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RESEARCH REACTOR CENTER

May 31, 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 October 28, 2015, 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 John L. Fruits, the facility Reactor Manager, at (573) 882-5319 or FruitsJ@missouri.edu.

Sincerely,

Ralph A. Butler, P.E.
Director

RAB/jlb

Enclosures

A020
NRR

1513 Research Park Drive Columbia, MO 65211 Phone: 573-882-4211 Fax: 573-882-6360 Web: www.murr.missouri.edu

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REFERENCE: Docket 50-186
University of Missouri-Columbia Research Reactor
Amended Facility Operating License 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 October 28, 2015

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.

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

Attached are responses to the October 28, 2015, request for additional information regarding the proposed Technical Specifications. Additionally, Technical Specification changes, as issued by Amendment No. 37 to the current facility operating license (NRC letter dated March 11, 2016), have also been incorporated into the revised proposed Technical Specifications.

If there are any questions regarding this response, please contact me at (573) 882-5319 or FruitsJ@missouri.edu. I declare under penalty of perjury that the foregoing is true and correct.

Sincerely,

John L. Fruits
Reactor Manager

ENDORSEMENT:
Reviewed and Approved,

Ralph A. Butler, P.E.
Director

DIANE STRUMPF
Notary Public - Notary Seal
State of Missouri
Commissioned for Boone County
My Commission Expires: January 31, 2018
Commission Number: 13398861

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

State of Missouri
County of Boone
subscribed and sworn to before me this 31st Day of May 2016
Diane Strumpf

Attachments:

1. Appendix A, Technical Specifications for the University of Missouri Research Reactor, Facility Operating License R-103, Docket 50-186, as revised.
2. Compliance Procedure CP-23, "DPS-929"
3. Form FM-152, "Fuel Element Inspection"
4. Compliance Procedure CP-17, "Emergency Generator Load Test"

1. *TS 1.0 Definitions:*

   a. *The proposed MURR TS definition for "Abnormal Occurrences" provides criteria that may not be consistent with the guidance provided ANSI/ANS-15.1 -2007, Section 6.7.2, as described below:*

      i. *Specification c, which states, in part, "unless the malfunction or condition is discovered during maintenance tests or periods of reactor shutdowns," is not consistent with the guidance in ANSI/ANS-15.1-2007, Section 6. 7 .2, item (1)(c)(iii), states, in part, "If the malfunction or condition is caused by maintenance, then no report is required."*

      Specification 1.1.c has been revised as follows:

      "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;"

      ii. *Specification e, which states, in part, "which could result in exceeding prescribed radiation exposure limits of personnel or environment or both;" is an exception not found in the guidance in ANSI/ANS-1 5.1-2007, Section 6.7.2, item (1)(c)(v).*

      Specification 1.1.e has been revised as follows:

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

   b. *The proposed MURR TS does not include a definition for "Containment," as provided in the guidance in ANSI/ANS-15.1-2007.*

      The Reactor Containment Building is defined in Specification 1.19 (old Specification 1.18). It has been revised as follows:

      "**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."

   c. *The proposed MURR TS does not include a definition for "Core Configuration," as provided in the guidance in ANSI/ANS-15.1-2007.*

      The following definition for "Core Configuration" has been added as Specification 1.8:

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

d. *The proposed MURR TS definition of "Excess Reactivity" does not contain the provision, as provided in the guidance in ANSI/ANS-15.1-2007, that the evaluation is performed at "reference core conditions."*

Specification 1.9 (old Specification 1.8) has been revised as follows:

"**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."

e. *The proposed MURR TS definition for "Operational Modes" states that the reactor can be "operated safely." The term "operated safely" is not defined in the TSs and could be subject to interpretation.*

MURR disagrees that the term "operated safely" could be subject to interpretation. Operating the reactor within the safety envelope established by the Safety Analysis Report and the Technical Specifications is operating safely. However, the word "safely" has been deleted from the definition of Operational Modes.

f. *The proposed MURR TS definition for "Reactor Operator" and "Senior Reactor Operator" states "certified." ANSI/ANS-15.1-2007 provides guidance that operators are "licensed."*

The second definition in ANSI/ANS-15.1-2007 is "**certified:** See Licensed." Based on this definition, MURR felt that the words "certified" and "licensed" are interchangeable. However, the definitions of "Reactor Operator" and "Senior Reactor Operator" have been revised as follows:

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

"**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."

g. *The proposed MURR TS definition for "Reactor Safety System" does not include one of the criteria provided in the guidance in ANSI/ANS-15.1-2007, which states, in part, "or to provide information for initiation of manual protective action."*

In reviewing other currently licensed research reactor Technical Specifications, not all facilities have this criterion in their definition of "Reactor Safety System." Specifically Oregon State University defines reactor safety systems as "Reactor safety systems are those systems, including their associated input channels, which are designed to initiate, automatically or manually, a

reactor scram for the primary purpose of protecting the reactor." MURR feels that this criterion is not required in its definition.

h. *The proposed MURR TS definition for "Reactor Shutdown" does not appear to be consistent with the guidance provided in NUREG-1537, which includes criteria for: 1) the shutdown reactivity ($1.00); 2) the reference core condition; and, 3) the reactivity worth of any installed experiments.*

Specification 1.26 (old Specification 1.25) has been revised as follows:

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

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

AND

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

i. *The proposed MURR TS definitions for "Shim Blade (Rod)" and "Regulating Blade (Rod)" do not indicate any safety function, e.g., whether the blades are scrammable.*

All safety functions for the control blades are described in detail in Sections 7.5.5 and 7.7, Rod Run-In System and Reactor Safety System, respectively, of the Safety Analysis Report. Specifically, regulating blade position (bottomed and 10% withdrawn) provides inputs to the rod run-in system; however, the regulating blade itself does not provide any safety functions. The shim blades provide both rod run-in and scram functions. The definition of "Rod Control" in ANSI/ANS-15.1-2007 does not use the term "scrammable" but instead uses the term "safety function," which MURR used in its definition. However, to provide greater clarification regarding control blade safety functions, the definitions for "Regulating Blade (Rod)" and "Shim Blade (Rod)," Specifications 1.28 and 1.37, respectively, have been revised as follows:

"**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."

"**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."

j. *The proposed MURR TS does not appear to include a definition for "Scram Time" as provided in the guidance in ANSI/ANS-15.1-2007.*

The following definition for "Scram Time" has been added as Specification 1.34:

"**Scram Time** - Scram time is the elapsed time between the initiation of a scram signal and specified movement of a control or safety device."

k. *The proposed MURR TS definition for "Unscheduled Shutdown" states, in part, "that occurs after all "Blade Full-In Lights" have cleared." The term "Blade Full-in Lights" does not appear to be clearly defined in the TSs.*

The "Blade Full-In Lights" are described in Section 7.5, Rod Control System, of the Safety Analysis Report. The lights are also listed on Table 7-1, item No. 6. These lights are illuminated when the shim blades are fully inserted. During a reactor startup, because of the inherent design and construction of this system, the "Blade Full-In Lights" do not extinguish immediately upon shim blade withdrawal. They typically will extinguish at a shim blade height of approximately 0.5 inches from fully inserted. MURR feels that a specific definition for the "Blade Full-In Lights" in the Technical Specifications is not required since their use in this case is strictly administrative for documenting an unscheduled shutdown.

2. *TS 2.0 Safety Limits and Limiting Safety System Settings:*

a. *The proposed MURR TS 2.1, Specifications a, b, and c, and accompanying TS Figures 2.0, 2.1 and 2.2, provide SLs that appear to represent the normal operating conditions associated with Core Power, Core Flow, Reactor Water Inlet Temperature, and Pressurizer Level, and thus appear to be LCOs. The guidance in NUREG-1537 appears to provide a SL of 530 degrees Celsius (°C) as an acceptable limit for the MURR fuel type. Explain the SL curves.*

As previously discussed with the NRC, MURR will use the guidance in NUREG-1537 and establish its Safety Limit (SL) based on NUREG-1313, "Safety Evaluation Report Related to the Evaluation of Low-Enriched Uranium Silicide-Aluminum Dispersion Fuel for Use in Non-Power Reactors." The new SL for MURR fuel is a temperature of 986 °F (530 °C).

b. *The proposed MURR TS 2.1, Specifications a, b, and c, differs from the current TS 2.1, Specifications a, b, and c, issued in License Amendment No. 36, by letter dated July 8, 2013 (ADAMS Accession No. ML 13133A349). Provide a justification for the revised wording.*

As described above, MURR will establish its Safety Limit for fuel and cladding temperature at 986 °F (530 °C) using the guidance of NUREG-1313; therefore, this question is no longer relevant.

c. *The proposed MURR TS 2.2, Limiting Safety System Setting (LSSS) provides the reactor power limit in units of percent of licensed power (e.g., 125% of full power). NRC staff is not clear if the measuring channel used in the control room display displays the same units (percent reactor power), and if the measuring channel is calibrated to percent reactor power.*

The measuring units on the reactor power meters (Power Range Level – Channels 4, 5 and 6) on the reactor control console, as shown on Figure 7.2 and listed on Table 7-1 of the Safety Analysis

Report (SAR), are in percent reactor power. The Power Range Channels are also described in detail in Section 7.4.2.3 of the SAR. The measuring channels are also calibrated in percent reactor power.

d. *The proposed MURR TS 2.2, LSSS states that a coolant flow rate of 1,625 gallons per minute (gpm) is required from "either" loop for operation in Mode I. SAR Section 4.6.4.1, indicates that the minimum flow is 1,625 gpm, whereas, your response to RAI A.20, by letter dated July 16, 2010 (ADAMS Accession No. ML 12354A237), states that 1,600 gpm is required for Mode II operation. Explain the difference.*

As described in Section 4.6.4.1 of the Safety Analysis Report (SAR) and the response to RAI A.20, 50 gpm of the total primary coolant flow rate is diverted to the Primary Coolant Demineralizer Loop in Mode I operation and 25 gpm is diverted in Mode II operation prior to flowing through the reactor core. Therefore, to ensure that 1600 gpm is flowing through the reactor core in Mode II operation, the flow rate as sensed by Flow Transmitters FT 912A and FT 912B must indicate at least 1625 gpm (25 gpm greater than what is flowing through the core) on whichever loop is in operation. Both primary coolant system loops must be in operation during Mode I but only one loop must be in operation in Mode II.

3. *TS 3.0 Limiting Conditions for Operation:*

a. *The proposed MURR TS 3.1, Specification c, Exception, states, in part, "the reactor may be operated with less than eight fuel assemblies." The NRC staff is not clear how operation in this mode is consistent with the definitions of the TS Operating Modes (Modes I, II, or Ill) as provided in the proposed MURR TS 2.2.*

Operation with less than eight (8) fuel assemblies is limited to a power level of 100 watts in Mode III. As stated in Note (1) to Specifications 3.2.f and 3.2.g, "Not required (a) below 50 kW operation with the natural convection flange and reactor pressure vessel cover removed or (b) in operation with the reactor subcritical by a margin of at least 0.015 $\Delta k/k$." Therefore, operation with less than eight (8) fuel assemblies in Mode III is by natural convection (natural convection flange and reactor pressure vessel cover removed) or with the reactor subcritical by a margin of at least 0.015 $\Delta k/k$.

b. *The proposed MURR TS 3.1, Specification d, provides a fuel burnup limit. NUREG-1537, Appendix 14.1, Section 4.1, item (6), provides guidance for surveillance requirements for burnup limits. There does not appear to be a corresponding SR in the proposed MURR TSs. Propose a SR, or, if a SR is not practical, propose moving the specification for fuel burnup to TS Section 5, Design Features.*

Specification 3.1.d was moved to Section 5 as suggested because a Surveillance Requirement is not practical. This specification is now Specification 5.3.c.

c. *The proposed MURR TS 3.1, Specification e, provides the LCO for inoperable (damaged) fuel that includes "anomalies" and a coolant channel dimensional check. NUREG-1537, Appendix 14.1, Section 3.1, item (6) (a), provides guidance that damaged fuel should have limits on growth, bowing, or bending, and detectable fission products. It also includes the guidance that the specification should reference the fuel manufacturer's guidance or recommendations for detecting deterioration. Explain the proposed LCO definition of fuel "anomalies," and if the additional criteria as provided by the guidance in NUREG-1537 (fuel manufacturer) should also be included in the proposed MURR TS 3.1, Specification e, to establish fuel operability.*

The manufacturer of MURR fuel elements provides no guidance or recommendations for detecting fuel element deterioration. Specification 3.3.c limits the iodine-131 concentration in the primary coolant system to 5×10^{-3} uCi/ml. This specification provides for the early detection of a leaking fuel element. 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 MURR 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). Therefore, MURR's definition of fuel anomalies is an increase in iodine-131 activity in the primary coolant system, a coolant channel dimension change of 10 mils or greater and failure of a visual inspection. Attached (Attachment 3) is form FM-152, "Fuel Element Inspection."

d. *The proposed MURR TS 3.2, Specification c, states, in part, "The shim blades shall be capable of insertion to 20% withdrawn position." The NRC staff reviewed the information provided in the associated TS Basis, which described the reactivity insertion at the 20% withdrawn position (91 %), and could not find a supporting analysis in the referenced SAR Section 13.2.2. Therefore, the basis for the 20% withdrawn position is not clear.*

