UNIVERSITY OF MISSOURI – COLUMBIA

RESEARCH REACTOR

LICENSE No. R-103

DOCKET No. 50-186

RESPONSES TO NRC REQUESTS FOR ADDITIONAL INFORMATION RELATING TO LICENSE RENEWAL

REDACTED VERSION*

SECURITY-RELATED INFORMATION REMOVED

***REDACTED TEXT AND FIGURES ARE BLACKED OUT OR DENOTED BY BRACKETS**

UNIVERSITY *of* MISSOURI

RESEARCH REACTOR CENTER

July 16, 2010

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

Reference: Docket 50-186
University of Missouri-Columbia Research Reactor
Amended Facility License R-103

Enclosed you will find the University of Missouri-Columbia Research Reactor's responses to the U.S. Nuclear Regulatory Commission's (NRC) request for additional information, dated June 1, 2010, regarding our renewal request for Amended Facility Operating License R-103, which was submitted to the NRC on August 31, 2006, as supplemented.

If you have any questions, please contact Leslie P. Foyto, the facility Reactor Manager, at (573) 882-5276 or foytol@missouri.edu.

Sincerely,

Ralph A. Butler, P.E.
Director

RAB/djr

Enclosures

A020
NRR

1513 Research Park Drive Columbia, MO 65211 Phone: 573-882-4211 Fax: 573-882-6360 Web: http://web.missouri.edu/~murrwww

Fighting Cancer with Tomorrow's Technology

UNIVERSITY *of* MISSOURI

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Attention: Document Control Desk
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REFERENCE: Docket 50-186
University of Missouri – Columbia Research Reactor
Amended Facility License R-103

SUBJECT: Written communication as specified by 10 CFR 50.4(b)(1) regarding responses to the "University of Missouri at Columbia – Request for Additional Information Re: License Renewal, Safety Analysis Report, 45-Day Response Questions (TAC No. MD3034)," dated June 1, 2010

On August 31, 2006, the University of Missouri-Columbia Research Reactor (MURR) submitted a request to the U.S. Nuclear Regulatory Commission (NRC) to renew Amended Facility Operating License R-103.

On June 1, 2010, the NRC requested additional information and clarification regarding the renewal request in the form of one hundred and sixty-seven (167) questions. MURR's responses to forty-seven (47) of those questions are attached. As discussed during a conference call on July 8, 2010 with the Jessie Quichocho, Research and Test Reactors Branch B Chief, and Duane Hardesty, Project Manager, this is just a partial response and a schedule to answer the remaining questions will be discussed and approved by the NRC.

If there are questions regarding this response, please contact me at (573) 882-5276 or foytol@missouri.edu. I declare under penalty of perjury that the foregoing is true and correct.

Sincerely,

Leslie P. Foyto
Reactor Manager

ENDORSEMENT:
Reviewed and Approved,

Ralph A. Butler, P.E.
Director

Enclosed: Standard Research Subcontract No. 00072718, "Reactor Assistance and Fuel Elements"

xc: Reactor Advisory Committee
Reactor Safety Subcommittee
Dr. Robert Duncan, Vice Chancellor for Research
Mr. Craig Basset, U.S. NRC
Mr. Alexander Adams, U.S. NRC

NOTARY PUBLIC NOTARY SEAL STATE OF MISSOURI

MARGEE P. STOUT
My Commission Expires
March 24, 2012
Montgomery County
Commission #08511436

1513 Research Park Drive Columbia, MO 65211 Phone: 573-882-4211 Fax: 573-882-6360 Web: http://web.missouri.edu/~murrwww

Fighting Cancer with Tomorrow's Technology

CHAPTER 1

1.1 *Section 1.6, Compliance With the Nuclear Waste Policy Act of 1982, Page 1-22. Provide details on the Missouri University Research Reactor (MURR) contract for disposition of fuel. Include the contract number and with whom the contract exists.*

Enclosed you will find a copy of the subcontract that exists between Battelle Energy Alliance, LLC (BEA) (Contractor) and the University of Missouri-Columbia (Subcontractor). The subcontract is issued under Prime Contract No. DE-AC07-05ID14517 between the Contractor and the United States Department of Energy (DOE) for the management and operation of the Idaho National Laboratory.

