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December 12, 2019

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

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

SUBJECT: Written communication as specified by 10 CFR § 50.4(b)(1) requesting U.S. Nuclear Regulatory Commission approval to revise the Technical Specifications appended to Renewed Facility Operating License No. R-103 pursuant to 10 CFR § 50.90

Enclosed is an application to amend Renewed Facility Operating License No. R-103 by revising University of Missouri Research Reactor (MURR) Technical Specification (TS) 1.26, "Reactor Secured;" TS 3.4, "Reactor Containment Building;" TS 4.4, "Reactor Containment Building;" and TS 6.4, "Procedures;" pursuant to 10 CFR § 50.90. The proposed, requested changes should be considered permanent changes to the MURR TSs. Additionally, the proposed, requested changes were favorably reviewed on December 4, 2019, by the Reactor Safety Subcommittee (RSS), a subcommittee of the Reactor Advisory Committee (RAC), in accordance with TS 6.2.a(4).

Enclosure 1 provides the basis for revising TS 1.26, Enclosure 2 provides the basis for revising TS 3.4, Enclosure 3 provides the basis for revising TS 4.4, and Enclosure 4 provides the basis for revising TS 6.4. Enclosure 5 contains the proposed, revised TS pages with track changes. Enclosure 6 contains the proposed, revised TS pages with changes accepted and revision bars.

MURR requests that the proposed license amendment be reviewed within 24 months from the date of this letter. If there are any questions regarding this license amendment request, please contact me at (573) 882-5118 or MeffertB@missouri.edu. I declare under penalty of perjury that the foregoing is true and correct.

A020
NRR

Sincerely,

Acting Reactor Manager
Jeremy Custer

Bruce A. Meffert
Reactor Manager

ENDORSEMENT:
Reviewed and Approved,

J. David Robertson
Reactor Facility Director

State of Missouri
County of Boone
Subscribed and sworn to before me this 12 day of December, 2019

Jacqueline L. Matyas
JACQUELINE L. MATYAS, Notary Public
My Commission Expires: March 26, 2023

NOTARY PUBLIC NOTARY SEAL STATE OF MISSOURI

JACQUELINE L. MATYAS
My Commission Expires
March 26, 2023
Howard County
Commission #15634308

cc: Reactor Advisory Committee
Reactor Safety Subcommittee
Isotope Use Subcommittee
Dr. Mark McIntosh, Vice Chancellor for Research, Graduate Studies and Economic Development
Mr. Geoffrey Wertz, U.S. Nuclear Regulatory Commission
Mr. William Schuster, U.S. Nuclear Regulatory Commission

Enclosures:

1. Basis for the Requested Change to Technical Specification 1.26
2. Basis for the Requested Change to Technical Specification 3.4
3. Basis for the Requested Change to Technical Specification 4.4
4. Basis for the Requested Change to Technical Specification 6.4
5. Proposed, revised Technical Specification pages – A-4, A-24, A-47, and A-70 (with track changes)
6. Proposed, revised Technical Specification pages – A-4, A-24, A-47, and A-70 (with accepted changes, revision bars)

Attachments:

1. MURR Safety Analysis Report Section 4.2.2.1
2. MURR Safety Analysis Report Section 9.1
3. MURR Safety Analysis Report Section 6.2
4. MURR Safety Analysis Report Section 7.8

Enclosure 1 – Basis for the Requested Change to Technical Specification 1.26

1.0 Introduction

The University of Missouri Research Reactor (MURR) is requesting a change to Technical Specification (TS) 1.26, "Reactor Secured." This change will revise condition b(4) of Specification 1.26 in order to allow work on uninstalled shim rod drive mechanisms (SRDMs), which are physically decoupled from the shim control blades (rods), thus providing greater flexibility in satisfying the requirements of TS 1.26 [Note: the SRDMs are also referred to as control rod drive mechanisms (CRDMs) in the MURR Safety Analysis Report (SAR)]. Allowing work on SRDMs that are physically decoupled from their associated shim control blades will reduce the time that reactor containment integrity is required by TS 3.4, "Reactor Containment Building," specifically Specification 3.4.b, while the reactor is shut down for maintenance. Minimizing the time that reactor containment integrity is required during reactor maintenance activities will help reduce the chance that equipment failure or human error would cause MURR to deviate from TS 3.4.b. Additionally, minimizing the time that reactor containment integrity is required during reactor maintenance will also allow greater scheduling flexibility for maintenance activities with a consequent efficiency increase in both personnel availability and reactor utilization.

The following license amendment request (LAR) provides justification that working on uninstalled SRDMs does not decrease the safety of reactor operations nor does it increase hazards or risks to the health and safety of the public.

2.0 Background

Original Facility Operating License No. R-103, issued on October 11, 1966, contained definition TS 1.1, "Reactor Secured," which stated:

> "The reactor shall be considered secured whenever it contains insufficient fuel in the reactor to establish criticality with all control rods removed or whenever all of the following conditions are met:
>
> (a) All shim rods are fully inserted.
>
> (b) The "Master Control" switch is on the "off" position and the key is removed and in the custody of a licensed operator.
>
> (c) No work is in progress involving handling of fuel or involving maintenance of the core structure, including control rods or their drives."

Amendment No. 2 to Amended Facility Operating License No. R-103, issued on July 9, 1974, contained definition TS 1.20, "Reactor Secured," which stated:

> "The reactor shall be considered secured whenever it contains insufficient fuel in the reactor core to establish criticality with all control rods removed or whenever all of the following conditions are met:
>
> a. All shim rods are fully inserted.

b. The "Master Control" switch is in the "off" position.

c. The "Master Control" switch key is removed and locked in the key box or is in the custody of a licensed operator.

d. No work is in progress involving transferring fuel in or out of the core.

e. No work is in progress involving control rods or control rod drives.

f. The reactor pressure vessel cover is secured in position and no work is in progress on the pressure vessel or its supports."

Amendment No. 14 to Amended Facility Operating License No. R-103, issued on April 15, 1981, revised definition TS 1.20, "Reactor Secured," to allow dummy load test connectors to be part of the definition. With dummy load test connectors installed, it is impossible to move the shim control blades, even inadvertently, from the reactor control console. Amendment No. 14 definition TS 1.20 stated:

"The reactor shall be considered secured whenever it contains insufficient fuel in the reactor core to establish criticality with all control rods removed or whenever all of the following conditions are met:

a. All shim rods are fully inserted.

b. One of the following conditions exists:

(1) The "Master Control" switch is in the "off" position with the key locked in the key box or in the custody of a licensed operator.

(2) A licensed operator is present in the Control Room and the dummy load control rod test connectors are installed.

c. No work is in progress involving transferring fuel in or out of the core.

d. No work is in progress involving the control rods or control rod drives with the exception of installing or removing dummy load control rod test connectors.

e. The reactor pressure vessel cover is secured in position and no work is in progress on the pressure vessel or its supports."

The existing TS definition of "Reactor Secured" became effective with the issuance of Renewed Facility Operating License No. R-103 on January 4, 2017. TS 1.26, "Reactor Secured," currently states:

"The reactor shall be considered secured when:

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

OR

b. Whenever all of the following conditions are met:

(1) All four shim rods are fully inserted;

(2) One of the two following conditions exists:

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

OR

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

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

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

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

3.0 Justification for Changing Technical Specification 1.26

When the "reactor secured" definition is met, the reactor core and its support equipment are in a condition where attaining criticality inadvertently is nearly impossible. The definition requires either there is insufficient fuel in the reactor core to attain criticality with optimum available conditions of moderation and reflection with all four (4) shim blades removed, or the four (4) shim blades are inserted and the SRDMs are in a condition to ensure that all four (4) shim blades cannot be inadvertently withdrawn.

American National Standard ANSI/ANS-15.1-2007 (R2013), "The Development of Technical Specifications for Research Reactors," Section 1.3, "Definitions," contains the following condition for work on control rods and control rod drives in its definition of "reactor secured:"

"(c) No work is in progress involving core fuel, core structure, installed control rods, or control rod drives unless they are physically decoupled from the control rods;"

MURR's proposed change to TS 1.26 to allow work on uninstalled SRDMs that are physically decoupled from the shim control blades aligns with the ANSI/ANS-15.1 definition of "reactor secured."

When MURR's Instrumentation Support staff work on SRDMs, the SRDMs are removed from the reactor, taken out of the reactor containment building, and placed on a test stand located inside the Instrumentation

Support shop. The following photos convey how the SRDMs are removed and physically decoupled from the shim control blades (rods) for maintenance. For additional information on the description of the shim control blades, their offset mechanisms, and their control drive mechanisms, see Section 4.2.2.1 of the MURR SAR. Section 4.2.2.1 has been provided as Attachment 1 to this LAR.

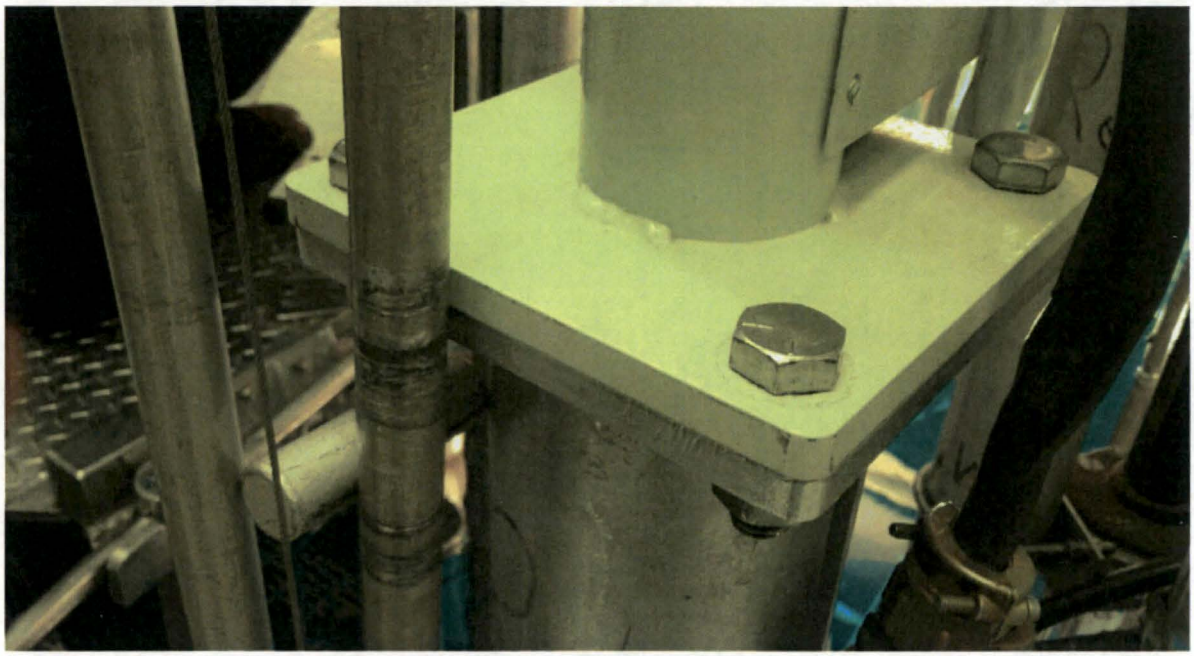


FIGURE 1
SRDM (green paint) Bolted and Installed on the Offset Mechanism Upper Housing (no paint)

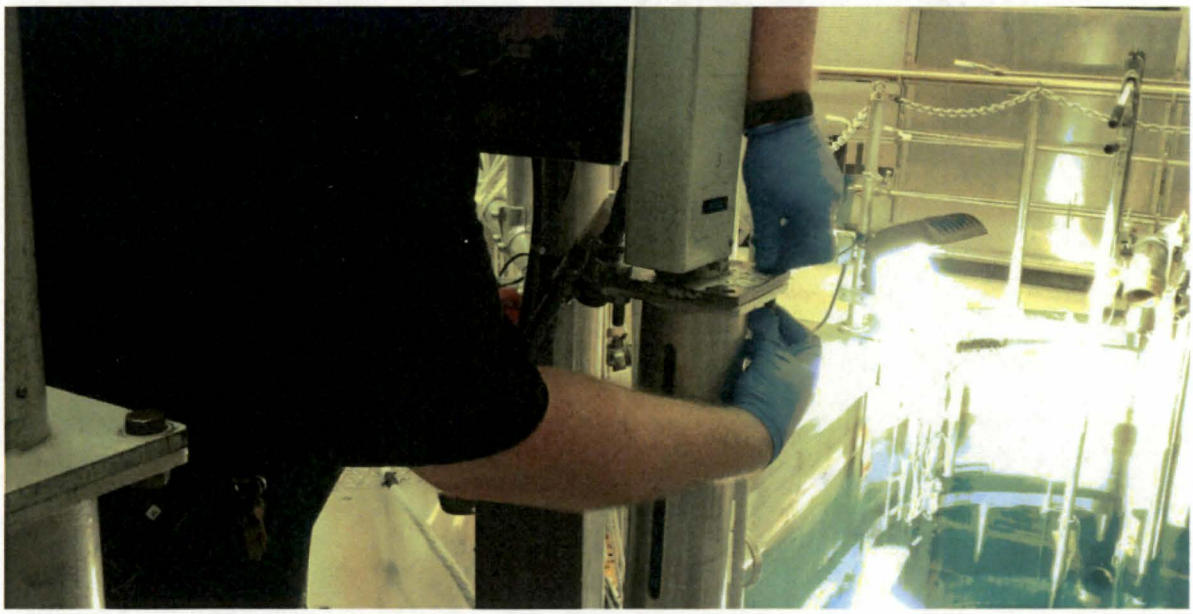


FIGURE 2
Operator Unbolting a SRDM (green paint) from the Offset Mechanism Upper Housing (no paint)

