May 31, 2016, MURR responded to questions from the October 28, 2015 request. Additionally, Technical Specification changes, as issued by Amendment No. 37 to the current facility operating license (NRC letter dated March 11, 2016), were incorporated into the revised proposed Technical Specifications.

On June 28, 2016, via a conference call between MURR and NRC staff, the NRC requested additional information/clarification regarding MURR's May 31, 2016 responses to the October 28, 2015 NRC request.

By letter dated July 1, 2016, MURR responded to the request for additional information/clarification from the June 28, 2016 conference call as well as submitted the revised proposed Technical Specifications.

By letter dated August 24, 2016, the NRC requested financial information in the form of one (1) question.

By letter dated August 31, 2016, MURR responded to the one (1) question from the August 24, 2016 request.

By letter dated September 7, 2016, the NRC requested additional information regarding the proposed Technical Specifications in the form of thirty (30) questions. Additionally, via multiple conference calls between MURR and NRC staff, additional information/clarification was requested. Below are the questions and MURR's responses to those questions from the NRC letter dated September 7, 2016 and the numerous conference calls. Additionally, the revised proposed Technical Specifications are also attached.

If there are any questions regarding this response, please contact me at (573) 882-5118 or MeffertB@missouri.edu. I declare under penalty of perjury that the foregoing is true and correct.

Sincerely,

Bruce Meffert
Reactor Manager

ENDORSEMENT:
Reviewed and Approved,

, Acting Director, for Ralph Butler

Ralph A. Butler, P.E.
Director

xc: Reactor Advisory Committee
Reactor Safety Subcommittee
Dr. Garnett S. Stokes, Provost
Dr. Mark McIntosh, Vice Chancellor for Research, Graduate Studies and Economic Development
Mr. Alexander Adams Jr., U.S. Nuclear Regulatory Commission
Mr. Geoffrey Wertz, U.S. Nuclear Regulatory Commission
Mr. Johnny Eads, U.S. Nuclear Regulatory Commission

Enclosure:

1. Appendix A, Technical Specifications for the University of Missouri Research Reactor, Facility Operating License R-103, Docket 50-186, as revised.

State of Missouri
County of Boone
Subscribed and sworn to before me this
7 **day of** November, 2016
Jacqueline L. Matyas
JACQUELINE L. MATYAS, Notary Public
My Commission Expires: March 26, 2019

NOTARY PUBLIC STATE OF MISSOURI NOTARY SEAL

JACQUELINE L. MATYAS
My Commission Expires
March 26, 2019
Howard County
Commission #15634308

1. *Proposed TS 1.0, "Definitions," provides definitions useful in the TSs. However, the NRC staff finds that the term "Protective Action," does not appear in the TS 1.0, "Definitions." Proposed TS 1.13 uses the term "protective action."*

   ANSI/ANS-15.1-2007, provides guidance that includes a definition for Protective Action. Furthermore, NUREG-1537, Part 1, Chapter 14, "Technical Specifications," Appendix 14.1, provides guidance that accepts the definitions in ANSI/ANS-15.1-2007, which includes Protective Action.

   Revise TS 1.0 to include Protective Action, or justify why no change is needed.

   The definition for "Protective Action" has been added to Section 1.0, as follows:

   "1.18 **Protective Action** - Protective action is the initiation of a signal or the operation of equipment within the reactor safety system in response to a parameter or condition of the reactor facility having reached a specified limit."

   The remainder of the definitions after Definition 1.18 have been renumbered.

2. *Proposed TS 3.4, "Reactor Containment Building," Specification b, provides exceptions for maintaining the reactor containment integrity. However, the NRC staff finds that the exceptions do not appear to include other activities which could pose a potential for release of radioactive material, such as movement of fueled experiment, control rod maintenance, or high worth experiments.*

   ANSI/ANS-15.1-2007, Section 3.4.1, "Operations that require containment or confinement," provides guidance that include: movement of fueled experiments with significant fission product inventory outside of containers, systems, or storage areas; core or control rod work that could cause a change in reactivity of more than one dollar; or movement of experiments that could cause a change of total worth of more than one dollar.

   Revise TS 3.4, Specification b, to include the guidance provided in ANSI/ANS-15.1-2007, or justify why no change is needed.

   Technical Specification 3.4.b states:

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

   (1) The reactor is secured,

   AND

   (2) Irradiated fuel with a decay time of less than sixty (60) days is not being handled."

Definition 1.26 (old Definition 1.25) states:

"**Reactor Secured** - The reactor shall be considered secured when:

a. There is insufficient fuel in the reactor core to attain criticality with optimum available conditions of moderation and reflection with all four (4) shim blades (rods) removed,

OR

b. Whenever all of the following conditions are met:

   (1) All four shim blades (rods) are fully inserted;

   (2) One of the two following conditions exits:

      i. The Master Control Switch is in the "OFF" position with the key locked in the key box or in custody of a licensed operator,

      OR

      ii. The dummy load test connectors are installed on the shim rod drive mechanisms and a licensed operator is present in the reactor control room;

   (3) No work is in progress involving the transfer of fuel in or out of the reactor core;

   (4) No work is in progress involving the shim blades (rods) or shim rod drive mechanisms with the exception of installing or removing the dummy load test connectors; and

   (5) The reactor pressure vessel cover is secured in position and no work is in progress on the reactor core assembly support structure."

ANSI/ANS-15.1-2007, Section 3.4.1, lists the following operations that require containment. Following each operation, is MURR's response to satisfying that requirement.

(1) "*Reactor operating*;" – MURR Technical Specification 3.4.b states that reactor containment integrity shall be maintained at all times except when the reactor is secured; therefore, this requirement is satisfied because the conditions of secured are more restrictive than the reactor not operating, such as shutdown.

(2) "*Movement of irradiated fuel or fueled experiments with significant fission product inventory outside of containers, systems, or storage areas*;" – MURR will revise Technical Specification 3.4.b(2) as follows: "Movement of irradiated fuel with a decay time of less than sixty (60) or fueled experiments with significant fission product inventory outside containers, systems, or storage areas."

(3) "*Core or control rod work that could cause a change in reactivity of more than one dollar*." – MURR feels that Specifications b(4) and b(5) of **Reactor Secured** more than adequately meets this specification since <u>no</u> work is allowed on the core or control rods, which is more restrictive than work that can cause a change in reactivity of more than one dollar (0.0074 $\Delta k/k$).

(4) "*Movement of experiments that could cause a change of total worth of more than one dollar;*" – MURR will add this specification to Technical Specification 3.4.b. MURR will add Technical Specification 3.4.b(3) as follows: "Movement of experiments that could cause a change of total worth greater than 0.0074 $\Delta k/k$."

Therefore, MURR Technical Specification 3.4.b has been revised as follows:

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

(1) The reactor is secured,

AND

(2) No movement of irradiated fuel with a decay time of less than sixty (60) days or fueled experiments with significant fission product inventory outside containers, systems, or storage areas,

AND

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

3. *Proposed TS 3.5, "Reactor Instrumentation," provides the reactor instrumentation needed for the reactor operators to operate MURR safely. However, the NRC staff finds that instrumentation for power level monitors does not appear to be included.*

NUREG-1537, Part 1, Chapter 14, "Technical Specifications," Appendix 14.1, item (8), "Control Systems and Instrumentation Requirements for Operation (Added by NRC)," provides guidance that TSs for non-power reactors should have redundant and accurate power level monitors that cover the range from subcritical source multiplication to above the full power level.

Revise TS 3.5, to include the guidance in NUREG-1537, or justify why no change is needed.

MURR Technical Specifications 3.2.f.1 and 3.2.g.1 state that three (3) "High Power Level" instrument channels, which include high power rod run-in and scram functions, are required for operation. In addition to the stated safety functions, these Power Range instrument channels each provide an indication of power over a range of 0 to 125%. Technical Specifications 3.2.f.2 and 3.2.g.2 state that two (2) "Reactor Period" instrument channels, which include low reactor period rod run-in and scram functions, are required for operation. In addition to the stated safety functions, these Intermediate Range instrument channels each provide an indication over a ten-decade range of reactor power from 10^{-8} to 200%. Technical Specification 3.5.a.1 states that a Source Range Nuclear Instrument Channel is required for operation. This channel provides an indication over a six-decade range of reactor power from 10^{-1} to 10^{5} cps. All of these instrument channels, which provide neutron flux measurement from reactor shutdown through full power level, are described in detail in Section 7.4 of the Safety Analysis Report. Although MURR feels that Specifications 3.2.f.1, 3.2.g.1, 3.2.f.2 and 3.2.g.2, as stated, adequately meets the guidance of Section 3.2, item (8) of Appendix 14.1, NUREG-1537, Part 1, Chapter 14, Technical Specification 3.5.a, and its bases, will be revised as follows to provide better clarity and remove any ambiguity:

"Specification:

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

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

(1) Required for reactor startup only."

"Bases:

a. The Power Range Nuclear Instrument Channels provide neutron monitors that provide reactor protective, alarm and indication functions over the power range (Ref. Section 7.4 of the SAR).

The Intermediate Range Nuclear Instrument Channels provide neutron monitors that provide reactor protective, interlock and indication functions over the intermediate range (Ref. Section 7.4 of the SAR).

The Source Range Nuclear Instrument Channel provides a neutron monitor that is very sensitive to neutrons and thus provides improved indication of the low neutron flux levels present during a reactor startup (Ref. Section 7.4 of the SAR).

The reactor pool temperature instrument is required to ensure that pool temperature does not increase to a level which would jeopardize the ability to cool in-pool components (Ref. Section 7.6.2.2 of the SAR)."

4. *Proposed TS 3.6, "Emergency Electrical Power System," provides requirements for the emergency electrical power system. However, the NRC staff finds that the term "vital equipment," used in the Objective section, does not appear to be not defined in the TSs.*

NUREG-1537, Part 1, Chapter 14, "Technical Specifications," Appendix 14.1, Section 1.2.2, "Format," provides guidance that any information used to support the TSs should be explicitly referenced.

Revise proposed TS 3.6, Objective, to define what components constitute vital equipment at MURR, or justify why no change is needed.

The term "vital equipment" has been removed from the Objective of Technical Specification 3.6. The Objective has been reworded similar to the currently-approved MURR Technical Specifications.

"Objective:
The objective of this specification is to ensure that adequate emergency electrical power is available in the event of a loss of normal electrical power."

5. *Proposed TS 3.8, "Experiements," Specification i, limits the amount of explosive materials which can be irradiated. However, the NRC staff finds that there does not appear to be a limit on the amount of explosive material allowed in the containment building.*

   NUREG-1537, Part 1, Chapter 14, "Technical Specifications," Appendix 14.1, Section 3.8.2, "Materials," provides guidance that the TS should include a limit the amount of explosive material in the reactor facility.

   Revise proposed TS 3.8, Specification i, to incorporate the guidance provided in NUREG-1537, or justify why no change is needed.

   Specification 3.8.i has been revised as follows:

   "Explosive materials shall not be irradiated nor shall they be allowed to generate in any experiment in quantities over 25 milligrams of TNT-equivalent explosives. Explosive materials shall be limited to a total quantity of 100 milligrams of TNT-equivalent explosives in the reactor containment building."

   Establishing an upper limit of 100 milligrams of TNT-equivalent explosives in the reactor containment building follows the guidance of Section 3.8.2 of Appendix 14.1, NUREG-1537, Part 1, Chapter 14.

6. *Proposed TS 3.8, "Experiments," Specification j, requires corrosive materials to be doubly encapsulated. However, the NRC staff finds that there does not appear to be any requirement associated the inspection of potentially damaged components as a result of a failure of encapsulation material.*

   NUREG-1537, Part 1, Chapter 14, "Technical Specifications," Appendix 14.1, Section 3.8.2, "Materials," provides guidance that the failure of an encapsulation of material that could damage the reactor should require removal and physical inspection of potentially damaged components.

   Revise proposed TS 3.8, Specification j, to incorporate the inspection guidance provided in NUREG-1537, or justify why no change is needed.

   Specification 3.8.j has been revised as follows:

   "Corrosive materials shall be doubly encapsulated in corrosion-resistant containers to prevent interaction with reactor components or pool water. Should a failure of the encapsulation occur that could damage the reactor, then the potentially damaged components shall be removed and inspected."

The revision follows the guidance of Section 3.8.2 of Appendix 14.1, NUREG-1537, Part 1, Chapter 14.

7. *Proposed TS 4.0, "Surveillance Requirements," provides general requirements for surviellances. However, the NRC staff finds that there does not appear to be a requirement for a surveillance test following a repair or replacement.*

   NUREG-1537, Part 1, Chapter 14, "Technical Specifications," Appendix 14.1, Section 4, "Surveillance Requirements," provides guidance that states that any time a reactor system or component is modified or repaired, the surveillance for that system should be performed as part of the operability check of the system or component. This should be done regardless of when the surveillance was last performed or when it is next due.

   Revise proposed TS 4.0 to include the guidance in NUREG-1537, or justify why no change is needed.

   "4.0 **General**

   Applicability:
   This specification applies to the surveillance requirements of any system related to reactor safety.

   Objective:
   The objective of this specification is to verify proper operation of any system related to reactor safety.

   Specification:
   a. Surveillance frequencies denoted herein are based on continuing operation of the reactor. Surveillance activities scheduled to occur during an operating cycle which cannot be performed with the reactor operating may be deferred to the end of that current reactor operating cycle. A reactor system or measuring channel shall not be considered operable until it is successfully tested. Any time a reactor system or component is modified or repaired, the surveillance for that system should be performed as part of the operability check of the system or component. This should be done regardless of when the surveillance was last performed or when it is next due. Surveillance intervals shall not exceed those defined by Specification 1.41. Discovery of noncompliance with any of the surveillance specifications listed in this Section shall limit reactor operations to that required to perform the surveillance.

   Bases:
   a. Experience has shown that surveillances will ensure performance and operability of any system related to reactor safety."

8. *Proposed TS 4.1, "Reactor Core Parameters," provides SRs for reactor core parameters. However, the NRC staff finds that the requirements do not appear to include a SR for the reactivity worth of the control blades.*

ANSI/ANS-15.1-2007, Section 4.2, "Reactor control and safety systems," item (1), provides guidance that the reactivity worth of the control rods, including peak worth, should be determined annually to biennially and following significant core configuration and or control rod changes.

Revise proposed TS 4.1 to incorporate the guidance in ANSI/ANS-15.1-2007, or justify why no changes are needed.

The following new Technical Specification (4.2.g) has been added:

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

The following new basis for new Specification 4.2.g has been added:

"g. Measurements of the reactivity worth of the shim blades have shown to vary slightly as a result of absorber burnup and only slightly with respect to operational core loading and experimental changes."

9. *Proposed TS 4.2, "Reactor Control and Reactor Safety Systems," provides SRs for the reactor control and safety systems. However, the NRC staff finds that this specification does not appear to include a SR for an operability test following maintenance or repairs.*

   ANSI/ANS-15.1-2007, Section 4.2, "Reactor control and safety systems," item (6) provides guidance that an operability test of the reactor control and safety systems should be performed following modifications or repairs.

   Revise proposed TS 4.2 to incorporate the guidance in ANSI/ANS-15.1-2007, or justify why no changes are needed.

The following new Technical Specification (4.2.m) has been added:

"m. Following any modifications or repairs on any portion of the reactor control and reactor safety systems, the modified or repaired portion of the system shall be satisfactorily tested before the system is considered operable."

The following new basis for new Specification 4.2.m has been added:

"m. Specification 4.2.m ensures that the modified or repaired system is satisfactorily tested prior to being considered operable."

10. *Proposed TS 4.2, "Reactor Control and Reactor Safety Systems," Specification h, requires a biennially test of the primary coolant relief valves. However, the NRC staff finds that there does not appear to be a corresponding LCO for the primary coolant relief valves. Proposed TS 5.2, "Reactor*

Coolant Systems," Specification k, requires the primary coolant system to contain at least two operable pressure relief valves.

ANSI/ANS-15.1-2007, Section 3.3, "Coolant systems," provides guidance that the minimum operating equipment should be specified in TS LCO Section 3.3.

Revise proposed TS 3.3, "Reactor Coolant Systems," to include the primary coolant relief valves, or justify why no change is needed.

Technical Specification 5.2.k specifically states, "The primary coolant system shall contain at least two (2) operable pressure relief valves." MURR feels that this Specification explicitly requires that the primary coolant system contain two (2) pressure relief valves AND that they shall be are operable. Creating a new Specification in Section 3.3 would merely be duplicating an already existing Specification.

11. *Proposed TS 4.3, "Reactor Coolant Systems," provides SRs for the reactor containment building. However, the NRC staff finds that the SR does not appear to require a leak-tightness test following modifications or repair.*

    ANSI/ANS-15.1-2007, Section 4.4.1, "Containment," item (4) provides guidance that a leak-tightness test should be performed following modifications or repairs that could affect the integrity of the containment boundary.

    Revise proposed TS 4.3 to include the guidance in ANSI/ANS-15.1-2007, or justify why no change is needed.

    The following new Technical Specification (4.4.b) has been added:

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

    The following new basis for new Specification 4.4.b has been added:

    "b. Measurement of the containment building leakage rate following any modification or repair that could affect the leak-tightness of the building ensures that the leakage rate of the structure will remain within the design limits outlined in Specification 5.5.c."

12. *Proposed TS 3.7, "Radiation Monitoring System and Airborne Effluents," Specification b, provides limits for the release of radionuclides from the main exhaust stack. However, the NRC staff finds that a corresponding SR for the TS 3.7, Specification b, appears to be missing. Additionally, the subject header, "Specification," appears to be missing.*

    ANSI/ANS-15.1-2007, Section 4, "Surveillance requirements," provides guidance that TS Section 4, SRs will prescribe the frequency and scope of surveillance to demonstrate the performance criterial

of the LCOs TS Section 3. ANSI/ANS-15.1-2007, Section 1.2.2, "Format," provides guidance that TS Sections 2, 3, and 4, should include the "Specification(s)" section.