It is a standard research subcontract (No. 00072718) for unclassified research and development work, not related to nuclear, chemical, biological, or radiological weapons of mass destruction or the production of special nuclear material for use in weapons of mass destruction. The period of performance of this subcontract is from January 1, 2008 to December 31, 2012.

CHAPTER 2

2.1 *Section 2.1.1, Site Location and Description, Page 2-1. The statement is made that personnel located within 1,500 feet of the facility can be rapidly evacuated if required. What is the basis of this statement?*

[redacted]

CHAPTER 3

3.1 *Section 3.2, Meteorological Damage, Page 3-28. The statement is made that the containment structure has been designed for area wind loads. Discuss the design wind loads and why the design of the containment is sufficient?*

While the specific architectural design factors used for assessing wind loading on the containment building are not available, several site and structural features aid in minimizing this load. As discussed in Section 3.2, Meteorological Damage, local terrain provides a significant load reduction. The containment building structure is designed and built for a peak internal pressure of 2.0 psig (13.8 kPa above atmosphere), which is the dominant architectural feature regarding loading. Among other elements, this feature results in eight (8) right angle shear walls external to the containment structure. The shear walls are enclosed to form the four (4) towers surrounding the containment structure. In addition to significant shear strength provided against wind loading, the resulting geometry presents an approximation to a more circular structure, rather than the base rectangular structure. This geometry results in significant reductions to both direct wind pressure and vortex induced negative pressure down-wind. A typical 3-second wind gust design speed for structures in the Midwest is 90 mph, and was the most likely design speed for this structure. Compare this with Table 2.10, "Normal and Extreme Winds for Columbia, Missouri," which shows that the maximum recorded 5-second wind gust speed has historically been 59 mph.

CHAPTER 4

4.2 *Section 4.2.1.2, Fuel Element Description, Page 4-10. Table 4-3 gives innermost and outermost fuel plate center radii which differ from what is stated in the text. Please discuss.*

The values stated in Section 4.2.1.2 – The nominal radius to the innermost plate is 2.77 inches (7.04 cm), and 5.76 inches (14.63 cm) to the outermost plate – are measurements from the inner radius of the innermost plate and the inner radius of the outermost plate, and not from the center radii.

The fuel plates are [redacted] thick. Therefore, the center radius of the innermost plate is 2.77 + 0.025 = 2.795 inches (7.099 cm), whereas the center radius of the outermost plate is 5.76 + 0.025 = 5.785 inches (14.694 cm).

4.4 *Section 4.2.1.3, Fuel Element Material of Construction, Page 4-11. Provide a reference or a discussion to how using other materials such as Al alloy 1100 for fuel cladding affects fuel performance.*

The following reference will be added to the list of references for Chapter 4: ATR Bond Study Comparison of 6061 Aluminum to ALCLAD 6061 in Bonding of Fuel Plates; Contract No. C29000113 Task Order 27; June 5, 1991.

4.6 *Section 4.2.2.2, Evaluation of the Control Blades, Page 4-18. A coefficient of linear expansion for aluminum is given in the text. Please confirm that the proper reference for this property is 4.19.*

The coefficient of linear expansion for aluminum is stated in Section 4.2.2.3, Evaluation of Control Blade Thermal Distortion, Page 4-18. The reference was improperly stated as Reference 4.18. The proper reference for this property is 4.19 – Principles of Nuclear Reactor Engineering, Samuel Glasstone, Table A.3, p. 843, 1955.

4.14 Section 4.5.3, Operating Limits.

a. *Page 4-39. The core temperature reactivity coefficient is given as -6.0x10^5. Confirm that this is typographical error regarding the positive exponent.*

Yes, it is a typographical error regarding the exponent. The value should be -6.0x10^{-5} Δk/k/°F.

CHAPTER 5

5.1 *Section 5.2.2, General Operating Conditions, Page 5-2. The inlet temperature to the reactor is listed as 120 degrees F, however it is listed as 140 degrees F in Table 4-13. Explain this difference in values.*

The reactor core inlet temperature of 140 °F listed in Table 4-13 is the design inlet temperature stated in Reference 4.30, "MURR Design Data, Volume I," Internuclear Company, 1962. Reference 4.30 provides all of the original design characteristics of the University of Missouri Research Reactor (MURR) prior to construction. The value of 120 °F listed in Section 5.2.2 is our normal operating inlet temperature.