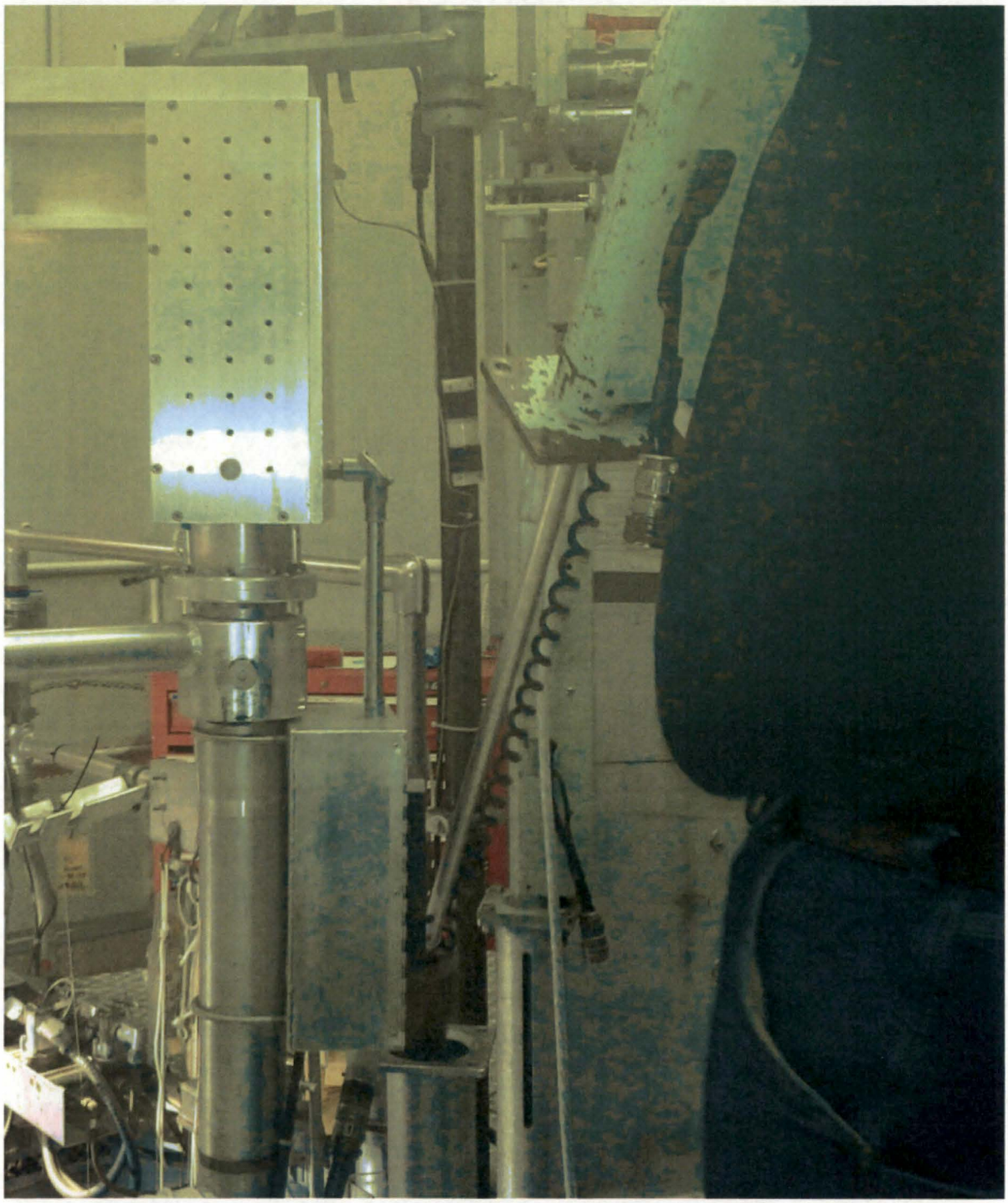


FIGURE 3
Operator Lifting a SRDM from the Offset Mechanism Upper Housing

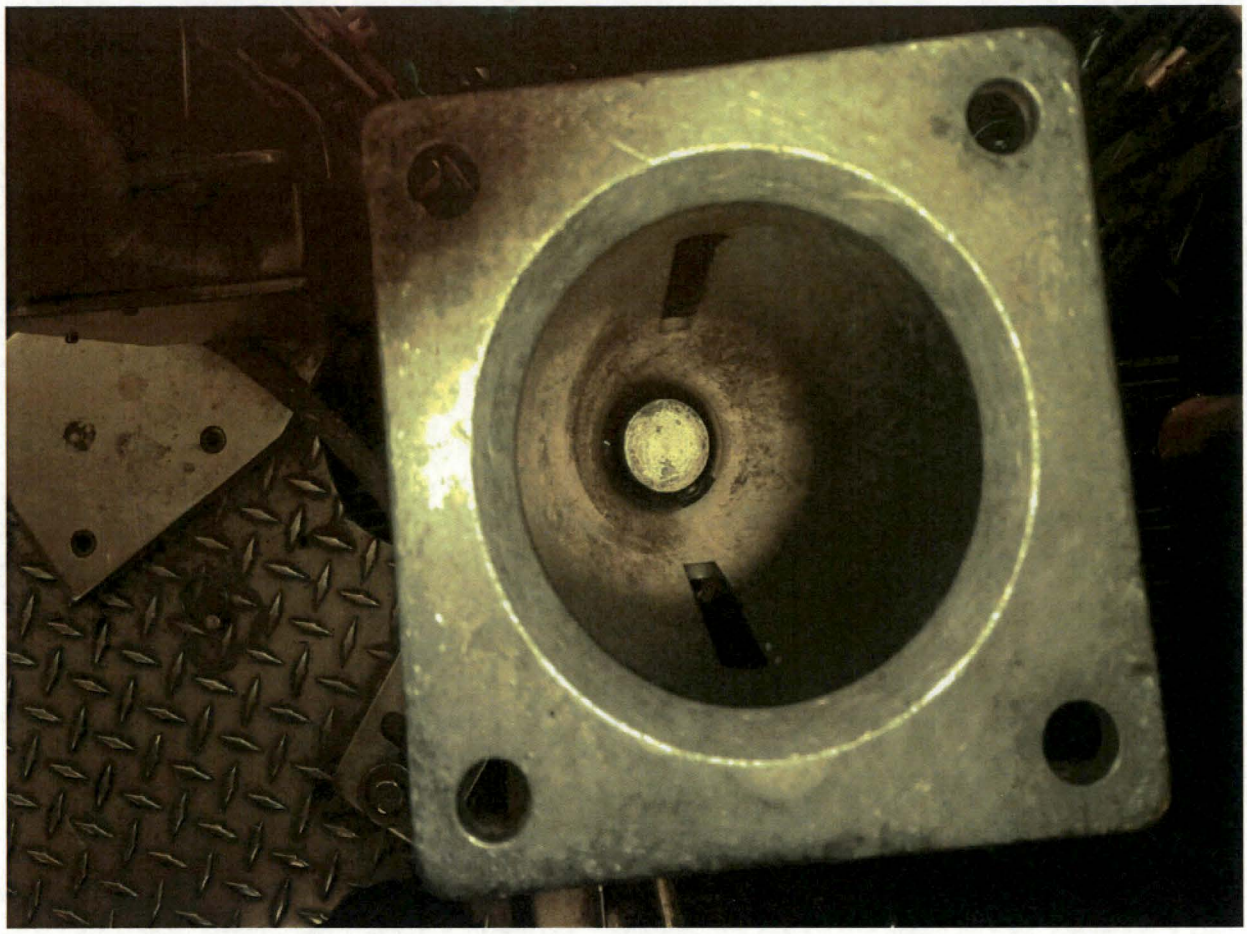


FIGURE 4
Offset Mechanism Upper Housing with SRDM Removed
(Note that the anvil where the electromagnet connects to the control blade lift-rod assembly is approximately 33 inches below the flange face in a 3-inch diameter pipe)

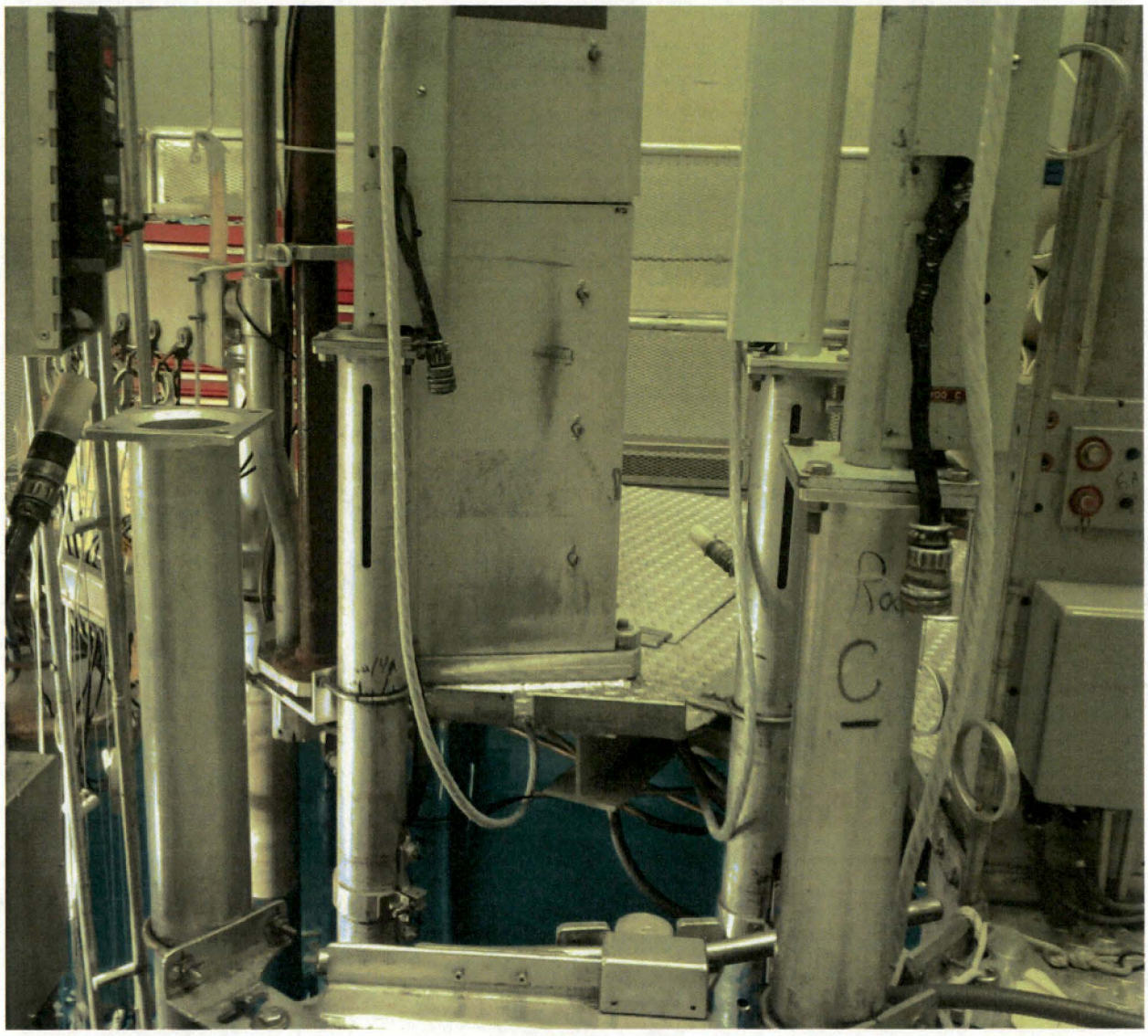


FIGURE 5
Comparison between three (3) Offset Mechanism Upper Housings with Installed SRDMs and Control Power Connectors Disconnected and One Offset Mechanism Upper Housing with SRDM Removed

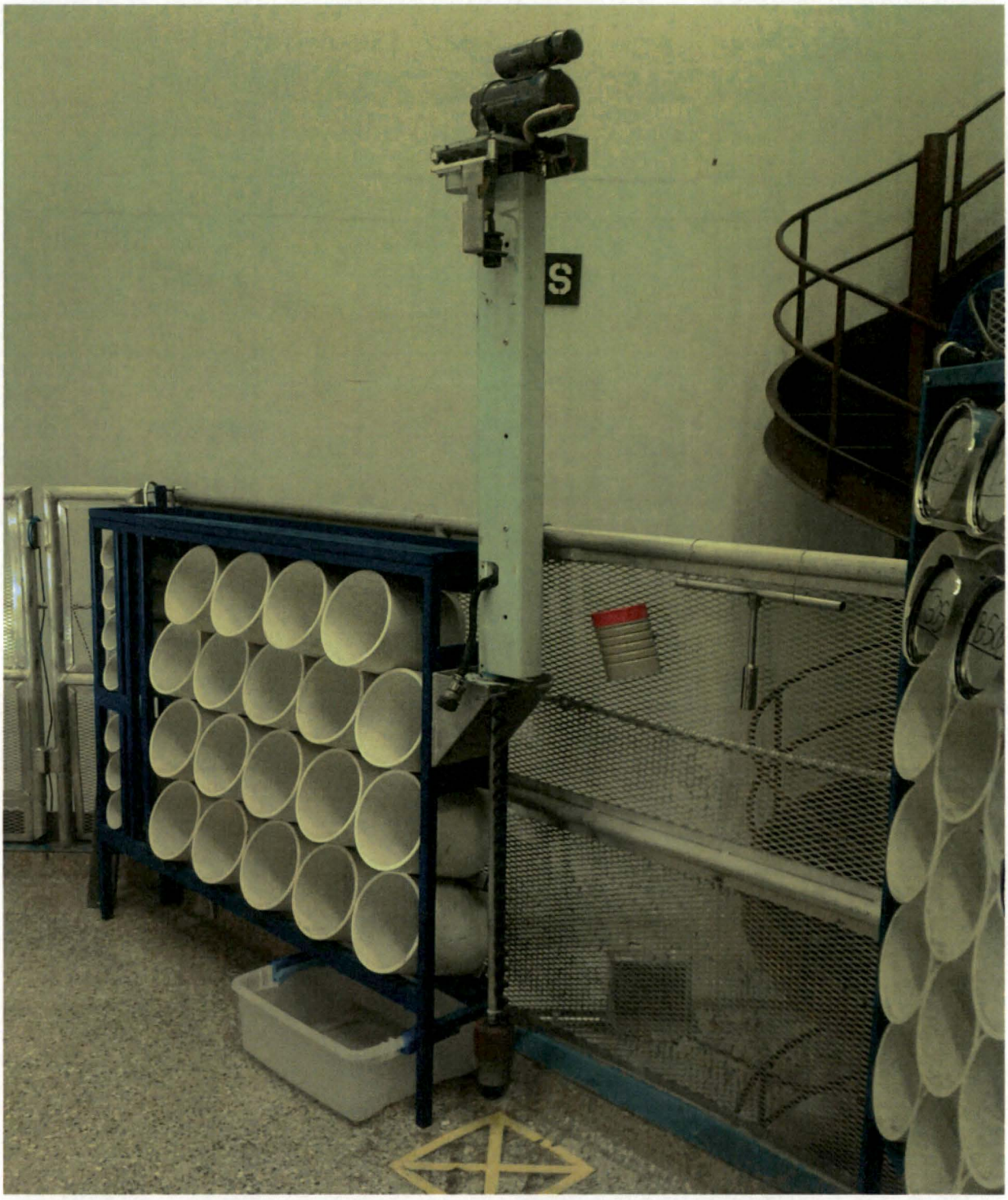


FIGURE 6
SRDM, which is Uninstalled and Physically Decoupled from its Shim Control Blade, on a Storage Stand Located in the Reactor Bridge Area