As stated in the MURR letter dated February 8, 2016, in response to the NRC's Request for Additional Information dated March 23, 2016 (follow-up information to the April 17, 2015 RAIs), Reactivity Insertion Accidents are the only MURR accidents where control blade worths and full insertion times are utilized. From the original configuration of MURR in 1966, the shim control blades were designed to be inserted to their 20% withdrawn position (or 80% inserted) in less than 0.7 seconds. This ensures prompt shutdown of the reactor in the event a reactor scram signal, manual or automatic, is received. The 20% withdrawn position is defined as 20% of the control blade full travel of 26 inches measured from the fully inserted position. Below the 20% withdrawn position the control blade fall is cushioned by a dashpot assembly. Approximately 91% of the control blade total worth is inserted at the 20% position. This is an original design feature of the reactor and its purpose has not been altered in over 49 years of operation. The same Technical Specification will remain in the relicensing Technical Specifications. Therefore, for accident analyses, the 80% inserted position is considered the full insertion position and only 91% of the total control blade worth is conservatively utilized in the analyses instead of 100%.

e. *The proposed MURR TS 3.2, Specifications d and e, provide a reactivity insertion rate limit for the regulating blade and shim blades (Δk/k/sec).*

   i. *The basis for the limit for the regulating blade is not provided or referenced. Indicate if a supporting analysis is available.*

   The maximum rate of reactivity insertion for the regulating blade has been changed to 1.5×10^{-4} Δk/k/sec. This is based on a regulating blade total reactivity worth limit of 6.0×10^{-3} Δk/k (Specification 5.3.d) and a blade travel speed of 40 inches/min.

   ii. *The associated SR requirements, TS 4.2, Specifications d and e, provide limits based on blade speed. The correlation between blade speed (TS 4.2) and the associated reactivity insertion rate (TS 3.2) is not clear.*

   Based on a total shim blade reactivity worth of 0.1838 Δk/k and a maximum shim blade travel speed of 2 inches/min in the inward direction, the maximum insertion rate of negative reactivity would be 2.4×10^{-4} Δk/k/sec. Based on a maximum shim blade travel speed of 1 inch/min in the outward direction, the maximum insertion rate of positive reactivity would be 1.2×10^{-4} Δk/k/sec (or 2.1×10^{-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 3.0×10^{-4} Δk/k/sec. The Continuous Rod Withdrawal Accident analyzed in Addenda 1 and 5 to the MURR Hazards Summary Report uses reactivity insertion rates of 2.78×10^{-4} Δk/k/sec and 3.0×10^{-4} Δk/k/sec, respectively. Based on these insertion rates, a Continuous Rod Withdrawal Accident will be terminated before the core is damaged.

f. *The proposed MURR TS 3.2, Specification f, provides a requirement for operability of the rod run-in function and provides a table of logic requirements.*

   i. *Item 1: The trip set point is indicated as "115% of full power (Max)." Confirm that the channel for the setpoint is calibrated in percent reactor power.*

   Yes, the trip set point for the High Power Level rod run-in is calibrated in percent reactor power.

   ii. *Item 8: The trip setpoint is indicated as "<10% withdrawn and bottomed." NRC staff is not clear as to how to interpret this set point.*

   As described in Section 7.5.5 of the Safety Analysis Report, regulating blade position provides the following two rod run-in functions: (1) Regulation Rod ≤ 10% Withdrawn, and (2) Regulating Rod Bottomed. Although the Rod Run-In function is "Regulating Blade Position," these are two separate inputs to the Rod Run-In System based on the following blade positions: 10% withdrawn or bottomed. The word "and" within the phrase "< 10% withdrawn and bottomed" does not mean that the regulating blade position has to be at both ≤ 10% withdrawn AND bottomed. If a transient would to occur that would cause the regulating blade to insert, then a rod run-in would initiate if regulating blade position reached 10% withdrawn. Should the transient continue to cause the regulating blade to insert to the bottomed position, then another rod run-in would

initiate. To help prevent any future confusion, the regulating blade portion of Specification 3.2.f has been revised as follows – the word "and" has been replaced by the word "or." Additionally, the correct set point is "≤ 10% withdrawn," not "<10% withdrawn." This has been corrected on the revised Technical Specifications.

8.	Regulating Blade Position	2	$2^{(2)}$	$2^{(2)}$	≤ 10% withdrawn or bottomed

g. *The proposed MURR TS 3.2, Specification g, provides a table for the required number of operable reactor safety system instrument channels.*

i. *Item 1: The trip set point is indicated as "125% of full power (Max)." Confirm that the channel setpoint is calibrated in percent reactor power.*

Yes, the trip set point for the High Power Level scram is calibrated in percent reactor power.

ii. *Items 3, 4, 5, and 9 provide trip set points based on flow rates in gpm. However, the associated TS footnotes (2), (3), and (4) suggest a delta-pressure (delta-P) based on a flow value in a table.*

(1) *Provide a reference for the flow value table.*

The table that footnotes (2), (3), and (4) refer to is the table in the Technical Specifications, specifically Specification 3.2.g.

Footnote (2) is specific to measuring primary coolant flow rate using flow transmitters and differential pressure sensors FT 912A, FT 912E, DPS 928A and DPS 928B. Section 7.6.4 of the Safety Analysis Report (SAR) describes these instruments in detail. As stated in the MURR letter dated November 30, 2010, in response to the NRC's Request for Additional Information dated June 1, 2010, specifically General RAI 0.1, "The two (2) tube-type, water-to-shell primary coolant heat exchangers have been replaced by two (2) water-to-water plate-type heat exchangers. The differential pressure sensors associated with each primary coolant heat exchanger have been replaced with a flow orifice/transmitter combination in the associated loops." Therefore, since submittal of the SAR in 2006, DPS 928A and DPS 928B have been replaced by FT 912G and FT 912H, respectively. All four (4) flow transmitters – FT 912A, FT 912E, FT 912G and FT 912H – now provide indication in gpm. MURR wishes to leave this footnote, which is worded the same in the currently licensed Technical Specifications, in order to provide for future flexibility.

Instrumentation originally and currently installed to monitor flow rate through the reactor core (Differential Pressure Across the Core) is a differential pressure sensor that sends a signal to a meter in the control room that reads out in pressure. An Alarm Unit within the current loop provides an input signal to the Reactor Safety System which scrams the reactor on low flow rate (See Section 7.6.4 of the SAR for a detailed description). Attached (Attachment 2) is Compliance Procedure CP-23, "DPS-929," which is the procedure used to satisfy the Technical Specification

Surveillance Requirement for this instrument. As you can see from the datasheet, calibration is performed by introduction of a pressure test signal with a corresponding response on the control room meter in psig. In addition to providing an additional method of determining primary coolant flow rate, the primary reason for monitoring differential pressure across the core is ensuring that no blockage or obstructions are present. Meter indication in pressure versus flow rate is much more accurate in providing this information since smaller pressure differentials will be more noticeable.

(2) Explain if the setpoint is based on flow or delta-P.

The set point is based on flow, which is determined using differential pressure across the reactor core.

(3) Indicate if the control room indications are provided in units of flow or delta-P.

As stated above, the indication for the DPS-929 meter in the reactor control room is in psig.

h. *The proposed MURR TS 3.3, provides requirements for reactor coolant systems.*

i. *Specification d, references 10 CFR Part 20, Appendix B, Table 3. For releases to the public, the NRC staff is questioning if the TS should refer to the effluent concentrations in 10 CFR Part 20, Appendix B, and Table 2?*

MURR believes that 10 CFR 20, Appendix B, Table 3 is the correct reference since Table 3 provides the limits on release to sanitary sewers. The Secondary Coolant System has a blowdown feature to help control secondary coolant conductivity. The coolant that is blowdown from the Secondary Coolant System is released straight to the sanitary sewer.

ii. *Specifications f and h, provide limits associated with the reactor coolant system pH ranges. For your consideration, the NRC staff has evaluated the use of a conductivity limit to control the pH and provided guidance by letter dated May 11, 2015 (ADAMS Accession No. ML 15114A433).*

Based on the NRC letter dated May 11, 2015 (TAC No. ME8511), the LCO for pool coolant system pH (Specification 3.3.h) has been removed since the MURR pool coolant system is an open pool system. The LCO for primary coolant system pH (Specification 3.3.e) will remain since the MURR primary coolant system is a closed system.

iii. *Specification i, provides an anti-siphon pressure specification. It also appears to provide additional requirements and operator actions that could be considered TS action statements, LCOs and SRs, in the case of a low-pressure alarm or low water level condition. Determine if the requirements for a low-pressure alarm or low water level condition constitute additional LCO or SRs.*

MURR agrees that Specification 3.3.i is operational guidance and should not be included in the Technical Specifications. This information was a carryover from the currently approved Technical Specifications. The LCOs that are specific to the Anti-Siphon System are 3.2.f.6, 3.3.a(1), and 3.5.b.2. No additional LCOs are required for the Anti-Siphon System.

i. *The guidance in NUREG-1537 and ANSI/ANS-15.1-2007, Section 3.3, "Coolant Systems," states, in part, "minimum operating equipment, or operating limits, or both, shall be specified for shutdown cooling or pump requirements." An LCO for shutdown cooling system minimum requirements does not appear to be included in the proposed MURR TS.*

Specification 3.3.a(3) provides the shutdown cooling system minimum requirement. MURR does not require forced cooling when shutdown, all decay heat is removed via the "in-pool convective cooling system." Section 5.8 of the SAR describes this system in detail. The Decay Heat Removal System is the "in-pool convective cooling system."

j. *The proposed MURR TS 3.4, provides requirements associated with the reactor containment building.*

   i. *The proposed MURR TS 3.4, uses the term "containment integrity" and "personnel airlock," whereas SAR, Section 6.2, "Containment System" appears to use the terms "integrity" and "pedestrian entry." Explain the difference in these terms.*

Section 6.2 also uses the term "containment integrity" as it applies to TS 3.4 (See SAR page 6-7, Section 6.2.3.2). The use of the term "integrity" in Section 6.2, such as the third paragraph of Section 6.2.3, describes the use of sealable closures in order to maintain containment integrity, i.e., sealable enclosures which ensure that sufficient integrity can be maintained to prevent the accidental release of radioactivity to the environment.

The terms "personal airlock" and "pedestrian entry" describe the same components and are used interchangeably at MURR; both terms were used in the original licensing basis documentation.

   ii. *The proposed MURR TS 3.4, requires operation of the exhaust system as a necessary component of "containment integrity." However, from information gained during the NRC staff site visit, it appears that operation of the exhaust system is also required to support the use of non-reactor facilities located in adjoining buildings, such as hot cells. Discuss the need to establish TS LCOs on the operation of the exhaust system, or on the operation of the non-reactor facilities, to provide ventilation of these non-reactor systems.*

As stated in the response to RAI 1.b, by MURR letter dated July 31, 2015, hot cells, glove boxes and fume hoods (processing units) at MURR are controlled in several ways with regard to the radiological aspects of their use and with respect to occupational and public dose considerations. Generally, processing units are controlled by their user group at MURR in accordance with an approved reactor license project authorization (RL Project). A 10 CFR 50.59 screen or evaluation is also performed in conjunction with the RL Project. This control process ensures appropriate

radiologic controls and facilitates the research or production aspects of MURR's mission to provide high quality radiopharmaceutical products to the research and medical communities.

Prior to a MURR research or production group utilizing a hot cell or glove box, evaluations are conducted by the Reactor Health Physics staff to ensure that the radioactive material used is appropriately shielded by the particular processing unit that will be used. Conversely, if a new isotope is identified for use at MURR, an evaluation occurs to determine if the existing fleet of processing units is sufficient to meet the radiation protection needs of the facility for both occupationally exposed staff and the general public. If no such facility exists, then a review process occurs (either within an existing project or during the creation of a new RL Project) as to what design characteristics are needed to provide the appropriate level of radiation protection to staff and to the general public prior to designing or procuring a new processing unit. Similarly, if higher activities are required for the process, existing hot cells or glove boxes will be evaluated in relationship to the characteristics of the proposed nuclides, including the chemistry and ergonomics of the process, necessary for the safe and effective utilization of the radioisotopes.

Within an RL Project evaluation consideration is given with regard to how occupational exposures will be minimized to the radiation workers based on the quantity of nuclides expected and chemical form to be utilized in the hot cell, glove box or fume hood. The RL Project defines and lists the quantities and chemical forms appropriate for the hot cell or glove box dependent on the specific processing unit shielding and ventilation capabilities. Historically, isotopes irradiated at MURR are metals or metallic compounds that are not subject to volatilization or aerosolization. Any heating during the processing of these isotopes is much less than the melting temperature of the metals or metallic compounds supplied for irradiation. Metallic compounds are usually in the form of nitrates or oxides and are thus more prone to decomposition rather than volatilization. In fact, the heating of these compounds during irradiation is considered during the Reactor Utilization Request (RUR) safety evaluation process prior to placing them in the reactor for irradiation to ensure that adverse heating conditions do not occur due to nuclear heating processes, as high temperatures would destroy the compound being irradiated, thus rendering them useless for processing and further utilization.