5.2 Section 5.2, Primary Coolant System

b. *Describe how oxygen and hydrogen from the radiolysis of primary coolant is controlled.*

BACKGROUND: In radiolysis of pure water, ionizing radiation produces hydrogen atoms (H) and hydroxyl radicals by reaction 1 and hydrated electrons (e^-_{aq}) by reaction 2.[1]

[1] $HOH \rightarrow H + OH$

[2] $2HOH \rightarrow H_3O^+ + OH + e^-_{aq}$

Comparison of the reaction rates of H and e^-_{aq} with hydrogen peroxide (a stable product of hydrolysis) for reactions 3 and 4, results in a ratio of rate constants $k\ e^-_{aq}(H_2O_2)/k_H(H_2O_2) = 500$ demonstrating the hydrated electron to be a much stronger reducing agent than hydrogen atoms.

[3] $H + H_2O_2 \rightarrow H_2O + OH$

[4] $e^-_{aq} + H_2O_2 \rightarrow OH^- + OH$

Depending on the type of ionizing radiation (alpha, beta and/or gamma), the purity of the water, and, to a lesser extent, the pH and temperature, numerous secondary free-radical reactions take place. Thirty-two (32) such reactions have recently been identified by Ershov and Gordeev.[2]

The stable product signatures resulting from the radiolysis of water are hydrogen gas ($H_{2(g)}$) and hydrogen peroxide (H_2O_2) and their respective g-values[3] at 54°C are 0.5 and 0.6.[4] The low g-values for these products are the result of the fast recombination rates of the precursor radicals on timescales of picoseconds to microseconds in water.[5]

RADIOLYSIS POTENTIAL AT THE MURR: The MURR operates at 10 MW approximately 150 hours per week. Consequently, the total gamma-ray activity resulting from neutron capture, fission products and activation products is substantial. Any resulting γ-radiolysis would be maximized in the reactor core and fuel storage areas in the reactor pool. In both locations, the

water is continuously purified via ion exchange maintaining a quality standard for purified water that meets ISO 3696 Grades 1 & 2. The water quality data for the MURR pool and primary coolant systems are given in the following table.

MURR Pool and Primary Coolant System Water Quality
(average & s.d. from 2009 weekly data)

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

The 8-element reactor core is cooled by a closed circulating water system (primary coolant system) containing 2,000 gallons (7,571 l) of deionized water with a flow rate of approximately 3,750 gpm (14,195 lpm). At 10 MW, the inlet and outlet temperatures are 120 °F (49 °C) and 136 °F (58 °C), respectively, giving an average operating temperature of 54°C. The primary coolant system also includes two heat exchangers, an anti-siphon tank, and a vent tank having a volume of 3391 in^3. During reactor operation, the vent tank contains water, water vapor, air and any collected gasses resulting from radiolysis.

The fuel-storage areas can accommodate up to 88 irradiated fuel elements at various stages in the fuel cycle. These elements are cooled by the pool water moving through the fuel-storage areas by pool circulation and convection. The pool water volume is 28,000 gallons (105,992 l) and is pumped at a flow rate of approximately 1,200 gpm (5,542 lpm). During 10 MW operation, the pool water temperature is 100 °F (38 °C).

Under routine MURR operating conditions, radiolysis of water in the pool and primary coolant systems will occur; however, given the high water quality, recombination of the initial radiolysis species (H atoms, hydroxyl radicals and hydrated electrons) occurs on picosecond timescales resulting in negligible production of the stable radiolysis products (H_2, H_2O_2 & O_2). Observation of hydrogen (H_2) or oxygen (O_2) gases in the reactor pool or primary coolant system vent tank requires these gases to be present at concentrations exceeding their solubility at the respective temperatures, 38 °C (pool water) and 54 °C (primary water). These solubility values are summarized in the following table.

Hydrogen (H_2) and oxygen (O_2) gas solubility in MURR pool and primary water at atmospheric pressure.

gas	pool water at 38 °C		primary water at 54 °C	
	g/L	g/106,000 L	g/L	g/7571 L
$H_{2(g)}$	0.0014	148.4	0.0012	9.1
$O_{2(g)}$	0.0066	699.6	0.0054	40.9

Supporting evidence exists for the conclusion that radiolysis products are negligible in the MURR pool and primary coolant systems based on unobserved hydrogen gas (H_2), a signature radiolysis product.