Revise proposed TS 4.3, to include a SR for TS 3.7, Specification b, and a "Specification" heading, or justify why no changes are needed.

A "Specification" heading has been added to Technical Specification 3.7.

MURR feels that this question was adequately addressed in the MURR letter to the NRC, dated May 31, 2016. The letter was in response to the NRC's Request for Additional Information, dated October 28, 2015. The following is the NRC's question and MURR's response to that question:

"h. The proposed MURR TS 4.7, provides SRs for the radiation monitoring systems and airborne effluents.

ii. TS LCO 3.7, Specification b, does not appear to have a corresponding SR within TS 4.7.

As described in Section 7.9.5, Off-Gas Radiation Monitoring System, of the SAR, all air exiting the facility is continuously monitored by the "Off-Gas Radiation Monitoring System." The continuous operation of this system, or compensatory measures necessary when the system is secured for calibration or maintenance, is discussed in Note (2) of Specification 3.7.a. The Off-Gas Radiation Monitoring System is displayed in the Control Room and monitored by licensed operators."

13. *Proposed TS 6.1, "Organization," Specification a, Figure 6.0, provides requirements for the MURR organization. However, the NRC staff finds that TS Figure 6.0 does not indicate the organization levels (e.g., Level 1, etc.).*

ANSI/ANS-15.1-2007, Section 6.1.1, "Structure," Figure 1, "Organizational chart," provides guidance for the organizational levels (e.g., Level 1, etc.).

Revise proposed TS 6.1, Figure 6.0, to include the organizational levels, or justify why no change is needed.

The organizational levels have been added to Figure 6.0. Additionally, the Office of the Provost has been replaced by the Office of the Chancellor. As the land grant university and the largest public research university in Missouri, there has been an increased emphasis on the research being conducted by the University of Missouri. Recognizing this increased emphasis the University has elevated the Office of Research from reporting to the Office of the Provost to the Office of the Chancellor. As MURR continues to report to the Office of Research, which now reports directly to the Office of the Chancellor, it is appropriate to revise the organizational chart to reflect this new reporting relationship.

14. *Proposed TS 6.1, "Organization," Specification b, provides a listing of the positions having responsibility for implementing the TSs. However, the NRC staff finds that the organization levels*

(i.e., Level 1, etc.) are not delineated within the list. Additionally, the NRC staff finds that the reactor operators and the organizational level described by Level 1 in ANSI/ANS-15.1-2007 do not appear to be listed.

ANSI/ANS-15.1-2007, Section 6.1.1, "Structure," provides guidance used to define the organizational levels (e.g., Level 1, Level 2, operating staff, etc.).

Revise proposed TS 6.1, Specification b, to include the organizational levels, including Level 1, and the reactor operators, or justify why no changes are needed.

The organizational levels have been added, including Level 1, and the Reactor Operations and Reactor Health Physics Staff.

The Office of the Chancellor has been added to Specification 6.1.b as follows:

"(1) Office of the Chancellor (Level 1): Shall be responsible for directing MU's research mission, the quality and effectiveness of all programs and dedicating university resources necessary to ensure that all research, education and service are conducted in accordance with applicable federal, state and local regulations and accreditation requirements."

The Reactor Operations Staff has been added to Specification 6.1.b as follows:

"(5) Reactor Operations Staff (Level 4): Shall be responsible for the manipulation of reactor controls, monitoring of instrumentation, and operation and maintenance of reactor-related equipment."

The Reactor Health Physics Staff has been added to Specification 6.1.b as follows:

"(6) Reactor Health Physics Staff (Level 4): Shall be responsible for directing research, training, and monitoring programs in order to protect personnel from radiation hazards and to assure compliance with federal, state, and MU regulations."

15. *Proposed TS 6.1, "Organization," Specification e, provides the requirements for the presence of a Senior Reactor Operator (SRO) at the facility. However, the NRC staff finds that the requirements associated with control rod relocations and movement of an experiment with a reactivity worth greater than one dollar do not appear to be listed in the specification. Furthermore, the NRC staff finds that the requirement associated with the "non-emergency power reduction," is not clearly understood by the NRC staff.*

    ANSI/ANS-15.1-2007, Section 6.1.3, "Staffing," item (3), provides guidance that an SRO should be at the facility during control rod relocations within the reactor core region and relocation of any experiment with reactivity worth greater than one dollar. ANSI/ANS-15.1-2007 does not provide guidance for a non-emergency power reduction.

Revise proposed TS 6.1, Specification e, to include the guidance in ANSI/ANS-15.1-2007 for control rod relocations within the reactor core region and relocation of any experiment with reactivity worth greater than one dollar, or justify why no change is needed. Explain the non-emergency power reduction requirement for the presence of an SRO in contrast to an emergency power reduction.

Specification 6.1.e has been revised as follows:

"e. Senior Reactor Operator licensed pursuant to 10 CFR 55 shall be present at the facility or readily available on call at all times during operation. Readily available on call means an individual who:

(1) Has been specifically designated and the designation known to the operator on duty;

(2) Can be rapidly contacted by phone, by the operator on duty; and

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

New Specification 6.1.f has been added as follows:

"f. Events requiring the presence of a Senior Reactor Operator at the facility are:

(1) Initial startup and approach to power;

(2) All fuel or control rod relocations within the reactor core region;

(3) Relocation of any experiment with a reactivity worth greater than 0.0074 $\Delta k/k$; and

(4) Recovery from an unplanned or unscheduled shutdown or significant power reduction."

These changes follow the Section 6.1.3 of ANSI/ANS-15.1-2007.

16. *Proposed TS 6.2, "Review and Audit," Specification a, provides a description of the Reactor Advisory Committee (RAC). However, the NRC staff finds that the composition and qualifications of the RAC do not appear to be described in the specification.*

ANSI/ANS-15.1-2007, Section 6.2.1, "Composition and qualifications," provides guidance for the composition and qualifications for the oversight committee, including the use of alternate members. The guidance in ANSI/ANS-15.1-2007 also includes: (1) the minimum number of members for a committee and subcommittee; (2) the background and expertise of the members; (3) the authority responsible for the appointment of members (e.g., Level 1); (4) the organizational level to which the committee reports (e.g., Level 1); and (5) the use of alternates.

Revise proposed TS 6.2, Specification a, to include the guidance in ANSI/ANS-15.1-2007, Section 6.2.1, or justify why no change is needed.

Technical Specification 6.2.a has been revised as follows:

"A Reactor Advisory Committee (RAC) shall provide independent oversight in matters pertaining to the safe operation of the reactor and with regard to planned research activities and use of the facility building and equipment. The RAC shall be composed of at least five (5) members who have knowledge of experimental activities, reactor operations, University business policy, or related subjects. The Committee members shall be appointed by, and report to, the Office of the Chancellor. The RAC shall review:"

MURR does not propose the use of qualified and approved alternates for the RAC.

17. *Proposed TS 6.2, "Review and Audit," Specification a.(1), provides requirements associated with changes to the MURR facility equipment, systems, and procedures. However, the NRC staff finds that this TS may be missing some of the criterial provided in the guidance in ANSI/ANS-15.1-2007, Section 6.2.3, "Review function," as described below:*

   a. *ANSI/ANS-15.1-2007, Section 6.2.3.(1) provides guidance that the proposed changes to equipment, systems or procedures should also include "tests and experiments."*

   b. *ANSI/ANS-15.1-2007, Section 6.2.3.(4), provides guidance that the review should include all proposed changes in technical specifications, license, or charter.*

   c. *ANSI/ANS-15.1-2007, Section 6.2.3.(1), provides guidance that the review should include a review of the "determinations" (as opposed to the actual change) that the proposed changes were allowed without prior NRC.*

   Revise proposed TS 6.2, Specification a.(1) to include the guidance in ANSI/ANS-15.1-2007, Section 6.2.3, or justify why no change is needed.

Technical Specifications 6.2.a(1) through (4) have been revised as follows:

"(1) Determinations that proposed changes to MURR equipment, systems, tests, experiments or procedures are allowed pursuant to 10 CFR 50.59.

(2) All new procedures and major revisions thereto having safety significance, proposed changes to reactor facility equipment, or systems having safety significance. Changes to procedures that do not change their original intent may be made without prior RAC review if approved by the TS-designated manager, either the Reactor Health Physics Manager or Reactor Manager, or a designated alternate who is a member of Reactor Health Physics or a Senior Reactor Operator, respectively. All such changes to the procedures shall be documented, reviewed pursuant to 10 CFR 50.59, and subsequently reviewed by the RAC;

(3) Proposed experiments significantly different from any previously reviewed or which involve a question pursuant to 10 CFR 50.59;

(4) Proposed changes in the Technical Specifications or the license;"

18. *Proposed TS 6.2, "Review and Audit," Specification a.(2), provides requirements associated with proposed experiments that are significantly different from any previously reviewed. However, the NRC staff finds that specification may not be consistent with the guidance in ANSI/ANS-15.1-2007.*

ANSI/ANS-15.1-2007, Section 6.2.3, "Review function," item (3), provides guidance that the review should include all new experiments or classes of experiments that could affect reactivity or result in the release of radioactivity.

Revise proposed TS 6.2.3, Specification a.(2), to include the guidance in ANSI/ANS-15.1-2007, Section 6.2.3, item (3), or justify why no change is needed.

MURR feels that adding the words "that could affect reactivity or result in a release of radioactivity" is redundant because all of the Specifications in Section 3.8 are reviewed when performing a safety evaluation for an experiment, not only the reactivity and release Specifications; therefore, the addition of these words is unnecessary.

19. *Proposed TS 6.2, "Review and Audit," Specification b, provides requirements for the RAC and subcommittee meetings. However, the NRC staff finds that this specification may not be consitent with the guidance provided in ANSI/ANS-15.1-2007, Section 6.2.3, as it appears to be missing requirements for the review and approval of the minutes.*

ANSI/ANS-15.1-2007, Section 6.2.3, "Review function," provides guidance that a written report or minutes of the findings and recommendations of the review group shall be submitted to Level 1 and the review and audit group members in a timely manner after the review has been completed.

Revise proposed TS 6.2, Specification b, to include the guidance in ANSI/ANS-15.1-2007, Section 6.2.3, or justify why no change is needed.

The following sentence has been added to the second paragraph of Technical Specification 6.2.b:

"Dissemination of the minutes to the Office of the Chancellor, the RAC and its subcommittees shall be done within three (3) months after the meetings."

20. *Proposed TS 6.2, "Review and Audit," Specification e.(1), provides requirements for the audit functions. However, the NRC staff finds that this specification may not be consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007, Section 6.2.4, "Audit function," item (4), as the emergency plan and security plan do not appear to be included.*

ANSI/ANS-15.1-2007, Section 6.2.4, item (4) provides guidance that the reactor facility emergency plan and implementing procedures be audited biennially. NUREG-1537, Part 1, Chapter 14, "Technical Specifications," Appendix 14.1, Section 6.2.4, provides guidance that all required plans, including the Emergency Plan and Security Plan, be audited.

Revise proposed TS 6.2, Specification e.(1), to include the guidance in ANSI/ANS-15.1-2007, Section 6.2.4, item (4) and NUREG-1537, Part 1, Chapter 14, "Technical Specifications," Appendix 14.1, Section 6.2.4, to include biennial audits of the emergency and security plans, or justify why no change is needed.

The requirements to audit the MURR Physical Security Plan and its Implementing Procedures are stated in Section B.7 of the Physical Security Implementing Procedures. Since these documents are classified as "Safeguards Information," only individuals that have met the requirements of 10 CFR 73.57, "Requirements for criminal history records checks of individuals granted unescorted access to a nuclear power facility, a non-power, or access to Safeguards Information" are allowed to review these documents. Very few members of the RAC and its subcommittees have gone through this vetting process nor have any experience in security and its regulations. MURR feels that the requirement of Section B.7 of the Implementing Procedures to annual audit of the Physical Security Plan and its Implementing Procedures by individuals that have been vetted and are familiar with the regulations regarding security is more than satisfactory to meet the intent of ANSI/ANS-15.1-2007, Section 6.2.4, item (4) and NUREG-1537, Part 1, Chapter 14, "Technical Specifications," Appendix 14.1.

As stated in proposed Technical Specification 6.4.c, the Reactor Manager shall annually review the Emergency Plan Implementing Procedures. Section 8.3 of the Emergency Plan states that the Emergency Plan and its Implementing Procedures shall be reviewed annually and revised as necessary. The Emergency Plan is an NRC-approved document and compliance with it is the same as compliance with the Technical Specifications. Therefore, MURR feels that it is redundant and not necessary to place the same requirement in two different NRC-approved documents that MURR must adhere to.

21. *Proposed TS 6.2, "Review and Audit," Specification e.(2), provides requirements for the audit findings to be immediately reported to the Reactor Facility Director. However, the NRC staff finds that this specification may not be consistent with the guidance in ANSI/ANS-15.1-2007, Section 6.2.4, "Audit function."*

ANSI/ANS-15.1-2007, Section 6.2.4, provides guidance that deficiencies uncovered that affect reactor safety shall immediately be reported to Level 1 management. A written report of the findings of the audit shall be submitted to Level 1 management and the review and audit group members within 3 months after the audit has been completed.

Revise proposed TS 6.2, Specification e.(2) to include the guidance in ANSI/ANS-15.1-2007, Section 6.2.4, or justify why no change is needed.

Technical Specification 6.2.e(2) has been revised as follows:

"(2) Audit findings which affect reactor safety shall be immediately reported to the Reactor Facility Director. A written report of the findings shall be submitted to the Reactor Facility

Director, the RAC and its subcommittees within three (3) months after the audit has been completed."

MURR feels that is more appropriate that audit findings which affect reactor safety immediately be reported to Level 2 management instead of Level 1 since Level 2 management is much more familiar with the day-to-day operations of the facility than Level 1 management. Level 2 management would also be able to immediately provide the necessary resources to correct any deficiencies.

22. *Proposed TS 6.4, "Procedures," Specification a.(6), provides requirements for procedures for implementation of the emergency plan. However, the NRC staff finds that this specification may not be consistent with the guidance in ANSI/ANS-15.1-2007, Section 6.4.(7).*

    ANSI/ANS-15.1-2007, Section 6.4, "Procedures," item (7), provides guidance that procedures shall include implementation of required plans such as emergency or security plans.

    Revise proposed TS 6.4, Specification a.(6), to include the security plan, or justify why no change is needed.

    Technical Specification 6.4.a(6) has been revised as follows:

    "(6) Implementation of the Emergency and Physical Security Plans."

23. *Proposed TS 6.4, "Procedures," Specification d, provides requirements for deviations from procedures. However, the NRC staff finds that this specification may not be consistent with the regulatory requirements of 10 CFR 50.59 which allows a licensee to make changes to the procedures as described in the final safety analysis report (as updated).*

    Revise proposed TS 6.4, Specification d, to include consideration of the requirments of 10 CFR 50.59, or justify why no change is needed.

    Technical Specification 6.4.d has been revised as follows:

    "d. Deviations from procedures required by this Specification may be enacted by a Senior Reactor Operator or member of Reactor Health Physics, as applicable. Such deviations shall be documented, reviewed pursuant to 10 CFR 50.59, and reported within 24 hours or the next working day to the Reactor Manager or Reactor Health Physics Manager or designated alternate."

24. *Proposed TS 6.6, "Reportable Events and Required Actions," Specification a.(2), provides reporting requirements for a safety limit violation. However, the NRC staff finds that this specification may not be consistent with the guidance in ANSI/ANS-15.1-2007, Section 6.6.1, "Action to be taken in case of safety limit violation," item (2), as it does not appear to require reporting to managment.*

    ANSI/ANS-15.1-2007, Section 6.6.1, item (2), provides guidance that a safety limit violation should be promptly reported to the Level 2, or designated alternates.

Revise proposed TS 6.6, Specification a.(2), to include the guidance in ANSI/ANS-15.1-2007, Section 6.6.1, item (2), or justify why no change is needed.

New Technical Specification 6.6.a(2) has been added as follows:

"(2) The safety limit violation shall be promptly reported to the Reactor Manager and Reactor Facility Director, or designated alternates;"

The remainder of the Specifications in Section 6.6.a have been renumbered.

25. *Proposed TS 6.6, "Reportable Events and Required Actions," Specification b.(2), provides reporting requirements for a release of radioactivity from the site greater than the allowable limits from the facilty boundary. However, the NRC staff finds that this specification may not be consistent with the guidance in ANSI/ANS-15.1-2007, Section 6.6.2, "Action to be taken in the event of an occurrence of the type identified in Secs. 6.7.2(1)(b) and 6.7.2(1)(c)," item (2), as it does not appear to require reporting to management.*

    ANSI/ANS-15.1-2007, Section 6.6.2, item (2), provides guidance that a release of radioactivity from the site above allowable limit should be promptly reported to the Level 2, or designated alternates.

    Revise proposed TS 6.6, Specification b.(2), to include the guidance in ANSI/ANS-15.1-2007, Section 6.6.2, item (2), or justify why no change is needed.

    New Technical Specification 6.6.b(2) has been added as follows:

    "(2) The release of radioactivity shall be promptly reported to the Reactor Manager and Reactor Facility Director, or designated alternates;"

    The remainder of the Specifications in Section 6.6.b have been renumbered.