Therefore, in RL Projects where there are no or minimal concerns with airborne contamination due to the inherently safe characteristics of the compound or element being irradiated, no language exists within the project documentation explicitly stating that this is not a concern. Conversely, if the RL Project does address the use or generation of nuclides capable of becoming airborne hazards, specific controls are established to ensure workers' and the public's safety is maintained.

Ultimately, all RL Projects are reviewed by the Isotope Use Subcommittee (IUS) of the Reactor Advisory Committee (RAC) for comprehensive scrutiny and approval, after review by Reactor Health Physics staff and review and approval by the Reactor Health Physics and Reactor Managers. In summary the RL Project process provides a conclusion of any analysis performed during the review process limiting the quantity of radionuclides used within any hot cell, glove box or fume hood with respect to workers' and the public's safety.

LCOs do exist for the new Iodine-131 Processing Hot Cells.

iii. *The MURR SAR, Section 6.2.6, describes two modes of operation of the containment system: 1) normal operation; and, 2) isolation upon activation of a high radiation signal at the reactor pool bridge or exhaust plenum.*

   (1) It is not clear if either mode is described in TS 3.4.

MURR does not consider the containment system as having two modes of operation. Section 6.2.6 of the SAR describes its operation and the physical components and equipment of the system. Section 7.8.2 also describes its operation and the instrumentation which actuates the containment system, causing the reactor containment building to isolate (Reactor Isolation).

During normal operation, the six (6) items listed under Specification 3.4.a are required to exist at <u>all times</u> unless the reactor is secured and irradiated fuel with a decay time of less than sixty (60) days is not being handled..

Although there is some interrelationship between the Containment System and "containment integrity," specifically the reactor containment building ventilation system's automatically-closing doors and automatically-closing valves, they are still two different systems or conditions.

   (2) It is not clear which equipment is necessary to ensure operability of each mode.

Equipment that is necessary to ensure that the containment system can perform a containment isolation are Specifications 3.2.g(19), which will cause a reactor scram if the reactor is operating, and Specification 3.4.a(3), which will ensure the containment building ventilation system's automatically-closing doors and automatically-closing valves will shut, if not already shut. Specification 4.4.a tests the operability of all of these components.

iv. *Specification a, Item (2), indicates the utility seal trench is filled to a required depth of 4.25 feet (51 inches). SAR Section 6.2.6, indicates a required depth of 65 inches. Explain the difference.*

Specification 3.4.a(2) provides the minimum water level that the utility entry seal trench must be filled to: 4.25 feet or 51 inches. This depth provides overpressure protection of the reactor containment building, which is designed to withstand a peak internal pressure of 2.0 psig. Section 6.2.6 of the SAR states "....the utility entry water seal is filled to a level of approximately 65 inches, and...." The statement does not use the word "required" since this level is a normal operational value and not a Technical Specification requirement.

v. *Specification a, Item (3), states, in part, "all of the reactor containment building ventilation system's automatically-closing doors and automatically closing valves are operable or placed in the closed position." The NRC staff is not clear regarding this TS requirement. A review*

of the associated TS basis or SAR Section 6.2, did not provide an explanation of the operability of the doors and valves. Explain the required operability of the doors and valves.

As stated in Specification 3.4.a(3), the reactor containment building ventilation system's automatically-closing doors and automatically-closing valves are required to be operable or placed in the closed position. As described in SAR Section 7.8.2.1, automatic or manual actuation of the Reactor Isolation System will cause these doors and valves to close if they are not already closed. Hence, operable, as it applies to the reactor containment building ventilation system's automatically-closing doors and valves, means that they will automatically close when required to.

vi. *Specification a, Item (6), refers to the MURR facility annual integrated leak rate test and as such is not a condition of equipment required to achieve containment integrity. It appears to be a surveillance requirement and not an LCO. Explain why this item should not be moved to the SRs.*

A Surveillance Requirement already exists for measuring the leakage rate of the reactor containment building – see Specification 4.4.a. However, MURR feels it is important to list that the most recent reactor containment building leakage rate test was satisfactory as a condition of containment integrity. Should the containment building leakage rate test fail, this requirement ensures that the reactor continues to be secured and irradiated fuel with a decay time of less than sixty (60) days will not be handled.

vii. *Specification b, is not clear to the NRC staff. It appears to contain an applicability statement, and not an LCO. Based on the guidance in NUREG-1537 and ANSI/ANS-15.1-2007, the LCO should be based on the operability of equipment necessary to ensure the operability of the reactor building containment integrity.*

Section 3 of ANSI/ANS-15.1-2007 states "Limiting conditions of operation (LCOs) are those administratively established constraints on equipment and <u>operational characteristics</u> that shall be adhered to during operation of the reactor. "Operational characteristics" is underlined to show emphasis.

MURR believes that ensuring containment integrity is maintained at all times unless the reactor is secured and irradiated fuel with a decay time of less than sixty (60) days is not being handled is a critical operating characteristic and should be an LCO. There is no other location with the Technical Specifications that this requirement can be placed.

viii. *Specification c, is not clear to the NRC staff. It appears to contain an applicability statement, an LCO, a setpoint, a SR, a TS exception, as well as some procedure guidance for the operators to manually actuate containment isolation. The guidance in NUREG-1537 and ANSI/ANS-15.1-2007, discusses the operability of equipment to ensure containment. Revise Specification (c) to state the LCO requirements and ensure that the associated SRs are properly stated in the appropriate section of the TSs.*

MURR feels that it is important to keep Specification 3.4.c as it is written. The LCO is "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." The exception allows for controlled evolutions in the reactor pool that are closely monitored. While MURR agrees that there is procedural guidance stated in this Specification while operating in the excepted state, MURR feels it is extremely important to instruct the operators on their required actions when operating in this state.

k. *The proposed MURR TS 3.5, Specification c, provides for overlap of adjacent ranges of nuclear instrument channels, but is not clear to the NRC staff as to which channels are affected, what the allowable ranges are, and whether these ranges are affected by the operational Mode.*

The overlap of the adjacent ranges of Nuclear Instrumentation applies to the three (3) Nuclear Instruments that provide the High Power Level and Reactor Period rod run-ins and scrams, Specifications 3.2.f.1, 3.2.f.2, 3.2.g.1 and 3.2.g.2, respectively. The ranges of the meters are described in Section 7.4 of the SAR, specifically Sections 7.4.2.1 (Source Range), 7.4.2.2 (Intermediate Range) and 7.4.2.3 (Power Range). The Source Range Channel has a six-decade range from 10^{-1} to 10^{5} cps. The Intermediate Range Channels have a ten-decade range from 10^{-8} to 200%. The Power Range Channels have a range of 0 to 125%. These ranges are not affected by operational Modes.

l. *The proposed MURR TS 3.7, Specification a, provides requirements for radiation monitoring channels.*

  i. *Setpoints and functions of the radiation monitors do not appear to be in the proposed MURR TS 3.7. The guidance in Table 14.1 in NUREG-1537, Appendix 14.1, provides information on the setpoints and functions of the radiation monitors.*

As stated in the response to RAI A.28, MURR letter dated September 30, 2010, "Isolation of the reactor containment building at 10 times the normal previously established radiation levels is necessary to allow for sample handling within the reactor pool or when removing samples from the pool. Normal pool surface radiation levels are around 20 mrem per hour while those at the containment building exhaust plenum are around 0.15 mrem per hour. 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 could not go unidentified for any significant period of time. Radiation monitor indications are recorded at set intervals in the reactor log book and any increase above normal would be identified by and responded to by Reactor Operations." The functions of the Reactor Bridge and Reactor Containment Building Exhaust Plenum Radiation Monitors are described in detail in Section 7.8 of the Safety Analysis Report.

The Off-gas (Stack) Radiation Monitor is used for quantitative analysis of radioactive releases in order to determine compliance with release limits or to prepare offsite dose calculations in an emergency type situation. The function of the Off-Gas (Stack) Radiation Monitor is described in detail in SAR Section 7.9.5.

ii. *Environmental monitors do not appear to be included in the proposed MURR TS. The guidance in NUREG-1537, Appendix 14.1, Section 3.7.1, item (4), "Environmental Monitors," provides that environmental monitors should be specified in the proposed MURR TS.*

New Specifications 3.7.c and 4.7.c have been added to include an environmental monitoring program.

m. *The proposed MURR TS 3.8, provides controls on experiments:*

i. *The specifications in proposed TS 3.8, do not appear to follow the format provided in the guidance in NUREG-1537 and ANSI/ANS-15.1-2007 which provides subsections for: "Reactivity Limits," "Materials," and "Failure and Malfunctions." Consider organizing the specifications to follow the guidance provided above.*

Section 3.8 has been reorganized following the guidance of ANSI/ANS-15.1-2007.

ii. *Specification c, does not appear to be supported by an analysis in the MURR SAR.*

(1) *Provide an analysis which explains the dose from the release being averaged over a year and not treated as an instantaneous release.*

Dose (and thus the experimental potential releases) is limited to 5000 mrem whole body and 50,000 mrem organ during the analysis of an experimental failure. It is not relevant over which time period this is delivered; the limiting factor is total dose.

(2) *The exception provided to "Fueled Experiments (see Specification 3.8.a)" is not explained in the Basis or described in the SAR. Provide an analysis or explanation for the exception.*

Fueled Experiments are excepted from this Specification because the amount of material in a fueled experiment will exceed this Specification; however, a specific analysis was performed for a failure of a fueled experiment (MURR letter dated April 15, 2016) which shows, by comparison, that the maximum TEDE and CDE for those occupationally-exposed during failure to applicable NRC dose limits in 10 CFR 20, the final values are well within the published regulatory limits and, in fact, lower than 25% of any occupational limit. For dose to the public in the unrestricted area, the overall TEDE is much less than 1 mrem, a value far below the applicable 10 CFR 20 regulatory limit.

iii. *Specification d, states "explosive materials shall not be irradiated," but then adds information relative to limiting an experiment to 25 milligrams of TNT-equivalent explosives. This specification is not clear to the NRC staff.*

RAI A.31 requested that MURR define what constitutes an explosive material. Currently approved MURR Specification 3.6.d states "Explosive materials shall not be irradiated or allowed to generate in any experiment in quantities over 25 milligrams." In reviewing the Technical Specifications from other facilities, the term "TNT-equivalent" was frequently used. Consequently, by letter dated June 1, 2010, MURR defined explosive materials to the standard of TNT-equivalent.

RAI 13.9.b requested that MURR provide an example of a calculation of an irradiation container that meets the requirement for containing the pressure by at least a factor of two from the detonation of 25 milligrams of TNT-equivalent material. By letter dated September 3, 2010, MURR provided the following analysis:

The detonation of TNT-equivalent material will release an energy equivalent of 4184 joules (~1000 calories) per gram of material. In the case of 25 milligrams, this would yield 104.6 joules (4184 j x 0.025 g) of energy. The pressure created in a confined space due to this energy release can be calculated using the Ideal Gas Equation PV = nRT, where:

P = Pressure (atmospheres)
V = Volume (cm^3)
n = Number of moles gas
R = Universal Gas Constant = 0.0821 $\left(\frac{l \cdot atm}{mol \cdot °K}\right)$
T = Temperature (°K)

in this case: PV = energy (joules)*

From PV = nRT $T = \frac{PV}{nR}$ therefore by substitution $T = \frac{\text{energy (j)}}{nR}$

From PV = nRT $P = \frac{nRT}{V}$ therefore by substitution $P = \frac{nR(\frac{j}{nR})}{V} = \frac{j}{V}$

$\frac{j}{V}$ can be converted to atmospheres by j = 9.87 $cm^3 \cdot atm$**

* Pressure (P) is force per unit area ($newtons/m^2$) and volume (V) is m^3, so, PV simplifies to newtons · meters = joules.

** 1 liter·atm = 101.325 j

$1 cm^3 \cdot atm$ = 0.101325 j

j = $1 cm^3 \cdot atm$ / 0.101325

$$j = 9.87\ cm^3 \cdot atm$$

A typical standard container that might be used at MURR to encapsulate 25 mg of TNT-equivalent material would be a 4-inch tall thin-walled cylinder manufactured of aluminum alloy 1100 with an inner diameter of 1-inch, a wall thickness of 0.025-inches, and an internal volume of approximately 51.5 cm^3.

Using the above formula $P = \frac{j}{V}$, the pressure created by the detonation of TNT-equivalent material in the confined volume of 51.5 cm^3 can be calculated as:

$$P = \frac{104.6\ j}{51.5\ cm^3} \times \frac{9.87 cm^3 \cdot atm}{j}$$

$P = 20.05$ atm, and converted to psi would be:

$$P = 20.05\ atm \times 14.7 \frac{psi}{atm} = 295\ psi$$

Additional pressure due to the compression of the air volume not occupied by the solid TNT-equivalent material would be negligible.

In order to confine the pressurization due to the detonation of 25 mg of TNT, the stress limit of the confining material cannot be exceeded. As stated earlier, a thin-walled aluminum cylinder of alloy 1100 would be a preferred encapsulation. The yield strength of aluminum alloy 1100 = 15.6 ksi (1 ksi = 1000 psi); therefore $\sigma_{yield} = 15{,}600$ psi.