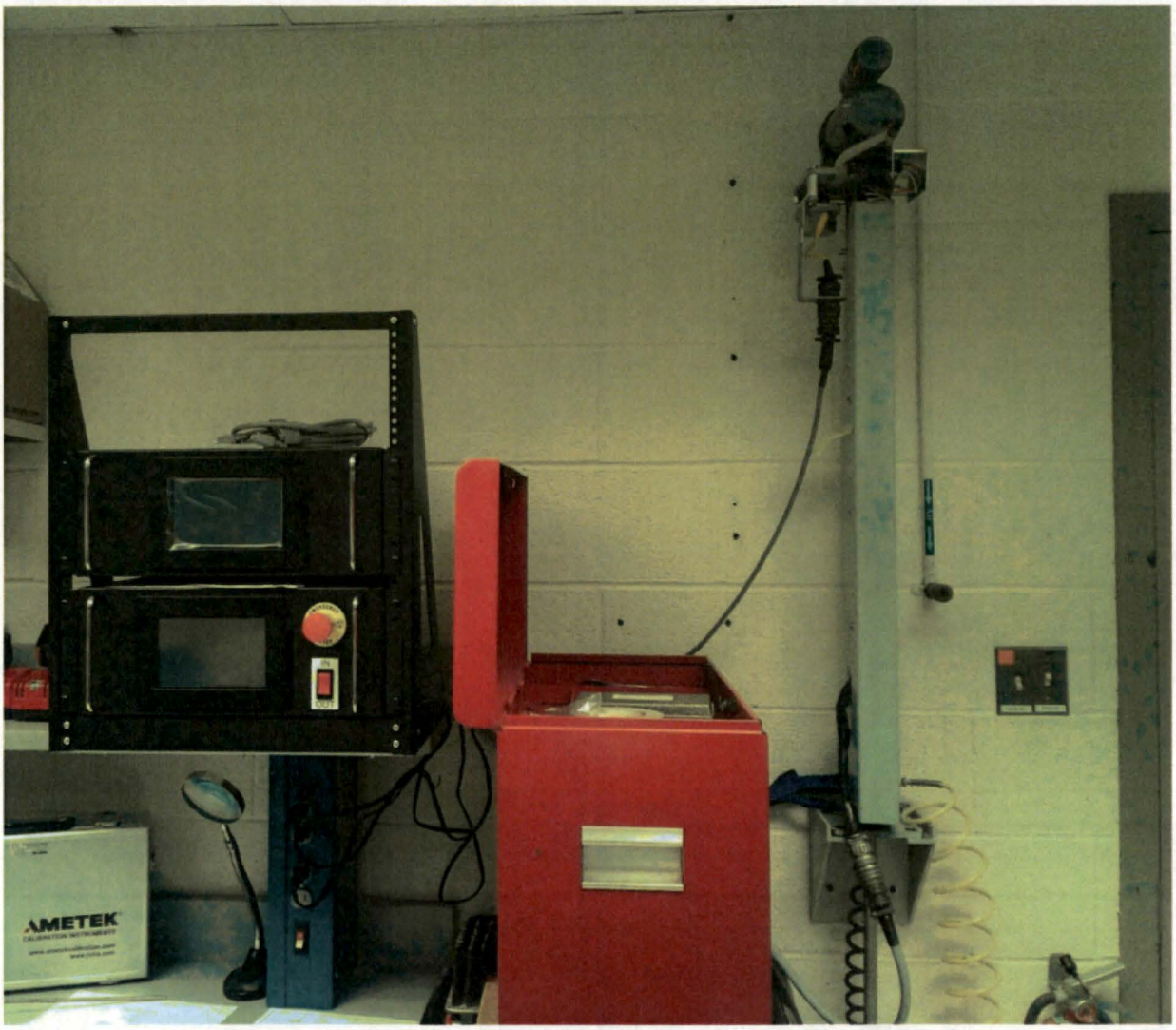


FIGURE 7
SRDM, which is Uninstalled and Physically Decoupled from its Shim Control Blade, on a Test Stand in the Instrumentation Support Shop ready for Preventative Maintenance and Bench Testing

As can be seen in the photographs and discussed in Section 4.4.4.2 of the MURR SAR, the only component that physically couples the SRDM to a shim control blade is an energized electromagnet. If the SRDM is electrically disconnected so that no electrical current can flow through the SRDM electromagnet, that SRDM can no longer change the position of its shim control blade – it is physically decoupled from the control blade even if the SRDM is installed in its offset mechanism upper housing. Furthermore, uninstalling (or removing) the SRDM from the offset mechanism upper housing provides an additional barrier that ensures the decoupling of the SRDM from its associated shim control blade. Therefore, TS 1.26, "Reactor Secured," can be revised as proposed below to allow for work to be performed on uninstalled and physically decoupled SRDMs.

4.0 Conclusion

Allowing work on uninstalled SRDMs, which are physically decoupled from their associated shim control blades, has no adverse effects to reactor safety. Therefore, MURR requests approval for the below revised Specification 1.26.

5.0 Proposed Revision to Technical Specification 1.26

Specification 1.26 currently states:

"The reactor shall be considered secured when:

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

OR

b. Whenever all of the following conditions are met:

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

(2) One of the two following conditions exists:

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

OR

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

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

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

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

Specification 1.26 will be revised as follows:

"The reactor shall be considered secured when:

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

OR

b. Whenever all of the following conditions are met:

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

(2) One of the two following conditions exists:

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 shim blades (rods) or installed shim rod drive mechanisms unless the shim rod drive mechanisms are physically decoupled from the shim blades (rods), 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."

Enclosure 2 – Basis for the Requested Change to Technical Specification 3.4

1.0 Introduction

The University of Missouri Research Reactor (MURR) is requesting a change to Technical Specification (TS) 3.4, "Reactor Containment Building." This change will revise condition (6) of Specification 3.4.a in order to clarify that containment integrity will still exist even if the negative pressure of the reactor containment building to the surrounding areas is less than 0.25 inches of water when the reactor containment building is isolated. This clarification will help eliminate any potential confusion about the condition of containment integrity when isolation of the reactor containment building occurs and the negative pressure decreases below 0.25 inches of water.

The following license amendment request (LAR) provides justification that the negative pressure requirement of this Specification is not applicable when the containment system has isolated the reactor containment building. In addition, the negative pressure requirement of Specification 3.4.a(6) is not needed to ensure the health and safety of the general public during a reactor isolation.

2.0 Background

The MURR TSs have contained condition (6) of Specification 3.4.a since the relicensing of MURR on January 4, 2017. Prior to January 4, 2017, there was no negative pressure requirement. The negative pressure requirement for containment integrity was established assuming normal operation of the reactor containment building ventilation system. When MURR and U.S. Nuclear Regulatory Commission (NRC) staff discussed the addition of this requirement during relicensing, no one considered that the verbiage of condition (6) was applicable only during normal ventilation system operation. However, when the quick-closing 16-inch isolation valves (designated 16A and 16B) close as a result of a manual or automatic reactor isolation, the hot exhaust line in the reactor containment building is isolated from the facility exhaust system [See Figure 9.1 on the next page taken from Chapter 9 of the MURR Safety Analysis Report (SAR)]. Without a constant air flow (approximately 2,500 cfm) out of the reactor containment building after valves 16A and 16B close, the negative pressure inside the containment building will slowly decrease until there is no pressure differential to the surrounding areas. The reactor containment building ventilation system was never designed to maintain a negative pressure during a reactor isolation, i.e., the reactor containment building isolated.

For a detailed description of MURR's ventilation supply and exhaust systems, refer to Section 9.1 of the MURR SAR, which is provided as Attachment 2. For a detailed description of MURR's containment system and the reactor containment building, refer to SAR Section 6.2, which is provided as Attachment 3. For a detailed description of MURR's reactor isolation system, refer to SAR Section 7.8, which is provided as Attachment 4.

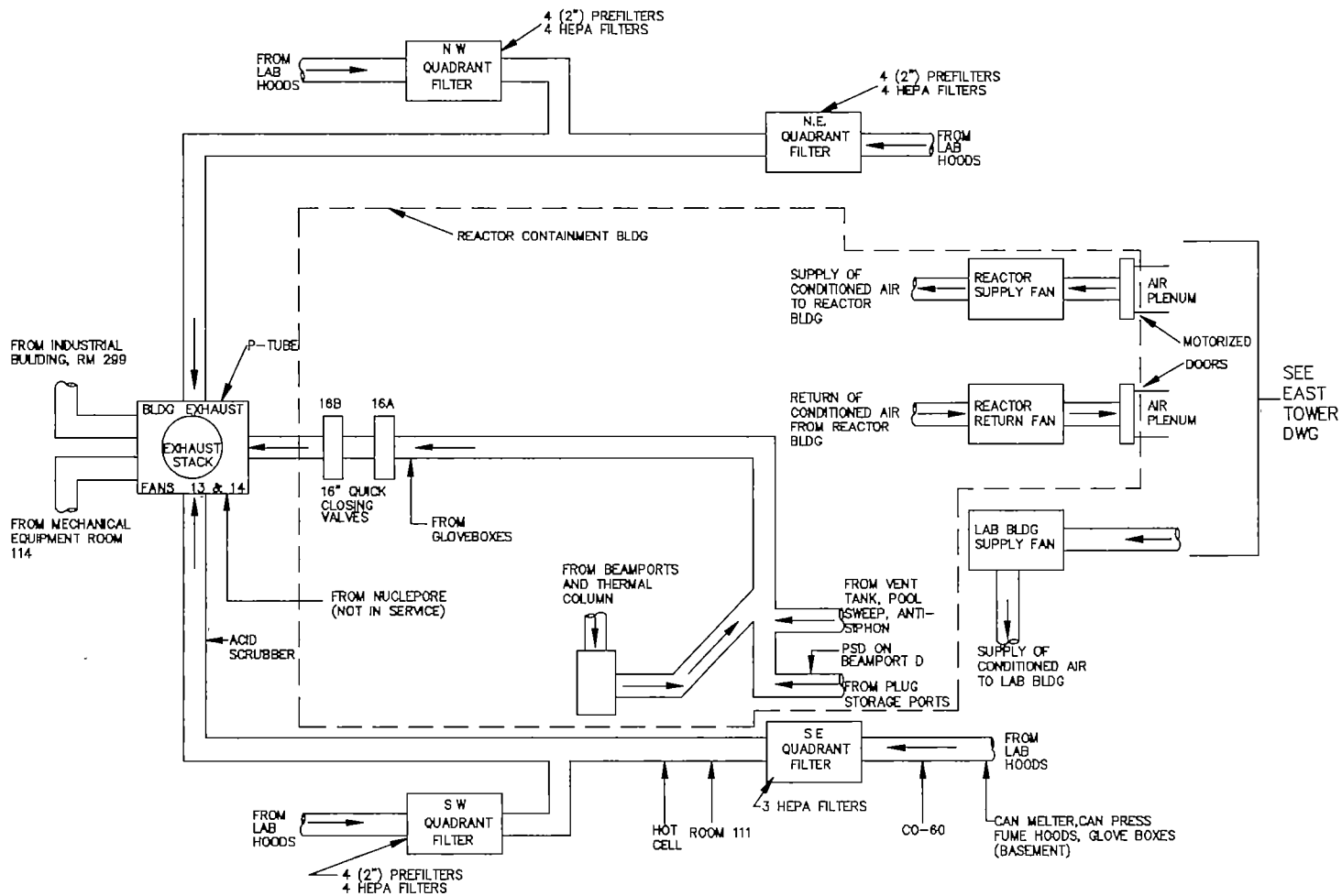


FIGURE 9.1
MURR Facility Ventilation System

3.0 Justification for Changing Technical Specification 3.4.a

The objective of TS 3.4, "Reactor Containment Building," currently states:

> "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."

Furthermore, the basis of Specification 3.4.a currently states:

> "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."

The MURR containment system was designed to isolate the reactor containment building atmosphere from the facility exhaust system along with isolating the containment building from all surrounding areas. Therefore, the original design of the system meets both the objectives of TS 3.4 and the basis of Specification 3.4.a. In addition, there are no MURR accident analyses that assume a negative pressure of the reactor containment building with respect to surrounding areas is required to mitigate an accident.

4.0 Conclusion

In effect, as Specification 3.4.a(6) is currently written, maintaining containment integrity during a reactor isolation is not possible because maintaining a negative pressure in the reactor containment building with respect to the surrounding areas with valves 16A and 16B closed is not feasible. Therefore, MURR requests approval for the below revised Specification 3.4.a.

5.0 Proposed Revision to Technical Specification 3.4.a

Specification 3.4.a currently states:

> "a. For reactor containment integrity to exist, the following conditions shall be satisfied:
>
> (1) The truck entry door is closed and sealed;
>
> (2) The utility entry seal trench is filled with water to a depth required to maintain a minimum water seal of 4.25 feet;
>
> (3) All of the reactor containment building ventilation system's automatically-closing doors and automatically-closing valves are operable or placed in the closed position;
>
> (4) The reactor mechanical equipment room ventilation exhaust system, including the particulate and halogen filters, is operating;
>
> (5) The personnel airlock is operable (one door shut and sealed);

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

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

Specification 3.4.a will be revised as follows:

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

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

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

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

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

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

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

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

No changes to the bases of TS 3.4 are required or requested.

Enclosure 3 – Basis for the Requested Change to Technical Specification 4.4

1.0 Introduction

The University of Missouri Research Reactor (MURR) is requesting a change to Technical Specification (TS) 4.4, "Reactor Containment Building." This change will revise the required periodicity of the reactor containment building leakage rate surveillance, as specified by TS 4.4.a, from annually to biennially thus affording MURR the potential to reduce adverse effects to reactor instrumentation and equipment, and the reactor containment building due to the high humidity and pressure used in conducting the leakage rate test. Some of these adverse effects could/have led to unscheduled shutdowns and undue stress to the reactor containment building structure. Unscheduled shutdowns reduce MURR's capacity to meet its mission and vision to improve the quality of life for all through nuclear science and technology. Undue stress to the reactor containment building could lead to increased containment building leakage, which would reduce its effectiveness to protect the public.

The following license amendment request (LAR) provides justification that measuring the reactor containment building leakage rate biennially is sufficient to reasonably assure proper operation of the containment system, thereby assuring that containment integrity is maintained when required so that the health and safety of the general public is assured.