26. *Proposed TS 6.6, "Reportable Events and Required Actions," Specification b.(3), provides the requirement that, reactor operation, following a shutdown due to a release of radioactivity greater than the allowable limits, may not resume until authorized by the Reactor Manager. However, the NRC staff finds that this specification may not be consistent with the guidance in ANSI/ANS-15.1-2007, Section 6.6.2, "Action to be taken in the event of an occurrence of the type identified in Secs. 6.7.2(1)(b) and 6.7.2(1)(c)," item (1).*

    ANSI/ANS-15.1-2007, Section 6.6.2, item (1), provides guidance that the reactor operations shall not be resumed unless authorized by Level 2 or designated alternates.

    Revise proposed TS 6.6, Specification b.(3) to include the guidance in ANSI/ANS-15.1-2007, Section 6.6.2, item (1), or justify why no change is needed.

Technical Specification 6.6.b(4) has been revised as follows:

"(4) If it is necessary to shut down the reactor to correct the occurrence, operations shall not be resumed until authorized by the Reactor Facility Director, or designated alternate; and"

27. *Proposed TS 6.6, "Reportable Events and Required Actions," Specification c.(1), provides reporting requirements for Abnormal Occurrences. However, the NRC staff finds that this specification may not be consistent with the guidance in ANSI/ANS-15.1-2007, Section 6.6.2, "Action to be taken in the event of an occurrence of the type identified in Secs. 6.7.2(1)(b) and 6.7.2(1)(c)," item (2), as it does not appear to require reporting to management.*

    ANSI/ANS-15.1-2007, Section 6.6.2, item (2), provides guidance that an abnormal occurrence be promptly reported to the Level 2, or designated alternates.

    Revise proposed TS 6.6, Specification c.(1), to include the guidance in ANSI/ANS-15.1-2007, Section 6.6.2, item (2), or justify why no change is needed.

    New Technical Specification 6.6.c(2) has been added as follows:

    "(2) The Abnormal Occurrence shall be promptly reported to the Reactor Manager and Reactor Facility Director, or designated alternates;"

    The remainder of the Specifications in Section 6.6.c have been renumbered.

28. *Proposed TS 6.6, "Reportable Events and Required Actions," Specification c.(3), provides the requirement that reactor operation, following a shutdown due to an Abnormal Occurrence, may not resume until authorized by the Reactor Manager. However, the NRC staff finds that this specification may not be consistent with the guidance in ANSI/ANS 15.1-2007, Section 6.6.2, "Action to be taken in the event of an occurrence of the type identified in Secs. 6.7.2(1)(b) and 6.7.2(1)(c)," item (1).*

    ANSI/ANS-15.1-2007, Section 6.6.2, item (1) provides guidance that the reactor operations shall not be resumed unless authorized by Level 2 or designated alternates.

    Revise proposed TS 6.6, Specification c.(3) to include the guidance in ANSI/ANS-15.1-2007, Section 6.6.2, item (1), or justify why no change is needed.

    Technical Specification 6.6.c(4) (old 6.6.c.3) has been revised as follows:

    "(4) The reactor shall be shut down or placed in a safe condition and return to normal reactor operations will not be allowed until authorized by the Reactor Facility Director, or alternate."

29. *Proposed TS 6.6, "Reportable Events and Required Actions," Specification d.(2), provides requirements for a written report if permanent changes occur in the facility organization involving the Office of the Provost or the Director's Office. However, the NRC staff finds that this specification*

may not be consistent with the guidance in ANSI/ANS-15.1-2007, Section 6.7.2, "Special reports," item (2)(a).

ANSI/ANS-15.1-2007, Section 6.7.2, item (2)(a).provides guidance that a written report is required when permanent changes occur in the facility organization involving Level 1 or 2 personnel. The NRC staff is not clear which positions are Level 1 or Level 2.

Revise proposed TS 6.6, Specification d.(2), to include the guidance in ANSI/ANS-15.1-2007, or justify why no change is needed.

Technical Specification 6.6.d(2) has been revised as follows:

"(2) Permanent changes in the facility organization involving the Office of the Chancellor or the Reactor Facility Director."

30. *Proposed TS 6.6, "Reportable Events and Required Actions," Specification e.(1) provides requirements for the Annual Reports. However, the NRC staff finds that this specification may not be consistent with the guidance in ANSI/ANS-15.1-2007, Section 6.7.1, "Operating reports," item (4), as annual reporting of new tests and experiments do not appear to be required.*

ANSI/ANS-15.1-2007, Section 6.7.1, item (4), provides guidance that Operating Reports include a tabulation of new test and experiments.

Revise proposed TS 6.6, Specification e.(1), to include the guidance in ANSI/ANS-15.1-2007, Section 6.7.1, item (4) to include new tests and experiments, or justify why no change is needed.

Proposed Technical Specification 6.6.e(5) states:

"(5) A summary of each modification to the reactor facility or change to the procedures, tests and experiments carried out under the conditions of 10 CFR 50.59;"

MURR feels that this Specification adequately meets the guidance of ANSI/ANS-15.1-2007, Section 6.7.1, "Operating reports," item (4).

The additional information/clarification that was requested during the multiple conference calls between MURR and NRC staff:

Comment AA1 – New TS 5.3.h has been added as follows:

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

Comment AA2 – TS 3.1.c was revised as follows:

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

Comment AA3 – In TS 3.1.d, the word "will" was replaced by "shall."

Comment AA4 – TS 5.3.k was revised as follows:

"j. The reactor shall have five (5) control blades between the pressure vessel and beryllium reflector. Four (4) of the control blades shall be made of boron and aluminum for coarse control (shim blades) of reactor power. One (1) control blade shall be made of stainless steel for fine control (regulating blade) of reactor power."

Comment WG6 & WG7 – TS 4.2.e and 4.2.f was revised as follows:

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

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

Comment AA10 & AA11 – TS 3.1.a and 3.2.b were revised as follows:

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

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

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

Comment WG13 – The Reactor Coolant Cleanup System is described in detail in Section 5.5 of the SAR. Although this section refers to a regeneration station (R-200), regeneration of resin has not been performed at MURR in over 30 years. The regeneration station is now used as a central transfer point, allowing resin to be transferred to or from any of the demineralizer tanks, including the resin storage tank. The regenerator tank is used for both dumping depleted resin and loading new resin.

Comment AA14 – The word "shall" has been added to the Specification section of TS 5.2.

Comment AA16 – TS 5.2.i has been revised as follows:
"i. The reactor shall have a natural convection coolant flow path for Mode III operation."
Additionally, all references for operation with the reactor subcritical by a margin of at least 0.15 $\Delta k/k$ have been removed from the Technical Specifications.

Comment AA17 – The word "operable" has been replaced by "operating" in TS 3.3.b(1).

Comment AA18 – TS 3.3.f has been revised as follows:
"f. The reactor shall not be operated if a radiochemical analysis of the secondary coolant system exceeds the limits of 10 CFR 20, Appendix B, Table 2, Column 2."
Additionally, new TS 3.3.e and its Basis have been added as follows:
"e. The reactor shall not be operated with forced circulation unless:
(1) The continuous secondary coolant system monitor is operating,

OR

(2) The secondary coolant system is sampled and analyzed for gross radioactivity at least daily."
"e. Specification 3.3.e provides for the early detection of a leaking reactor coolant system heat exchanger so that corrective actions can be taken."

Comment AA19 – New TS 3.3.d and its Basis has been added as follows:
"d. The reactor shall not be operated if a radiochemical analysis of the pool coolant system indicates gross radioactivity twice the historical average."
"d. Detection of the deterioration of components in the pool coolant system and in-pool experimental facilities is provided by Specification 3.3.d (NUREG-1537)."

Comment AA20 – TS 3.3.g, 3.3.h and 3.3.i have been changed from monthly to quarterly.

Comment AA22 – The basis for the Decay Heat Removal System as stated in Section 5.8.1 of the SAR, is the original design basis for fabrication of the system as designed by Internuclear in the 1960s. The 30 days of continuous 10 MW operation is an ultra-conservative, upper bound assumption in the design because MURR never operates greater than 7 days continuous.

Comment AA23, AA24 & AA25 – The word "shall" has been inserted into the Specification section of TS 5.5.

Comment AA26 – Last sentence of TS 5.5.c has been moved to TS 4.4.a.

Comment AA27 – The word "shall" replaced "must" in TS 3.4.a.

Comment AA29 – The words "special maintenance" has been replaced by "repairs or modifications" in TS 4.4.a.

Comment AA37 & AA38 – The "certain control system functions" of TS 3.2.h have been added to its Basis.

Comment AA41 – TS 4.2.i has been revised as follows:

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

Comment AA44 & AA45 – TS 3.5.b has been revised as follows:

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

Comment AA47, AA48, AA49 and AA50 – TS 3.5.c has been deleted.

Comment AA54 – TS 5.4.a has been revised as follows:

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

Comment AA55 – TS 5.4.b has been revised as follows:

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

Comment AA57 – The word "shall" replaced "will" in TS 4.1.d.

Comment AA59 – The absolute reactivity worth of each secured removable experiment is limited to 0.006 $\Delta k/k$ – a value which has been shown not to cause any fuel damage if accidentally moved while the reactor is operating. By Definition 1.36, secured experiments are rigidly held in place by mechanical means and cannot be moved (inserted or removed) while the reactor is operating. In addition, since these experiments are rigidly help in place, accidental simultaneous removal of multiple secured experiments is not a credible scenario. Due to these reasons, MURR feels that there is no need to put a limit on the total reactivity worth of all secured experiments. The limits on core excess reactivity and shutdown margin will limit the total worth of experiments that can be installed outside of the beryllium reflector.

Comment WG62 – TS 3.8.l has been revised as follows:

"l. Fluids utilized in beamport loop experiments shall be of types which will not chemically react in the event of leakage and shall be maintained at pressure and temperature conditions such that the integrity of the beam tube will not be impaired in the event of loop rupture."

Comment WG64 – The words "Appendix B, Table 1" has been deleted from TS 3.8.n.

Comment AA65 – TS 3.8.p has been revised as follows:

"p. Experiments shall be designed such that a failure of an experiment will not lead to a direct failure of another experiment, a failure of a reactor fuel element, or to interfere with the action of the reactor safety and reactor control systems or other operating components."

Comment AA66 – TS 4.8.a has been revised as follows:

"a. The criteria of Specification 3.8 shall be evaluated and found acceptable prior to inserting an experiment in the reactor or its experimental facilities."

Comment WG68 – The word "operable" has been replaced by "operating" in TS 3.7.a.

Comment AA70 – The word "will" has been replaced by "shall" in TS 3.7.a note 1.

Comment AA71 – TS 3.7.a note 2 has been revised as follows:

"(2) The stack radiation monitor may be placed out of service for up to two (2) hours for calibration and maintenance. During this out-of-service time, no experimental or maintenance activities shall be conducted which are likely to result in the release of airborne radioactivity."

Comment WG73 – The Basis for TS 3.7.b has been revised as follows:

"b. For the purposes of Specification 3.7.b, air effluents for particulates and halogens with half-lives greater than 8 days are limited to the Air Effluent Concentrations (AEC) without the inclusion of a dilution multiplier to minimize any chance of reconcentration at the receptor site resulting in doses in excess of the direct exposures via air concentrations. Data from Soldat, JD (Health Physics 9, p. 1170, 1963), suggest a reconcentration factor of approximately 400 for the Iodine-131 milk/man pathway; however, dilution of the stack effluent to the nearest residence due north of MURR (760 meters), the prevailing wind direction, is approximately 1900, thus giving a safety factor (ratio) of 4.75. This value is also conservative in that the wind blows from 360 degrees around MURR throughout the year and thus this value represents a worst case scenario to only the maximally exposed receptor point.

For Argon-41, the primary air effluent from MURR, dispersion calculations are based on standard reference material and experimental data obtained at the reactor showing that concentrations under average conditions will be 0.008 of the AEC limits in the unrestricted area surrounding the reactor facility. Also, dilution factors under conservative conditions are in the range of 5×10^4 under both average and stable conditions at ground level from the facility building. For normal short burst releases at the facility which are five to ten seconds in duration and occur on an average of ten times per day five days per week the effect on the average concentration is less than 1% when averaged over a one-day period.

It is concluded that these concentrations as specified will not constitute a hazard to the health and safety of the public."

Comment WG76, AA77 & AA78 – The words "shall be" have been added to TS 6.1.b(1), 6.1.b(2), 6.1.b(3) and 6.1.b(4).

Comment AA78 – The word "will" has been replaced by "shall" in TS 6.1.c.

Comment AA82 – The "office of the Provost" has been replaced by the "Office of the Chancellor" as the Level 1. As the land grant university and the largest public research university in Missouri, there

has been an increased emphasis on the research being conducted by the University of Missouri. Recognizing this increased emphasis the University has elevated the Office of Research from reporting to the Office of the Provost to the Office of the Chancellor. As MURR continues to report to the Office of Research, which now reports directly to the Office of the Chancellor, it is appropriate to revise the organizational chart to reflect this new reporting relationship.

Comment AA90 & WG92 – The words "are," "will" and may have been replaced by "shall" in TS 6.2.b. TS 6.2.b has been revised as follows:

"b. The RAC may appoint subcommittees consisting of knowledgeable members of the public, students, faculty, and staff of MU when it deems it necessary in order to effectively discharge its primary responsibilities. When subcommittees are appointed, these subcommittees shall consist of no less than three (3) members with no more than one (1) student appointed to each subcommittee. The subcommittees may be authorized to act on behalf of the RAC.

The RAC and its subcommittees shall maintain minutes of meetings in which the items considered and the committees' recommendations are recorded. Dissemination of the minutes to the Office of the Chancellor, the RAC and its subcommittees shall be done within three (3) months after the meetings. Independent actions of the subcommittees shall be reviewed by the parent committee at the next regular meeting. A quorum of the committee or the subcommittees consisting of at least fifty percent of the appointed members shall be present at any meeting to conduct the business of the committee or subcommittee. Additionally, reactor facility staff shall not constitute greater than fifty percent of the quorum. Reactor facility staff shall not constitute a majority of the RAC. The RAC shall meet at least quarterly.

A meeting of a subcommittee shall not be deemed to satisfy the requirement of the parent committee to meet at least once during each calendar quarter."

Comment AA91 – Minimum RAC membership is listed in TS 6.2.a.

Comment AA93 – The sentence "Additionally, reactor facility staff shall not constitute greater than fifty percent of the quorum." has been added to TS 6.2.b.

Comment WG94 – The words "timely manner" has been changed to "within three (3) months" in TS 6.2.b.

Comment AA102 – MURR has historically interpreted "radiological control" to include procedures that involve the use or handling of byproduct material at quantities sufficient to present a radiological hazard. As such MURR has required all byproduct procedures where radiological controls are required to ensure the safety of the worker to be reviewed by the Health Physics Manager and a subcommittee of the Reactor Advisory Committee. We believe that MURR's historical interpretation encompasses the intent of the NRC's proposal to include the wording "use of byproduct material."

Comment AA105 – The words "and subsequently confirmed in writing" have been added to TS 6.6.a(3).

Comment AA106 – The word "will" has been replaced by "shall" in TS 6.6.a(5).

Comment AA109 – The words "and subsequently confirmed in writing" have been added to TS 6.6.b(3).

Comment AA111 – The word "will" has been replaced by "shall" in TS 6.6.b(5).

Comment AA113 – The word "violation" has been replaced by "Abnormal Occurrence" in 6.6.c(1).

Comment AA114 – The words "and subsequently confirmed in writing" have been added to TS 6.6.c(1).

Comment AA15 & AA116 – The word "will" has been replaced by "shall" in TS 6.6.c(3) and 6.6.c(4).

Comment AA122 – The word "are" has been replaced by "shall" in TS 6.7(a).

Comment AA123 – New TS 6.7.a(5) has been added as follows:
"(5) Reviews and reports pertaining to a violation of a safety limit, limiting safety system setting, or limiting conditions for operations."

Comment AA124 – The word "are" has been replaced by "shall" in TS 6.7(b).

Comment AA129 – Specification 1.35 has been revised as follows:
"1.35 **Scram Time** - Scram time is the elapsed time between the initiation of a scram signal and insertion of the shim blades to the 20% withdrawn position."

Comment AA130 – Specification 1.36 has been revised as follows:
"1.36 **Secured Experiment** - A secured experiment is any experiment, experimental apparatus, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining forces must be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces that are normal to the operating environment of the experiment, or by forces that can arise as a result of credible malfunctions."

Comment AA131, AA1132 & AA133 – TS 4.0 has been revised as follows:
"4.0 **General**

Applicability:
This specification applies to the surveillance requirements of any system related to reactor safety.

Objective:
The objective of this specification is to verify proper operation of any system related to reactor safety.

Specification:

a. Surveillance frequencies denoted herein are based on continuing operation of the reactor. Surveillance activities scheduled to occur during an operating cycle which cannot be performed with the reactor operating may be deferred to the end of that current reactor operating cycle. A reactor system or measuring channel shall not be considered operable until it is successfully tested. Any time a reactor system or component is modified or repaired, the surveillance for that system should be performed as part of the operability check of the system or component. This should be done regardless of when the surveillance was last performed or when it is next due. Surveillance intervals shall not exceed those defined by Specification 1.41. Discovery of noncompliance with any of the surveillance specifications listed in this Section shall limit reactor operations to that required to perform the surveillance.

Bases:

a. Experience has shown that surveillances will ensure performance and operability of any system related to reactor safety."

Comment WG136 – General statement in Section 5 has been deleted.

Other Changes:

1. Page A-7 – Specification 1.35 has been revised to Specification 1.36 in Definition 1.44, Unsecured Experiment.

2. Page A-7 – Surveillance Interval "Daily" has been added to Specification 1.41 as follows:

   "g. Daily - interval not to exceed 1 calendar day."