The hoop stress limit in a cylindrical container with thin walls is represented by one-half the pressure times the ratio of the capsule diameter to wall thickness, or:

$$\sigma = \frac{pd}{2t}$$

where:

σ	=	maximum wall stress
p	=	total pressure in the container
d	=	inner diameter of container
t	=	wall thickness

In order to determine if a particular container could confine the expected pressure, the maximum stress in the container wall would need to be less than or equal to the yield strength of the material, or:

$$\frac{pd}{2t} \leq \sigma_{yield}$$

In order to determine if a particular container would meet MURR's more conservative specification, the maximum stress in the container wall would need to be *a factor of two* less than the yield strength of the material, or:

$$\frac{pd}{2t} \leq \frac{\sigma_{yield}}{2}$$

Solving this equation for $\frac{d}{t}$ provides a method for evaluating an encapsulation material using this diameter to thickness ratio:

$$\frac{pd}{2t} \leq \frac{\sigma_{yield}}{2}$$

$$\frac{d}{t} \leq \frac{1}{p}\sigma_{yield}$$

then:

$$\frac{d}{t} \leq \frac{\sigma_{yield}}{p}$$

If you were to confine the 295 psi calculated pressure in a 51.5 cm^3 volume using aluminum alloy 1100, the maximum diameter to thickness ratio for the cylinder at this pressure using the *factor of two* model would be:

$$\frac{\sigma_{yield}}{p} = \frac{15600 \text{ psi}}{295 \text{ psi}} = 53$$

As described above, the typical standard container, which is a 4-inch tall thin-walled cylinder manufactured of aluminum alloy 1100 with an inner diameter of 1-inch and a wall thickness of 0.025-inches, the diameter to thickness ratio would be:

$$\frac{d}{t} = \frac{1 \text{ inch}}{0.025 \text{ inches}} = 40$$

Therefore, this would be an acceptable encapsulation to contain the potential pressurization by a *factor of two.*

iv. Specification e, states "other experiments," without providing a definition or explanation.

"Other experiments" consist of the secured experiments as stated in the last sentence of Specification 3.8.e.

v. Specifications f, g, and l, do not appear to be LCOs consistent with the guidance in NUREG-1537, but rather appear to be procedural controls.

MURR feels that Specifications 3.8.f, 3.8.g and 3.8.l are very important and would like for them to remain.

vi. Specification n, uses the description "first-of-a-kind" which does not appear to be described in the TS, or have an associated TS definition.

The phrase "first-of-a-kind" was introduced into the currently approved MURR Technical Specifications by Amendment No. 8. Amendment No. 8 authorized MURR to change the limitations on fueled experiments, specifically allowing an increase in the inventory of iodines. As stated in the NRC Safety Evaluation (page 3), dated February 24, 1978, "The license will be required to provide instruments for all first-of-a-kind fueled experiments to measure the surface temperature of the experiment for comparison of actual to calculated values to detect abnormal conditions. These requirements provide adequate assurance that the experiment will not melt releasing the stored activity to the containment or environment." MURR interprets this to mean any new fueled experiment of a type that has not been previously analyzed and measured.

vii. Specification o, requires both HEPA and charcoal filters as well as continuous monitoring for increases in radioactive material accumulation. However, there is no LCO to control the presence of these filters nor is there an identification of the monitoring system needed to support their use.

As described in the response to RAI No. 10.b. – Information/Clarification Needed, by MURR letter dated February 8, 2016, fueled experiments have only been previously performed in two experimental facilities within MURR: in the Graphite Reflector Region and in the Thermal Column. These experimental facilities are described in detail in Chapter 10 of the SAR. The fueled experiment performed in the Thermal Column was a vented fueled experiment that was authorized by the issuance of Amendment No. 8, dated February 24, 1978. The experiment consisted of four (4) fission plates; metal plates with a thin coating of uranium oxide on one side. The uranium was highly-enriched with a total plate loading of 1.93 grams of uranium per plate. Four such plates were used with a provision that the sizes of the plates and the mass of uranium could be adjusted to maintain the experiment within the Technical Specification (TS) total inventory limits for I-131 through I-135 and Sr-90.

The fission plates were situated in an irradiation canister that was at the front (nearest to the reactor core) of the Thermal Column. Thin poly-carbonate material (film) fed from a large roll would pass by the fission plates and the fission fragments would create small holes in the film. The film was then removed, etched and used as a fine filter media. The entire experiment was vented through charcoal and HEPA filters to the facility ventilation system exhaust stack as required by the TSs. The charcoal and HEPA filters were continuously monitored for radiation levels by radiation monitoring equipment, also as required by the TSs. This experiment was decommissioned in the mid-1990s. Since then, no other vented fueled experiment has been performed at MURR.

MURR would like to continue to keep this Specification in order to provide experimental flexibility. Specific LCOs for the control of filtration or a radiation monitoring system do not exist since currently there is no vented fueled experiment that contains these inventories.

viii. Specification p, uses the term "secured removable experiment" which does not appear to be described or have an associated TS definition.

Specification 1.29 provides a definition of "removable experiment." Specification 1.35 provides a definition of a "secured experiment." A "secured removable experiment" meets the requirements of both a secured experiment and a removable experiment.

4. *TS 4.0 Surveillance Requirements:*

a. *Proposed MURR TS LCOs: 3.1.a, 3.1.c, 3.1.d, 3.1.e, 3.2.a, 3.5.c, 3.3.b, 3.3.e, 3.3.h, 3.3.i, 3.4.a, 3.4.b, and 3.5.c, do not appear to have corresponding SRs.*

Specification 4.1.a (SR for Specification 3.1.a) has been revised as follows:

"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 > 0.0013 Δk/k."

Specification 4.1.c (SR for Specification 3.1.c) has been added as follows:

"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."

Specification 3.1.d:

Specification 3.1.d was moved to Section 5 as suggested because a Surveillance Requirement is not practical. This specification is now Specification 5.3.d. (See response to RAI 3.b above).

Specification 3.1.e:

The Surveillance Requirement for Specification 3.1.d (old Specification 3.1.e) is Specification 4.1.d.

Specification 3.2.a:

Specification 4.2.a (SR for Specification 3.2.a) has been added as follows:

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

Specification 3.5.c:

MURR feels that a Surveillance Requirement for Specification 3.5.c is not practical since the overlap is only observed during a reactor startup.

Specification 4.3.b (SR for Specification 3.3.b) has been added as follows:

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

Specification 3.3.e:

The Surveillance Requirement for Specification 3.3.e is Specification 4.3.f.

Specification 3.3.h:

The Surveillance Requirement for Specification 3.3.h is Specification 4.3.g.

Specification 3.3.i:

Specification 3.3.i has been deleted as suggested by RAI 3.h.iii.

Specification 4.3.c (SR for Specifications 3.4.a and 3.4.b) has been added as follows:

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

b. The proposed MURR TS 4.1, provides SRs for reactor core parameters:

i. Specification a, does not define the criteria for determining when "significant core configuration and/or control blade change," exists.

Specification 4.1.a is worded as per ANSI/ANS-15.1-2007. ANSI/ANS-15.1-2007 does not define any criteria for determining when a "significant core configuration and/or control blade change" exists. MURR feels that "control blade change" is self-explanatory and does not require any further criteria. In reviewing other facility Technical Specifications, some facilities do not define the criteria for a "significant core configuration change" while others define "significant change" as >\$0.25. MURR will use half the reactivity worth of an unsecured experiment, or 0.0013 $\Delta k/k$, as a "significant core configuration change."

ii. Specification c, does not indicate the criteria for acceptable results of the fuel inspections.

This Surveillance Requirement is not intended to provide any criteria for acceptable results. This Specification merely provides the frequency of when a fuel inspection needs to be conducted. Specification 3.1.d provides the criteria for an acceptable fuel inspection. See the response to RAI 3.c above.

c. *The proposed MURR TS 4.2, provides SRs for reactor control and reactor safety systems:*

i. *Specification a, states "drop time," which does not appear to correspond to any of the LCOs listed in TS 3.2.*

Specification 3.2.c provides an LCO of 0.7 seconds for shim blade insertion to the 20% withdrawn position. This is also known as a shim blade "drop time" at MURR.

ii. *The proposed MURR TS 4.2, Specifications b and j, do not appear to have any corresponding LCOs.*

The LCO for Specification 4.2.b is Specification 3.2.a, where all control blades, including the regulating blade, shall be operable during reactor operation. Operable, as it applies to the control blades, also means that the control blades do not exhibit abnormal swelling or abnormalities.

ANSI/ANS-15.1-2007 does not provide an LCO for Specification 4.2.j.

iii. *Specifications d and e, appear to require a limit based on withdraw and insertion speed; whereas, the proposed MURR TS 3.2, Specifications d and e, provide limits based on reactivity rate.*

Specifications 4.2.d and 4.2.e were added to the revised Technical Specifications, by MURR letter dated January 27, 2014, as directed by the NRC. As described above in the responses to RAI 3.e.i and 3.e.ii, reactivity insertion rates are based on control blade withdrawal and insertion speeds. Performing surveillances on control blade speeds ensures that the reactivity insertion rate LCOs are met.

iv. *Specification g, states, in part, "that the reactor safety system shall be channel tested." The NRC staff is not clear if this SR refers to all specifications (a. through h.) of proposed MURR TS 3.2.*

Only Specification 3.2.g applies to the reactor safety system. Specification 4.2.g applies to Specification 3.2.g.

d. *The proposed MURR TS 4.3, Specifications d and g, do not appear to have corresponding LCOs and associated limits.*

The corresponding LCO for Specification 4.3.d (now 4.3.e) is Specifications 3.3.d. The corresponding Specification for 4.3.g (now 4.3.h) is Specifications 3.5.b.1 and 5.2.k.

e. *The proposed MURR TS 4.4, Specification a, states, in part, "No special maintenance shall be performed just prior to the test." The NRC staff is unclear as the basis for this statement in the SR.*

As stated in the second sentence of the bases of Specification 4.4.a, "No special maintenance will be performed prior to the test so that the results demonstrate the historic integrity of the containment structure." If special maintenance were performed on the reactor containment building immediately prior to the annual surveillance, the validity of the containment structure integrity prior to that maintenance might be questioned.

f. *The proposed MURR TS 4.5, Specification c, provides an exception to performing the surveillance that is not clearly described, and the basis for the exception is not provided.*

This exception is discussed in the response to RAI A.40, by MURR letter dated October 29, 2010, "The basis for not requiring channel testing of nuclear instrumentation within 2 hours following a reactor shutdown is a combination of factors. Providing an unsafe condition does not exist or a failure of the nuclear instrumentation channel has not occurred as stipulated in the TS, the nuclear instrumentation channels were 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 core to ensure that proper observation of subcritical multiplication and that criticality can be observed during the subsequent startup. In effect, the nuclear instrumentation system has been in continuous operation for this period and thus additional testing is not required."

g. *The proposed MURR TS 4.6, provides SRs for the emergency electrical power system:*

   i. *Specification a, states, in part, "The operability of the emergency power generator." However, the NRC staff is not clear as to the criteria for operability of the emergency power generator.*

Operability of the emergency power generator is determined by its ability to sense a loss of the Normal Electrical Power System and automatically start upon that loss.

   ii. *TS LCO 3.6, Specification a, refers to the emergency electrical power system. Neither Specification a or b, include the emergency electrical power system. Explain the emergency electrical power system, LCO 3.6, Specification a, surveillance test.*

As discussed in Section 8.2, Emergency Electrical Power System, of the SAR, the majority of the system is load based and is in continuous operation regardless of the source of electrical power. The only exceptions to this are the Automatic Transfer Switch (ATS) and the Emergency Power Generator. On a loss of normal electrical power the emergency power generator starts and assumes the load as the ATS transfers to the emergency power generator supply. All loads on the Emergency Electrical Power System remained energized. The surveillance test for this system is broken down into two sections:

1) *"The operability of the emergency power generator shall be verified on a weekly basis."* The emergency power generator has proven to be reliable since its installation in 1989.

The Emergency Power Generator is automatically exercised and run for at least 30 minutes on a weekly basis. This weekly check has been demonstrated sufficient to ensure the Emergency Power Generator will start and run when required.

2) *"The ability of the emergency power generator to assume the emergency electrical load shall be verified on a semiannual basis."* The entire Emergency Electrical Power System is verified to function as designed by performance of Compliance Procedure CP-17, "Emergency Generator Load Test" (Attachment 4). The surveillance tests the entire operation of the Emergency Electrical Power System including the ATS and Emergency Power Generator.

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

i. *Specification a, states, in part, "radiation monitors shall be verified operable by monthly radiation source checks or channel tests." The NRC staff is not clear if radiation source checks and channel tests represent equivalent surveillance tests.*

Radiation source checks and channel tests are equivalent surveillance tests – 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 – and only dependent upon the type of radiation detector and availability of an appropriate source material and source strength.

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.

i. *The proposed MURR TS 4.8, Specification a, states, in part, "the criteria of Specification 3.8, shall be evaluated." However, the criteria for the acceptability of the evaluation are not provided.*

The criteria for experiment acceptability are discussed in Section 10.4, Experiment Review, of the SAR. Criteria are also discussed the response to RAI 10.3.a, by MURR letter dated March 11, 2011. These criteria are used to verify that each experiment complies with all of the applicable Technical Specifications.