2.0 Background

The MURR TSs have contained an annual requirement to measure reactor containment building leakage rate since initial licensing in 1966. From 1966 to 1978, either a pressure decay or reference volume technique was used while applying 2.0 pounds per square inch gauge (psig) pressure to the interior of the containment building.

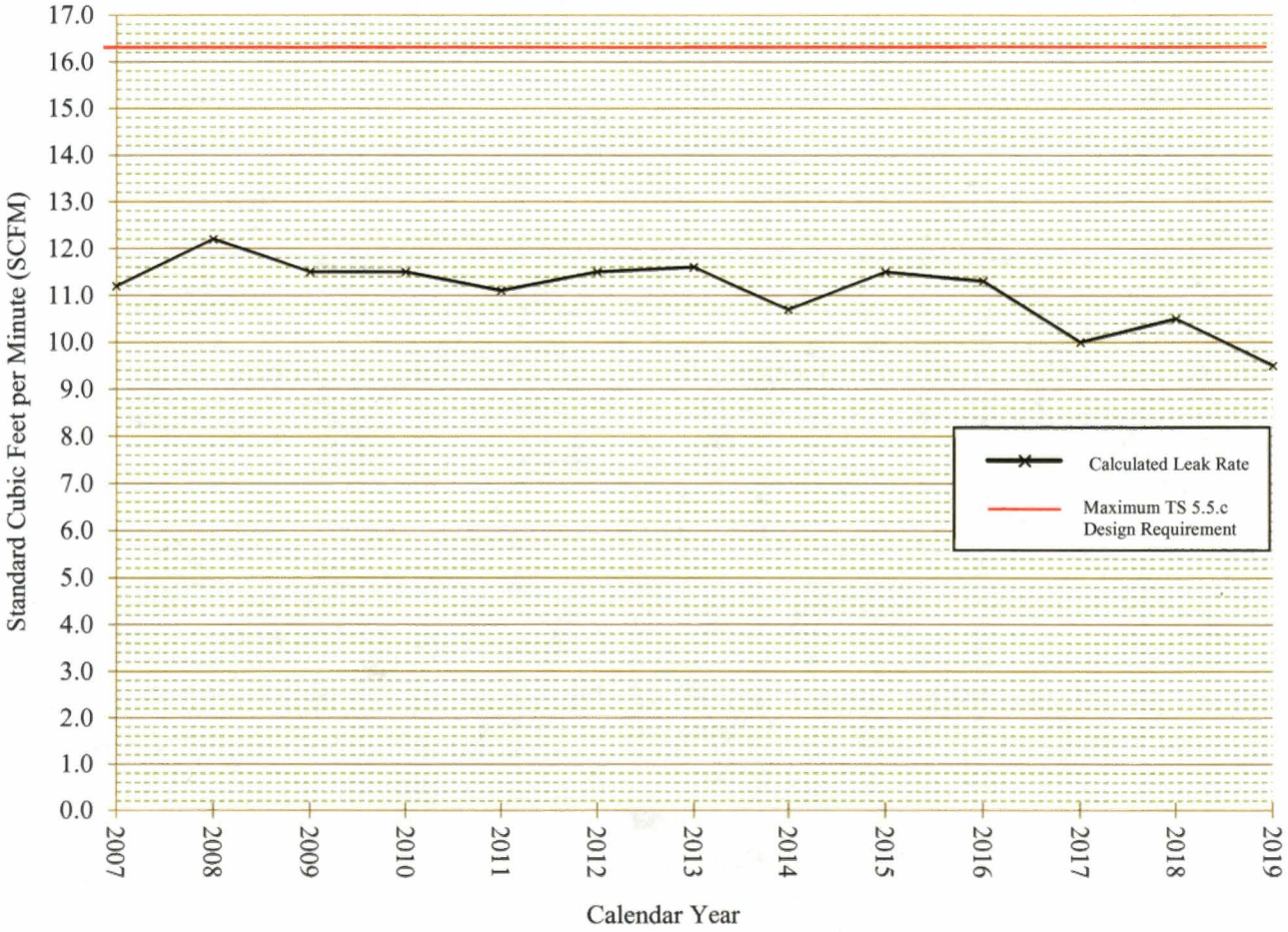
Amendment No. 10 to Amended Facility Operating License No. R-103, issued on July 13, 1978, authorized MURR to perform the reactor containment building leakage rate test at 1.0 psig using the make-up flow technique to avoid unnecessarily stressing the containment structure or evacuating the overpressure relief protection water seal trench, which would invalidate the test results. Amendment No. 10 Specification 4.2.c stated:

> "c. The containment building leakage rate shall not exceed 16.3 ft^3/min. (STP) with an overpressure of one pound per square inch gauge or 10% of 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 makeup flow, pressure decay, or reference volume techniques."

Since 1978, MURR has only used the make-up flow technique at 1.0 psig to measure the reactor containment building leakage rate. During the performance of this technique, steam is introduced into the containment building to raise the relative humidity above 90% in order to reduce the evaporation rate of reactor pool water, which affects the accuracy of the test. Although steam is secured prior to starting the measurements, the relative humidity in the containment building remains greater than 80% for the duration of the test, which totals approximately 10 hours. In addition, the 1.0 psig air pressure is maintained for 6-8 hours during the test. Therefore, all electric motors, relays, switches, meters, radiation detectors, and other electronic instrumentation and control equipment, as well as computers that are left in the containment

building, must withstand the high humidity and pressure condition for several hours. The high humidity level creates a sticky film on surfaces including electrical contact faces. The high-pressure condition forces moisture into detector and motor enclosures, and it may take several hours or days for the humidity level inside those enclosures to return to normal levels. There are a few select nuclear instrumentation drawers that are removed prior to conducting the test due to their sensitivity to higher-than-normal humidity; however, it would be impractical and impossible to remove all of the electrical equipment that could be affected during the test from the containment building. MURR suspects that the high humidity and pressure condition may have contributed to equipment failures that occurred after the leakage rate test in years 2006, 2013, 2018, and 2019. Most of these failures eventually caused unscheduled shutdowns of the reactor.

From 1978 through 2006, the average reactor containment building leakage rate was 8.1 standard cubic feet per minute (SCFM), and there was some variance through the years as MURR used different pressure and humidity instruments, as well as converting from a 1980s VAX computer program to using Microsoft Excel. Since 2007, the MURR process, software, and instrumentation used for measuring containment building leakage rate have been very consistent. Leakage rate values from 2007 to 2019 range from 9.5 to 12.2 SCFM, well below the Specification 5.5.c design requirement of 16.3 SCFM – see graph below. The red horizontal line at the top of the graph designates the TS leakage rate limit of 16.3 SCFM at a containment building pressure of 1.0 psig.



Containment Building Leak Rate Historical Data (in SCFM) – Calendar Years 2007 to 2019

In the post-measurement analysis, MURR management always looks for any upward trend in leakage rate compared to the last measurement, and the slope of the increase in rate is evaluated. The data is then extrapolated based on that slope to ensure the subsequent year's leakage rate will fall well below the design limit of Specification 5.5.c. If this LAR is approved, MURR would conduct the same analysis but extrapolate the leakage rate out two (2) years from the current measurement and make the same determination.

In addition to measuring reactor containment building leakage rate, MURR conducts preventive maintenance procedure BC1-Q1, "Containment Doors Gasket Inspection," on a quarterly basis to help identify any potential sealing surface defects prior to the sealing surface degrading to a point of significantly affecting containment building leakage rate.

3.0 Justification for Changing Technical Specification 4.4.a

The objective of surveillance TS 4.4, "Reactor Containment Building," currently states:

> "The objective of this specification is to reasonably assure proper operation of the containment system."

Additionally, the objective of TS 3.4, "Reactor Containment Building," currently states:

> "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."

Furthermore, the basis of Specification 4.4.a currently states:

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

Historic reactor containment building leakage rate data indicates that if only biennial measurements were made, there is a reasonable assurance that the leakage rate would remain within the design limits of Specification 5.5.c, specifically the leakage rate would remain below 16.3 SCFM at STP with an overpressure of 1.0 psig.

Moreover, American National Standard ANSI/ANS-15.1-2007 (R2013), "The Development of Technical Specifications for Research Reactors," Section 4.4.1(2), establishes the following surveillance frequency for "Containment:"

> "(2) Integrated leak rate test: Annually to biennially;"

Changing Specification 4.4.a to a biennial requirement will reduce the cyclic stress occurring to the reactor containment building which should reduce the potential for cracking of the containment structure. In

addition, changing to a biennial requirement will reduce humidity and pressure cycles on electrical equipment which should reduce electrical component failures and unscheduled shutdowns in the future.

4.0 Conclusion

Measuring the reactor containment building leakage rate biennially is sufficient to reasonably assure proper operation of the containment system, thereby assuring that reactor containment integrity is maintained when required so that the health and safety of the general public is assured. Therefore, MURR requests approval for the below revised Specification 4.4.a.

5.0 Proposed Revision to Technical Specification 4.4.a

Specification 4.4.a currently states:

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

Specification 4.4.a will be revised as follows:

"a. The reactor containment building leakage rate shall be measured biennially. The test shall be performed by the make-up flow, pressure decay, or reference volume techniques. No repairs or modifications shall be performed just prior to the test."

The basis for Specification 4.4.a currently states:

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

The basis for Specification 4.4.a will be revised as follows:

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

Enclosure 4 – Basis for the Requested Change to Technical Specification 6.4

1.0 Introduction

The University of Missouri Research Reactor (MURR) is requesting a change to Technical Specification (TS) 6.4, "Procedures." This change will adjust the required periodicity of procedure reviews by the Reactor Manager and the Reactor Health Physics Manager, as specified by Specification 6.4.c, from annually to biennially, thus greatly reducing the administrative burden of reviewing well-developed, time-tested, and mature MURR TS-required procedures.

The following license amendment request (LAR) provides justification that reviewing procedures for normal operations of the reactor, the Emergency Plan implementing procedures, radiological control procedures, and procedures for the preparation for shipping and the shipping of byproduct material biennially is sufficient to assure the continued effectiveness of MURR TS-required procedures.

2.0 Background

Original Facility Operating License No. R-103, issued on October 11, 1966, contained administrative TS 9.1, which stated:

> "Written procedures shall be in effect for normal operations of the reactor, emergencies to the reactor or facility which could result in significant radioactive releases, and for radiological control. Copies of these procedures shall be available in the reactor control room and shall be reviewed and approved annually by the Reactor Supervisor."

Since original licensing, Amendment Nos. 3, 4, and 30, as well as the relicensing of MURR on January 4, 2017, have changed the wording and numbering of this Specification; however, the essence of an annual review of TS-required procedures by the responsible manager has remained in the TSs.

MURR staff take these procedure reviews seriously and exert much effort into conducting thorough annual reviews. However, the annual review is rarely the initiator of a substantive change to the procedures. Changes initiated from annual reviews are normally grammatical, editorial, and/or formatting changes that do not change the technical steps in how a procedure guides the operator or technician to complete a task.

Substantive changes to the procedures are normally initiated by modifications to the facility and equipment, component replacements, or facility corrective actions. MURR administrative procedures AP-RR-015, "Work Control Procedure," and AP-RO-115, "Modification Records," direct the organization to ensure maintenance, operating, and other procedures are updated prior to completing and closing a work package or modification record. Furthermore, the MURR Corrective Action Program (CAP) Review Committee, per administrative procedure AP-RR-001, "Corrective Action Program," ensures that procedures are revised prior to closing a CAP report where the corrective action depends on a procedure revision.

3.0 Justification for Changing Technical Specification 6.4.c

American National Standard ANSI/ANS-15.1-2007 (R2013), "The Development of Technical Specifications for Research Reactors," Section 6.4, "Procedures," does not contain any requirement for periodic reviews of procedures.

ANSI/ANS-15.1-2007 (R2013), Section 6.2.4, "Audit Function," contains the following criteria:

> "(4) the reactor facility emergency plan and implementing procedures: at least once every other calendar year (interval between audits not to exceed 30 months)."

Additionally, ANSI/ANS-15.1-2007 (R2013), Section 6.2.3, "Review function," states:

> "The following items shall be reviewed:
>
> (1) determinations that proposed changes in equipment, systems, test, experiments, or procedures are allowed without prior authorization by the responsible authority, for example, 10 CFR 50.59 or 10 CFR 830;
>
> (2) all new procedures and major revisions thereto having safety significance, proposed changes in reactor facility equipment, or systems having safety significance;..."

A cursory review of the 23 other NRC-licensed research reactors' TS revealed only one (1) facility where a review of TS-required procedures was required annually. Otherwise, most other NRC-licensed research reactors have wording in Section 6.4 of their TSs which closely resembles Section 6.4 of ANSI/ANS-15.1-2007 (R2013), which has no requirement to periodically review TS-required procedures.

Although this proposed TS change will revise the required periodicity of reviewing the Emergency Plan implementing procedures, the NRC-approved MURR Emergency Plan requires that a more restrictive, annual review of the plan and its implementing procedures be conducted, as does the NRC-approved MURR Physical Security Plan and its implementing procedures. Therefore, MURR will continue to conduct annual reviews of emergency preparedness and security procedures even though the revised Specification 6.4.c indicates only a biennial review requirement for Emergency Plan implementing procedures.

4.0 Conclusion

Most MURR TS-required procedures have been in place for many years. Due to the maturity of MURR procedures, an annual review is an unnecessary burden on MURR staff. As stated above, the annual review does not normally reveal any technical weaknesses in the procedures. Reviewing TS-required procedures biennially is sufficient to assure the continued effectiveness of MURR procedures to safely operate the reactor, work with radioactive materials, and prepare radioactive shipments and ship byproduct material. Therefore, MURR requests approval for the below revised Specification 6.4.c.

5.0 Proposed Revision to Technical Specification 6.4.c

Specification 6.4.c currently states:

> "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."

Specification 6.4.c will be revised as follows:

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

No changes to the bases of TS 6.4 are required or requested.

1 **DEFINITIONS** - Continued

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

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

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

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

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

OR

b. Whenever all of the following conditions are met:

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

(2) One of the two following conditions exists:

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

OR

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

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

(4) No work is in progress involving the shim blades (rods) or installed shim rod drive mechanisms unless the shim rod drive mechanisms are physically decoupled from the shim blades (rods), 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.

3.4 Reactor Containment Building

Applicability:
This specification applies to the reactor containment building.

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

Specification:

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

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

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

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

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

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

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

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

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

(1) The reactor is secured,

AND

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

AND

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

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 biennially ~~annually~~, plus or minus four (4) months. The test shall be performed by the make-up flow, pressure decay, or reference volume techniques. No repairs or modifications shall be performed just prior to the test.