3. Page A-7 – Surveillance Interval "Within a shift" has been revised to:

   "h. Within a shift - interval not to exceed the reactor shift."

4. Page A-39 – Specification 1.40 has been revised to Specification 1.41 in TS 4.0, General.

5. Page A-44 – The bases for Technical Specification 4.3.c improperly referenced Specification 4.2.b. It has been corrected to reference Specification 4.3.c.

6. Page A-44 – The bases for Technical Specification 4.3.h improperly referenced Specification 4.3.g. It has been corrected to reference Specification 4.3.h.

7. Page A-45 – The bases for Technical Specification 4.4.d has been corrected to reference Specification 4.4.d because of the addition of new Technical Specification 4.4.b.

8. Page A-70 – Changed "Director's Office" to "Reactor Facility Director" in Technical Specification 6.6.d(2).

9. The reactivity value for a significant core configuration change has been revised from 0.0013 to 0.002 $\Delta k/k$ in TS 4.1.a and 4.1.b. This corresponds to approximately $0.25, or 1/4 of beta effective.

Attachment 1

APPENDIX A

TECHNICAL SPECIFICATIONS

FOR

THE UNIVERSITY OF MISSOURI RESEARCH REACTOR

FACILITY OPERATING LICENSE No. R-103
DOCKET No. 50-186

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Introduction

The Technical Specifications represent the administrative controls, equipment availability, operational conditions and limits, and other requirements imposed on reactor facility operation in order to protect the environment and the health and safety of the facility staff and the general public in accordance with 10 CFR 50.36.

This document is divided into the following six sections:

Section 1 - Definitions
Section 2 - Safety Limits (SL) and Limiting Safety System Settings (LSSS)
Section 3 - Limiting Conditions for Operations (LCO)
Section 4 - Surveillance Requirements
Section 5 - Design Features
Section 6 - Administrative Controls

Specific limitations and equipment requirements for safe reactor operation and for dealing with abnormal situations are called specifications. These specifications, typically derived from the facility descriptions and safety considerations contained in the Safety Analysis Report (SAR), represent a comprehensive envelope of safe operation. Only those operational parameters and equipment requirements directly related to preserving that safe envelope are listed in the Technical Specifications. Procedures or actions employed to meet the requirements of these Technical Specifications are not included in the Technical Specifications. Normal operation of the reactor within the limits of the Technical Specifications will not result in off-site radiation exposure in excess of 10 CFR 20 guidelines.

Specifications in Sections 2, 3, 4 and 5 provide related information in the following format shown:

- **Applicability** - This indicates which components are involved;
- **Objective** - This indicates the purpose of the specification(s);
- **Specification(s)** - This provides specific data, conditions, or limitations that bound a system or operation. This is the most important statement in the Technical Specifications; and
- **Bases** - This provides the background or reasoning for the choice of specification(s), or references a particular section of the SAR that does.

Section 6, Administrative Controls, simply state the applicable specification(s).

Although the applicability, objective and bases provide important information, only the "specification(s)" statement is governing.

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1 DEFINITIONS

1.1 **Abnormal Occurrences** - An abnormal occurrence is any of the following which occurs during reactor operation:

a. Operation with actual safety system settings for required systems less conservative than specified in Section 2.2, Limiting Safety System Settings;

b. Operation in violation of Limiting Conditions for Operations established in Section 3.0;

c. A reactor safety system component malfunction which renders or could render the reactor safety system incapable of performing its intended safety function. If the malfunction or condition is caused by maintenance, then no report is required;

d. An unanticipated or uncontrolled change in reactivity in excess of 0.006 $\Delta k/k$. Reactor trips resulting from a known cause are excluded;

e. Abnormal and significant degradation in reactor fuel or cladding, or both, primary coolant boundary, or containment boundary (excluding minor leaks) where applicable; or

f. An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence or development of an unsafe condition involving operation of the reactor.

1.2 **Center Test Hole** - The center test hole is that volume in the flux trap occupied by the removable experiment sample canister.

1.3 **Channel** - A channel is the combination of sensor, line, amplifier, and output devices that are connected for the purpose of measuring the value of a parameter.

1.4 **Channel Calibration** - A channel calibration is an adjustment of the channel such that its output corresponds with acceptable accuracy to known values of the parameter that the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip, and shall be deemed to include a channel test.

1.5 **Channel Check** - A channel check is a qualitative verification of acceptable performance by observation of channel behavior. This verification, where possible, shall include comparison of the channel with other independent channels or systems measuring the same variable.

1 **DEFINITIONS** - Continued

1.6 **Channel Test** - A channel test is the introduction of a simulated input signal into channel and the observation of proper channel response. When applicable, the test shall include verification of proper safety trip operation.

1.7 **Control Blade (Rod)** - A control blade (rod) is either a shim blade (rod) or the regulating blade (rod). The words blade and rod can be used interchangeably.

1.8 **Core Configuration** - The core configuration includes the number, type, or arrangement of fuel elements, reflector elements, and control blades occupying the core region.

1.9 **Excess Reactivity** - Excess reactivity is that amount of positive reactivity that would exist if all of the control blades were moved to the fully withdrawn position from the point where the reactor is exactly critical ($K_{eff} = 1$) at reference core conditions.

1.10 **Experiment** - An experiment is any operation, hardware, or target (excluding devices such as detectors or foils) which is designed to investigate non-routine reactor characteristics or which is intended for irradiation within an irradiation facility. Hardware rigidly secured to a core or shield structure so as to be part of their design to carry out experiments is not normally considered an experiment.

1.11 **Flux Trap** - The flux trap is that portion of the reactor through the center of the core bounded by the 4.5-inch inside diameter tube and 15 inches above and below the reactor core horizontal center line.

1.12 **Irradiated Fuel** - Irradiated fuel is any fuel element which has been irradiated and used to an integrated power of:

a. Greater than 0.10 megawatt-day;

OR

b. Less than or equal to 0.10 megawatt-day but greater than 1.0 kilowatt-day and with a decay time of less than 7 days since last irradiation;

OR

c. Less than or equal to 1.0 kilowatt-day and with a decay time of less than 24 hours since last irradiation.

1.13 **Limiting Safety System Settings** - Limiting Safety System Settings (LSSS) are settings for automatic protection devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting shall be so chosen that automatic protective action will correct the most severe abnormal situation anticipated before a safety limit is exceeded.

1 **DEFINITIONS** - Continued

1.14 **Movable Experiment** - A movable experiment is one which is designed with the intent that it may be moved into, out of, or in the near proximity of the reactor while the reactor is operating.

1.15 **Operable** - Operable means a component or system is capable of performing its intended function.

1.16 **Operating** - Operating means a component or system is performing its intended function.

1.17 **Operational Modes** - The reactor may be operated in any of three (3) operating modes, depending upon the configuration of the reactor coolant systems and the protective system set points.

a. Operational Mode I - Reactor can be operated at a thermal power level of ten megawatts or less.

b. Operational Mode II - Reactor can be operated at a thermal power level of five megawatts or less.

c. Operational Mode III - Reactor can be operated at a thermal power level of fifty kilowatts or less.

1.18 **Protective Action** - Protective action is the initiation of a signal or the operation of equipment within the reactor safety system in response to a parameter or condition of the reactor facility having reached a specified limit.

1.19 **Reactivity Worth of an Experiment** - The reactivity worth of an experiment is the value of the reactivity change that results from the experiment, being inserted into or removed from its intended position.

1.20 **Reactor Containment Building** - The reactor containment building is a reinforced concrete structure within the facility site which houses the reactor core, pool, and irradiated fuel storage facilities that is designed to (1) be at a negative internal pressure to ensure in-leakage, (2) control the release of effluents to the environment, and (3) mitigate the consequences of certain analyzed accidents or events.

1.21 **Reactor Core** - The reactor core shall be considered to be that volume inside the reactor pressure vessels occupied by eight or less fuel elements.

1.22 **Reactor Operator** - A reactor operator is an individual who is licensed to manipulate the controls of a reactor.

1 **DEFINITIONS** - Continued

1.23 **Reactor in Operation** - The reactor shall be considered in operation unless it is either shutdown or secured.

1.24 **Reactor Safety System** - The reactor safety system is that combination of sensing devices, electronic circuits and equipment, signal conditioning equipment, and electro-mechanical devices that serves to either effect a reactor scram, or activates the engineered safety features.

1.25 **Reactor Scram** - A reactor scram is the insertion of all four (4) shim blades (rods) by gravitational force as a result of removing the holding current from the shim rod drive mechanism electromagnets.

1.26 **Reactor Secured** - The reactor shall be considered secured when:

a. There is insufficient fuel in the reactor core to attain criticality with optimum available conditions of moderation and reflection with all four (4) shim blades (rods) removed,

OR

b. Whenever all of the following conditions are met:

(1) All four shim blades (rods) are fully inserted;

(2) One of the two following conditions exits:

i. The Master Control Switch is in the "OFF" position with the key locked in the key box or in custody of a licensed operator,

OR

ii. The dummy load test connectors are installed on the shim rod drive mechanisms and a licensed operator is present in the reactor control room;

(3) No work is in progress involving the transfer of fuel in or out of the reactor core;

(4) No work is in progress involving the shim blades (rods) or shim rod drive mechanisms with the exception of installing or removing the dummy load test connectors; and

(5) The reactor pressure vessel cover is secured in position and no work is in progress on the reactor core assembly support structure.

1 **DEFINITIONS** - Continued

1.27 **Reactor Shutdown** - The reactor is shutdown when:

a. It is subcritical by at least 0.0074 $\Delta k/k$ in the reference core condition with the reactivity worth of all installed experiments included,

AND

b. All four (4) of the shim blades (rods) are fully inserted and power is unavailable to the shim rod drive mechanism electromagnets.

1.28 **Reference Core Condition** - Reference core condition is the condition of the core when it is at ambient temperature (cold) and the reactivity worth of xenon is negligible ($< 0.002\ \Delta k/k$).

1.29 **Regulating Blade (Rod)** - The regulating blade (rod) is a low worth control blade (rod) used for very fine adjustments in the neutron density in order to maintain the reactor at the desired power level. The regulating blade (rod) may be controlled by the operator with a manual switch or push button, or by an automatic controller. The regulating blade (rod) does not have scram capability nor will it insert on a rod run-in signal.

1.30 **Removable Experiment** - A removable experiment is any experiment which can reasonably be anticipated to be moved during the life of the reactor.

1.31 **Research Reactor** - A research reactor is defined as a device designed to support a self-sustaining neutron chain reaction for research, development, educational, training, or experimental purposes and that may have provisions for the production of radioisotopes.

1.32 **Research Reactor Facility** - A research reactor facility includes all areas within which the owner or operator directs authorized activities associated with the reactor.

1.33 **Rod Run-In System** - The rod run-in system is that combination of sensing devices, electronic circuits and equipment, signal conditioning equipment, and electro-mechanical devices that serves to effect a rod run-in. A rod run-in is the automatic insertion of the shim blades at a controlled rate should a monitored parameter exceed a predetermined value. This system is not part of the reactor safety system, as defined by Specification 1.24; however, it does provide a protective function by introducing shim blade insertion to terminate a transient prior to actuating the reactor safety system.

1.34 **Safety Limits** - Safety Limits (SL) are limits placed upon important process variables which are found to be necessary to reasonably protect the integrity of

1 **DEFINITIONS** - Continued

the principal physical barriers which guard against the uncontrolled release of radioactivity.

1.35 **Scram Time** - Scram time is the elapsed time between the initiation of a scram signal and insertion of the shim blades to the 20% withdrawn position.

1.36 **Secured Experiment** - A secured experiment is any experiment, experimental apparatus, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining forces must be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces that are normal to the operating environment of the experiment, or by forces that can arise as a result of credible malfunctions.

1.37 **Senior Reactor Operator** - A senior reactor operator is an individual who is licensed to direct the activities of reactor operators and manipulate the controls of a reactor.

1.38 **Shim Blade (Rod)** - A shim blade (rod) is a high worth control blade (rod) used for coarse adjustments in the neutron density and to compensate for routine reactivity losses. The shim blade (rod) is magnetically coupled to its drive mechanism allowing it to scram when the electromagnet is de-energized. The shim blade (rod) also provides rod run-in functions.

1.39 **Shall, Should, and May -** The word "shall" is used to denote a requirement; the word "should" is used to denote a recommendation; and the word "may" is used to denote permission, neither a requirement nor a recommendation.

1.40 **Shutdown Margin** - Shutdown margin is the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and reactor safety systems starting from any permissible operating condition and with the most reactive shim blade and the regulating blade in the fully withdrawn positions, and that the reactor will remain subcritical without further operator action.

1.41 **Surveillance Intervals** - Surveillance intervals are the maximum allowable intervals established to provide operational flexibility and not reduce frequency. Established frequencies shall be maintained over the long term. The surveillance interval is the time between a check, test or calibration, whichever is appropriate to the item being subjected to the surveillance, and is measured from the date of the last surveillance. Allowable surveillance intervals shall not exceed the following:

1 **DEFINITIONS** - Continued

a. Biennial - interval not to exceed 2.5 years.

b. Annual - interval not to exceed 15 months.

c. Semiannual - interval not to exceed 7.5 months.

d. Quarterly - interval not to exceed 4 months.

e. Monthly - interval not to exceed 6 weeks.

f. Weekly - interval not to exceed 10 days.

g. Daily - interval not to exceed 1 calendar day.

h. Within a shift - interval not to exceed the reactor shift.

1.42 **True Value** - The true value is the actual value of a parameter.

1.43 **Unscheduled Shutdown** - An unscheduled shutdown is defined as any unplanned shutdown, that occurs after all "Blade Full-In Lights" have cleared, caused by actuation of the reactor safety system, rod run-in system, operator error, equipment malfunction, or a manual shutdown in response to conditions that could adversely affect safe operation, not including shutdowns that occur during testing or checkout operations.

1.44 **Unsecured Experiment** - An unsecured experiment is any experiment which is not secured as defined by Specification 1.36, or the moving parts of secured experiments when they are in motion.

2 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limits

Applicability:
This specification applies to the reactor fuel.

Objective:
The objective of this specification is to define the maximum reactor fuel element temperature that can be permitted to ensure that the integrity of the fuel cladding is maintained to guard against an uncontrolled release of radioactivity.

Specification:

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

Bases:

a. Maintaining the integrity of the fuel cladding requires that the cladding remain below the blistering temperature of 986 °F (530 °C). For all operating conditions that avoid either a Departure from Nucleate Boiling (DNB), or exceeding the Critical Heat Flux (CHF), or the Onset of Flow Instability (OFI), fuel cladding temperatures remain substantially below the fuel blistering temperature (NUREG-1313).

2.2 Limiting Safety System Settings

Applicability:
This specification applies to the set points for the reactor safety system channels monitoring reactor power level, primary coolant flow rate, reactor inlet water temperature and pressurizer pressure.

Objective:
The objective of this specification is to assure that automatic protective action is initiated to prevent a safety limit from being exceeded.

Specification:

a. **Mode I Operation**

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

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

b. **Mode II Operation**

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

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

c. **Mode III Operation**

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

Bases:

a. - b. The limiting safety system settings (LSSS) are set points which, if exceeded, will cause the reactor safety system to initiate a reactor scram. The LSSS were chosen such that the true value of any of the four safety-related variables, i.e., reactor power level, core flow rate, reactor inlet water temperature and pressurizer pressure will not exceed the operating limits under the most severe anticipated transient. Section 4.6.4 of the SAR and Amendment No. 36 present analyses to show that the LSSS for Mode I and II operation meet this criterion.

2.2 **Limiting Safety System Settings** - Continued

c. For Mode III operation, the high power scram set point of 125% of full power will occur at 62.5 kW, thus, there is a margin of 87.5 kW between the LSSS and the operating limit of 150 kW.

3 LIMITING CONDITIONS FOR OPERATIONS

General: Limiting Conditions for Operations (LCOs) are those administratively established constraints on equipment and operational characteristics that shall be adhered to during operation of the facility. The LCOs are the lowest functional capability or performance level required for safe operation.

3.1 Reactor Core Parameters

Applicability:
This specification applies to the reactor core and fuel elements used in the reactor core.

Objective:
The objective of this specification is to assure that the reactor can be controlled and shut down at all times and that the fuel elements are operated within acceptable design considerations thus ensuring fuel element integrity is maintained.

Specification:

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

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

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

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

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

Bases:

a. Specification 3.1.a provides additional assurance that Specification 3.1.b is satisfied (Ref. Section 4.5 of the SAR).

3.1 **Reactor Core Parameters** - Continued

b. Specification 3.1.b assures that a shutdown margin, as defined by Specification 1.40, is maintained (Ref. Section 4.5 of the SAR).

c. Operation at a power level greater than 100 watts requires a full core of eight (8) fuel elements to assure the validity of the operating limit curves and other safety analyses. When it may be important to conservatively determine the actual critical core loading, Specification 3.1.c allows operation with less than eight (8) fuel elements up to a power level not to exceed 100 watts. This maximum power limit is low enough to ensure no fuel damage will occur. This provides for a conservative approach to criticality with less than eight (8) new fuel elements.

Typically, the first approach to critical would be with a number of fuel elements insufficient to achieve criticality but be able to observe subcritical multiplication. Then one additional fuel element would be added at a time in between approaches to critical. The reactor would be operated in this manner only to perform necessary conservative approaches to criticality.

d. Specification 3.1.d assures that fuel elements which have been inspected and found to be defective are no longer used for reactor operation. Specification 5.3.c restricts the peak fissions per cubic centimeter burnup to values that have been correlated to result in less than 10% swelling of the fuel plates. 10% swelling of a fuel plate would roughly equate to an increase in plate thickness of 5 mils. Assuming a worst-case scenario where two adjacent fuel plates swell towards the same coolant channel gap, this would cause a decrease in the nominal coolant channel gap of 10 mils (Note: Nominal coolant channel gap is 80 mils, with a lower fabrication tolerance of 72 mils).