1. If radiolysis without recombination, in excess of negligible quantities, were to occur on the surfaces of the irradiated fuel elements stored in the pool, hydrogen gas would be constantly produced beyond its solubility limit and ultimately emanate as a gas, rise to the pool surface and be exhausted from the containment building by the pool-sweep system. However, the routine emanation (continuous or intermittent) of gas from the irradiated fuel

elements stored in the MURR pool is NOT observed supporting the conclusion that radiolysis, without recombination, of MURR pool water is negligible.

2. If radiolysis without recombination, in excess of negligible quantities, were to occur in the primary coolant system, hydrogen gas would be constantly produced beyond its solubility limit and ultimately be collected in the primary coolant system vent tank causing it to pressurize and sporadically off-gas to the pool and be exhausted from the containment building by the pool-sweep system. However, off-gassing of the vent tank is NOT observed supporting the conclusion that radiolysis, without recombination, of primary coolant system water is negligible.

3. The same arguments apply for O_2 gas produced through radiolysis in the pool and primary coolant systems. However, some O_2 will already be present in pool and primary water from dissolved air. Consequently any O_2 produced by radiolysis would be additive and induce gas emanation at that point when the sum of O_2 from the two sources (dissolved air and radiolysis) exceed the solubility limit for O_2. As stated in the previous paragraphs (1 & 2), off-gassing in the pool or the primary coolant system vent tank is NOT observed supporting the conclusion that radiolysis, without recombination, in these systems is negligible.

SUMMARY: Neither hydrogen or oxygen gas emanation is observed in the irradiated fuel-storage areas in the MURR pool nor from the primary coolant system vent tank indicating that radiolysis, without recombination, of MURR pool and primary water is negligible. Furthermore, any gasses emanating from the pool and primary coolant systems are captured by the pool sweep system and exhausted from the MURR containment building without being re-circulated and with large dilution factors.

REFERENCES:

[1]Hart, EJ. , "The Hydrated Electron" (1964) Science 146, p19-25.
[2]Ershov, BG, Gordeev, AV. "A model for radiolysis of water and aqueous solutions."
[3]The g-value is the number of molecules, ions, atoms or radicals formed per 100eV absorbed energy.
[4]Elliot, AJ. "Rate constants and G-values for the simulation of the radiolysis of light water over the range 0 to 300 °C" (1994) AECL-11073.
[5]Janik, D, Janik I, Bartels, M. "Neutron and β/γ radiolysis of water up to supercritical conditions. 1. β/γ yields for H_2, H atom, and hydrated electron" (2007) J.Phys.Chem A, 111, p7777-7786.

5.5 *Section 5.5.6, Reactor Coolant Cleanup System, General Description. The statement is made that any malfunctions or leaks in the reactor coolant cleanup system will not lead to radiation exposure to personnel or releases to the environment that exceed the regulatory requirements or facility ALARA Program guidelines. Discuss the bases for reaching this conclusion.*

Any release of water containing radionuclides from the Reactor Coolant Cleanup System would drain into sumps located in the floors of the coolant cleanup rooms. The presence of water in the sumps would activate switches that would automatically pump the water to the Liquid Waste Disposal System (as described in Section 9.11.4). Activation of these switches will also cause an audible and visual alarm in the Control Room. This water is tested for radionuclides prior to release into the sanitary sewage system. Three means of detection of leaks can be used. The first involves the volume of water being discharged into the Liquid Waste Disposal System. The increased volume going into this system would be detected by the Operations Staff while on their periodic routine patrols. This would lead to an investigation of the potential source of the water. The second involves the increase in activity that would be present in water in the Liquid Waste

Disposal System when it is analyzed prior to release into the sanitary sewage system. This again would lead to an investigation to determine the source of the water, especially since water from the pool and primary coolant systems have known gamma signatures that would make them identifiable via gamma spectroscopy. The third indication of leakage would be from the need to add greater than normal amounts of make-up water to the Pool or Primary Coolant System. This again would lead to an investigation as to the source of the anomaly of the pool or primary coolant system volumes.