5. *TS 5.0 Design Features:*

a. *The proposed MURR TS 5.1, Specification a, contains a footnote, which seems redundant to information in the Basis, or referenced in the SAR.*

The footnote has been deleted.

b. *The proposed MURR TS 5.2, Exception a, is not clear, and it appears to be redundant TS 2.1, Specification a for Mode II operation.*

MURR assumes that the question is referring to TS 2.2.a, not 2.1.a as stated above. The exception is important because this allows MURR to perform maintenance on the shutdown leg of the primary or pool coolant system when in Mode II operation. If a component is removed for maintenance, such as a pump or heat exchanger in the shutdown leg, and the reactor was in operation in Mode II, then MURR would be in violation of Specification 5.2.a. even if the shutdown system is secured in a manner such as to assure system integrity.

c. *The proposed MURR TS 5.2, Exception b, exempts certain components from Specification 5.2.e. However, the response to RAI A.49, by letter dated October 29, 2010 (ADAMS Accession No. ML 12355A023), stated that the size of components of concern did not present a hazard to the PCS. As such, the NRC staff is not clear if these components are considered major components as defined by TS 5.2, Specification e, and if so, then it appears that the TS 5.2, Exception b, is not necessary.*

MURR still feels that this exception is warranted. As stated in the response to RAI A.49, the exception allows the licensee to evaluate the materials of these smaller components 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. Following an acceptable evaluation, the results of the evaluation are documented under the existing 10 CFR 50.59 process. Where appropriate, the use of excepted materials is considered more advantageous and with fewer failure modes than would be the isolators or modifications needed to fully comply with Technical Specification 5.3.e.

To provide greater clarification, the Bases for Specification 5.2.e has been revised as follows:

"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 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."

6. *TS 6.0 Administrative Controls:*

   a. *The proposed MURR TS 6.1.3, does not appear to include the guidance in ANSI/ANS-15.1-2007, Section 6.1.3, that a designated senior reactor operator shall be available in an "on call" status.*

   Specification 6.1.e has been revised as follows:

   "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, and shall be present at the facility during all startups and approaches to power, recovery from an unplanned or unscheduled shutdown or non-emergency power reduction, and refueling activities. 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)."

   b. *The NRC staff are unable to find any requirements in the proposed MURR TSs, Section 6, which are consistent with the guidance provided in ANSI/ANS-15.1-2007, Section 6.1.4, and the ANSI/ANS-15.4-1998, regarding the selection and training of personnel.*

   The following specification has been added as Specification 6.1.f:

   "f. 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."

c. *The proposed MURR TS 6.2, Specification a. (1), does not appear to be consistent with the guidance in ANSI/ANS-15.1-2007, Section 6.2.3, that the review should include all changes to the facility.*

In re-reviewing Section 6.2.3 of ANSI/ANS-15.1-2007, MURR could not find where it states "the review should include all changes to the facility." 6.2.3(2) states "all new procedures and major revisions thereto having safety significance, proposed changes in reactor facility equipment, or systems having safety significance;" We feel that the currently proposed MURR Specification 6.2.a(1) meets the requirements of ANSI/ANS-15.1-2007 Section 6.2.3(2). Proposed Specification 6.2.a(1) states:

"(1) Proposed changes to the MURR equipment, systems, or procedures when such changes have safety significance, or involve an amendment to the facility operating license, a change in the Technical Specifications incorporated in the license, or a question pursuant to 10 CFR 50.59. 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 Health Physics or a Senior Reactor Operator, respectively. All such changes to the procedures shall be documented and subsequently reviewed by the RAC;"

d. *The proposed MURR TS 6.2.2, does not appear to be consistent with the guidance of ANSI/ANS-15.1-2007, Section 6.2.2, states, in part, "that operating staff may not constitute a majority of the Reactor Advisory Committee."*

The second paragraph in Specification 6.2.b has been revised as follows:

"The RAC and its subcommittees are to maintain minutes of meetings in which the items considered and the committees' recommendations are recorded. Independent actions of the subcommittees are to 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 must be present at any meeting to conduct the business of the committee or subcommittee. Reactor facility staff may not constitute a majority of the RAC. The RAC shall meet at least quarterly."

e. *The NRC staff are unable to find any requirements in the proposed MURR TSs, Section 6, which are consistent with the guidance in ANSI/ANS-15.1-2007, Section 6.8.2, regarding records to be retained for at least one operator certification cycle.*

The following specification has been added as Specification 6.7.c:

"a. 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."

7. *TS Bases:*

   a. *The NRC staff review of the proposed MURR TS Bases identified the following:*

      i. *The proposed MURR TS Bases, listed below appear to need a description or to provide a SAR reference to the referenced TS:*

         (1) TS 3.1, Specifications a and b;

The Basis for Specifications 3.1.a and 3.1.b is Section 4.5 of the SAR; this reference has been added to Specification 3.1.a and 3.1.b Bases.

(2) TS 3.2, Specifications a and d;

The Basis for Specifications 3.2.a and 3.2.d is Section 4.5 of the SAR; this reference has been added to Specification 3.2.a and 3.2.d Bases.

(3) TS 3.5, Specifications a, c and d;

The Basis for Specification 3.5.a is Sections 7.4 and 7.6.2.2 of the SAR; these references have been added to Specification 3.5.a Basis.

The Basis for Specification 3.5.c is Section 7.4 of the SAR; this reference has been added to Specification 3.5.c Basis.

(4) TS 3. 7, Specification b; and

MURR feels that the description in the Basis of Specification 3.7.b is adequate.

(5) TS 3.8, Specifications a, b, c, f, g, h, and i.

The Basis for Specification 3.8.a is the new "Fueled Experiment Failure" analysis that was submitted to the NRC by MURR letter dated April 15, 2016. This analysis will eventually go into Section 13.2.6 of the SAR.

The Basis for Specification 3.8.b is to prevent accidental voiding in these regions which could introduce an uncontrolled reactivity addition based on their respective void coefficients. The Basis is Section 4.5 of the SAR; this reference has been added to Specification 3.8.b Basis.

The Basis for Specification 3.8.c is 10 CFR 20.

MURR feels that the Bases for Specifications 3.8.f and 3.8.g are adequate.

The Basis for Specification 3.8.h is to prevent accidental voiding in the reactor reflector region by preventing the surface temperature of a submerged irradiated experiment from exceeding the saturation temperature of the cooling medium. Cooling also reduces the likelihood of failure from internal or external heat generations. Section 4.5 of the SAR addresses the void coefficient and Section 13.2.6 mentions temperature. These references have been added to Specification 3.8.h Basis.

The Basis for Specification 3.8.i is Section 13.2.6 of the SAR; this reference has been added to Specification 3.8.i Basis.

ii. *The proposed MURR TS 2.2.a Basis, does not appear to reference the analysis submitted in License Amendment 36.*

Amendment No. 36 has been added as a reference to the bases of Specifications 2.2.a and 2.2.b.

iii. *The proposed MURR TS 3.1, Specification c Basis, does not appear to provide a reference for a supporting analysis for the Exception provided for operation with less than eight fuel assemblies. Provide a reference to an analysis supporting operation with less than eight fuel assemblies.*

The reference for operation with less than eight fuel assemblies is "Low Power Testing Program for the Missouri University Research Reactor 6.2 Kilogram Core," October 20, 1971.

iv. *The proposed MURR TS 3.2.h Basis, does not appear to provide any information relative for the interlocks cited in the table of interlocks.*

The basis for Specification 3.2.h is Sections 7.5.3.1 and 7.5.4 of the SAR; these references have been added to Specification 3.2.h basis.

v. *The proposed MURR TS 5.2.e Basis, does not appear to incorporate the material introduced by the response to RAI A.49, by letter dated October 29, 2010 (ADAMS Accession No. ML 12355A023), concerning hazards to the reactor coolant system.*

The basis for Specification 5.2.e has been revised as follows to include material from the response to RAI A.49:

"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 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."

vi. *The proposed MURR TS 3.8.a Basis, references SAR Section 13.2.6. However, the SAR only compares the iodine and strontium isotope limits to the Maximum Hypothetical Accident analysis, which assumes that the failure occurs in the coolant system with no dose consequence. The NRC staff questions whether the TS Basis should reference a safety analysis that considers failures of fueled experiments that could occur in other locations of the facility.*

MURR is not clear on what this question is asking. The dose consequences from a "Fueled Experiment Failure" have been calculated for the restricted and unrestricted areas (MURR letter dated April 15, 2016). This accident scenario is now the new Maximum Hypothetical Accident. The assumed fission product activity for fueled experiments is limited by Specification 3.8.f.

APPENDIX A

TECHNICAL SPECIFICATIONS

FOR

THE UNIVERSITY OF MISSOURI RESEARCH REACTOR

FACILITY OPERATING LICENSE R-103
DOCKET 50-186

ATTACHMENT 1

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.

ATTACHMENT 1

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ATTACHMENT 1

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1.0 DEFINITIONS

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

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

b. Operation in violation of Limiting Conditions for Operation 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.

1.0 **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 rods 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.

1.0 **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 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 **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.

1.19 **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.

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

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

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

1.0 **DEFINITIONS** - Continued

1.23 **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.

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

1.25 **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 shim 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 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.

1.0 **DEFINITIONS** - Continued

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

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

AND

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

1.27 **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 Δk/k).

1.28 **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.

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

1.30 **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.

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

1.32 **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 1.23; however, it does provide a protective function by introducing shim blade insertion to terminate a transient prior to actuating the reactor safety system.

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

1.0 **DEFINITIONS** - Continued

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

1.34 **Scram Time** - Scram time is the elapsed time between the initiation of a scram signal and specified movement of a control or safety device.

1.35 **Secured Experiment** - A secured experiment is any experiment which is rigidly held in place by mechanical means with sufficient restraint to withstand any anticipated forces to which the experiment might be subjected to.

1.36 **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.

1.37 **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.

1.38 **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.

1.39 **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.

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

a. Annual - interval not to exceed 15 months.

b. Semiannual - interval not to exceed 7.5 months.

1.0 **DEFINITIONS** - Continued

c. Quarterly - interval not to exceed 4 months.

d. Monthly - interval not to exceed 6 weeks.

e. Weekly - interval not to exceed 10 days.

f. Within a shift - must be done during a reactor shift.

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

1.42 **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.

1.43 **Unsecured Experiment** - An unsecured experiment is any experiment which is not secured as defined by Specification 1.35, or the moving parts of secured experiments when they are in motion.

2.0 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, 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	1,625 gpm either loop (Minimum)
Reactor Inlet Water Temperature	155 °F (Maximum)
Pressurizer Pressure	75 Psia (Minimum)

b. **Mode II Operation**

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

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.

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.

3.0 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. The reactor core excess reactivity above reference core condition shall not exceed 0.098 $\Delta k/k$.

b. The reactor shall be subcritical by a margin of at least 0.02 $\Delta k/k$ with the most reactive shim blade and the regulating blade in their fully withdrawn positions.

c. The reactor core shall consist of eight (8) fuel elements.
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. The reactor will 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).

b. Specification 3.1.b assures that a shutdown margin, as defined by Specification 1.39, 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 fuel elements up at a power level not to exceed 100 watts. This maximum power limit

3.1 **Reactor Core Parameters** - Continued

is low enough to ensure no fuel damage will occur. This provides for a conservative approach to criticality with less than eight 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.

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} Δk/k/sec.

e. The maximum rate of reactivity insertion for the four shim blades operating simultaneously shall not exceed 3.0×10^{-4} Δk/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	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) Not required (a) below 50 kW operation with the natural convection flange and reactor pressure vessel cover removed or (b) in operation with the reactor subcritical by a margin of at least 0.015 $\Delta k/k$.
(2) Not required during calibration measurements of the regulating blade.

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

	Reactor Safety System Instrument Channel	**Number Required (N)** Mode I	Mode II	Mode III	**Trip Set Point**
1.	High Power Level	3	3	3	125% of full power (Max)
2.	Reactor Period	2	2	2	8 Seconds (Min)
3.	Primary Coolant Flow	4	2	2(1)	1,625 gpm(2) (Min)
4.	Differential Pressure Across the Core	1	0	0	3,200 gpm(3) (Min)
5.	Differential Pressure Across the Core	0	1	1(1)	1,600 gpm(3) (Min)

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**
6.	Primary Coolant Low Pressure	4	4	$4^{(1)}$	75 psia$^{(5)}$ (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$^{(4)}$ (Min)
10.	Differential Pressure Across the Reflector	1	0	0	2.52 psi (Min) 8.00 psi (Max)
11.	Differential Pressure Across the Reflector	0	1	0	0.63 psi (Min) 2.00 psi (Max)
12.	Pressurizer High Pressure	1	1	$1^{(1)}$	95 psia (Max)
13.	Pressurizer Low Water Level	1	1	$1^{(1)}$	16 inches below centerline (Min)
14.	Pool Low Water Level	0	0	1	23 feet (Min)
15.	Primary Coolant Isolation Valves 507A/B Off Open Position	1	1	$1^{(1)}$	Either valve off open position
16.	Pool Coolant Isolation Valve 509 Off Open Position	1	1	0	Valve 509 off open position
17.	Power Level Interlock	1	1	1	Scram as a result of incorrect selection of operating mode
18.	Facility Evacuation	1	1	1	Scram as a result of actuating the facility evacuation system

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**
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(6)	2(6)	2(6)	Scram as a result of removing the center test hole removable experiment test tubes or strainer

(1) Not required (a) below 50 kW operation with the natural convection flange and pressure vessel cover removed or (b) in operation with the reactor subcritical by a margin of at least 0.015 $\Delta k/k$.