3.2 Reactor Control and Reactor Safety Systems

Applicability:
This specification applies to the reactor control and reactor safety systems.

Objective:
The objective of this specification is to reasonably assure proper operation of the reactor control system, thus avoiding conditions which could jeopardize the integrity of the fuel element cladding or endanger personnel health and safety, and to specify the minimum number of reactor safety system instrument channels that must be operable for safe reactor operation.

Specification:

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

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

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

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

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

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

	Rod Run-In Function	Number Required (N) Mode I	Mode II	Mode III	Trip Set Point
1.	High Power Level	3	3	3	115% of full power (Max)
2.	Reactor Period	2	2	2	10 Seconds (Min)
3.	Pool Low Water Level	1	1	0	27 feet (Min)
4.	Vent Tank Low Level	1	1	0	1 foot below centerline (Min)

3.2 **Reactor Control and Reactor Safety Systems** - Continued

	Rod Run-In Function	**Number Required (N)** Mode I	Mode II	Mode III	**Trip Set Point**
5.	Rod Not-In-Contact With Magnet	4	4	4	Magnet disengaged from any rod
6.	Anti-Siphon System High Level	1	1	1(1)	6 inches above valves (Max)
7.	Truck Entry Door	1	1	1	Loss of entry door seal pressure
8.	Regulating Blade Position	2	2(2)	2(2)	≤ 10% withdrawn or bottomed
9.	Manual Rod Run-In	1	1	1	Push button on Control Console

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

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

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

	Reactor Safety System Instrument Channel	**Number Required (N)** Mode I	Mode II	Mode III	**Trip Set Point**
1.	High Power Level	3	3	3	125% of full power (Max)
2.	Reactor Period	2	2	2	8 Seconds (Min)
3.	Primary Coolant Flow	4	2	2(1)	1,625 gpm(2) (Min)
4.	Differential Pressure Across the Core	1	0	0	3,200 gpm(3) (Min)

3.2 **Reactor Control and Reactor Safety Systems** - Continued

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

3.2 **Reactor Control and Reactor Safety Systems** - Continued

	Reactor Safety System Instrument Channel	Number Required (N) Mode I	Mode II	Mode III	Trip Set Point
18.	Facility Evacuation	1	1	1	Scram as a result of actuating the facility evacuation system
19.	Reactor Isolation	1	1	1	Scram as a result of actuating the reactor isolation system
20.	Manual Scram	1	1	1	Push button on Control Console
21.	Center Test Hole	2[5]	2[5]	2[5]	Scram as a result of removing the center test hole removable experiment test tubes or strainer

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

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

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

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

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

3.2 **Reactor Control and Reactor Safety Systems** - Continued

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

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

Bases:

a. Specification 3.2.a ensures that the normal method of reactivity control is used during reactor operation (Ref. Section 4.5 of the SAR).

b. Specification 3.2.b provides a restriction on the maximum neutron flux tilting that can occur in the core to ensure the validity of the power peaking factors described in Section 4.5 of the SAR.

c. Specification 3.2.c assures prompt shutdown of the reactor in the event a scram signal is received as analyzed in Section 13.2.2 of the SAR. The 20% level is defined as 20% of the shim blade full travel as measured from the fully inserted position. Below the 20% level, the fall of the shim blade is cushioned by a dashpot assembly. Approximately 91% of the shim blade total worth is inserted at the 20% level.

d. Specification 3.2.d limits the rate of reactivity addition by the regulating blade to provide for a reasonable response from operator control (Ref. Section 4.5 of the SAR). This Specification is based on a regulating blade total reactivity worth limit of 6.0×10^{-3} $\Delta k/k$ (Specification 5.3.d) and a regulating blade travel speed of 40 inches per minute.

e. Specification 3.2.e assures that power increases caused by control rod motion will be safely terminated by the reactor safety system. The continuous control rod withdrawal accident is analyzed in Section 13.2.2 of the SAR. Based on a total shim blade reactivity worth of 0.1838 $\Delta k/k$ and a maximum shim blade travel

3.2 **Reactor Control and Reactor Safety Systems** – Continued

speed of 2 inches per minute in the inward direction, the maximum insertion rate of negative reactivity would be 2.4 x 10^{-4} Δk/k/sec. Based on a maximum shim blade travel speed of 1 inch per minute in the outward direction, the maximum insertion rate of positive reactivity would be 1.2 x 10^{-4} Δk/k/sec (or 2.1 x 10^{-4} Δk/k/sec at the peak worth region of the shim blade bank). Both values are less than Specification 3.2.e limit of 3.0 x 10^{-4} Δk/k/sec. The continuous rod withdrawal accident analyzed in Addenda 1 and 5 to the MURR Hazards Summary Report used reactivity insertion rates of 2.78 x 10^{-4} Δk/k/sec and 3.0 x 10^{-4} Δk/k/sec, respectively.

f. The specifications on high power level and short reactor period are provided to introduce shim blade insertion on a reactor transient before the reactor safety system trip is actuated.

The low pool level rod run-in provides assurance that the radiation level from direct core radiation above the pool will not exceed 2.5 mR/h (Ref. Section 11.1.5.1 of the SAR).

The vent tank low level rod run-in prevents reactor operation with a vent tank level which could result in the introduction of air into the primary coolant system (Ref. Section 9.13 of the SAR).

The anti-siphon system high level rod run-in provides assurance that the introduction of air to the invert loop is sufficiently rapid to prevent a siphoning action following a rupture of the primary coolant piping (Ref. Section 6.3 of the SAR).

The rod not-in-contact with magnet rod run-in assures the reactor cannot be operated in violation of Specification 3.2.b due to a dropped shim blade.

The specification on the truck entry door prohibits reactor operation without the door's contribution to containment integrity as required by Specification 3.4.a.

The regulating blade rod run-ins ensure termination of a transient which, in automatic control, is causing a rapid insertion of the regulating blade.

g. The specifications on high power level, primary coolant flow, primary coolant pressure and reactor inlet water temperature provide for the limiting safety system settings outlined in Specifications 2.2.a, 2.2.b and 2.2.c. In Mode I and Mode II operation, the core differential temperature is approximately 17 °F; therefore, the reactor outlet water temperature scram set point at 175 °F provides a backup to the high reactor inlet water temperature scram. The core differential pressure scram provides a backup to the primary coolant low flow scrams.

3.2 **Reactor Control and Reactor Safety Systems** - Continued

The reactor period scram assures protection of the fuel elements from a continuous control blade withdrawal accident as analyzed in Section 13.2.2 of the SAR.

With the reflector plenum natural convection valve V547 in the open position and a pool coolant flow rate at 850 gpm, the pool coolant low flow scram assures the adequate cooling of the reactor pool, reflectors, control rods, and the flux trap (Ref. Section 5.3.5 of the SAR). The reflector high and low differential pressure scram provides a backup to the low pool coolant flow scram.

The pressurizer high pressure scram provides assurance that the reactor will be shut down during a high pressure transient before the relief valve set point or the pressure limit of the primary coolant system is reached as analyzed in Section 13.2.9.4 of the SAR.

The pressurizer low level scram provides assurance that the reactor will be shut down on a loss of coolant accident before pressurizer level decreases sufficiently to introduce nitrogen gas into the primary coolant system.

The pool water low level scram assures that the radiation level above the reactor pool from direct core radiation remains below 2.5 mR/h (Ref. Section 11.1.5.1 of the SAR).

The reactor scrams caused by the primary and pool coolant isolation valves (V507A/B and V509) leaving their full open position provide the first line of protection for a loss of flow accident (in their respective system) initiated by an inadvertent closure of the isolation valve(s).

The power level interlock (PLI) scram provides assurance that the reactor cannot be operated with a power level greater than that authorized for the mode of operation selected on the Power Level Switch. The PLI scram also provides the interlocks to assure that the reactor cannot be operated in Mode I with a primary or pool coolant low flow scram bypassed.

The facility evacuation and reactor isolation scrams provide assurance that the reactor is shut down for any condition which initiates or leads to the initiation of a facility evacuation or an isolation of the reactor containment building.

The manual scram provides assurance that the reactor can be shut down by the operator if an automatic function fails to initiate a reactor scram or if the operator detects an impending unsafe condition prior to the initiation of an automatic scram.

3.2 **Reactor Control and Reactor Safety Systems** - Continued

The center test hole scram provides assurance that the reactor cannot be operated unless the removable experiment sample canister or the strainer is inserted and latched in the center test hole. This is required anytime the reactivity worth of the center test hole removable experiment sample canister and the contained experiments or the strainer exceeds the limit of Specification 3.8.b (Ref. Section 13.2.2 of the SAR). The center test hole scram may be bypassed if the total reactivity worth of the removable experiment sample canister and the contained experiments or the strainer does not exceed the limit of Specification 3.8.b and is authorized by the Reactor Manager.

h. Specification 3.2.h assures that certain system conditions have been met prior to conducting a reactor startup (Master Control Switch 1S1 in the "ON" position; No Nuclear Instrument anomaly; Shim rods bottomed and in contact with their electromagnets; Source range level indication greater than 20 cps or intermediate range level recorder indication greater than 2 x 10^{-5}% power; and Thermal Column door closed) or placing the reactor in automatic control at power (Reactor period as indicated by Intermediate Range Channels 2 and 3 greater than 35 seconds; Indicated reactor power level greater than the "auto control prohibit" set point on the wide range neutron flux monitor recorder; Regulating blade position greater than 60% withdrawn; and Range Selector Switch 1S2 in the 5-kW red scale position or above) (Ref. Sections 7.5.3.1 and 7.5.4 of the SAR).

3.3 Reactor Coolant Systems

Applicability:
This specification applies to the reactor coolant systems.

Objective:
The objective of this specification is to protect the integrity of the reactor fuel and to prevent the release of fission product radioisotopes.

Specification:

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

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

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

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

   OR

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

c. The reactor shall not be operated if a radiochemical analysis of the primary coolant system indicates an iodine-131 concentration of greater than 5×10^{-3} μCi/ml.

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

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

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

   OR

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

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

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

3.3 **Reactor Coolant Systems** - Continued

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

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

Bases:

a. The first line of protection against a loss of core water resulting from a rupture of the primary coolant system is provided by the check valve on the inlet line and by the invert loop and the anti-siphon system on the outlet line. Upon opening, the anti-siphon isolation valves will admit a fixed volume of air to the highest point of the invert loop, thus preventing the reactor core from becoming uncovered by breaking any potential siphon which may have been created by the pipe rupture (Ref. Section 6.3 of the SAR).

The primary coolant isolation valves are located on the inlet and outlet primary coolant lines as close as practicable to the biological shield. Proper operation of these valves is not required for protection of the integrity of the fuel elements; however, their operation provides a means for isolation of the in-pool portions of the primary coolant from the remainder of the system.

The in-pool convective cooling system is not required for core protection (Ref. Section 13.2.9.3 of the SAR); however, its operation is desirable to prevent the formation of steam in the loop and to reduce thermal cycling of the reactor fuel.

b. - c. The primary coolant system with an iodine-131 concentration of 5×10^{-3} μCi/ml would contain a total iodine-131 inventory of 0.038 Ci in the system. Based on the iodine-131 activity in the reactor core provided in Section 13.2.1.2 of the SAR, this iodine-131 concentration would equate to less than 0.000022 % of the total core iodine-131 inventory in the primary coolant. Specifications 3.3.b and 3.3.c provide for the early detection of a leaking fuel element so that corrective actions can be taken to prevent the release of fission products.

d. Detection of the deterioration of components in the pool coolant system and in-pool experimental facilities is provided by Specification 3.3.d (NUREG-1537).

e. Specification 3.3.e provides for the early detection of a leaking reactor coolant system heat exchanger so that corrective actions can be taken.

f. Secondary coolant system activity is limited to ensure releases are maintained below the limits of 10 CFR 20.

3.3 **Reactor Coolant Systems** - Continued

g. - i. Experience at many research reactor facilities has shown that maintaining the conductivity and pH within the specified limits provides acceptable control of corrosion and limits concentrations of particulate and dissolved containments that could be made radioactive by neutron irradiation (NUREG-1537).

3.4 Reactor Containment Building

Applicability:
This specification applies to the reactor containment building.

Objective:
The objective of this specification is to assure that containment integrity is maintained when required so that the health and safety of the general public is not endangered as a result of reactor operation.

Specification:

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

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

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

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

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

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

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

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

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

   (1) The reactor is secured,

   AND

   (2) No movement of irradiated fuel with a decay time of less than sixty (60) days or fueled experiments with significant fission product inventory outside containers, systems, or storage areas,

   AND

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

3.4 **Reactor Containment Building** - Continued

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

Bases:

a. - b. Specifications 3.4.a and 3.4.b assure that the reactor containment building can be isolated at all times except when plant conditions are such that the probability of a release of radioactivity is negligible.

c. Radiation monitors located at the reactor bridge and in the reactor containment building ventilation exhaust plenum supply input signals to meters located in the reactor control room. A containment isolation will occur when radiation levels in these areas exceed a predetermined value. During operations such as the removal of experiments or equipment from the pool, the radiation level at the level of the reactor bridge or in the exhaust plenum can increase significantly for short periods. To prevent inadvertent containment isolations, it may be necessary to raise the set point on the reactor bridge or exhaust plenum monitor. During periods in which the set point is raised to more than one decade above the normal reading, the radiation level in the area of the monitor will be continuously monitored. Thus, should the radiation level increase from unknown causes or from material which could be released to the unrestricted environment, the reactor containment building can be quickly isolated by manually actuating the isolation system.

3.5 Reactor Instrumentation

Applicability:
This specification applies to the instruments that provide information which must be available to the operator during reactor operation.

Objective:
The objective of this specification is to ensure that sufficient reliable information is presented to the operator to assure safe operation of the reactor.

Specification:

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

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

(1) Required for reactor startup only.

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

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

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

3.5 **Reactor Instrumentation** - Continued

Bases:

a. The Power Range Nuclear Instrument Channels provide neutron monitors that provide reactor protective, alarm and indication functions over the power range (Ref. Section 7.4 of the SAR).

The Intermediate Range Nuclear Instrument Channels provide neutron monitors that provide reactor protective, interlock and indication functions over the intermediate range (Ref. Section 7.4 of the SAR).

The Source Range Nuclear Instrument Channel provides a neutron monitor that is very sensitive to neutrons and thus provides improved indication of the low neutron flux levels present during a reactor startup (Ref. Section 7.4 of the SAR).

The reactor pool temperature instrument is required to ensure that pool temperature does not increase to a level which would jeopardize the ability to cool in-pool components (Ref. Section 7.6.2.2 of the SAR).

b. The maximum primary coolant pressure of 110 psig assures that the system design pressure of 125 psig is not exceeded.

Maintaining the minimum anti-siphon system pressure ensures that the system will adequately perform its intended function (Ref. Section 6.3 of the SAR).

The reactor pool temperature limit provides an operating limit to assure the adequate cooling of the reactor fuel or pool components during all modes of operation.

3.6 **Emergency Electrical Power System**

Applicability:
This specification applies to the emergency electrical power system.

Objective:
The objective of this specification is to ensure that adequate emergency electrical power is available in the event of a loss of normal electrical power.

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

Bases:
a. On a loss of normal electrical power, the emergency electrical power system will supply power to the containment ventilation isolation doors, personnel entry doors, facility ventilation exhaust fans, emergency lighting panel, and reactor instrumentation and control systems. The emergency electrical power system is not required for protection of the integrity of the fuel elements on a loss of normal electrical power. In the extremely unlikely event of a simultaneous loss of normal electrical power and fuel element failure, the operation of the emergency electrical power system would be required to provide for continuous containment isolation (Ref. Section 13.2.7 of the SAR).

3.7 Radiation Monitoring Systems and Airborne Effluents

Applicability:
This specification applies to radiation monitoring information which must be available to the reactor operator during reactor operation and the release of gaseous and particulate activity from the facility ventilation exhaust stack.

Objective:
The objective of this specification is to assure that sufficient radiation monitoring information is available to the reactor operator during reactor operations and exposure to the public resulting from the radioactivity released from the reactor facility to the unrestricted environment will not exceed the limits of 10 CFR 20.

Specification:
a. The reactor shall not be operated unless the following radiation monitoring channels are operating:

	Channel	**Minimum Numbers Operating** Mode I	Mode II	Mode III
1.	Reactor Bridge Radiation Monitor	1[(1)]	1[(1)]	1[(1)]
2.	Reactor Containment Building Exhaust Plenum Radiation Monitor	1	1	1
3.	Off-Gas (Stack) Radiation Monitor	1[(2)]	1[(2)]	1[(2)]

(1) The trip setting may be temporarily set upscale during periods of maintenance and sample handling. During these periods, the radiation monitor indication shall be closely observed.

(2) The stack radiation monitor may be placed out of service for up to two (2) hours for calibration and maintenance. During this out-of-service time, no experimental or maintenance activities shall be conducted which are likely to result in the release of airborne radioactivity.