Any personnel exposure to these liquids would be no different than what is encountered during much of the day-to-day operations of the reactor. Exposure to primary or pool water is a normal potential in operating the reactor and steps are taken to minimize any exposures to these sources. It is not uncommon to perform maintenance on the pool and primary coolant systems during normal operation of the reactor. Sections of either system can be isolated to allow work on the systems, thus any leakage would not be outside of the normal parameters of exposure that would occur during a maintenance outage. Dosimetry records and routine monitoring results show that no unusual exposures, either to staff or the public, have occurred during the normal performance of maintenance of these systems. This normal maintenance on these systems can in fact involve larger volumes of pool and primary water than a postulated leakage from the Reactor Coolant Cleanup System.

CHAPTER 6

6.1 *Section 6.2.2, Reactor Containment Building, Pages 6-2 to 6-5. Describe the containment building isolation set points.*

As stated by MURR Technical Specification (TS) 3.5.b, "While containment integrity is required, the reactor containment building shall be automatically isolated if the activity in the ventilation exhaust plenum or at the reactor bridge indicates an increase of 10 times above previously established levels at the same operating condition. Exception: The containment isolation set point may temporarily be increased to avoid an inadvertent scram and isolation during controlled evolutions such as experiment transfers or minor maintenance in the reactor pool area. The pool area shall be continuously monitored, and, if necessary, a manual containment isolation actuated, until the automatic set point is reset to its normal value."

There are four (4) radiation detectors associated with the containment building isolation system:

1. Two (2) are located in the containment building exhaust plenum and are designated:

   - "Containment Building Exhaust Plenum No. 1"
   - "Containment Building Exhaust Plenum No. 2"

2. Two (2) are located at the reactor pool upper bridge and are designated:

   - "Reactor Pool Upper Bridge"
   - "Reactor Pool Upper Bridge – ALARA"

The table below lists the detector location, the current set point and the basis for that set point. When controlled evolutions are conducted in the pool area, the "Reactor Pool Upper Bridge" detector is bypassed, as allowed by TS 3.5.b, however, the "Reactor Pool Upper Bridge – ALARA" will still provide a containment isolation if its set point of 10 R/hr is exceeded.

Detector Location	Current Set Point	Set Point Basis
Containment Building Exhaust Plenum No. 1	3.0 mR/hr	10 x normal operating background
Reactor Pool Upper Bridge – ALARA	10K mR/hr	As determined by Health Physics
Reactor Pool Upper Bridge	50 mR/hr	10 x normal operating background
Containment Building Exhaust Plenum No. 2	3.0 mr/hr	10 x normal operating background

CHAPTER 9

9.1 *Section 9.3, Fire Protection and Programs. The SAR discusses why one fire exit from the containment building is acceptable according to National Fire Protection Association (NFPA) 101 "Life Safety Code". Provide a reference for the 300 ft used in the evaluation and the issue date of NFPA 101 that supports these statements.*

Section 9.3, Fire Protection Systems and Programs, refers to a travel distance limit of 300 ft to any one exit. This was based on a special purpose (low hazard) Industrial Occupancy, which is shown in Table A-5-6.1, Common Path, Dead-End, and Travel Distance Limits (by occupancy) of NFPA 101, issued 1996. As of NFPA 101, issued 2000, this same table is now Table A.7.6.1.

9.2 *Please describe the areas of the facility that are under the jurisdiction of the reactor license. Section 9.5.2, Rooms, Spaces, and Equipment states that any room or area designated by MURR management and approved by Health Physics may be used for radioactive materials.*

The areas of the facility that are under the jurisdiction of the Reactor License include the Laboratory Building, which includes the Reactor Containment Building, and all attached structures to the Laboratory Building except the North Office Addition.