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

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

(4) Flow orifice ΔP (psi) corresponding to the flow value in the table.

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

(6) Not required if reactivity worth of the center test hole removable experiment test tubes 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 certain control system logic functions have been satisfied	1
2.	Automatic Control Prohibit	Prevents placing the reactor in automatic control unless certain control system logic functions 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).

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.

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.

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

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 Technical 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 and, 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.

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

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

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.

The center test hole scram provides assurance that the reactor cannot be operated unless the removable experiment test tubes 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 test tubes 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 test tubes and the contained experiments or the strainer does not exceed the limit of Specification 3.8.b and is authorized by the Reactor Manager.

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

h. Specification 3.2.h assures that certain system conditions have been met prior to conducting a reactor startup or placing the reactor in automatic control at power (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 operable,

   OR

   (2) The primary coolant system is sampled and analyzed at least once every four 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 secondary coolant system exceeds the limits of 10 CFR 20, Appendix B, Table 3, for radioisotopes with half-lives greater than 24 hours.

e. 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 month.

f. 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 month.

g. 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 month.

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

3.3 **Reactor Coolant Systems** - Continued

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

e. - g. 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 must 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 operable;

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

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

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

3.4 **Reactor Containment Building** - Continued

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:

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

(1) Required for reactor startup only.

b. Sufficient instrumentation shall be provided to assure that the following limits are not exceeded during steady-state 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 and (a) below 50 kW with the natural convection flange and reactor pressure cover removed or (b) with the reactor subcritical by a margin of at least 0.015 $\Delta k/k$.

c. A minimum of one decade of overlap shall exist between adjacent ranges of nuclear instrument channels.

3.5 **Reactor Instrumentation** - Continued

d. The reactor shall not be started up unless:

(1) The Source Range Channel is indicating a neutron count rate of at least 1 count per second and the Wide Range Monitor is indicating a power level greater than 1 watt,

OR

(2) The Source Range Channel is indicating a neutron count rate of at least 2 counts per second and is verified just prior to startup by a neutron test source or movement on the Source Range meter demonstrating that the channel is responding to neutrons.

Bases:

a. 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 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.

c. Specification 3.5.c ensures that, during a startup, the reactor power level is continuously monitored over the entire range (Ref. Section 7.4 of the SAR).

d. Specification 3.5.d provides for adequate neutron flux level monitoring to ensure that subcritical multiplication and criticality can be observed during a startup.

3.6 Emergency Electrical Power System

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

Objective:
The objective of this specification is to assure emergency electrical power is available to vital equipment.

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. Therefore, on a loss of normal electrical power, the emergency electrical power system is not required for protection of the integrity of the fuel elements. 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.

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

	Channel	**Minimum Numbers Operable** **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 will be closely observed.

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

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

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

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

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

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

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

AND

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

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.

b. Dispersion calculations based upon standard reference material and experiment data obtained at the reactor show that argon-41 concentrations under average conditions will be 0.008 of the AEC limits in the unrestricted area surrounding the reactor facility. 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.

The normal short burst releases at the facility are five to ten seconds in duration and occur on an average of ten times per day five days per week. The short bursts affect the concentration by 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 not exceed 0.0025 $\Delta k/k$.

e. The absolute value of the reactivity worth of all unsecured experiments which are in the reactor shall not exceed 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.

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.

3.8 **Experiments** - Continued

j. Corrosive materials shall be doubly encapsulated in corrosion-resistant containers to prevent interaction with reactor components or pool water.

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

l. Fluids utilized in loop experiments placed in the beamports 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.

<u>Failure and Malfunctions Specification</u>:

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, Appendix B, Table 1. 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 control system or other operating components.

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

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

3.8 **Experiments** - Continued

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.

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.

3.8 **Experiments** - Continued

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.

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.

3.8 **Experiments** - Continued

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 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 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.

4.0 SURVEILLANCE REQUIREMENTS

General: 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. If the reactor is not operated for a reasonable time, a reactor system or measuring channel surveillance requirement may be waived during the associated period. Prior to reactor system or measuring channel operation, the surveillance shall be performed for each reactor system or measuring channel for which surveillance was waived. A reactor system or measuring channel shall not be considered operable until it is successfully tested. Surveillance intervals shall not exceed those defined by Specification 1.40. Discovery of noncompliance with any of the surveillance specifications listed in this Section shall limit reactor operations to that required to perform the surveillance.

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 > 0.0013 $\Delta k/k$.

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 > 0.0013 $\Delta k/k$.

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 will be inspected for anomalies.

4.1 **Reactor Core Parameters** - Continued

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 each six months so that every blade is inspected every two years. 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 withdrawal and insertion speeds of the regulating blade shall be verified on an annual basis.

f. The withdrawal and insertion speeds of each shim blade shall be verified on an annual basis.

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

h. The reactor safety system shall be channel tested before each reactor startup involving a refueling, a shutdown greater than 24 hours or quarterly.

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

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

k. A thermal power verification of power range indication, using coolant flows and

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

differential temperatures, shall be performed weekly when the reactor is operating above 2 MW.

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 normally 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. - j. 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).

k. 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.

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 at two-year intervals, 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 4.2.b 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.g.

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. No special maintenance shall be performed just prior to the test.

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

c. 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 special maintenance will be performed prior to the test so that the results demonstrate the historic integrity of the containment structure.

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

c. Specification 4.4.c 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.

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 biannually.

   AND

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

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 biannual basis will provide information that environmental limits are not being exceeded. Collecting and analyzing film badges, thermoluminescent dosimeters, or other devices on a biannual 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 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 filter banks ensures that the filters will perform as analyzed.

5.0 DESIGN FEATURES

General: Major alterations to safety-related components or equipment shall not be made prior to appropriate safety reviews.

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 the University Research Park, an 84-acre tract of land approximately one mile southwest of the University of Missouri 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 University Research Park is owned and operated by the University of Missouri. 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 reactor coolant systems: primary, pool, and secondary. The following design features apply to these coolant systems:

a. The reactor coolant systems shall consist of not less than a reactor pressure vessel, a primary pressurizer, two primary coolant circulation pumps, two primary coolant heat exchangers, two pool coolant circulation pumps, one pool coolant heat exchanger, and one 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 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 anti-siphon isolation valves.

i. The reactor shall have a natural convection coolant flow path for Mode III operation except for operation with the reactor subcritical by a margin of at least 0.015 $\Delta k/k$.

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 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 Section 5 of the SAR. The reactor can be safely operated at 10 MW 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 10 MW 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 fuel element shall have a maximum uranium-235 loading of 775 grams.

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

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

j. The reactor shall have five control blades between the pressure vessel and beryllium reflector. Four blades shall be for coarse control (shim blades) and one for fine control (regulating blade) of reactor power.

k. The reactor shall have the following experimental facilities:
   1. Six beam tubes which penetrate the graphite reflector;
   2. A center test hole located in the flux trap;

5.3 **Reactor Core and Fuel** - Continued

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.

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 R-103, Change No. 6 to Facility License 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.- g. The MURR 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 and 5.3.g require fuel content and dimensions of the fuel elements to be in accordance with the design and fabrication specifications (Ref. Section 4.2.1 of the SAR).

h. Specification 5.3.h assures that the reactor fuel is properly positioned in the pressure vessel during operation (Ref. Section 4.2.5 of the SAR).

i. Specification 5.3.i assures proper neutron reflection as required by design (Ref. Section 4.2.3 of the SAR).

j. Specification 5.3.j assures reactivity of the reactor is properly controlled as required by design (Ref. Section 4.2.2 of the SAR).

k. Specification 5.3.k assures that the reactor consists of the experimental facilities as required by design (Ref. Chapter 10 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.

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

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

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

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

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 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 containment building will have sufficient integrity 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 (Ref. Sections 6.2.10 and 13.2.1 of the SAR).

5.5 **Reactor Containment Building** - Continued

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.

6.0 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) Reactor Facility Director: 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.

(2) Reactor Manager: To safeguard the public and facility personnel from undue radiation exposure, the Reactor Manager is responsible for:

i. Compliance with Technical Specifications and license requirements regarding reactor operation, maintenance and surveillance; and

ii. Oversight of the experiment review process.

(3) Reactor Health Physics Manager: To safeguard the public and facility personnel from undue radiation exposure, the Reactor Health Physics Manager is 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.

c. At a minimum during reactor operation, there shall be two 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 must 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;

6.1 **Organization** - Continued

(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, and shall be present at the facility during all startups and approaches to power, recovery from an unplanned or unscheduled shutdown or non-emergency power reduction, and refueling activities. 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. 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 review:

(1) Proposed changes to the MURR equipment, systems, or procedures when such changes have safety significance, or involve an amendment to the facility operating license, a change in the Technical Specifications incorporated in the license, or a question pursuant to 10 CFR 50.59. 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 Health Physics or a Senior Reactor Operator, respectively. All such changes to the procedures shall be documented and subsequently reviewed by the RAC;

6.2 **Review and Audit** - Continued

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

(3) The circumstances of reportable occurrences and violations of the Technical Specifications or license and the measures taken to prevent a recurrence;

(4) Violations of internal procedures or operating abnormalities having safety significance; and

(5) Reports from audits required by the Technical Specifications.

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

The RAC and its subcommittees are to maintain minutes of meetings in which the items considered and the committees' recommendations are recorded. Independent actions of the subcommittees are to 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 must be present at any meeting to conduct the business of the committee or subcommittee. Reactor facility staff may 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.

6.2 **Review and Audit** - Continued

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 years; and

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.

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

6.4 **Procedures** - Continued

(6) Implementation of the Emergency Plan.

b. Written procedures shall be in effect for radiological control, and the preparation for shipping and the shipping of byproduct material produced under the facility operating license.

c. The Reactor Manager shall approve and annually review the procedures for normal operations of the reactor and the Emergency Plan implementing procedures. The Reactor Health Physics Manager shall approve and annually review the radiological control procedures and the procedures for the preparation for shipping and the shipping of byproduct material.

d. Deviations from procedures required by this Specification may be enacted by a Senior Reactor Operator or member of Health Physics, as applicable. Such deviations shall be documented 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

Approved experiments shall be carried out in accordance with established and approved procedures. Procedures related to experiment review and approval shall include the following:

a. All new experiments or class of experiments shall be reviewed by the RAC and approved in writing by the Reactor Manager.

b. 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. Safety Limit Violation - In the event of a safety limit violation, the following actions shall be taken:

(1) The reactor shall be shut down and reactor operation shall not be resumed until authorized by the NRC pursuant to 10 CFR 50.36(c)(1);

(2) The safety limit violation shall be promptly reported to the NRC. Prompt reporting of the violation shall be made by MU, by telephone or email, to the NRC Operations Center no later than the following working day;

6.6 **Reportable Events and Required Actions** - Continued

(3) A detailed follow-up report shall be prepared. The report shall include the following:

i. Applicable circumstances leading to the violation including, when known, the causes and contributing factors;

ii. Date and approximate time of the occurrence;

iii. Effect of the violation upon the reactor and associated systems;

iv. Effect of the violation on the health and safety of the facility staff and general public; and

v. Corrective actions to prevent recurrence.

(4) The follow-up report will 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 NRC. Prompt reporting of the violation shall be made by MU, by telephone or email, to the NRC Operations Center no later than the following working day;

(3) If it is necessary to shut down the reactor to correct the occurrence, operations shall not be resumed until authorized by the Reactor Manager; and

(4) A detailed follow-up report shall be prepared. The follow-up report will 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 violation shall be made by MU, by telephone or email, to the NRC Operations Center no later than the following working day;

(2) A detailed follow-up report shall be prepared. The follow-up report will be submitted within fourteen (14) days to the NRC Document Control Desk; and

(3) 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 Manager.

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 Provost or the Director's Office.

e. Annual Report - An annual operating report shall be submitted to the NRC within sixty (60) days following the end of each calendar year. The report shall include the following information for the preceding year:

(1) A brief narrative summary of (a) operating experience (including operations designed to measure reactor characteristics), (b) changes in the reactor facility design, performance characteristics, and operating procedures related to reactor safety occurring during the reporting period, and (c) results of surveillance tests and inspections;

(2) A tabulation showing the energy generated by the reactor (in megawatt-days);

(3) The number of emergency shutdowns and inadvertent scrams, including the reasons therefore and corrective action, if any, taken;

(4) Discussion of the major maintenance operations performed during the period, including the effects, if any, on the safe operation of the reactor;

6.6 **Reportable Events and Required Actions** - Continued

(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 are to 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; and

(4) Updated drawings of the reactor facility.

b. Five Year Records - The following records are to be maintained for a period of at least five 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);

6.7 **Records** - Continued

(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.