3.7 **Radiation Monitoring Systems and Airborne Effluents** - Continued

b. The maximum discharge rate through the ventilation exhaust stack shall not exceed the following:

Type of Radioactivity	**Max. Concentration Averaged Over One Year**	**Max. Controlled Instantaneous Release Concentration**
Particulates and halogens with half-lives greater than 8 days	AEC	AEC
All other radioactive isotopes	350 AEC	3,500 AEC

AEC = Air Effluent Concentration as listed in Appendix B, Table 2, Column 1 of 10 CFR 20, "Standards for Protection Against Radiation."

c. An environmental monitoring program shall be carried out and shall include, as a minimum:

(1) Analysis of samples from surface waters from the surrounding areas, and vegetation or soil,

AND

(2) Placement of film badges, thermoluminescent dosimeters, or other devices at control points.

Bases:

a. The radiation monitors provide information of an impending or existing danger from radiation so that corrective action can be initiated to prevent the spread of radioactivity to the surroundings and so that there will be sufficient time to evacuate the facility should it be necessary to do so.

Isolation of the reactor containment building at 10 times the normal previously established radiation levels is necessary to allow for sample handling within the reactor pool or when removing samples from the pool. Normal pool surface radiation levels are approximately 20 mR/h while those at the containment building exhaust plenum are around 0.15 mR/h. Operational experience has demonstrated that the 10 times factor provides sufficient margin to minimize inadvertent reactor scrams without allowing for the potential of unacceptable exposure rates to personnel in containment. Ten times the routine dose rates equate to 200 mrem at the bridge monitor and 1.5 mrem at the exhaust plenum. Dose rates at this level do not constitute an unreasonable risk and would not go unidentified for any significant period of time.

3.7 **Radiation Monitoring Systems and Airborne Effluents** - Continued

b. For the purposes of Specification 3.7.b, air effluents for particulates and halogens with half-lives greater than 8 days are limited to the Air Effluent Concentrations (AEC) without the inclusion of a dilution multiplier to minimize any chance of reconcentration at the receptor site resulting in doses in excess of the direct exposures via air concentrations. Data from Soldat, JD (Health Physics 9, p. 1170, 1963), suggest a reconcentration factor of approximately 400 for the Iodine-131 milk/man pathway; however, dilution of the stack effluent to the nearest residence due north of MURR (760 meters), the prevailing wind direction, is approximately 1900, thus giving a safety factor (ratio) of 4.75. This value is also conservative in that the wind blows from 360 degrees around MURR throughout the year and thus this value represents a worst case scenario to only the maximally exposed receptor point.

For Argon-41, the primary air effluent from MURR, dispersion calculations are based on standard reference material and experimental data obtained at the reactor showing that concentrations under average conditions will be 0.008 of the AEC limits in the unrestricted area surrounding the reactor facility. Also, dilution factors under conservative conditions are in the range of 5×10^4 under both average and stable conditions at ground level from the facility building. For normal short burst releases at the facility which are five to ten seconds in duration and occur on an average of ten times per day five days per week the effect on the average concentration is less than 1% when averaged over a one-day period.

It is concluded that these concentrations as specified will not constitute a hazard to the health and safety of the public.

c. Collecting and analyzing water, and soil or vegetation samples will provide information that environmental limits are not being exceeded. Film badges, thermoluminescent dosimeters, or other devices placed at control points provide a measurement of radiation. The continuation of the environmental program will verify that operation of the facility presents no significant risk to the health and safety of the general public.

3.8 **Experiments**

Applicability:
This specification applies to all experiments which directly utilize neutrons or other radiation produced by the reactor. Radioactive sources shall meet the requirements for experiments.

Objective:
The objective of this specification is to prevent an accident which would jeopardize the safe operation of the reactor or would constitute a hazard to the safety of the facility staff and general public.

Reactivity Limits Specification:

a. The absolute value of the reactivity worth of each secured removable experiment shall be limited to 0.006 $\Delta k/k$.

b. The absolute value of the reactivity worth of all experiments in the center test hole shall be limited to 0.006 $\Delta k/k$.

c. Each movable experiment or the movable parts of any individual experiment shall have a maximum absolute reactivity worth of 0.001 $\Delta k/k$.

d. The absolute value of the reactivity worth of each unsecured experiment shall be limited to 0.0025 $\Delta k/k$.

e. The absolute value of the reactivity worth of all unsecured experiments which are in the reactor shall be limited to 0.006 $\Delta k/k$.

Materials Specification:

f. Each fueled experiment shall be limited such that the total inventory of iodine-131 through iodine-135 in the experiment is not greater than 150 Curies and the maximum strontium-90 inventory is no greater than 300 millicuries.

g. Fueled experiments containing inventories of iodine-131 through iodine-135 greater than 1.5 Curies or strontium-90 greater than 5 millicuries shall be in irradiation containers that satisfy the requirements of Specification 3.8.s or be vented to the facility ventilation exhaust stack through high efficiency particulate air (HEPA) and charcoal filters which are continuously monitored for an increase in radiation levels.

h. Each non-fueled experiment that is intended to produce iodine-131 shall be limited such that the inventory of iodine-131 is not greater than 150 Curies.

3.8 **Experiments** - Continued

i. Explosive materials shall not be irradiated nor shall they be allowed to generate in any experiment in quantities over 25 milligrams of TNT-equivalent explosives. Explosive materials shall be limited to a total quantity of 100 milligrams of TNT-equivalent explosives in the reactor containment building.

j. Corrosive materials shall be doubly encapsulated in corrosion-resistant containers to prevent interaction with reactor components or pool water. Should a failure of the encapsulation occur that could damage the reactor, then the potentially damaged components shall be removed and inspected.

k. Cryogenic liquids shall not be used in any experiment within the reactor pool.

l. Fluids utilized in beamport loop experiments shall be of types which will not chemically react in the event of leakage and shall be maintained at pressure and temperature conditions such that the integrity of the beam tube will not be impaired in the event of loop rupture.

m. The normal operating procedures shall include controls on the use or exclusion of corrosive, flammable, and toxic materials in experiments or in the reactor containment building. These procedural controls shall include a current list of those materials which shall not be used and the specific controls and procedures applicable to the use of corrosive, flammable, or toxic materials which are authorized.

Failure and Malfunctions Specification:

n. Where the possibility exists that the failure of an experiment could release radioactive gases or aerosols into the containment building atmosphere, the experiment shall be limited to that amount of material such that the airborne concentration of radioactivity when averaged over a year will not exceed the limits of 10 CFR 20. Exception: Fueled experiments that produce iodine-131 through iodine-135 and non-fueled experiments that are intended to produce iodine-131 (See Specifications 3.8.f and 3.8.h).

o. Experiments shall be designed and operated so that identifiable accidents such as a loss of primary coolant flow, loss of experiment cooling, etc., will not result in a release of fission products or radioactive materials from the experiment.

p. Experiments shall be designed such that a failure of an experiment will not lead to a direct failure of another experiment, a failure of a reactor fuel element, or to interfere with the action of the reactor safety and reactor control systems or other operating components.

3.8 **Experiments** - Continued

q. No experiments shall be placed in the reactor pressure vessel or water annulus surrounding the center test hole other than for reactor calibration.

r. Cooling shall be provided to prevent the surface temperature of a submerged irradiated experiment from exceeding the saturation temperature of the cooling medium.

s. Irradiation containers to be used in the reactor, in which a static pressure will exist or in which a pressure buildup is predicted, shall be designed and tested for a pressure exceeding the maximum expected pressure by at least a factor of two (2).

t. The maximum temperature of a fueled experiment shall be restricted to at least a factor of two (2) below the melting temperature of any material in the experiment. First-of-a-kind fueled experiments shall be instrumented to measure temperature.

Other Specification:

u. Only movable experiments in the center test hole shall be removed or installed with the reactor operating. All other experiments in the center test hole shall be removed or installed only with the reactor shutdown. Secured experiments shall be rigidly held in place during reactor operation.

v. Non-fueled experiments that are intended to produce iodine-131 shall be processed in hot cells that are vented to the exhaust stack system through charcoal filters which are continuously monitored for an increase in radiation levels.

Bases:

a. Specification 3.8.a provides assurance that any inadvertent insertion/removal or credible malfunction of a secured removable experiment would not introduce positive reactivity whose consequences would lead to radiation exposures in excess of the 10 CFR 20 limits. The step reactivity insertion is analyzed in Section 13.2.2 of the SAR.

b. The reactivity worth of experiments in the center test hole is limited by Specification 3.8.b such that the introduction of the maximum reactivity worth of all experiments would not result in damage to the fuel plates as analyzed in Section 13.2.2 of the SAR.

c. Specification 3.8.c provides assurance that the movement of movable experiments or movable parts of any experiment will not introduce reactivity transients more severe than one that can be controlled without initiating a reactor safety system action as analyzed in Section 13.2.2 of the SAR.

3.8 **Experiments** - Continued

d. Specification 3.8.d prevents the installation of an unsecured experiment which could introduce, as a positive step change, sufficient reactivity to place the reactor in a transient that would cause a violation of a limit as analyzed in Section 13.2.2 of the SAR.

e. Specification 3.8.e assures that the reactivity worth of all unsecured experiments shall not exceed the maximum value authorized for a single secured removable experiment.

f. Specification 3.8.f restricts the generation of hazardous materials to levels that can be handled safely and easily. Analysis of fueled experiments containing a greater inventory of fission products has not been completed, and therefore their use is not permitted (Ref. Section 13.2.6 of the SAR).

g. Specification 3.8.g restricts the generation of hazardous materials to levels that can be handled safely and easily. Analysis of fueled experiments containing a greater inventory of fission products has not been completed, and therefore their use is not permitted (Ref. Section 13.2.6 of the SAR).

h. Specification 3.8.h provides assurance that the processing of iodine-131 can be performed safely and that equipment necessary for accident mitigation has been installed (Ref. Amendment No. 37).

i. Specification 3.8.i is intended to reduce the likelihood of damage to reactor or pool components resulting from the detonation of explosive materials (Ref. Section 13.2.6 of the SAR).

j. Specification 3.8.j provides assurance that no chemical reaction will take place to adversely affect the reactor or its components.

k. The extremely low temperatures of the cryogenic liquids present structural problems that enhance the potential of an experiment failure. Specification 3.8.k provides for the proper review of proposed experiments containing or using cryogenic materials.

l. Specification 3.8.l provides assurance that the integrity of the beamports will be maintained for all loop-type experiments.

m. Specification 3.8.m assures that corrosive materials which are chemically incompatible with reactor components, highly flammable materials, and toxic materials are adequately controlled and that this information is disseminated to all reactor users.

3.8 **Experiments** - Continued

n. The limitation on experiment materials imposed by Specification 3.8.n assures that the limits of 10 CFR 20, Appendix B, are not exceeded in the event of an experiment failure.

o. - p. Specifications 3.8.o and 3.8.p provide guidance for experiment safety analysis to assure that anticipated transients will not result in radioactivity release and that experiments will not jeopardize the safe operation of the reactor.

q. Specification 3.8.q is intended to reduce the likelihood of accidental voiding in the reactor core or water annulus surrounding the center test hole by restricting materials which could generate or accumulate gases or vapors (Ref. Section 4.5 of the SAR).

r. Specification 3.8.r is intended to reduce the likelihood of reactivity transients due to accidental voiding in the reactor or the failure of an experiment from internal or external heat generation (Ref. Sections 4.5 and 13.2.6 of the SAR).

s. Specification 3.8.s is intended to reduce the likelihood of damage to the reactor and/or radioactivity releases from experiment failure (Ref. Section 13.2.6 of the SAR).

t. Specification 3.8.t is intended to reduce the likelihood of damage to the reactor and/or radioactivity releases from experiment failure.

u. Specification 3.8.u is intended to limit the experiments that can be moved in the center test hole while the reactor is operating to those that will not introduce reactivity transients more severe than one that can be controlled without initiating reactor safety system action (Ref. Section 13.2.2 of the SAR).

v. Specification 3.8.v provides assurance that the processing of iodine-131 can be performed safely and that equipment necessary for accident mitigation has been installed (Ref. Amendment No. 37).

3.9 Auxiliary Systems

Applicability:
This specification applies to the reactor auxiliary systems.

Objective:
The objective of this specification is to provide for the operation of certain auxiliary systems and thus further protect the reactor fuel and personnel.

Specification:

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

b. The reactor shall not be operated unless the emergency pool fill system is operable.

Bases:

a. Specification 3.9.a provides for an adequate supply of primary grade water for reactor plant make-up during all modes of operation.

b. The emergency pool fill system is capable of supplying water at approximately 1,000 gpm to the reactor pool. This supply assures that the water level in the reactor pool will remain above the reflector in case a 6-inch beamport or a 6-inch pool coolant line is sheared (Ref. Sections 13.2.9.1 and 13.2.9.2 of the SAR).

3.10 Iodine-131 Processing Hot Cells

Applicability:
This specification applies to the equipment needed to safely process iodine-131.

Objective:
The objective of this specification is to reasonably assure that the health and safety of the staff and public is not endangered as a result of processing iodine-131.

Specification:

a. The facility ventilation exhaust system shall be operable when processing iodine-131 in the iodine-131 processing hot cells.

b. The facility ventilation exhaust system shall maintain the iodine-131 processing hot cells at a negative pressure with respect to the surrounding areas when processing iodine-131.

c. Processing of iodine-131 shall not be performed in the iodine-131 processing hot cells unless the following minimum number of radiation monitoring channels are operable.

	Radiation Monitoring Channel	**Number**
1.	Off-Gas (Stack) Radiation Monitor	1
2.	Iodine-131 Processing Hot Cells Radiation Monitor	1[(1)]

(1) Exception: When the required radiation monitoring channel becomes inoperable, then portable instruments may be substituted for the normally installed monitor in Specification 3.10.c.2 within one (1) hour of discovery for a period not to exceed one (1) week.

d. At least three (3) charcoal filter banks each having an efficiency of 99% or greater shall be operable when processing iodine-131 in the iodine-131 processing hot cells.

Bases:

a. Specification 3.10.a requires that the facility ventilation exhaust system is in operation when processing iodine-131 in the iodine-131 processing hot cells to ensure proper dilution of effluents to prevent exceeding the limits of 10 CFR 20 Appendix B.

b. Specification 3.10.b assures that the iodine-131 processing hot cells are maintained at a negative pressure with respect to the surrounding areas ensures safety for the facility staff.

3.10 **Iodine-131 Processing Hot Cells** - Continued

c. Specification 3.10.c assures that the radiation monitors provide information to operating personnel regarding routine release of radioactivity and any impending or existing danger from radiation. Their operation will provide sufficient time to take the necessary steps to prevent the spread of radioactivity to the surroundings. The Stack Radiation Monitor continuously monitors the air exiting the facility through the exhaust stack for airborne radioactivity. The Iodine-131 Processing Hot Cells Radiation Monitor is a six (6) detector system; two (2) detectors serving each one of the three (3) hot cells. For each hot cell, one (1) detector is located at the processor's work area where the hot cell manipulators are installed and the other is located in the bay above the hot cell next to the exhaust charcoal filters.

d. The potential radiation dose to staff and individuals at the Emergency Planning Zone boundary and beyond have been calculated following an accidental release of iodine-131 activity. These calculations are based on the facility ventilation exhaust system directing all iodine-131 processing hot cell effluents through charcoal filtration with an efficiency of 99% or greater prior to being released through the facility exhaust stack.

4 SURVEILLANCE REQUIREMENTS

4.0 General

Applicability:
This specification applies to the surveillance requirements of any system related to reactor safety.

Objective:
The objective of this specification is to verify proper operation of any system related to reactor safety.

Specification:

a. Surveillance frequencies denoted herein are based on continuing operation of the reactor. Surveillance activities scheduled to occur during an operating cycle which cannot be performed with the reactor operating may be deferred to the end of that current reactor operating cycle. A reactor system or measuring channel shall not be considered operable until it is successfully tested. Any time a reactor system or component is modified or repaired, the surveillance for that system should be performed as part of the operability check of the system or component. This should be done regardless of when the surveillance was last performed or when it is next due. Surveillance intervals shall not exceed those defined by Specification 1.41. Discovery of noncompliance with any of the surveillance specifications listed in this Section shall limit reactor operations to that required to perform the surveillance.

Bases:

a. Experience has shown that surveillances will ensure performance and operability of any system related to reactor safety.

4.1 Reactor Core Parameters

Applicability:
This specification applies to the surveillance requirements of the reactor core parameters.

Objective:
The objective of this specification is to verify reactor core parameters which are directly related to reactor safety.

Specification:

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

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

c. The reactor core shall be verified to consist of eight (8) fuel elements after a refueling for a reactor startup.
Exception: The reactor may be operated to 100 watts above shutdown power on less than eight (8) elements for the purposes of reactor calibration or multiplication measurement studies.

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

Bases:

a. - b. Annual measurements, coupled with measurements made after changes that can affect reactivity values, provide adequate assurance that core behavior resulting from configuration changes are adequately characterized.

c. Operation at a power level greater than 100 watts requires a full core of eight (8) fuel elements to assure the validity of the operating limit curves and other safety analyses.

d. The specified fuel element inspections along with the continuous primary coolant system fission product monitoring and the weekly radiochemical analysis of the primary coolant provide for the detection of anomalies resulting from reactor operation and reduces the possibility of fission product release to the primary coolant system. Inspecting the fuel elements at the end of their life has the added advantage of allowing for the decay of the fuel elements and, thus, reduction of exposure to personnel.

4.2 Reactor Control and Reactor Safety Systems

Applicability:
This specification applies to the surveillance requirements on the reactor control and reactor safety systems.

Objective:
The objective of this specification is to reasonably assure proper operation of the reactor control system and the reactor safety system instrument channels.

Specification:

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

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

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

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

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

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

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

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

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

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

4.2 **Reactor Control and Reactor Safety Systems** - Continued

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

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

m. Following any modifications or repairs on any portion of the reactor control and reactor safety systems, the modified or repaired portion of the system shall be satisfactorily tested before the system is considered operable.