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

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

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

Bases:

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

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

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

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

6.4 **Procedures** - Continued

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

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

6.5 **Experiment Review and Approval**

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

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

(2) Substantive changes to previously approved experiments shall be made only after review by the RAC and approved in writing by the Reactor Manager.

6.6 **Reportable Events and Required Actions**

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

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

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

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

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

1 **DEFINITIONS** - Continued

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

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

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

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

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

OR

b. Whenever all of the following conditions are met:

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

(2) One of the two following conditions exists:

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

OR

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

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

(4) No work is in progress involving the shim blades (rods) or installed shim rod drive mechanisms unless the shim rod drive mechanisms are physically decoupled from the shim blades (rods), 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.

3.4 Reactor Containment Building

Applicability:
This specification applies to the reactor containment building.

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

Specification:

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

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

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

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

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

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

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

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

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

(1) The reactor is secured,

AND

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

AND

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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 biennially, plus or minus four (4) months. The test shall be performed by the make-up flow, pressure decay, or reference volume techniques. No repairs or modifications shall be performed just prior to the test.

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

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

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

Bases:

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

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

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

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

6.4 **Procedures** - Continued

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

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

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

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

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

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

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

4.2.2 Control Blades

4.2.2.1 Description

The reactivity of the reactor is controlled by five neutron-absorbing control blades, which are all located external to the pressure vessels. Each control blade is coupled to a control blade drive mechanism by means of a support and guide extension (offset mechanism). This offset mechanism is shown in Figure 4.5. Four of the control blades, referred to as the shim blades, are used for coarse adjustments to the neutron density within the reactor core. The fifth control blade is a regulating blade. The low reactivity worth of this blade allows for very fine adjustments in the neutron density in order to maintain the reactor at the desired power level.

The shim blades are constructed of formed boral plate which is, by weight, nominally 50% boron carbide and 50% aluminum. The boron carbide-aluminum mixture is clad with 0.0375 inches (0.9525 mm) of aluminum-alloy 1100 for a nominal blade thickness of 0.175 inches (4.445 mm). All four sides of the blades have a 0.25-inch (6.35-mm) aluminum frame where the blade thickness is 0.25 inches (6.35 mm). The active length of the neutron absorbing material is 34 inches (86.35 cm), and the overall blade length is approximately 40 inches (101.6 cm). The upper 6 inches (15.25 cm) of the shim blade is a 0.25-inch (6.35-mm) thick aluminum mounting plate that is curved to the shape of the blade. Each shim blade occupies approximately 72° of a circular arc around the pressure vessel.

The regulating blade is constructed of stainless steel. The blade has an overall length of approximately 30 inches (76.2 cm) and occupies approximately 18° of the circular arc.

The shim blades are positioned by four control blade drive mechanisms mounted on the upper bridge over the reactor pool surface. Each control blade drive mechanism consists of a 0.02-HP, 115-volt, one-amp, single-phase, 60-cycle motor connected to a lead screw assembly through a reduction gear box. The lead screw assembly converts the rotating motion of the drive motor to the linear motion of the control blades. A ball nut coupled directly to a drive tube is driven inward and outward by the lead screw. Connected to the bottom end of the drive tube is an electromagnet which engages a cadmium-plated carbon steel anvil attached above the water level to the end of a lift-rod assembly. The lift-rod assembly allows the positioning of a shim blade through a support and guiding mechanism (offset mechanism) mounted on a pedestal attached to the reflector tank. Limit switches mounted on each control blade drive mechanism stop the drive motor at the top and bottom of travel. These limit switches also provide indication on the reactor control console that the drive mechanism is either in the full-in or full-out position.

The shim blade can be withdrawn when the electromagnet is energized. When the reactor is scrammed, the electromagnet is de-energized, releasing the anvil, and allowing the shim blade and lift-rod assembly to drop. Loss of electrical power to the control blade system results in a reactor scram and safe shut down of the reactor. A dash pot assembly cushions the fall of the shim blade during the final 20% of travel.

The regulating blade is positioned by a control blade drive mechanism similar to the ones used for the shim blades with a few exceptions. A scram signal does not insert the regulating blade; therefore, an electromagnet and an anvil are not required. The lift-rod assembly is pinned directly to the drive tube.

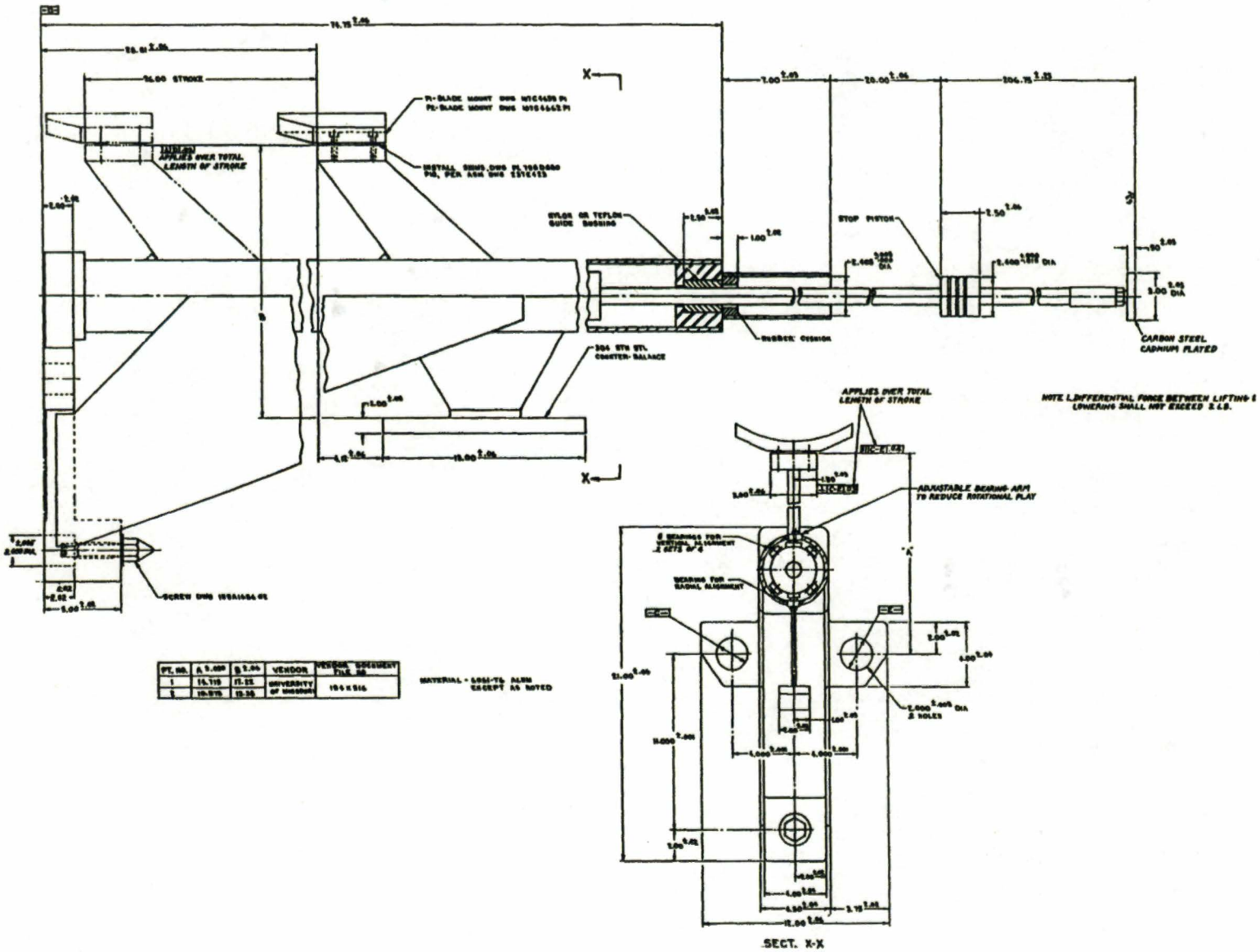

FIGURE 4.5
OFFSET MECHANISM

Also, the regulating blade is driven at a faster speed than the shim blades, requiring a different reduction gear box.

Position indication for the shim blades and the regulating blade is provided by an encoder transducer mounted on each control blade drive mechanism. Analog signals from the shaft encoder are converted into digital form by the Rod Position Indication (RPI) chassis mounted into the control room instrument panel. Digital position indication is displayed on the RPI chassis and an Operator Display Assembly (ODA) installed in the reactor control console. The RPI system is described in greater detail in Section 7.5.6, Control Rod Position Indication.

9.1 Heating, Ventilation, and Air-Conditioning Systems

9.1.1 Introduction

The Missouri University Research Reactor (MURR) heating, ventilation, and air-conditioning systems provide heated and conditioned air for the laboratory and reactor containment buildings, creating an acceptable working atmosphere for personnel and equipment. In addition, the facility ventilation system (Figure 9.1) is designed[1] to perform the following radiological control functions:

- Maintain the reactor containment and laboratory buildings at a slightly negative pressure with respect to the surrounding environment to prevent the spread of radioactive contamination;
- Provide the necessary air exchanges to ensure that concentrations of radioactive gases in the laboratory and reactor containment buildings are maintained at levels which are below the limits of 10 CFR 20, Appendix B, Table 1 for restricted areas;
- Ensure that maximum dilution of potentially contaminated air is attained, resulting in minimum concentrations of radioactive gases being released to the environment; and
- Continuously monitor all radioactive gases discharged through the facility ventilation exhaust stack.

The ventilation system also prevents the uncontrolled release of radioactive materials to the environment in the event of an accident. This function is discussed in greater detail in Chapter 6, Engineered Safety Features.

9.1.2 Description of Operation

9.1.2.1 Supply System

Fresh air is supplied to both the laboratory and reactor containment buildings through louver dampers on the north and south facades of the reactor building east tower. Most of the supply air entering the east tower is diverted to the laboratory building; a smaller portion is for make-up air to the containment building. Figure 9.2 shows the air flow paths through the reactor containment building east tower.

Outside air is preheated, as necessary, by a steam coil and then filtered before entering the laboratory building via the laboratory building supply fan (SF-1). The supply air is then directed to two receiving plenums (hot and cold decks), each containing an independent coil system. Steam is supplied to the hot deck to heat the supply air; chill water from the air-conditioning system is supplied to the cold deck to cool the air. The heated and conditioned air from these plenums is then circulated throughout the laboratory building via a double duct air distribution system. A ceiling register and mixing box in each laboratory and office allows the hot and cold air from the distribution system to be combined and circulated within that space to create a suitable environment for personnel comfort and equipment cooling.

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[1] The original safety evaluation of the MURR is documented in the Preliminary Hazards Report (Ref. 9.1), the Hazards Summary Report (Ref. 9.2), and Hazards Summary Report, Addenda 1-5 (Ref. 9.3-9.7).