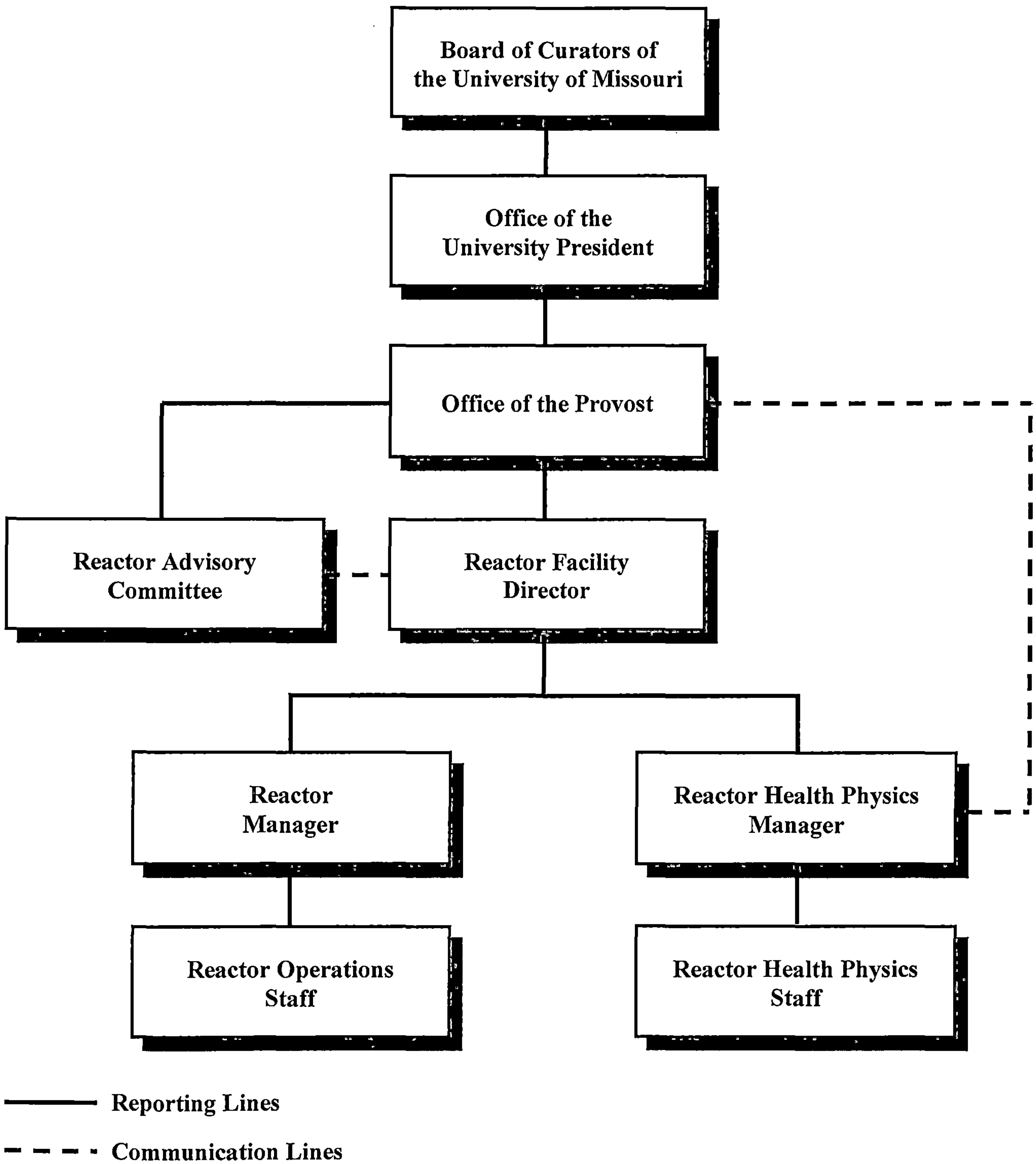


FIGURE 6.0
UNIVERSITY OF MISSOURI RESEARCH REACTOR (MURR)
ORGANIZATION

ATTACHMENT 2

Compliance Check DPS-929	**Frequency:** Semi-annually
Plant Conditions: Reactor Shutdown Primary Coolant System Shutdown	**Number of Men Needed:** 3 **Estimated Time:** 1 hour

Test Equipment, Tools and Materials

1. MURR Island Test Rig
2. Walkie Talkies or Phone
3. Dummy Load Test Connectors
4. Shorted Relays
5. Jumpers
6. Crystal Engineering Gauge / Heise Gauge

References

Technical Specifications 2.1, 2.2, 3.3, 3.4, 3.9, 4.4, 5.2 and 5.4
Prints 41, 42, 138, 139 and 156
OP-RO-410, "Primary Coolant System"
OP-RO-460, "Pool Coolant System – Two Pump Operation"
OP-RO-461, "Pool Coolant System – One Pump Operation"

Procedure

DPS-929 CALIBRATION CHECK

NOTE: The Crystal Engineering Gauge should be used during the performance of this Compliance Procedure. If this gauge is unavailable, then the Heise gauge may be used as an acceptable substitute. Verify the gauge being used has a current calibration sticker.

NOTE: Ensure that there are no air bubbles in the lines of the MURR Island Test Rig prior to connecting the rig to the D/P Transmitter.

1. Remove flow transmitter DPS-929 from service as follows:
   ___ a. OPEN DPS-929 Equalizing Valve 568H-EQ.
   ___ b. CLOSE DPS-929 H/P Isolation Valve 568H-HP.
   ___ c. CLOSE DPS-929 L/P Isolation Valve 568H-LP.
2. Connect MURR Island Test Rig as follows:
   ___ a. OPEN the following valves:
   L/P Vent
   L/P Isolation
   H/P Isolation
   ___ b. CHECK CLOSED the following valves:
   Bypass
   H/P Drain
   Air Isolation

   NOTE: Ensure proper connections are made between transmitter and MURR Island Test Rig to prevent damage to the transmitter.

   ___ c. Install test lines tubing as follows:
   Connect L/P side of MURR Island Test Rig to L/P side of transmitter.
   Connect H/P side of MURR Island Test Rig to H/P side of transmitter.

APPROVED: ______________
Reactor Manager

3. Connect Crystal Engineering Gauge / Heise Gauge as follows:
____ a. CHECK the regulator backed off.
____ b. CHECK OPEN Crystal Engineering Gauge vent valve.
____ c. VERIFY the mechanical zero of the Crystal Engineering Gauge and adjust as necessary to establish zero.
____ d. CHECK CLOSED Crystal Engineering Gauge test outlet valve.
____ e. Connect service air to Crystal Engineering Gauge inlet connection.
____ f. Connect hose between Crystal Engineering Gauge test connection and MURR Island Test Rig.
____ g. OPEN air isolation valve on the MURR Island Test Rig.
____ h. OPEN Crystal Engineering Gauge test outlet valve.
____ i. CLOSE Crystal Engineering Gauge vent valve.
____ j. OPEN Crystal Engineering Gauge air inlet valve.

5. Place flow transmitter DPS-929 in test as follows:
____ a. OPEN DPS-929 L/P Test Fitting Isolation Valve 599AC.
____ b. OPEN DPS-929 H/P Test Fitting Isolation Valve 599AB.
c. CLOSE DPS-929 Equalizing Valve 568H-EQ.

NOTE:	An Electronic Technician will perform Steps 6a and 6c.

6. Test flow transmitter as follows:
____ a. Connect digital ammeter to the test jacks.
____ b. Raise pressure on Crystal Engineering Gauge with the regulator to raise pressure on the transmitter. STOP at the listed calibration data points AND RECORD data on datasheet.
____ c. Disconnect digital ammeter.

7. Remove test equipment from service as follows:
____ a. Back off Crystal Engineering Gauge regulator.
____ b. CLOSE Crystal Engineering Gauge air inlet valve.
____ c. Vent pressure off MURR Island Test Rig by OPENING Crystal Engineering Gauge Vent Valve.
____ d. When pressure is vented off, CLOSE air isolation valve on the MURR Island Test Rig and remove air hose.
____ e. OPEN Valve 568H-EQ.
____ f. CLOSE Valve 599AB.
____ g. CLOSE Valve 599AC.
____ h. CLOSE L/P vent valve on the MURR Island Test Rig.
____ i. CLOSE L/P isolation valve on the MURR Island Test Rig.
____ j. CLOSE H/P isolation valve on the MURR Island Test Rig.
____ k. Remove H/P and L/P test lines between the MURR Island Test Rig and DPS-929 transmitter.

8. Place flow transmitter DPS-929 back on service as follows:
____ a. OPEN DPS-929 H/P Isolation Valve 568H-HP (vent the transmitter if necessary).
____ b. OPEN DPS-929 L/P Isolation Valve 568H-LP (vent the transmitter if necessary).
____ c. CLOSE DPS-929 Equalizing Valve 568H-EQ.

____9. Install Dummy Load Test Connectors.
____10. Place Primary Coolant System online per OP-RO-410, "Primary Coolant System."
____11. Place Pool Coolant System online per OP-RO-460, "Pool Coolant System – Two Pump Operation," or OP-RO-461, "Pool Coolant System – One Pump Operation."
____12. Install jumper Y-2 (reactor loop B low flow FT-912G scram bypass).
____13. Install jumper Y-12 (close In-Pool Heat Exchanger Isolation Valve 546A).
____14. Install jumper G-3 (green leg of safety system bypass).
____15. Install jumper G-11 (reactor loop B low flow FT-912H scram bypass).
____16. Install jumper G-18 (close In-Pool Heat Exchanger Isolation Valve 546B).

___17. Remove relay K-30 (reactor loop low flow FT-912A scram and annunciator bypass) and install shorted relay in K-30 position.
___18. Remove relay K-38 (reactor loop low flow FT-912E scram and annunciator bypass) and install shorted relay in K-38 position.
___19. Place Primary Demineralizer Pump P-513A switch in the 'OFF' position.
___20. Place Primary Coolant Demineralizer Inlet Valve 527E switch in the 'CLOSE' position (closes valve 527E).
___21. Place Master Control Switch 1S1 to 'ON' position.
___22. Place Magnet Current Switch 1S14 to 'ON' position.
___23. RESET scram TAAs.

CAUTION:	Do not allow primary system pressure, as read at pump discharge, to exceed 110 psig. If necessary, use Pump Bypass Valves 538A and 538B to decrease flow. **(TS 3.4.b)**

___24. Decrease flow rate through Primary Coolant System Heat Exchangers 503A and 503B in small incremental steps, by alternately throttling CLOSED on Heat Exchanger Outlet Valves 540A and 540B, until scram occurs.
___25. RECORD Primary Coolant System Heat Exchanger 'A' and 'B' flow rates and DPS-929 meter reading at scram setpoint on datasheet.
___26. VERIFY scram TAAs and magnet current zero.
___27. Remove jumper Y-12 AND VERIFY Valve 546A OPENS.
___28. Remove jumper G-18 AND VERIFY Valve 546B OPENS.
___29. Remove jumper G-3.
___30. Install jumper G-4 (green leg of safety system bypass with exception of power level interlock).
___31. Install jumper Y-1 (yellow leg of safety system bypass).
___32. ENSURE scram TAAs will not reset.
___33. Remove jumper Y-1.
___34. Remove jumper G-4.
___35. Install jumper G-20 (reactor loop B low flow annunciator FT-912H Valve 546 A/B control).
___36. Install jumper Y-22 (Valve 546 A/B control FT-912A).
___37. Place Master Control Switch 1S1 to 'TEST' position.
___38. Place IN Pool HX Valve 546A switches to 'MAN/CLOSE' position.
___39. Place IN Pool HX Valve 546B switches to 'MAN/CLOSE' position
___40. ENSURE DPS-929 is below scram set point.
___41. Place Valve 546A switches in 'AUTO/OPEN' position (Valve 546A will open).
___42. Place Valve 546B switches in 'AUTO/OPEN' position (Valve 546B will open).
___43. Place Master Control Switch 1S1 to 'ON' position.
___44. ENSURE DPS-929 is below scram set point.
___45. Place Master Control Switch 1S1 to 'TEST' position.
___46. Place Valve 546A switches in 'MAN/CLOSE' position.
___47. Place Valve 546B switches in 'MAN/CLOSE' position.
___48. Restore primary coolant flow of 1,825 gpm to 1,950 gpm in each loop and 3,650 gpm to 3,900 gpm total flow.
___49. Place Valve 546A switches in 'AUTO/OPEN' position.
___50. Place Valve 546B switches in 'AUTO/OPEN' position.
___51. Place Master Control Switch 1S1 to 'ON' position.
___52. Secure Primary Coolant System Circulation Pumps P501A/B at the same time.
___53. CLOSE HX503B Inlet Valve 510F.
___54. START Primary Coolant System Circulation Pump P-501A.
___55. Place Primary Coolant System Flow Bypass Switch 2S41 to 'HX503B' position.
___56. Place Power Level Switch 1S8 to '5MW' position.
___57. Place NI channel 6 drawer switch to 'ZERO' position.

____58. Remove 10 MW module AND install 5 MW module for NI Channel 6.
____59. Place NI Channel 6 drawer switch to 'OPERATE' position AND reset drawer to clear NUCLEAR INSTRUMENT ANOMALY on Annunciator.
____60. RESET scram TAAs.

CAUTION:	Do not allow primary system pressure, as read at pump discharge, to exceed 110 psig. If necessary, use Pump Bypass Valves 538A and 538B to decrease flow.