Bases:

a. Specification 4.2.a assures that the control blades will be verified operable within a shift.

b. Measurement of the drop time of each of the four (4) shim blades is made quarterly to demonstrate that the blades are capable of performing properly. In over 40 years of operation, to date, the shim blades have never failed to meet Specification 3.2.c.

c. Periodic inspection of the shim blades provides detection of singular blade abnormalities and any potential generic blade design deficiencies. Specification 4.2.c further assures that the reactor will not be operated using shim blades with suspected generic design deficiencies.

d. Specification 4.2.d assures that shim blade heights will be verified within a shift.

e. - f. The drive mechanisms for the regulating and shim blades are constant speed mechanical devices and withdrawal and insertion speeds should not vary except as a result of mechanical wear. The surveillance is chosen to provide a significant margin over expected failure or wear rates of these mechanical devices.

g. Measurements of the reactivity worth of the shim blades have shown to vary slightly as a result of absorber burnup and only slightly with respect to operational core loading and experimental changes.

h. - k. Experience has shown that the identified frequencies will ensure performance and operability for each of these systems or components (NUREG-1537 and ANSI/ANS-15.1-2007).

4.2 **Reactor Control and Reactor Safety Systems** - Continued

l. Thermal power verification will ensure that indicated reactor power level is correct. Because of the small primary coolant differential temperature at 10 MW (about 17 °F), these verifications will not be performed below 2 MW.

m. Specification 4.2.m ensures that the modified or repaired system is satisfactorily tested prior to being considered operable.

4.3 Reactor Coolant Systems

Applicability:
This specification applies to the surveillance requirements on the reactor coolant systems.

Objective:
The objective of this specification is to reasonably assure proper operation of the reactor coolant systems.

Specification:

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

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

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

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

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

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

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

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

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

Bases:

a. The past 40 years of operation of the anti-siphon system, primary coolant isolation valves and in-pool convective cooling system has shown that monthly testing is adequate to provide assurance of continued operability.

4.3 **Reactor Coolant Systems** - Continued

b. The primary coolant system fuel element failure monitor channel check will assure that the instrument is operable. Semiannual channel calibration will assure that long-term drift of the instrument will be corrected.

c. The weekly radiochemical analysis will provide assurance that a fuel element leak will be discovered so that corrective action can be taken to prevent the release of fission products. Specification 4.3.c establishes the frequency of verification of compliance with Specification 3.3.c.

d. - g. Experience has shown that the frequency of measurements on the reactor coolant systems for gross radioactivity, conductivity and pH adequately maintain the water quality at such a level to minimize corrosion and maintain safety.

h. Satisfactory performance of both relief valves during the testing program over the past 40 years has demonstrated the reliability of the valves and the assurance of operability gained by the testing frequency outlined in Specification 4.3.h.

4.4 Reactor Containment Building

Applicability:
This specification applies to the surveillance requirements on the containment system.

Objective:
The objective of this specification is to reasonably assure proper operation of the containment system.

Specification:

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

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

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

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

Bases:

a. Annual measurement of the containment building leakage rate has proven adequate to ensure that the leakage rate of the structure will remain within the design limits outlined in Specification 5.5.c. No repairs or modifications will be performed prior to the test so that the results demonstrate the historic integrity of the containment structure.

b. Measurement of the containment building leakage rate following any modification or repair that could affect the leak-tightness of the building ensures that the leakage rate of the structure will remain within the design limits outlined in Specification 5.5.c.

c. The reliability of the containment actuation (reactor isolation) system has proven that monthly verification of its proper operation is sufficient to assure operability.

d. Specification 4.4.d assures that containment integrity is verified to exist to limit the leakage of contained potentially radioactive air in the event of any reactor accident to ensure exposures are maintained below the limits of 10 CFR 20.

4.5 Reactor Instrumentation

Applicability:
This specification applies to the surveillance requirements of the reactor instrumentation systems.

Objective:
The objective of this specification is to reasonably assure proper operation of the reactor instrumentation systems.

Specification:

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

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

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

Bases:

a. - b. Semiannual channel calibration of the instrument channels and instrumentation will assure that long-term drift of the channels and instrumentation will be corrected.

c. The nuclear instrumentation channel test will assure that the channels are operable.

The basis for not requiring channel testing within two (2) hours following a reactor shutdown is a combination of factors. Provided that an unsafe condition does not exist or a failure of the nuclear instrumentation channel has not occurred, the nuclear instrumentation channels are observed to be operable during the previous reactor operating period and continue to indicate appreciable neutron strength during the short shutdown period. Within the 2-hour period following the shutdown, operational history demonstrates that there remains sufficient source neutron strength in the reactor core to ensure that proper observation of subcritical multiplication and that criticality can be observed during the subsequent startup. In effect, the nuclear instrumentation channels have been in continuous operation for this period and thus additional testing is not required.

4.6 Emergency Electrical Power System

Applicability:
This specification applies to the surveillance requirements of the emergency electrical power system.

Objective:
The objective of this specification is to reasonably assure proper operation of the emergency electrical power system.

Specification:

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

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

Bases:

a. The emergency power generator tests provide assurance that the generator is operable.

b. The semiannual electrical load test has proven satisfactory in providing reasonable assurance that the emergency power generator electrical control and distribution system will remain operable.

4.7 Radiation Monitoring Systems and Airborne Effluents

Applicability:
This specification applies to the surveillance requirements of the radiation monitoring instrumentation.

Objective:
The objective of this specification is to reasonably assure proper operation of the radiation monitoring instrumentation.

Specification:

a. Radiation monitoring instrumentation required by Specification 3.7.a shall be verified operable by monthly radiation source checks or channel tests.

b. Radiation monitoring instrumentation required by Specification 3.7.a shall be channel calibrated on a semiannual basis.

c. Surveillance of the environmental monitoring program shall include:

   (1) A collection of water, and vegetation or soil samples semiannually,

   AND

   (2) A collection of film badges, thermoluminescent dosimeters, or other devices semiannually.

Bases:

a. Experience has shown that monthly verification of operability of the radiation monitoring instrumentation is adequate assurance of proper operation over a long time period.

b. Semiannual channel calibration of the radiation monitoring instrumentation will assure that long-term drift of the channels will be corrected.

c. Collecting and analyzing water, and soil or vegetation samples on a semiannual basis will provide information that environmental limits are not being exceeded. Collecting and analyzing film badges, thermoluminescent dosimeters, or other devices on a semiannual basis will provide information that radiation limits are not being exceeded.

4.8 Experiments

Applicability:
This specification applies to the surveillance requirements of experiments installed in the reactor or its experimental facilities.

Objective:
The objective of this specification is to prevent the conduct of experiments which may damage the reactor or release excessive amounts of radioactive materials as a result of experiment failure.

Specification:

a. The criteria of Specification 3.8 shall be evaluated and found acceptable prior to inserting an experiment in the reactor or its experimental facilities.

b. The reactivity worth of an experiment shall be estimated or measured, as appropriate, before reactor operation with said experiment.

Bases:

a. - b. Experience has shown that experiments which are reviewed by the staff and the Reactor Advisory Committee can be conducted without endangering the safety of the reactor or exceeding the limits specified in the Technical Specifications.

4.9 Auxiliary Systems

Applicability:
This specification applies to the surveillance requirements of the reactor auxiliary systems.

Objective:
The objective of this specification is to reasonably assure proper operation of the auxiliary systems.

Specification:

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

b. The operability of the emergency pool fill system shall be tested on a semiannual basis.

Bases:

a. Specification 4.9.a assures that an adequate supply of primary grade water is available for make-up during all modes of operation.

b. The University of Missouri-Columbia water supply system provides a virtually unlimited source of raw water for the emergency pool fill system. Water supply is maintained at a high pressure by automatically-controlled pumping stations. The above test, in light of the reliability of the emergency pool fill system, provides assurance that Specification 3.9.b is satisfied.

4.10 Iodine-131 Processing Hot Cells

Applicability:
This specification applies to the surveillance requirements of the equipment needed to safely process iodine-131.

Objective:
The objective of this specification is to reasonably assure proper operation of the equipment needed to safely process iodine-131.

Specification:

a. An operability test of the facility ventilation exhaust system shall be performed monthly.

b. A channel check of the facility ventilation exhaust system to maintain the iodine-131 processing hot cells at a negative pressure with respect to the surrounding areas shall be verified daily prior to any process.

c. The radiation monitors as required by Specification 3.10.c shall be calibrated on a semiannual basis.

d. The radiation monitors as required by Specification 3.10.c shall be checked for operability with a radiation source at monthly intervals.

e. The efficiency of the iodine-131 processing hot cells charcoal filter banks shall be verified biennially or following major maintenance. It shall be verified that the charcoal filter banks have a removal efficiency of 99% or greater for iodine.

Bases:

a. Experience has shown that monthly tests of the facility ventilation exhaust system are sufficient to assure proper operation.

b. Verifying that the iodine-131 processing hot cells are at negative pressure with respect to the surrounding areas prior to use ensures personnel safety.

c. Semiannual channel calibration of the radiation monitoring instrumentation will assure that long-term drift of the channels will be corrected.

d. Experience has shown that monthly verification of operability of the radiation monitoring instrumentation is adequate assurance of proper operation over a long time period.

e. Biennial verification of the filter banks ensures that the filters will perform as analyzed.

5 DESIGN FEATURES

5.1 Site Description

Applicability:
This specification applies to the site of the University of Missouri Research Reactor (MURR) facility.

Objective:
The objective of this specification is to identify the location of the MURR facility.

Specification:

a. The MURR facility is situated on a 7.5-acre lot in the central portion of Research Commons, an 84-acre tract of land approximately one mile southwest of the University of Missouri (MU) at Columbia's main campus. This campus is located in the southern portion of Columbia, the county seat and largest city in Boone County, Missouri.

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

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

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

Bases:

a. The MURR facility site location and description are strictly defined in Chapter 2 of the SAR. The location of the MURR facility and Research Commons is owned and operated by MU. Based on the information provided in Chapter 2, and throughout the SAR, the site is well suited for the location of the facility when considering the relatively benign operating characteristics of the reactor.

5.2 Reactor Coolant Systems

Applicability:
This specification applies to the reactor coolant systems.

Objective:
The objective of this specification is to assure proper coolant for safe operation.

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

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

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

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

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

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

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

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

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

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

5.2 **Reactor Coolant Systems** - Continued

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

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

Exceptions:

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

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

Bases:

a. - k. The reactor coolant systems are described and analyzed in Chapter 5 of the SAR. The reactor can be safely operated at ten megawatts with the coolant systems as described.

Specification 5.2.a as excepted, permits reactor operation at 50% of full power in the event of a major component failure in which repairs cannot be accomplished in a reasonable period of time. The reactor was designed and has extensive safe operating history for operation at 50% of ten megawatts cooling capacity. In this event, the shutdown system shall be secured in a manner such as to assure system integrity.

Specification 5.2.e assures strength and corrosion resistance of the coolant system components and excepts some smaller components, such as instrumentation of the system, which are not commercially available in the materials specified. The size of these components is such that a failure would not result in a hazard to safe operation of the reactor.

Aluminum alloys and stainless steels are well-suited for service in the chemical environment and temperature/pressure conditions of the coolant systems. The major purpose in specifying these materials is to minimize or prevent corrosion, whereas aluminum and its alloys are also particularly well-suited for service in a neutron-rich environment. The use of exception b is intended primarily to apply to instrumentation components that are not commercially available in the materials specified. It is also an acknowledgement that these components perform better and more reliably using materials other than aluminum alloys and stainless steels.

Other non-instrumentation components can also be considered under this exception. Examples would be the carbon face materials in pump mechanical seals, cobalt-alloyed valve disc facings, rubber valve diaphragms, and the

5.2 **Reactor Coolant Systems** - Continued

beryllium reflector. These materials are evaluated with regard to corrosion potential, both individually and in galvanic potential with their surroundings, fatigue or cycle lifetime, temperature and pressure service reliability, and potential for dissolution, erosion, and activation in the coolant.

5.3 Reactor Core and Fuel

Applicability:
This specification applies to the reactor core and fuel elements.

Objective:
The objective of this specification is to specify the general reactor core configuration and to assure that the fuel elements are of a type designed for use in the reactor.

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

a. The average reactor core temperature coefficient of reactivity shall be more negative than -6.0×10^{-5} $\Delta k/k/°F$.

b. The average reactor core void coefficient of reactivity shall be more negative than -2.0×10^{-3} $\Delta k/k/\%$ void.

c. The peak burnup for UAl_X dispersion fuel shall not exceed a calculated 2.3×10^{21} fissions per cubic centimeter.

d. The regulating blade total reactivity worth shall be a maximum of 6.0×10^{-3} $\Delta k/k$.

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

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

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

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

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

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

k. The reactor shall have five (5) control blades between the pressure vessel and beryllium reflector. Four (4) of the control blades shall be made of boron and aluminum for coarse control (shim blades) of reactor power. One (1) control

5.3 **Reactor Core and Fuel** – Continued

blade shall be made of stainless steel for fine control (regulating blade) of reactor power.

l. The reactor shall have the following experimental facilities:
   1. Six (6) beam tubes which penetrate the graphite reflector;
   2. A center test hole located in the flux trap;
   3. A portion of the graphite reflector;
   4. A bulk pool consisting of the water region above and outside the graphite reflector; and
   5. A thermal column.

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

Bases:

a. Specification 5.3.a limits one of the parameters which assures that core damage will not occur following any credible step reactivity insertion as analyzed in Section 13.2.2 of the SAR.

b. The average core void coefficient of reactivity also limits the step reactivity insertion accident as analyzed in Section 13.2.2 of the SAR.

c. Specification 5.3.c restricts the peak fissions per cubic centimeter burnup to values that have been correlated to result in less than 10% swelling of the fuel plates. It has been found that fuel plate swelling of less than 10% has no detrimental effect on fuel plate performance (Ref.: Change No. 4 to Facility License No. R-103, Change No. 6 to Facility License No. R-103, and Application dated September 12, 1986 with supplements).

d. The regulating blade total reactivity worth is limited by Specification 5.3.d such that any condition resulting in the step insertion of the maximum worth of 6×10^{-3} $\Delta k/k$ will not result in fuel plate damage.

e.- h. The MURR reactor fuel elements are one of a configuration (aluminide UAl_X dispersion fuel system) successfully and extensively used for many years in test and research reactors. Specifications 5.3.e, 5.3.f, 5.3.g and 5.3.h require fuel content, materials and dimensions of the fuel elements to be in accordance with the design and fabrication specifications (Ref. Section 4.2.1 of the SAR).

i. Specification 5.3.i assures that the reactor fuel is properly positioned in the pressure vessel during operation (Ref. Section 4.2.5 of the SAR).

5.3 **Reactor Core and Fuel** – Continued

j. Specification 5.3.j assures proper neutron reflection as required by design (Ref. Section 4.2.3 of the SAR).

k. Specification 5.3.k assures reactivity of the reactor is properly controlled as required by design (Ref. Section 4.2.2 of the SAR).

l. Specification 5.3.l assures that the reactor consists of the experimental facilities as required by design (Ref. Chapter 10 of the SAR).

m. Specification 5.3.m ensures that, during a startup, the reactor power level is continuously monitored over the entire range (Ref. Section 7.4 of the SAR).

5.4 **Fuel Storage**

Applicability:
This specification applies to the storage of reactor fuel at times when it is not in the reactor core.

Objective:
The objective of this specification is to assure that fuel which is stored shall not become critical and will not reach an unsafe temperature.

Specification:
The following design features apply to fuel storage:

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

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

Bases:
a. - b. The limits imposed by Specifications 5.4.a and 5.4.b are conservative and assure safe fuel storage.

5.5 Reactor Containment Building

Applicability:
This specification applies to the building in which the reactor is located.

Objective:
The objective of this specification is to assure adequate restriction to the accidental release of radioactivity to the environment.

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

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

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

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

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

Bases:

a. No credible accident scenario has been identified which can result in a significant overpressure condition in the reactor containment building. However, Specification 5.5.a assures that a sufficient free volume exists to prevent any pressure buildup in the reactor containment building (Ref. Section 6.2.2.2 of the SAR).

b. Specification 5.5.b assures a sufficient stack height for more than adequate atmospheric dispersion.

c. Specification 5.5.c assures that the reactor containment building will have sufficient integrity to limit the leakage of contained potentially radioactive air in

5.5 **Reactor Containment Building** - Continued

the event of any reactor accident to ensure exposures are maintained below the limits of 10 CFR 20 (Ref. Sections 6.2.10 and 13.2.1 of the SAR).

d. Specification 5.5.d assures safe and secure storage of fresh fuel.

5.6 Emergency Electrical Power System

Applicability:
This specification applies to the facility emergency electrical power system.

Objective:
The objective of this specification is to assure adequate emergency electrical power in the event of normal electrical power failure.

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

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

Bases:
a. The emergency electrical power system is described in Section 8.2 of the SAR. Specification 5.6.a assures that a system exists to provide the necessary electrical power to monitor the reactor systems and assure personnel safety in the event of a normal power failure to the reactor facility.

6 ADMINISTRATIVE CONTROLS

6.1 Organization

a. The organizational structure of the University of Missouri-Columbia (MU) relating to the University of Missouri Research Reactor (MURR) shall be as shown in Figure 6.0.

b. The following positions shall have direct responsibility in implementing the Technical Specifications as designated throughout this document:

(1) Office of the Chancellor (Level 1): Shall be responsible for directing MU's research mission, the quality and effectiveness of all programs and dedicating university resources necessary to ensure that all research, education and service are conducted in accordance with applicable federal, state and local regulations and accreditation requirements.