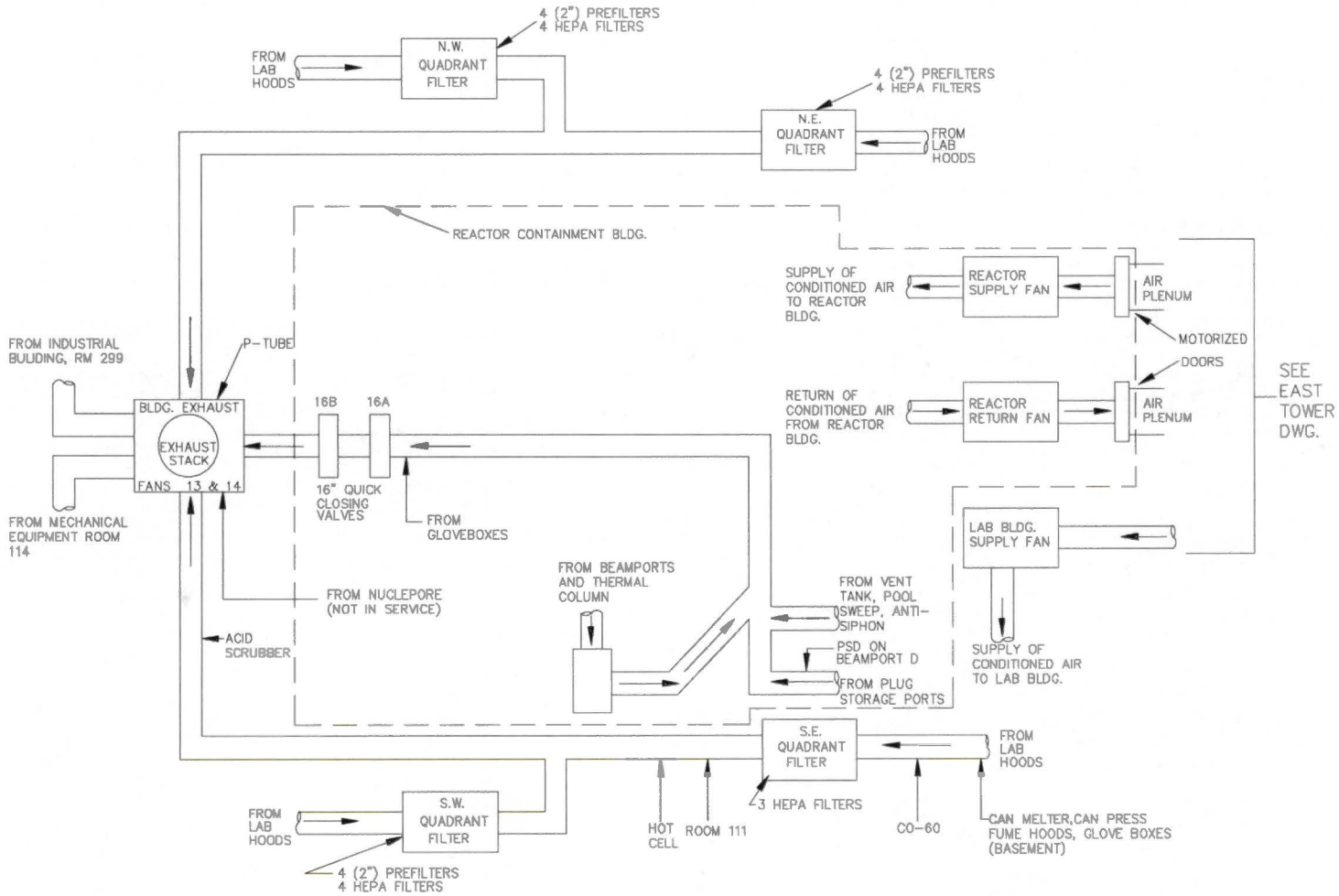

FIGURE 9.1
MURR FACILITY VENTILATION SYSTEM

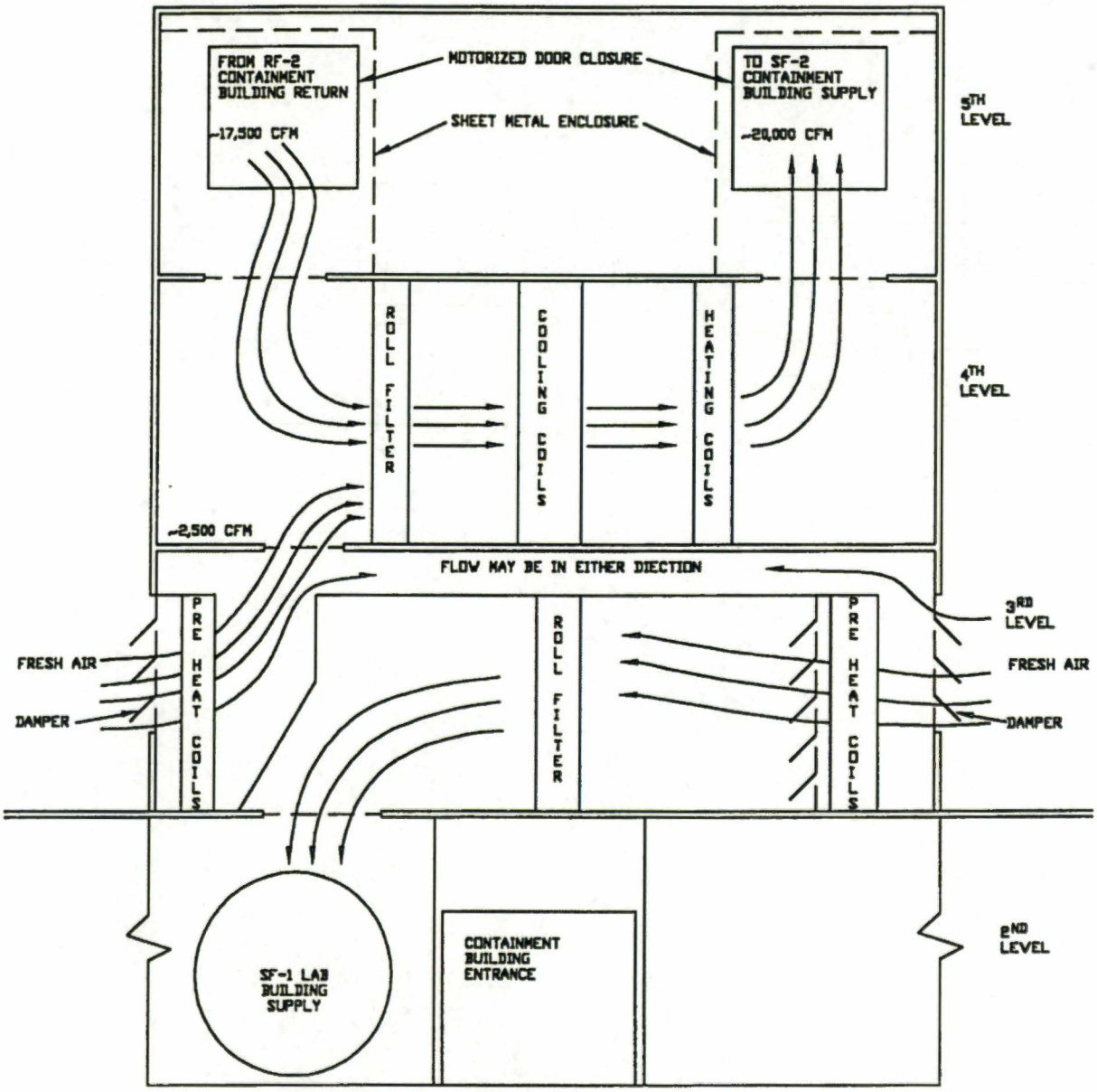

FIGURE 9.2
MURR CONTAINMENT AIR FLOW PATHS

Additional fresh air is supplied to the laboratory building through two roof top air handlers (RTAHs) located on the laboratory roof above the north and south outer corridors. Each RTAH has a heating and a cooling coil to heat and/or condition the supply air before its discharge into the laboratory building through registers in the ceilings of these corridors.

A third RTAH services an addition attached to the southwest corner of the laboratory building which houses the 275-kW emergency diesel generator. Fresh heated and/or conditioned air is supplied to this space through the RTAH and subsequently removed by a roof ventilator.

Ventilation for the reactor containment building is primarily recirculation with only a small amount of air being exhausted through a 16-inch line. Air from the containment building is returned to a plenum in the east tower at a rate of approximately 17,500 cubic feet per minute (cfm) by the reactor containment building return fan (RF-2). Approximately 2,500 cfm of fresh make-up air is mixed with the return air before passing through a dust filter and a set of heating and cooling coils. The mixed air then collects in a supply plenum before entering the containment building via the reactor containment building supply fan (SF-2) at a rate of approximately 20,000 cfm. The supply and return fans are interlocked such that securing SF-2 will automatically secure RF-2, thus preventing an extreme negative pressure condition from occurring within the containment building. However, operation of SF-2 with RF-2 secured is allowed.

The supply and return plenums for the reactor containment building each contain two isolation doors. One door is air-operated-to-open, gravity-to-close, and is designated as a "back-up door." The other door is driven open and closed by a 3-phase motor connected to a chain drive assembly. The motorized door for the supply plenum is designated as Door 504 and the motorized door for the return plenum is designated as Door 505. Closure of either door will secure both RF-2 and SF-2, preventing operation of the fans with no flow path. The isolation doors are discussed in greater detail in Chapter 6, Engineered Safety Features.

9.1.2.2 Exhaust System

Exhaust air from both the laboratory and reactor containment buildings is combined in an exhaust plenum prior to being discharged to the atmosphere. Since air from both buildings is never mixed until this point, potentially contaminated air is diluted by mixing with uncontaminated air, resulting in minimum concentrations of radioactive gases being released to the environment.

The exhaust system for the laboratory building is divided into four quadrants, each servicing approximately one quarter of the building. Each quadrant consists of a stainless steel filter housing containing a bank of pre-filters and a bank of high efficiency particulate air (HEPA) filters. Air is ducted from the quadrants to an exhaust plenum located in the reactor building west tower. Laboratory building exhaust fans (EF-13 and EF-14), located within the exhaust plenum, discharge the air through the facility exhaust stack to the atmosphere. The top of the exhaust stack is approximately 70 feet (21 m) above grade level of the containment building. One exhaust fan is in operation while the other is in standby. This condition is indicated by a green light on the fan failure alarm panels located in the reactor control room and the facility lobby (Room 202). Any condition other than one fan in "fast speed" and the other in "stand-by" will de-energize the green light and indicate an abnormal condition. Malfunction of the operating fan will automatically start the standby fan and de-energize the green light indicating a loss of the operating fan.

Failure of both fans, or significantly degraded flow, actuates a pressure switch which initiates an audible alarm in the reactor control room.

Exhaust air from the mechanical equipment room (Room 114), MURR Industrial Building (Room 299), hot cell, and the pneumatic tube system passes through separate filtration systems before discharging into the exhaust plenum.

Reactor containment building exhaust air discharges to the exhaust plenum at approximately 2,500 cfm through a 16-inch line which penetrates the west containment building wall. This line contains two quick-closing isolation valves designated 16A and 16B. Exhaust air from areas which produce radioactive gases or airborne contamination is ducted to the 16-inch line. These areas include the reactor pool sweep system, the beamport storage ports (hot storage ports), the thermal column, and the beamport vestibules. The hot exhaust line isolation valves (16A and 16B) are discussed in greater detail in Chapter 6, Engineered Safety Features.

9.1.3 Surveillance

The reactor control room Fan Failure Alarm Panel is periodically tested for operability.

6.2 Containment System

6.2.1 Introduction

The containment system is designed to completely isolate the reactor containment building, thereby preventing or mitigating any uncontrolled release of radioactive materials to the environment during an accident.[1] Redundancy is incorporated into the system to ensure that no single component or circuit failure will render any portion of the containment system inoperative. Isolation of the reactor containment building can be automatically initiated by radiation detectors located at the reactor pool upper bridge and in the containment building exhaust plenum. Isolation can be manually activated by switches in the reactor control room or the facility lobby (Room 202). Actuation of the Room 202 switch will also cause a facility evacuation.

The instrumentation necessary for the containment system to perform its intended function is described in greater detail in Section 7.8, Engineered Safety Features Actuation Systems.

6.2.2 Reactor Containment Building

6.2.2.1 Description

The reactor containment building is a five-level poured concrete building with 12-inch (0.3-m) thick reinforced exterior walls configured to form the shape of a cube, with each side being approximately 60 feet (18.3 m) long. All horizontal and vertical seams created in the pouring are sealed with a polyvinylchloride (PVC) plastic waterstop of the dumbell type. The waterstops were manufactured to meet the Corp. of Engineers Specification, CRD-C572.61. The outside surfaces of the concrete walls are finished with removable aluminum sheet siding 16 feet (4.9 m) long by 4 feet (1.2 m) wide. The siding is attached to angle standoffs on the walls. Centered on the exterior of each wall are two pilaster support columns closed at the back to form a tower. Located within each tower is support equipment necessary for the operation of the reactor facility. The pilaster support columns, comprising the side walls of the towers, are built into the main building structure to achieve the requisite structural strength. Below grade, within the containment structure, is a space (Room 101B) extending to the north that is 15 feet (4.6 m) high by 37 feet (11.3 m) deep by 40 feet (12.2 m) wide that contains various experimental equipment (Fig. 6.1). The walls and ceiling of this space are also of poured concrete with the pour joints sealed with the same type of PVC plastic waterstop.

6.2.2.2 Design Basis

The reinforced concrete walls of the reactor containment building have been designed to withstand a peak internal pressure of 2.0 psig (13.8 kPa above atmosphere). This design pressure was not predicated on any one accident scenario, rather the pressure was selected to envelope all postulated accident scenarios.

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[1] The original safety evaluation of the MURR is documented in the Preliminary Hazards Report (Ref. 6.1), the Hazards Summary Report (Ref. 6.2), and Hazards Summary Report, Addenda 1-5 (Ref. 6.3-6.7).

It is difficult to envision a pressure buildup within the containment structure to 2.0 psig (13.8 kPa above atmosphere) because the maximum design temperature of the bulk water in the reactor is only 160 °F (71 °C) (Ref. 6.8). In the event of a rupture in the primary coolant system piping, the coolant in the reactor core region would remain subcooled relative to atmospheric pressure and the water would not flash to steam as in the case of a highly pressurized water reactor. In addition, the one foot (0.3 m) diameter column of water immediately above the one foot (0.3 m) diameter reactor pressure vessel is calculated to absorb 108 MW-sec of energy before reaching 212 °F (100 °C).[1] Therefore, any large, positive reactivity insertion leading to an uncontrolled power excursion which results in the formation of steam in the reactor pressure vessel and a rupture in the primary coolant system within the reactor pool volume would be quickly quenched by the pool water.

The energy release necessary to cause a 2.0-psig (13.8-kPa above atmosphere) equilibrium loading within the reactor containment building has been calculated.[1] The calculation includes the following assumptions:

1. No heat sinks;
2. Initial containment building air temperature: 75 °F (24 °C) with 50% relative humidity;
3. Initial pool water temperature: 100 °F (38 °C);
4. Containment building volume: 240,000 ft^3 (6,796 m^3); and
5. Final condition of 100% saturated air in the containment building.

The amount of steam that must be released from the pool to cause a 2.0-psig (13.8-kPa above atmosphere) increase within the containment structure corresponds to 1,040 MW-sec of energy release. Under the above conditions, the final equilibrium temperature within the containment building would be 113 °F (45 °C). This amount of energy release is considerably greater than would be expected as a result of any accident to this type of reactor.[1]

However, the possibility that a severe fuel meltdown causing a molten aluminum water (Al-H_2O) reaction which releases a considerable amount of energy does exist. The specific conditions under which such a reaction might occur are widely discussed in literature pertaining to reactor hazards.[1] It is currently believed that the occurrence of an Al-H_2O reaction requires first, that the aluminum be present in a finely divided form such as droplets or particles of less than 500 microns, and second, that the temperature of the aluminum be greater than its melting point.

The amount of aluminum in the fuel plates of eight fuel assemblies (complete core loading) is 29.4 kg, more than 75% of which is cladding. If this quantity of aluminum were to react completely with water, it would produce 441 MW-sec of energy and release approximately 1,340 ft^3 (38 m^3) of hydrogen at standard conditions. If the hydrogen were to completely recombine, an additional 523 MW-sec of energy would be released. This total release of 964 MW-sec is less than the 1,040 MW-sec of corresponding energy in steam which would have to be released to cause a 2.0-psig (13.8-kPa above atmosphere) equilibrium overpressure in the reactor containment building.

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[1] See Footnote on Page 1.

To postulate that the total reaction of the aluminum and the recombination of the evolved hydrogen would occur is an excessively conservative and unrealistic assumption. A more realistic, but still exceedingly conservative, approach would be to assume that only 1.3% of the aluminum would react. This assumption would be consistent with the MHA which postulates the melting of four number-1 fuel plates in the reactor core. The four number-1 fuel plates contain 78.58 grams of uranium-235 (235U). Considering the total reactor core inventory of 6.2 kg of 235U, during the MHA, approximately 1.3% of the core would melt. Therefore, the amount of aluminum which would react with the coolant would be 0.3822 kg (1.3% of 29.4 kg). This would equate to a total energy release of only 5.7 MW-sec.

In conclusion, the reactor containment building design pressure of 2.0 psig (13.8 kPa above atmosphere), based on the previous calculations, is completely adequate for all postulated accident scenarios. The design of the containment engineered safety feature gives reasonable assurance that it will not interfere with reactor operation or shutdown.

6.2.3 Penetrations and Closures

Penetrations through the reactor containment building concrete structure are limited in order to minimize potential leakage points. Existing penetrations include a utility entry water seal, a pedestrian entry, an entry for heavy equipment, supply and exhaust air ducts, a hot exhaust line, and two steel penetration plates which allow electrical lines and the pneumatic tube system to enter and exit the containment building.

Three lines used for the reactor containment building leakage rate test also enter the containment structure. Two of these lines enter through one of the penetration plates while the third line penetrates the containment building wall.

The pedestrian and heavy equipment entries and the supply and exhaust air ducts include sealable closures which ensure that sufficient integrity can be maintained to prevent the accidental release of radioactivity to the environment.

Table 6-1 lists all of the penetrations through the reactor containment building concrete structure and their corresponding sealing methods.