Following is a list of all of the areas under the Reactor License:

- The Laboratory Building, which includes the Reactor Containment Building
- The Cooling Tower
- Temporary Office Building 1
- Temporary Office Building 2
- Temporary Office Building 3
- Temporary Office Building 4
- Temporary Office Building 5
- MURR Industrial Building
- Shipping and Receiving Building

9.3 Section 9.5, Possession and Use of Byproduct, Source, and Special Nuclear Material.

a. *Describe how licensed radioactive material is kept segregated between the broad-scope materials license and the reactor license. Also, describe the process for transferring materials between the licenses.*

Licensed material is segregated between the Reactor License and the Broad Scope Material License via the use of research or production specific Project Authorizations which fall under the

purview of either the Radiation Safety Committee (RSC), for the Broad Scope Material License, or the Isotope Use Subcommittee (IUS), for the Reactor License. These Project Authorizations provide the administrative controls for the use of radioactive materials throughout the facility as well as provide the administrative controls necessary to segregate the Reactor Licensed radioactive materials from other materials used and stored in the facility. Transfers of materials are not a regular occurrence between these two licenses but do occur periodically. However, when a material transfer occurs a dedicated transfer form is used which documents the transfer between one project and another. This is usually completed by the Health Physics Staff in concert with the authorized supervisor of the project or projects involved with the transfer. Records are available of the transfers.

9.3 Section 9.5, Possession and Use of Byproduct, Source, and Special Nuclear Material.

b. *It is unclear from your application if the current license possession limits are to be carried over to the renewal license? Please state and justify the license material possession limits.*

The current license possession limits will be carried over to the renewal license. The current license limits are listed below. The justification for these limits is that they are required to operate the reactor in a safe and reliable manner and achieve the mission of research and development as authorized pursuant to Section 104c of the Act.

To receive, possess and use up to 60 kilograms of contained uranium-235 of any enrichment, provided that no more than 5 kilograms of this amount be unirradiated; up to 80 grams of plutonium-beryllium neutron source; up to 20 grams of plutonium-239 in the form of sheets enclosed in aluminum for use in connection with operation of the reactor; up to 40 grams of plutonium enriched to 90% plutonium-242 in the form of a rod sealed in a stainless steel can for use in connection with operation of the reactor; and to possess, but not separate, such special nuclear material as may be produced by the operation of the facility.

To receive, possess, and use in connection with operation of the reactor a source of 100 curies of antimony-beryllium; and to possess, use, but not separate except for byproduct material produced in reactor experiments, such byproduct materials as may be produced by operation of the facility.

To receive, possess, and use in connection with the operation of the reactor up to 20 kilograms each of natural uranium and thorium; and up to 50 kilograms of depleted uranium for instructional and experimental purposes.

9.5 *Section 9.8, Pool Skimmer System. Discuss if loss of pool water caused by a failure in the pool skimmer system could effect reactor operations and personnel.*

A loss of pool water caused by a failure of the pool skimmer system would most likely require the reactor to be shutdown if it occurred during operation. A leak in the pool skimmer system would be very similar to a leak in the pool coolant system, but much less severe since the skimmer system piping is only 2 inches in diameter whereas the pool coolant system piping is 6 inches. Additionally, the skimmer pump is only periodically operated to remove any surface debris. Furthermore, use of the system to lower pool water level is only performed with the reactor shutdown.

As described in Section 13.2.9.1, Leak in the Pool Coolant System, severity of a leak would be dependent upon location. The only area that is of consequence is a leak between the skimmer return check valve 516A and where the piping returns to the reactor pool. A leak here would be

similar to an unisolatable section of the pool coolant system, but once again not as severe. A leak in any other section of the skimmer system would be isolable and would have minimal impact. Any leakage would be collected in sumps located in the floor of the mechanical equipment room (Room 114). The presence of water in the sumps would activate switches that would automatically pump the water to the Liquid Waste Disposal System (as described in Section 9.11.4). Activation of these switches will also cause an audible and visual alarm in the Control Room.

A leak in a section of the pool skimmer system which can be isolated would result in some water loss from the reactor pool, but would be minimal and easily collected and contained. Any personnel exposure from a small leak would be no different than what is encountered during much of the day-to-day operations of the reactor.

Should a leak occur in that section of piping between check valve 516A and the reactor pool, the emergency actions would be identical as described in Section 13.2.9.1. If the leak is in the unisolatable section, water can be added to the pool as described in Section 13.2.9.2.2. Personnel exposure would most likely be higher than what would be received from a small leak, but the addition of water to the pool would keep radiation levels within reason.

9.6 *Section 9.14, Compressed Air System. The SAR states that there are no functions or malfunctions of the compressed air system that could initiate a reactor accident, prevent safe reactor shutdown, or initiate the uncontrolled release of radioactive material. Discuss the basis for this statement.*

As stated in section 9.14, the compressed air systems are broken down into four subsystems, each having interconnections for redundant supplies but separated such that failure of a single subsystem will not affect the operation of the other subsystems.