(2) Reactor Facility Director (Level 2): Shall be responsible for establishing the policies that minimize radiation exposure to the public and to radiation workers, and that ensures that the requirements of the license and Technical Specifications are met.

(3) Reactor Manager (Level 3): To safeguard the public and facility personnel from undue radiation exposure, the Reactor Manager shall be responsible for:

i. Compliance with Technical Specifications and license requirements regarding reactor operation, maintenance and surveillance; and

ii. Oversight of the experiment review process.

(4) Reactor Health Physics Manager (Level 3): To safeguard the public and facility personnel from undue radiation exposure, the Reactor Health Physics Manager shall be responsible for:

i. Compliance with Technical Specifications and license requirements regarding radiation safety, byproduct material handling and the shipment of byproduct material; and

ii. Implementation of the Radiation Protection Program.

(5) Reactor Operations Staff (Level 4): Shall be responsible for the manipulation of reactor controls, monitoring of instrumentation, and operation and maintenance of reactor-related equipment.

6.1 **Organization** - Continued

(6) Reactor Health Physics Staff (Level 4): Shall be responsible for directing research, training, and monitoring programs in order to protect personnel from radiation hazards and to assure compliance with federal, state, and MU regulations.

c. At a minimum during reactor operation, there shall be two (2) facility staff personnel at the facility. One of these individuals shall be a Reactor Operator or a Senior Reactor Operator licensed pursuant to 10 CFR 55. The other individual shall be knowledgeable of the facility.

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

(1) Management personnel;

(2) Reactor Health Physics personnel; and

(3) Reactor Operations personnel.

e. A Senior Reactor Operator licensed pursuant to 10 CFR 55 shall be present at the facility or readily available on call at all times during operation. Readily available on call means an individual who:

(1) Has been specifically designated and the designation known to the operator on duty;

(2) Can be rapidly contacted by phone, by the operator on duty; and

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

f. Events requiring the presence of a Senior Reactor Operator at the facility are:

(1) Initial startup and approach to power;

(2) All fuel or control rod relocations within the reactor core region;

(3) Relocation of any experiment with a reactivity worth greater than 0.0074 $\Delta k/k$; and

(4) Recovery from an unplanned or unscheduled shutdown or significant power reduction.

6.1 **Organization** - Continued

g. The selection, training, and requalification of operations personnel should be in accordance with the requirements of ANSI/ANS-15.4-2007, "Selection and Training of Personnel for Research Reactors." Qualification and requalification of licensed reactor operators shall be performed in accordance with a U.S. Nuclear Regulatory Commission (NRC) approved program.

6.2 **Review and Audit**

a. A Reactor Advisory Committee (RAC) shall provide independent oversight in matters pertaining to the safe operation of the reactor and with regard to planned research activities and use of the facility building and equipment. The RAC shall be composed of at least five (5) members who have knowledge of experimental activities, reactor operations, University business policy, or related subjects. The Committee members shall be appointed by, and report to, the Office of the Chancellor. The RAC shall review:

(1) Determinations that proposed changes to MURR equipment, systems, tests, experiments or procedures are allowed pursuant to 10 CFR 50.59.

(2) All new procedures and major revisions thereto having safety significance, proposed changes to reactor facility equipment, or systems having safety significance. Changes to procedures that do not change their original intent may be made without prior RAC review if approved by the TS-designated manager, either the Reactor Health Physics Manager or Reactor Manager, or a designated alternate who is a member of Reactor Health Physics or a Senior Reactor Operator, respectively. All such changes to the procedures shall be documented, reviewed pursuant to 10 CFR 50.59, and subsequently reviewed by the RAC;

(3) Proposed experiments significantly different from any previously reviewed or which involve a question pursuant to 10 CFR 50.59;

(4) Proposed changes in the Technical Specifications or the license;

(5) The circumstances of reportable occurrences and violations of the Technical Specifications or license and the measures taken to prevent a recurrence;

(6) Violations of internal procedures or operating abnormalities having safety significance; and

(7) Reports from audits required by the Technical Specifications.

6.2 **Review and Audit** - Continued

b. The RAC may appoint subcommittees consisting of knowledgeable members of the public, students, faculty, and staff of MU when it deems it necessary in order to effectively discharge its primary responsibilities. When subcommittees are appointed, these subcommittees shall consist of no less than three (3) members with no more than one (1) student appointed to each subcommittee. The subcommittees may be authorized to act on behalf of the RAC.

The RAC and its subcommittees shall maintain minutes of meetings in which the items considered and the committees' recommendations are recorded. Dissemination of the minutes to the Office of the Chancellor, the RAC and its subcommittees shall be done within three (3) months after the meetings. Independent actions of the subcommittees shall be reviewed by the parent committee at the next regular meeting. A quorum of the committee or the subcommittees consisting of at least fifty percent of the appointed members shall be present at any meeting to conduct the business of the committee or subcommittee. Additionally, reactor facility staff shall not constitute greater than fifty percent of the quorum. Reactor facility staff shall not constitute a majority of the RAC. The RAC shall meet at least quarterly.

A meeting of a subcommittee shall not be deemed to satisfy the requirement of the parent committee to meet at least once during each calendar quarter.

c. Any additions, modifications or maintenance to the systems described in these Specifications shall be made and tested in accordance with the specifications to which the system was originally designed and fabricated or to specifications approved by the NRC.

d. Following a favorable review by the NRC, the RAC, or the Reactor Facility Management, as appropriate, and prior to conducting any experiment, the Reactor Manager shall sign an authorizing form which contains the basis for the favorable review.

e. Audits:

(1) Audits of the following functions shall be conducted by an individual or group without immediate responsibility in the area to be audited:

i. Facility Operations, for conformance to the Technical Specifications and license conditions, at least annually;

ii. Operator Requalification Program, for compliance with the approved program, at least every two (2) years; and

6.2 **Review and Audit** - Continued

iii. Corrective Action items associated with reactor safety, at least annually.

(2) Audit findings which affect reactor safety shall be immediately reported to the Reactor Facility Director. A written report of the findings shall be submitted to the Reactor Facility Director, the RAC and its subcommittees within three (3) months after the audit has been completed.

6.3 **Radiation Safety**

a. The Reactor Health Physics Manager shall be responsible for the implementation of the Radiation Protection Program. The requirements of the Radiation Protection Program are established in 10 CFR 20. The program should use the guidelines of American National Standard "Radiation Protection at Research Reactor Facilities," ANSI/ANS-15.11-1993 (R2004).

6.4 **Procedures**

a. Written procedures shall be in effect for operation of the reactor, including the following:

(1) Startup, operation, and shutdown of the reactor;

(2) Fuel loading, unloading and movement within the reactor;

(3) Maintenance of major components of systems that could have an effect on reactor safety;

(4) Surveillance checks, calibrations and inspections that may affect reactor safety;

(5) Administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect reactor safety or core reactivity; and

(6) Implementation of the Emergency and Physical Security Plans.

b. Written procedures shall be in effect for radiological control, and the preparation for shipping and the shipping of byproduct material produced under the facility operating license.

c. The Reactor Manager shall approve and annually review the procedures for normal operations of the reactor and the Emergency Plan implementing

6.4 **Procedures** - Continued

procedures. The Reactor Health Physics Manager shall approve and annually review the radiological control procedures and the procedures for the preparation for shipping and the shipping of byproduct material.

d. Deviations from procedures required by this Specification may be enacted by a Senior Reactor Operator or member of Reactor Health Physics, as applicable. Such deviations shall be documented, reviewed pursuant to 10 CFR 50.59, and reported within 24 hours or the next working day to the Reactor Manager or Reactor Health Physics Manager or designated alternate.

6.5 **Experiment Review and Approval**

a. Approved experiments shall be carried out in accordance with established and approved procedures. Procedures related to experiment review and approval shall include the following:

(1) All new experiments or class of experiments shall be reviewed by the RAC and approved in writing by the Reactor Manager.

(2) Substantive changes to previously approved experiments shall be made only after review by the RAC and approved in writing by the Reactor Manager.

6.6 **Reportable Events and Required Actions**

a. <u>Safety Limit Violation</u> - In the event of a safety limit violation, the following actions shall be taken:

(1) The reactor shall be shut down and reactor operation shall not be resumed until authorized by the NRC pursuant to 10 CFR 50.36(c)(1);

(2) The safety limit violation shall be promptly reported to the Reactor Manager and Reactor Facility Director, or designated alternates;

(3) The safety limit violation shall be promptly reported to the NRC. Prompt reporting of the violation shall be made by MU, by telephone, and subsequently confirmed in writing, or email, to the NRC Operations Center no later than the following working day;

(4) A detailed follow-up report shall be prepared. The report shall include the following:

6.6 **Reportable Events and Required Actions** - Continued

i. Applicable circumstances leading to the violation including, when known, the causes and contributing factors;

ii. Date and approximate time of the occurrence;

iii. Effect of the violation upon the reactor and associated systems;

iv. Effect of the violation on the health and safety of the facility staff and general public; and

v. Corrective actions to prevent recurrence.

(5) The follow-up report shall be submitted within fourteen (14) days to the NRC Document Control Desk.

b. Release of Radioactivity - Should a release of radioactivity greater than the allowable limits occur from the reactor facility boundary, the following actions shall be taken:

(1) Reactor conditions shall be returned to normal or the reactor shall be shut down;

(2) The release of radioactivity shall be promptly reported to the Reactor Manager and Reactor Facility Director, or designated alternates;

(3) The release of radioactivity shall be promptly reported to the NRC. Prompt reporting of the violation shall be made by MU, by telephone, and subsequently confirmed in writing, or email, to the NRC Operations Center no later than the following working day;

(4) If it is necessary to shut down the reactor to correct the occurrence, operations shall not be resumed until authorized by the Reactor Facility Director, or designated alternate; and

(5) A detailed follow-up report shall be prepared. The follow-up report shall be submitted within fourteen (14) days to the NRC Document Control Desk.

c. Other Reportable Occurrences - In the event of an Abnormal Occurrence, as defined by Specification 1.1, the following actions shall be taken:

(Note: Where components or systems are provided in addition to those required by these Technical Specifications, the failure of the extra components or systems

6.6 **Reportable Events and Required Actions** – Continued

is not considered reportable provided that the minimum numbers of components or systems specified or required perform their intended reactor safety function.)

(1) The Abnormal Occurrence shall be promptly reported to the NRC. Prompt reporting of the Abnormal Occurrence shall be made by MU, by telephone and subsequently confirmed in writing or email, to the NRC Operations Center no later than the following working day;

(2) The Abnormal Occurrence shall be promptly reported to the Reactor Manager and Reactor Facility Director, or designated alternates;

(3) A detailed follow-up report shall be prepared. The follow-up report shall be submitted within fourteen (14) days to the NRC Document Control Desk; and

(4) The reactor shall be shut down or placed in a safe condition and return to normal reactor operations shall not be allowed until authorized by the Reactor Facility Director, or alternate.

d. Other Reports - A written report shall be submitted to the NRC Document Control Desk within thirty (30) days of:

(1) Any significant change(s) in the transient or accident analyses as described in the SAR; and

(2) Permanent changes in the facility organization involving the Office of the Chancellor or the Reactor Facility Director.

e. Annual Report - An annual operating report shall be submitted to the NRC within sixty (60) days following the end of each calendar year. The report shall include the following information for the preceding year:

(1) A brief narrative summary of (a) operating experience (including operations designed to measure reactor characteristics), (b) changes in the reactor facility design, performance characteristics, and operating procedures related to reactor safety occurring during the reporting period, and (c) results of surveillance tests and inspections;

(2) A tabulation showing the energy generated by the reactor (in megawatt-days);

(3) The number of emergency shutdowns and inadvertent scrams, including the reasons therefore and corrective action, if any, taken;

6.6 **Reportable Events and Required Actions** - Continued

(4) Discussion of the major maintenance operations performed during the period, including the effects, if any, on the safe operation of the reactor;

(5) A summary of each modification to the reactor facility or change to the procedures, tests and experiments carried out under the conditions of 10 CFR 50.59;

(6) A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as measured at or prior to the point of such release or discharge;

(7) A description of any environmental surveys performed outside the reactor facility; and

(8) A summary of radiation exposures received by facility staff, experimenters, and visitors, including the dates and time of significant exposure, and a brief summary of the results of radiation and contamination surveys performed within the facility.

6.7 **Records**

Records of the following activities shall be maintained and retained for the periods specified below. The records may be in the form of logs, data sheets, or other suitable forms or documents. The required information may be contained in single or multiple records, or a combination thereof.

a. Lifetime Records - The following records shall be retained for the lifetime of the reactor facility: (Note: Applicable annual reports, if they contain all of the required information, may be used as records in this section.)

(1) Gaseous and liquid radioactive effluents released to the environs;

(2) Off-site environmental-monitoring surveys required by the Technical Specifications;

(3) Radiation exposure for all monitored personnel;

(4) Updated drawings of the reactor facility; and

(5) Reviews and reports pertaining to a violation of a safety limit, limiting safety system setting, or limiting conditions for operations.

6.7 **Records** - Continued

b. Five Year Records - The following records shall be maintained for a period of at least five (5) years or for the life of the component involved, whichever is shorter:

(1) Normal reactor facility operation (but not including supporting documents such as checklists, log sheets, etc. which shall be maintained for a period of at least one year);

(2) Principal maintenance operations;

(3) Reportable occurrences;

(4) Surveillance activities required by the Technical Specifications;

(5) Reactor facility radiation and contamination surveys required by applicable regulations;

(6) Experiments performed with the reactor;

(7) Fuel inventories, receipts and shipments;

(8) Approved changes to operating procedures; and

(9) Records of meetings and audit reports of the review and audit group.

c. Operator Licensing Records - Record of training and requalification of licensed reactor operators and senior reactor operators shall be retained at all times the individual is employed or until the license is renewed.

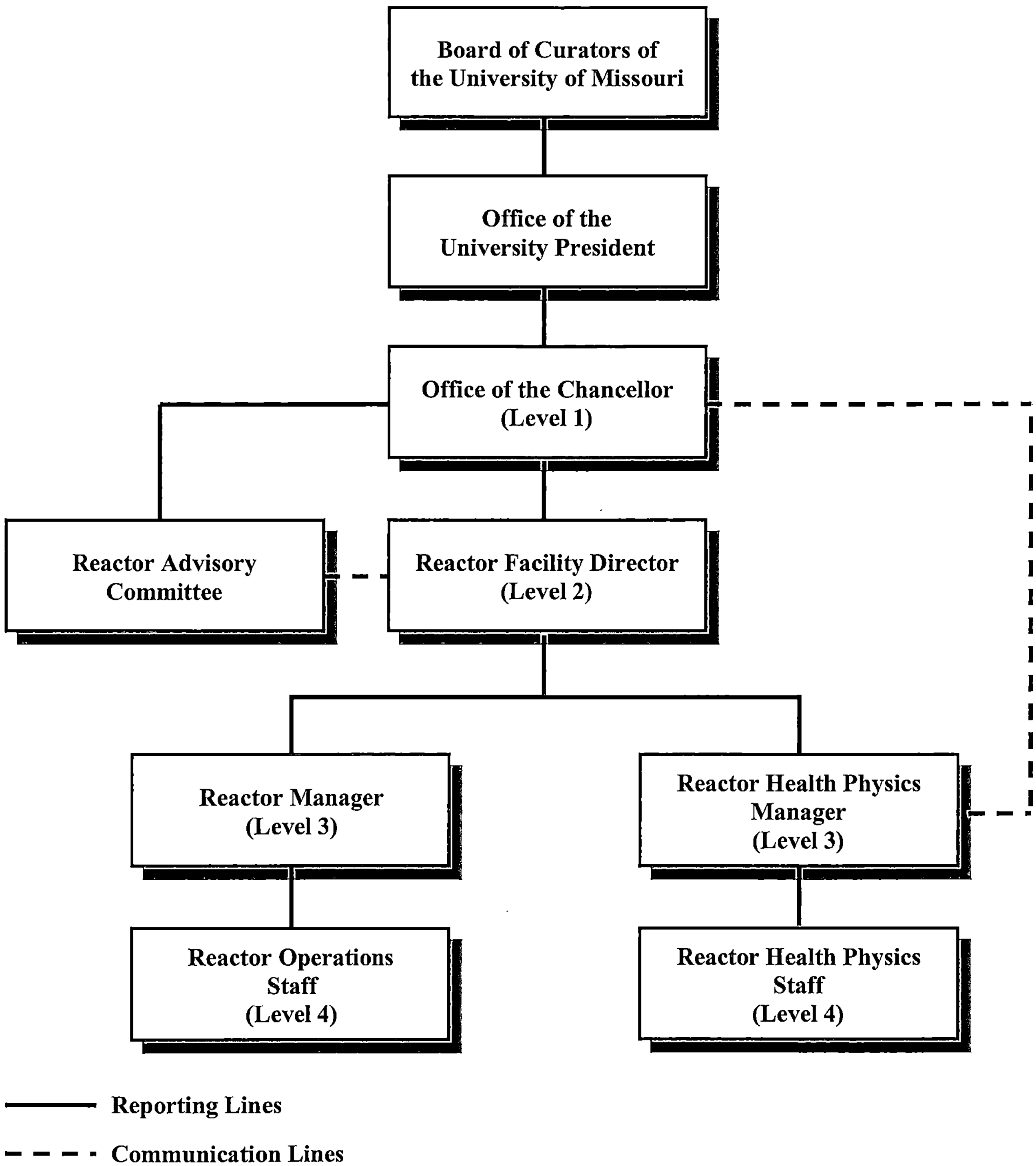

FIGURE 6.0
UNIVERSITY OF MISSOURI RESEARCH REACTOR (MURR)
ORGANIZATION