6.2.3.1 Utility Entry Water Seal

The utility entry water seal (seal trench) is a water-filled trap which provides 2.0-psig (13.8-kPa above atmosphere) overpressure relief protection for the reactor containment building. The seal trench also provides a path for site utilities to enter and exit the containment building. The seal trench forms a water seal between the reactor containment building and the adjacent laboratory structure. In the event that an overpressure condition within the containment building exceeds 2.0 psig (13.8 kPa above atmosphere), the water is elevated out of the seal trench, overflowing onto the laboratory basement floor. This creates a path for pressure within the containment building to be released into the laboratory building. Once the pressure is relieved, the water level will drop and reseal the reactor containment building. The water in the seal trench also serves as an absorbent for radioactive particulate or gas

Water from the domestic cold water system is manually added to the seal trench as required to make up for losses due to evaporation. Level is maintained at approximately 65 inches (1.7 m) for all operating conditions. The seal trench is also equipped with a float assembly that initiates a "Utility Trench Seal Water Lo Level" annunciator alarm on low water level.

TABLE 6-1
REACTOR CONTAINMENT BUILDING PENETRATIONS

Penetration	Sealing Method	No.
Utility Entry[a]	Water	1
Pedestrian Entry (Airlock)	Door and Inflatable Gasket	2
Heavy Equipment Entry	Door and Inflatable Gasket	1
Supply Air Duct[b]	Door and Inflatable Gasket	1
Exhaust Air Duct[b]	Door and Inflatable Gasket	1
Pneumatic Tube System	None Required[c]	N/A
Electrical Lines	Sealed Connectors	N/A
Leakage Rate Test Connections		
1. Instrument Line	Manual Valve	1
2. Pressurization Line	Threaded Plug and Sealant	1
3. Depressurization Line	Bolted Flange and Gasket	1
Hot Exhaust Line	Automatic Valve	2

[a] All site utilities which enter or exit the reactor containment building through the utility entry water seal are listed in Section 6.2.3.1.

[b] Redundancy for the ventilation plenums is provided by backup doors. Operation of the backup doors, including the sealing method, is described in Section 6.2.4.

[c] Entry for the pneumatic tube system is described in Section 6.2.3.6.

The following services or lines enter or exit the reactor containment building through the seal trench:

a. 6-inch fire protection water line;

b. 6-inch emergency pool fill line;

c. 1½-inch gas line (purged and blank flanged - no longer in use);

d. 2-inch compressed air line;

e. 2-inch domestic cold water line;

f. 2-inch domestic hot water line;

g. ¾-inch domestic hot water return line;

h. 1 ½-inch containment sump discharge;

i. 1 ½-inch sanitary sump discharge;

j. ½-inch demineralized water line;

k. d-inch cooling water discharge line (from the experimental facilities);

l. 1¼-inch vacuum line;

m. ¾-inch PVC alternate air supply line to the supply and exhaust

n. plenum backup doors from the emergency air system;

o. ¾-inch PVC (capped - no longer in use); and

p. ¾-inch PVC (capped - no longer in use).

6.2.3.2 Pedestrian Entry

The pedestrian entry for the reactor containment building (Fig. 6.2) consists of two electric-motor-driven horizontal sliding doors and an intervening vestibule (airlock). The outer door, designated as Door 276, and the inner door, designated as Door 277, allow the entrance of personnel from the laboratory building main corridor, through the airlock, to the second level of the containment building. The airlock is a portion of the containment system.

The doors are constructed of steel and designed to withstand a 2.0-psig (13.8-kPa above atmosphere) overpressure. Each door is suspended from an overhead rail by two adjustable one-ton trolleys. A 3-phase motor connected through a gear reducer to a chain drive assembly drives the door open and closed. When a door is in the fully closed position, a rotary limit switch actuates an air supply valve, inflating a gasket mounted in the door facing, sealing the door. A manually-operated throw-out clutch disengages the motor and gear reducer from the chain drive assembly allowing manual operation of the door at zero differential pressure.

The pedestrian entry control system is designed and interlocked such that one door is always closed and sealed, ensuring that containment integrity is maintained.

6.2.3.3 Heavy Equipment Entry

The heavy equipment entry consists of an electric-motor-driven horizontal sliding door which allows entrance from the laboratory building basement to the reactor containment building beamport floor. This entry, serviced by the laboratory building 15-ton (13,608-Kg) freight elevator, provides an access for heavy and large equipment to be moved into and out of the containment building at below-grade level (Fig. 6.1). The closure for this entry is designated as the truck entry door (Door 101).

Door 101 is constructed of steel and suspended from an overhead rail by two adjustable one-ton (907-Kg) trolleys. A 3-phase motor connected through a gear reducer to a chain drive assembly drives the door open and closed. When the door is in the fully-closed position, a rotary limit switch actuates an air supply valve, inflating a gasket mounted in the door facing, thus sealing the door. A track built into the beamport floor guides the door up against the inflatable gasket. A manually-operated throw-out clutch disengages the motor and gear reducer from the chain drive assembly allowing manual operation of the door at zero differential pressure. A pressure switch monitors gasket seal pressure and initiates a "Truck Entry Door Open Rod Run-In" annunciator alarm and rod run-in upon depressurization of the gasket. Door 101 is maintained in the closed and sealed position at all times during reactor operation.

6.2.3.4 Supply and Exhaust Air Ducts

The supply and exhaust air ducts contain two electric-motor-driven horizontal sliding doors located in the ventilation plenums on the fifth level of the reactor containment building (Fig. 6.3). When these doors are in the open position, a path is created for containment air to be circulated through the east tower and then back into the containment building. The openings in the containment building wall for each plenum are approximately 4 feet by 4 feet (1.2 m x 1.2 m). The motorized isolation door for the supply plenum is designated as Door 504 and the motorized isolation door for the exhaust plenum is designated as Door 505.

The doors are constructed of ¼-inch (0.64-cm) thick metal plate welded to a steel frame. Each door is suspended from a 12½-foot (3.8-m) long, 6-inch (15.2-cm) I-beam by two adjustable ½-ton (453-Kg) trolleys. To insure proper alignment, the doors travel in a guided slot mounted along the bottom of the door opening. A 3-phase, 440-volt, 1-HP motor connected through a gear reducer to a chain drive assembly drives the door open and closed. When a door is in the fully-closed position, a rotary limit switch energizes a solenoid-operated three-way valve, inflating a gasket mounted in the door facing, thus sealing the door. A manually-operated throw-out clutch disengages the motor and gear reducer from the chain drive assembly allowing manual operation of the door at zero differential pressure.

The supply and exhaust air doors are normally kept in the open position during reactor operation. Actuation of the reactor isolation or facility evacuation switches located in the reactor control room, or the facility evacuation switch located in the facility lobby (Room 202), will close Door 504 and Door 505. A radiation level greater than the set point of either of the reactor bridge radiation monitors, or either of the exhaust plenum radiation monitors, will also close both doors automatically. A closure signal from any of the above causes two parallel relays to de-energize, closing a contact associated with each relay, and completing the circuit to the closing coil of each motor. The motors will drive the isolation doors to the closed position, actuating their rotary limit switches and energizing the solenoid-operated three-way valves, thus sealing the doors. It takes approximately 7 seconds from the initiation of the isolation signal until the isolation doors are closed and sealed.

6.2.3.5 Electrical Entry

All electrical connections from equipment external to the reactor containment building, as well as electrical power supplies which must enter the containment building, pass through two steel penetration plates located in the containment structure walls. One penetration plate is located in the east wall, and the other is located in the south wall. Each penetration plate contains sealed connectors which allow electrical lines to enter the containment building with minimal air leakage. There are no through-the-containment-wall electrical conduits.

6.2.3.6 Pneumatic Tube System Entry

The pneumatic tube system enters the reactor containment building through a penetration plate located in the east containment structure wall. Eight (8) lines [four 1½-inch (3.8-cm) sample carrier tubes and four 1½-inch (3.8-cm) air-vacuum driver tubes] pass through the penetration plate into the containment building and converge adjacent to the biological shield. The four sample carrier tubes (two on the north side and two on the south side) enter the reactor pool through the biological shield and terminate within the graphite reflector elements surrounding the reactor core.

The pneumatic tube facility is a closed system that extends from the outer building laboratories (which contain the sending-receiving stations) into the containment building, terminating near the reactor core. Since there is no leakage path into the pneumatic tubes after penetrating the containment wall, the pneumatic tube system is considered an extension of the laboratories into the reactor containment building. The pneumatic tubes are designed to withstand 2.0-psig (13.8-kPa above atmosphere) overpressure with zero leakage.

6.2.3.7 Building Leak Rate Test Penetrations

The reactor containment building leak rate test uses three lines which penetrate the containment structure. Two lines consist of pipes welded to the penetration plate located in the containment building east wall. One line is used for pressurizing the containment building and the second line is used for attaching the instruments necessary for measuring internal pressure and leakage rate. A third line runs through a steel collar positioned in the concrete of the containment building east wall and allows for depressurizing the containment structure upon completion of the leak rate test. The line consists of a 4-inch pipe welded to the collar. This line is blank-flanged within the reactor containment building when not in use.

6.2.3.8 Hot Exhaust Line

The hot exhaust line is a 16-inch pipe which discharges potentially contaminated gases from the reactor containment building. Exhaust air from areas which produce radioactive gases or airborne contamination is ducted to this 16-inch line which penetrates the west wall of the containment building just below the ceiling level and discharges to the facility exhaust plenum located in the west tower. A 12-inch (0.3-m) long steel collar surrounding the 16-inch pipe is cast into the containment building wall. The 16-inch pipe is welded to this collar. Two quick-closing 16-inch isolation valves, designated valve 16A and valve 16B, are located on this line within the containment building.

Isolation valve 16A is an air-operated-to-open, spring-to-close, butterfly valve located approximately 12 feet (3.7 m) from the east containment building wall. Isolation valve 16B is an air-operated-to-open, air-operated-to-close, butterfly valve located adjacent to the west containment building wall. Air is supplied to the valve operators from the facility main air compressors and the emergency air compressor. An additional emergency air compressor and accumulator tank dedicated to valve 16B ensures that three possible air sources are available to operate this valve. Closure time of both valves is approximately 3 seconds.

The hot exhaust line isolation valves are normally kept open during reactor operation. Actuation of the reactor isolation or facility evacuation switches located in the reactor control room, or the facility evacuation switch located in the facility lobby (Room 202) will close valves 16A and 16B. Detection of a radiation level greater than the set point of either of the reactor bridge radiation monitors, or either of the exhaust plenum radiation monitors, will also close both valves automatically. Two solenoid-operated valves, installed in series, control the air supply to valve 16A. A closure signal will de-energize both solenoid valves, causing air to be vented from the actuator, and allowing the spring to close the valve. Actuation of either solenoid-operated valve will close valve 16A. The air supply to valve 16B is controlled by four solenoid operated valves. Two valves, installed in series, control the air supply to the "open side" of the actuator, and two valves, installed in parallel, control the air supply to the "close side" of the actuator. A closure signal will de-energize all four solenoid operated valves, venting the "open side" and allowing air to be supplied to the "close side" of the actuator. Should one of the parallel solenoid-operated valves fail to operate, flow control valves installed on each valve's vent port ensure that air is supplied to the "close side" of valve 16B's actuator at a faster rate than is vented from the failed solenoid valve.

6.2.4 Backup Doors

A second set of isolation doors, designated the backup doors, are located in the reactor containment building supply and exhaust plenums thereby providing redundancy for containment building isolation. When the backup doors are shut, the steel plenum chamber above the door becomes part of the containment system.

The backup doors are constructed of ¼-inch (0.64-cm) thick metal plate and reinforced to withstand 2.0-psig overpressure (13.8-kPa above atmosphere) (Fig. 6.4). Each door is held open against gravity by a double acting pneumatic cylinder. Air is supplied to the pneumatic cylinders from the facility main air compressors and the emergency air compressor. A 3/8-inch (0.95-cm) rubber gasket installed in the door facing creates a seal for the backup doors when in the closed position.

The backup doors are normally kept open during reactor operation. A radiation level greater than the set point of either of the reactor bridge radiation monitors, or either of the exhaust plenum radiation monitors, will close both isolation doors automatically. Two solenoid-operated valves, installed in series, control the air supply to each pneumatic cylinder. A closure signal will de-energize both solenoid valves causing air to be vented from the pneumatic cylinder, and allowing gravity to close the isolation door. Actuation of either of the solenoid-operated valves will close the backup door.