____61. Lower flow rate by throttling HX-503A Outlet Valve 540A until the scram occurs.
____62. RECORD primary coolant system flow rate and DPS-929 meter reading on datasheet.
____63. VERIFY scram TAAs and magnet current zero.
____64. Raise flow rate by throttling Valve 540A to minimize cycling of Valves 546 A/B.
____65. Place Magnet Current Switch 1S14 to 'OFF' Position.
____66. Place Primary Coolant System Flow Bypass Switch 2S41 to '10 MW' position.
____67. Place Power Level Switch 1S8 to '10 MW' position.
____68. Place NI channel 6 drawer switch to 'ZERO' position.
____69. Remove 5 MW module AND install 10 MW module for NI Channel 6.
____70. Place NI Channel 6 drawer switch to 'OPERATE' position AND reset drawer to clear NUCLEAR INSTRUMENT ANOMALY on Annunciator.
____71. OPEN Valve 510F.
____72. START Primary Coolant System Circulation Pump P-501B.
____73. Place Valve 527E switch in the 'OPEN' position.
____74. START Primary Coolant Demineralizer Pump.
____75. Place Master Control Switch 1S1 to 'TEST' position.
____76. Place Valve 546A switches in 'MAN/CLOSE' position (resets DPS-929).
____77. Place Valve 546B switches in 'MAN/CLOSE' position (resets DPS-929).
____78. Place Valve 546A switches in 'AUTO/CLOSE' position.
____79. Place Valve 546B switches in 'AUTO/CLOSE' position.
____80. Place Master Control Switch 1S1 to 'ON' position.
____81. Restore primary coolant flow of 1,825 gpm to 1,950 gpm in each loop and 3,650 gpm to 3,900 gpm total flow.
____82. Secure Primary Coolant System per OP-RO-410, "Primary Coolant System."
____83. Secure Pool Coolant System per OP-RO-460, "Pool Coolant System – Two Pump Operation," or OP-RO-461, "Pool Coolant System – One Pump Operation."
____84. Remove jumper G-20.
____85. Remove jumper G-11.
____86. Remove jumper Y-22.
____87. Remove jumper Y-2.
____88. Remove shorted relay from K-38 position and install relay K-38.
____89. Remove shorted relay from K-30 position and install relay K-30.
____90. Remove Dummy Load Test Connectors and connect control rod drive mechanism cables.
____91. ENSURE all jumpers, shorted relays and Dummy Load Test Connectors are removed.
____92. Sign and date datasheet.
____93. Perform CP-23 Post Maintenance Valve Line-Up Checksheet.
____94. Log CP complete in Console Log Book and Maintenance Day Book.

Date Completed:____________________

LSRO Signature:__________________________________

ATTACHMENT 2

CP-23 Datasheet
DPS-929

PERFORMED BY:______________________________
____
____

____________________________ ________________
Lead Senior Reactor Operator Date

DPS-929

Crystal Engineering Reading	Meter	Milliamps*	
5 psig	________psig	________	(5.64 – 6.36)
10 psig	________psig	________	(7.64 – 8.36)
15 psig	________psig	________	(9.64 – 10.36)
20 psig	________psig	________	(11.64 – 12.36)
25 psig	________psig	________	(13.64 – 14.36)
30 psig	________psig	________	(15.64 – 16.36)
35 psig	________psig	________	(17.64 – 18.36)

<u>DPS-929 (Yellow & Green Leg) Mode I</u>

Loop A Flow____________gpm

Loop B Flow____________gpm

Total Flow____________gpm
3350 - 3450 gpm

DPS Meter Reading______psi

<u>DPS-929 Mode II</u>

Low Flow Scram____________gpm
1625 - 1675

DPS Meter Reading__________psi

* If the above data does not fall within the MA bandwidth prescribed, first verify the present scram set point then calibrate the transmitter in accordance with procedure.

APPROVED:____________________
Reactor Manager

CP-23 Post Maintenance Valve Line-Up Checksheet

Date:___________________

VALVE	DESCRIPTION	POSITION	PERFORMER	VERIFIER
568H-HP	DPS-929 HP Isolation	OPEN		
568H-LP	DPS-929 LP Isolation	OPEN		
568H-EQ	DPS-929 Equalizing Isolation	CLOSED		
599AB	DPS-929 HP Test Isolation	CLOSED		
599AC	DPS-929 LP Test Isolation	CLOSED		
540A	HX-503A Outlet	THROTTLED		
540B	HX-503B Outlet	THROTTLED		
510F	HX-503B Inlet	OPEN		
538A	P-501A Bypass	CLOSED		
538B	P-501B Bypass	CLOSED		

Approved:___________________
Reactor Manager

ATTACHMENT 3

FM-152
Revision 2

FUEL ELEMENT INSPECTION

Visual Inspection of End Plate Surfaces

Element No.__________ Element From Core__________ MWD__________

Comments:____

____

____

____

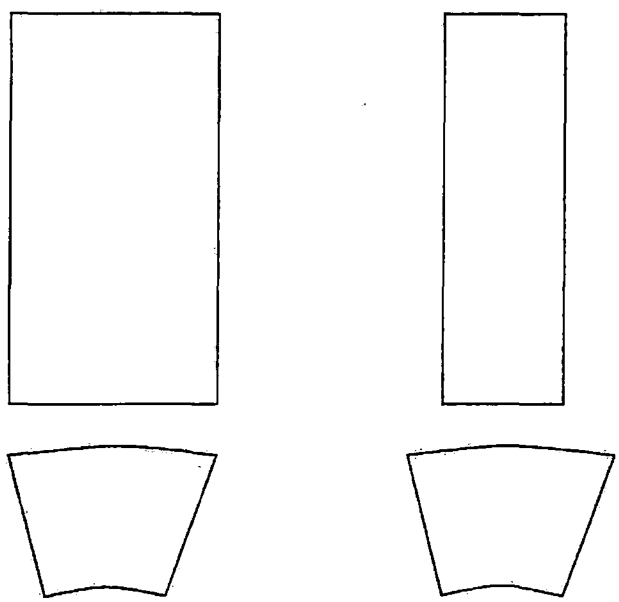
Inspected By:______________________________
(Print Name)

Inspected By:______________________________ ______________________________
(Signature) (Date)

Reviewed By:______________________________ ______________________________
(Assistant Reactor Manager-Physics) (Date)

Reviewed By:______________________________ ______________________________
(Reactor Manager) (Date)

Plate 24 Plate 1



Top Bottom

ATTACHMENT 3

FM-152
Revision 2

FUEL ELEMENT INSPECTION

Water Gap Inspection

Element No.____________ Element From Core_____________ MWD___________

Measured diameter of inspection rods_________ / ________ Checked By:___________________
(Diameter must be greater than or equal to 0.0620 inches) (Print Name)

NOTE:	Inspector shall mark either the GO or NO-GO blanks as appropriate. Successful completion of the water gap inspection requires that only GO blanks are marked.

	GO	NO-GO	Comments
Gap 1 (inner-most)	________	________	______________________________
Gap 2	________	________	______________________________
Gap 3	________	________	______________________________
Gap 4	________	________	______________________________
Gap 5	________	________	______________________________
Gap 6	________	________	______________________________
Gap 7	________	________	______________________________
Gap 8	________	________	______________________________
Gap 9	________	________	______________________________
Gap 10	________	________	______________________________
Gap 11	________	________	______________________________
Gap 12	________	________	______________________________
Gap 13	________	________	______________________________
Gap 14	________	________	______________________________
Gap 15	________	________	______________________________
Gap 16	________	________	______________________________
Gap 17	________	________	______________________________
Gap 18	________	________	______________________________
Gap 19	________	________	______________________________
Gap 20	________	________	______________________________
Gap 21	________	________	______________________________
Gap 22	________	________	______________________________
Gap 23	________	________	______________________________

Inspected By:______________________ ________________________ _____________
(Print Name) (Signature) (Date)

Reviewed By:______________________________ ____________________________
(Assistant Reactor Manager-Physics) (Date)

Reviewed By:______________________________ ____________________________
(Reactor Manager) (Date)

FM-152
Revision 2

FUEL ELEMENT INSPECTION

Guidelines

Purpose

To comply with our Technical Specifications requirement that one out of every eight (8) spent fuel elements be inspected for anomalies (T.S. Section 5.5 and 3.8c). Additionally, fuel elements may be inspected if damage to an element is suspected.

Equipment Needed

- Fuel Inspection Rig (usually hangs over the rail on the north side of the bridge)
- Fuel Inspection Board (wooden board that is usually stored on the Fifth Level)
- Two (2) Long Poles with Welding Rod attached to its end
- Calipers

Precautions

Before a fuel element can be raised up for inspection, Health Physics personnel need to be present at the bridge for verifying radiation levels.

The fuel element to be inspected is transferred to the Fuel Inspection Rig and raised to a level approved by Health Physics personnel monitoring the evolution.

EPDs and ring badges shall be worn by the Operator inspecting the fuel element.

Ensure the welding rod is not bent. If it is bent, replace with a new one (can use either Al or SS welding rod).

The welding rod thickness needs to be equal to or greater than 0.062" to comply with T.S. requirements.

Visual Inspection Procedure Guidelines

The outside surfaces of plates 1 and 24 shall be visually inspected from bridge level and comments recorded on Page One of the form.

Water Gap Procedure Guidelines

By lying down on the Fuel Inspection Board, insert the Welding Rod into each of the 23 fuel gaps to inspect gap thickness.

NOTE: Using two Welding Rods will make it easier to step through the gaps without losing your place.

Verify the gap thickness between the fuel plates and record the conditions on the form.

Compliance Check: Emergency Generator Load Test	**Frequency:** Semi-Annually
Plant Conditions: Reactor Secured Primary, Pool and Secondary Coolant Systems Shutdown	**Number of Men Needed:** 3 **Estimated Time:** 1 Hour

Test Equipment, Tools and Materials:

1. Walkie Talkies

References:

Technical Specifications 3.10, 4.5 and 5.6
Technical Manual 159, Cummins Power Systems - Model 275NT6
OP-RO-520, "Emergency Diesel Generator"
OP-RO-730, "Facility Exhaust System"
Print No. 522, Sheet 4 of 7, "Electrical Distribution Emergency Electrical Power System"

Procedure:

____1. Perform Emergency Diesel Generator Preoperational Checks, per OP-RO-520.
____2. Station operators in the control room, at the emergency generator and at Substation 'B.'
____3. NOTIFY MU Police Department (MUPD) of potential alarms due to electrical testing.
____4. Prop open the back-up doors.
____5. Place the Reactor Isolation horns cutout switch in '<u>Cutout</u>'.
____6. Announce to the facility that a test of the Emergency Electrical Distribution System will occur.
____7. Trip the '<u>Auto Transfer Switch</u>' breaker at Substation 'B.'
____8. VERIFY that the emergency generator starts and assumes electrical load.

<u>NOTE</u>: EF-13/14 fast speed operation is preferred to maintain a normal ventilation line-up.

<u>NOTE</u>: Switching fans to fast speed will occasionally result in a fan trip, causing the non-running fan to start in slow speed. The breaker on the Emergency Distribution Center may or may not trip.

____9. PLACE the running exhaust fan (EF-13/14) in fast speed.
____10. After the emergency generator has been running for at least 20 minutes, RECORD normal operating parameters on the Datasheet.
____11. After operating data has been recorded and the emergency generator has been running <u>at least 25 minutes</u>, RESET and CLOSE the '<u>Auto Transfer Switch</u>' breaker at Substation 'B.'
____12. Announce to the facility that a reduction in facility exhaust will occur.
____13. SHIFT the exhaust fan (EF-13/14) to SLOW speed.
____14. VERIFY that the load shifts back to '<u>Normal</u>' supply approximately ten (10) minutes after closing the Auto Transfer Switch breaker.

APPROVED: ______________________
Reactor Manager

NOTE:	Switching fans to fast speed will occasionally result in a fan trip, causing the non-running fan to start in slow speed. The breaker on the Emergency Distribution Center may or may not trip.

____15. SHIFT the exhaust fan (EF-13/14) to FAST speed.
____16. Announce to the facility that facility exhaust has returned to normal line-up.
____17. VERIFY the emergency generator secures approximately five (5) minutes after returning the load to 'Normal' supply.
____18. VERIFY all emergency distribution breakers not designated as spares are 'Shut/On.'
____19. IF the air filter indicator is in the red band after testing is complete, THEN change the air filter.
____20. Remove props from back-up doors.
____21. Place the Reactor Isolation horns cutout switch to 'Normal.'
____22. ENSURE alarms are reset at keypad.
____23. NOTIFY MUPD that testing is complete AND VERIFY their alarm has cleared.
____24. Announce to the facility that electrical system testing is complete and to regard all further alarms.
____25. Restart Waste Heat Pump AXP-2 (if necessary).
____26. Log CP Complete in Console Log Book and Maintenance Day Book.

Date Completed:____________________

LSRO Signature:_________________________________

DATASHEET

EMERGENCY GENERATOR

Date:________________________

Performed by:__________________________________

____

____

RPM____________________ Hertz____________________
1760-1830 59-61

Oil Press.________________
~ 50 psi

Phase
Volts: 1.______________ 2.______________ 3.______________
460-500 (Green) 460-500 (Green) 460-500 (Green)

Amps: 1.______________ 2.______________ 3.______________
(Green) (Green) (Green)

Air Filter Replaced? Yes / No

Signed:____
Lead Senior Reactor Operator

APPROVED:________________________
Reactor Manager