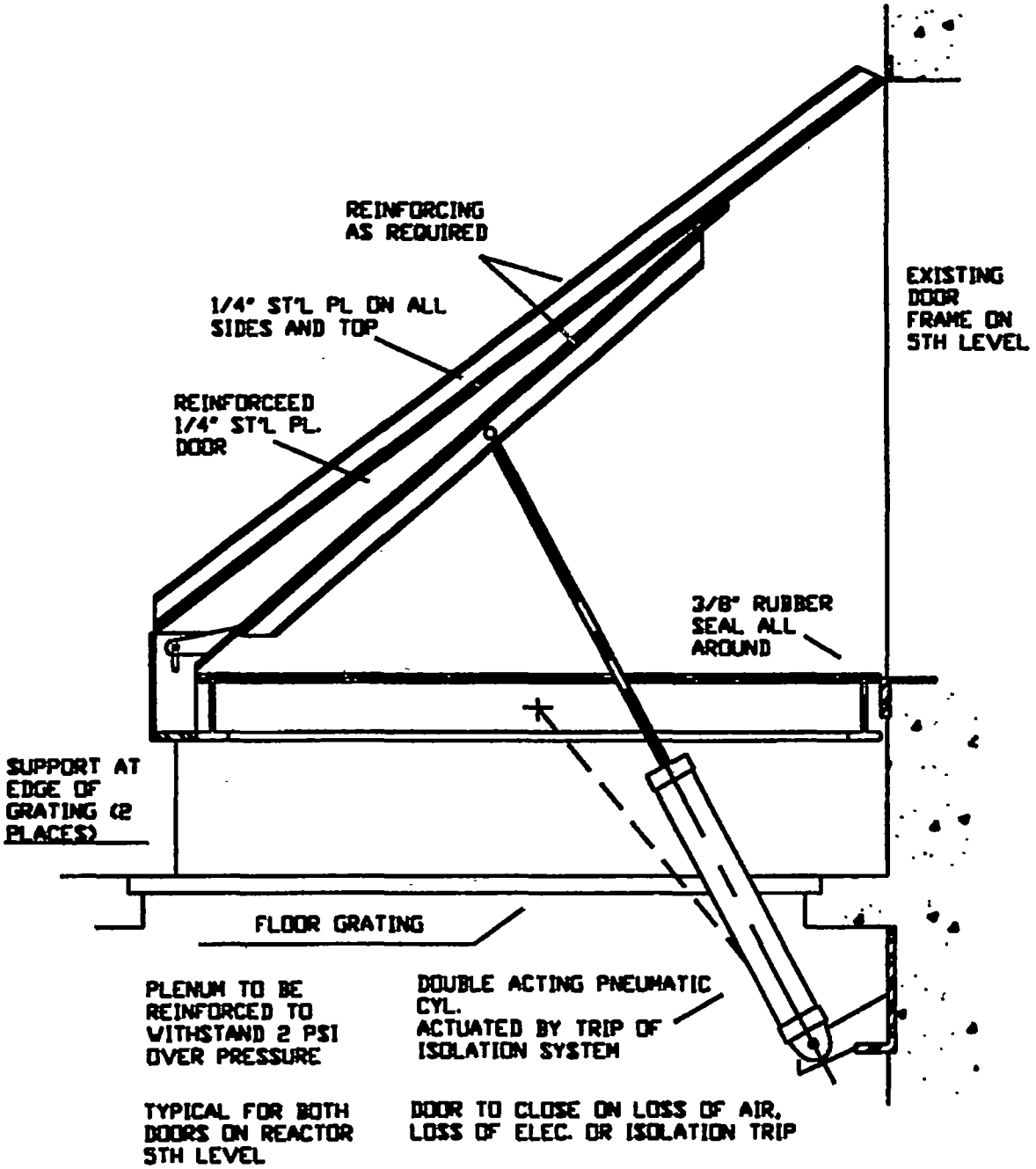

FIGURE 6.4
BACKUP DOOR

6.2.5 Compressed Air and Inflatable Gaskets

To ensure proper operation, all reactor containment building closures with inflatable gaskets are dependent upon a continuous supply of compressed air. Two main air compressors located in the mechanical area (Room 278) of the laboratory building provide the normal supply of compressed air to the inflatable gaskets. This supply of compressed air enters the containment structure through the seal trench. Should the main air system become inoperative, an alternate supply of compressed air is provided by an emergency air compressor located on the fifth level of the containment building (Fig. 6.3). If supply air pressure decreases to less than 70 psig (482.6 kPa above atmosphere), the emergency air system will assume the compressed air load and cause a check valve in the air supply line near the seal trench to shut, thus isolating the main air system external to the containment building. The check valve ensures that the emergency air compressor will only supply compressed air to selected loads within the containment building. These loads include:

a. Personnel Airlock Door 276 Sealing Gasket;
b. Personnel Airlock Door 277 Sealing Gasket;
c. Motorized Ventilation Isolation Door 504 Sealing Gasket;
d. Motorized Ventilation Isolation Door 505 Sealing Gasket;
e. Truck Entry Door (Door 101) Sealing Gasket;
f. Ventilation Exhaust Valve 16A Actuator;
g. Ventilation Exhaust Valve 16B Actuator; and
h. Backup Doors.

The emergency air compressor is powered by the emergency electrical power system to ensure that containment integrity is maintained during a loss of normal electrical power to the reactor facility.

6.2.6 Description of Operation

During normal operation of the reactor, the following conditions exist: one pedestrian entry door is closed and sealed, the truck entry door is closed and sealed, the supply and exhaust air doors are open, the backup doors are open, the hot exhaust line valves are open, the utility entry water seal is filled to a level of approximately 65 inches, and pressure within the reactor containment building is slightly negative with respect to the surrounding laboratory building.

In the event of detection of a high radiation level at the reactor pool bridge or in the exhaust plenum, the containment system will provide complete isolation of the reactor containment building. The reactor operator may also initiate a containment building isolation from the control console in the reactor control room or a facility evacuation and subsequent containment building isolation from the facility lobby (Room 202).

6.2.7 System Redundancy and Reliability

To ensure compliance with Criterion 54 of Appendix A of Title 10, Chapter I of the Code of Federal Regulations, Part 50 (10 CFR 50) and IEEE-279-1971, Single Failure Criterion, redundancy is incorporated into the containment system. This ensures that no single component or circuit failure will render the system

inoperable. System reliability is achieved through instrumentation and equipment which fail-safe when de-energized. Characteristics of the containment system which provide the redundancy and reliability are as follows:

(1) All relays which energize to perform their intended function are paralleled by a second relay such that, if one fails to energize, the unaffected relay will operate as required;

(2) All relays which de-energize to perform their intended function are assumed to be fail-safe;

(3) The solenoid-operated valves which control the compressed air supply to the hot exhaust line isolation valves (valve 16A and valve 16B) de-energize to shut the isolation valves, therefore, they are designed to be fail-safe;

(4) Hot exhaust line isolation valve 16A is spring-to-close which is designed fail-safe; isolation valve 16B is air-operated-to-close and supplied by three different compressed air sources; the isolation valves are installed in series to provide redundancy;

(5) Power to operate the supply and exhaust air doors (Door 504 and Door 505) is supplied by both the normal and emergency electrical power systems; therefore, the doors will still close in case of a simultaneous containment isolation and loss of normal electrical power; and

(6) The solenoid-operated valves which control the compressed air supply to the backup doors de-energize to shut the isolation doors, therefore, they are designed to be fail-safe; the backup doors provide redundancy for Door 504 and Door 505.

At least two major failures must occur before the containment system would be unable to perform its intended function. Containment building isolation is needed only in the event that fission products or other large sources of radioactivity have been released into the reactor containment building. This circumstance requires a major system failure; an event which itself is highly unlikely. A failure of the containment system must occur simultaneously with this event for the system to fail to perform its function.

6.2.8 Limiting Conditions for Operation

The limiting conditions for operation of the reactor containment building and their bases are discussed in the Technical Specifications. The objective of this specification is to reasonably ensure that the health and safety of the general public are not endangered as a result of reactor operation.

6.2.9 Surveillance

The reactor containment building is designed to hold a peak internal overpressure of 2.0 psig with a leakage rate over a 24-hour period not exceeding 10% of the contained volume or 16.3 ft^3/min (STP) with an overpressure of 1.0 psig. To ensure that these design limits are maintained, a containment building leakage rate test is performed as stated in the Technical Specifications.

The reactor isolation and facility evacuation system and its associated radiation monitors are tested for operability as stated in the Technical Specifications. The reactor containment building instrumentation and control systems, testing, surveillance provisions and intervals, and related Technical Specifications reasonably ensure that, if required, the containment engineered safety feature will be available and operable.

6.2.10 Design Features

The specifications and bases for the design features of the reactor containment building are discussed in the Technical Specifications. The objective of these specifications is to ensure adequate restriction in the event of an accidental release of radioactivity to the environment. The design and functional features of the reactor containment building ensure that exposures will be maintained below the limits of 10 CFR 20. Radiological consequences for postulated events are provided in Chapter 13, Accident Analyses.

Note: Figures 6.1, 6.2 and 6.3 are floor plans and are not included since they are not required to support this License Amendment Request.

7.8 Engineered Safety Features Actuation Systems

7.8.1 Introduction

The Engineered Safety Features Actuation Systems receive input signals from monitoring instruments and initiate the operation of the engineered safety features (ESFs). The ESFs are systems which are designed to mitigate the consequences of the Maximum Hypothetical Accident (MHA) and the Loss of Coolant Accident (LOCA) and to keep radiological exposures to the operating staff and the general public within the limits of 10 CFR 20. The Engineered Safety Features Actuation Systems' designs give reasonable assurance of reliable operation, if required. The two systems which are identified as ESFs for the MURR are the Containment System and the Anti-Siphon System. These systems are described in greater detail in Chapter 6, Engineered Safety Features.

The MHA and LOCA are discussed in detail in Chapter 13, Accident Analyses.

7.8.2 Containment Actuation (Reactor Isolation) System

The Containment Actuation System (CAS) is designed to completely isolate the reactor containment building, thereby preventing or mitigating any uncontrolled release of radioactive materials to the environment during an accident. Isolation of the reactor containment building can be automatically initiated by radiation detectors located at the reactor pool upper bridge and in the containment building exhaust plenum. Isolation can be manually actuated by switches in the reactor control room or the facility lobby (Room 202). The radiation monitors are described in detail in Section 7.9.

Manual or automatic actuation of the CAS causes the following actions to occur:

1. Reactor will scram;
2. All normally-open reactor containment building penetrations with automatic sealable closures will close;
3. An audible alarm will sound throughout the containment building; and
4. A flashing light at the entrance to the containment building personnel airlock will illuminate.

To ensure system reliability, the CAS is normally energized. This ensures that any system failure such as a relay coil "burnout" or a short in the system would place the CAS in a tripped condition. Redundancy is incorporated into the actuation system to ensure that no single component or circuit failure will render any portion of the system inoperative. Characteristics of the CAS which provide the redundancy and reliability are as follows:

(a) All relays which energize to perform their intended function are paralleled by a second relay such that if one fails to energize, the unaffected relay will operate as required;

(b) All relays which de-energize to perform their intended function are assumed to be fail-safe;

(c) Four (4) independent radiation detectors can initiate automatic isolation of the reactor containment building, two at each location (reactor pool upper bridge and containment building ventilation exhaust plenum);

(d) The radiation monitors are isolated from each other such that the operation of one does not affect the operation of another;

(e) The radiation monitors are powered by independent low voltage power supplies and are therefore not subject to a single failure event; and

(f) The emergency electrical power system provides a continuous supply of electrical power to the radiation monitors and the reactor isolation system in the event of a loss of normal electrical power.

At least two major failures must occur before the CAS would be unable to perform its intended function. This is not a credible event.

7.8.2.1 Automatic Initiation

Isolation of the reactor containment building can be automatically initiated by radiation detectors located at the reactor pool upper bridge and in the containment building ventilation exhaust plenum. A radiation level which is one decade above background at these locations will cause relays 2K1A and 2K1B to de-energize. Relay 2K1A is actuated by the containment building exhaust plenum No.1 and the reactor pool upper bridge ALARA radiation monitors. Relay 2K1B is actuated by the containment building exhaust plenum No. 2 and reactor pool upper bridge radiation monitors. De-energizing either of these relays will cause the following actions to occur:

1. A reactor scram is initiated by the "green leg" of the Reactor Safety System by a process input string to E3B of the NCLUs;

2. Parallel relays R2A and R2B and relay 2K2 in the CAS will de-energize;

3. Containment building ventilation exhaust plenum backup doors will close; and

4. "Bldg Air Plenum & Bridge Hi Activity Scram" annunciator alarm will be initiated.

De-energizing either relay R2A or R2B (see No. 2 above) of the CAS will cause the following actions to occur:

1. Containment building ventilation supply and exhaust duct electric-motor-driven isolation doors 504 and 505 will close;

2. Containment building hot exhaust line isolation valves 16A and 16B will close;

3. The reactor isolation horns will activate; and

4. The warning light at the entrance to the containment building personnel airlock will illuminate.

These radiation monitors feature a failure mode that occurs if they do not receive an input signal from the detector assembly for a specified time period (approximately one minute). This failure will de-energize relays 2K1A and 2K1B, thus preventing reactor operation with a failed radiation monitor.

7.8.2.2 Manual Initiation

Isolation of the reactor containment building can be manually initiated from either of two locations: the reactor control room or the facility lobby (Room 202). Manual initiation from the reactor control room is performed by the actuation of either of two switches located on the reactor control console: reactor isolation switch 1S15 and facility evacuation switch 1S16.

Actuation of the reactor isolation switch will cause the following actions to occur:

1. Relay 2K2 will de-energize and cause:
   a) A reactor scram from the "yellow leg" of the reactor safety system by a process input string to E4A of the NCLUs; and
   b) Initiation of the "Evacuation or Isolation Scram" annunciator alarm; and
2. All actions caused by relays R2A and R2B de-energizing.

Actuation of the facility evacuation switch will cause the following actions to occur:

1. Parallel relays R3A and R3B will de-energize and open contacts in the CAS thereby de-energizing relays 2K2, R2A, and R2B; and
2. Evacuation horns installed at strategic locations throughout the reactor facility laboratory building will activate.

A second facility evacuation switch is located at the reception desk in the facility lobby (Room 202).

Any postulated emergency necessitating the evacuation of all personnel from the reactor facility does not require instantaneous action, therefore, the actuation of the facility evacuation system is an administrative decision and no automatic trip function is required.

7.8.2.3 Surveillance

The CAS and its associated radiation monitors are tested for operability as stated in the Technical Specifications. The objective of this specification is to ensure proper operation of the system.

7.8.3 Anti-Siphon Actuation System

The Anti-Siphon Actuation System functions as a backup system to the various safety instrumentation and equipment (e.g., pressure sensors, pump and valve interlocks, etc.) which ensures that the reactor core does not become uncovered during a LOCA. The system is designed to admit a fixed volume of air to the high

point of the reactor outlet piping, or invert loop, instantaneously establishing the pressure in this area at equal to or greater than atmosphere. This prevents a siphon action from being created due to a rupture of the primary coolant piping.

The Anti-Siphon Actuation System is automatically actuated upon detection of primary coolant system low pressure. System pressure is monitored by two electronic pressure transmitters (PT 944A and PT 944B) located on the 12-inch primary coolant piping between the reactor pressure vessel and primary coolant isolation Valve 507A. Should primary coolant system pressure decrease below a predetermined value, PT 944A and PT 944B will de-energize relays 2K13 and 2K28, respectively, either of which will cause the following actions to occur:

1. Reactor will scram - 2K13 will open a contact in the process input string to E4A and 2K28 will open a contact in the process string to E3B of the Reactor Safety System NCLUs;
2. Primary Coolant Circulation Pumps 501A and 501B will stop;
3. Primary Coolant Isolation Valves 507A and 507B will close; and
4. Anti-Siphon System Isolation Valves 543A and 543B will open.

Redundancy is incorporated into the Anti-Siphon Actuation System to ensure that no single component or circuit failure will render any portion of the system inoperative. System reliability is achieved through instrumentation and equipment which fail-safe when de-energized. Characteristics of this system which provide the redundancy and reliability are:

(a) All relays which de-energize to perform their intended function are assumed to be fail-safe;

(b) Redundant primary coolant system pressure monitors PT 944A and PT 944B, either of which will initiate operation of this system are provided; and

(c) Redundant relays 2K13 and 2K28, either of which will initiate operation of this system are provided.

At least two major failures must occur before the Anti-Siphon Actuation System would be unable to perform its intended function. This is not a credible assumption.

7.8.3.1 Surveillance

The actuation of this system is periodically tested for operability as stated in the Technical Specifications. The objective of this specification is to ensure proper operation of this system.