The Main Air System supplies compressed air laboratory needs and contains the main compressors for each of the other subsystems. In the event of a failure of the compressors or other component within the system, each subsystem is isolated from the Main Air System via in-line check valves and has a backup compressor to provide service for that subsystem.

The Instrument Air System serves to control facility air conditioning and heating. It is normally divorced from the other subsystems through isolation valves. If this system was cross-connected with the other subsystems during maintenance and a failure were to occur, the associated check valve would prevent interactions between the other subsystems. No reactor safety features are associated with the facility heating and air conditions system

The Valve Operation Air System serves to provide compressed air to remotely operated pool and primary coolant valves. Each of these valves is air-operated to the appropriate operating position and spring-operated to the fail-safe condition upon a loss of compressed air. A loss of compressed air would cause the valve to spring operate to the fail-safe position and may initiate an associated reactor scram, but would in no case prevent the safety system from functioning as required.

The Emergency Air System provides air for operation of the hot exhaust line isolation valves and backup doors and to supply the inflatable gaskets which maintain the reactor containment building closures. Included in this system is a backup air compressor and associated check valve to prevent subsystem failure if the Main Air System is compromised. The hot exhaust line isolation valves are in series and of different types – Valve 16A is spring-to-close and air to open

which is designed fail-safe, while valve 16B is air-to-open and air-to-close. Valve 16B has an additional independent compressor should all of the three previous compressors fail. In the event that all air pressure is lost, Valve 16A would spring-to-close and isolate containment from the hot exhaust system. Air would also be lost to all of the inflatable seal gaskets and cause a breach in containment integrity. However it would take an additional failure of the primary system coolant boundary and a fuel cladding failure before fission products could be released. A failure of the containment system must occur simultaneously with another event for the system to fail to perform its function.

CHAPTER 11

11.1 *Section 11.1.1.1.1, Argon-41 from the Pneumatic Tube System. Discuss any potential failures of this system that could release Argon-41 to the containment or laboratory building and the potential radiological consequence to facility staff.*

There are no credible failures of the pneumatic tube system that could release Argon-41 to the containment or laboratory buildings. The pneumatic tube system is described in detail in Section 10.3.4, Pneumatic Tube System.

Directional air flow, supplied by two turbo-compressors, moves small high density polyethylene sample carriers, or "rabbits," between the sending-receiving station and the terminus. A cabinet located on the basement level of the laboratory building adjacent to the turbo-compressors contains solenoid valves which control direction of the air flow. The air flow is designed to pull the rabbit from place to place rather than push it. This arrangement ensures air flow is always into the pneumatic tube system should a leak develop in the sample carrier tubing. Additionally, the solenoid-operated control valves are positioned (de-energized state) such that a continuous flow path for air exists through the sample carrier tubing even when the pneumatic tube system is secured. This continuous flow path, in conjunction with the laboratory building exhaust system, maintains the pneumatic system at a negative pressure regardless if the system is in use or not.

11.2 *Section 11.1.2.5, Radiation Protection Training. The SAR states that Class I training is for materials under the MURR broad scope materials license. Please discuss training for persons involved with material under the MURR reactor license.*

This statement in Section 11.1.2.5 under Class I training should read "This level of training is required for individuals requesting permission to direct or supervise the work of others in utilizing radioactive materials under the Reactor License." In fact, all training at MURR applies to and is sufficient to allow work with byproduct material for either the Broad Scope Material or Reactor License. This has historically been the case and this practice will continue into the future.

11.3 *Section 11.1.5.6.1, Restricted Area. The SAR and Table 11-18 indicate that film badges are used for area radiation monitoring throughout the facility. Confirm that film badges are being used or provide a description of the dosimetry currently in use.*

The dosimetry currently utilized for area radiation monitoring within MURR are Optically Stimulated Luminescence (OSL) Dosimeters provided by Landauer, Inc. This is the same dosimeter type utilized throughout the facility to monitor whole body exposures for MURR employees. These dosimeters are normally changed on a monthly basis. Table 11-18 and any other references to film badges should reflect the above noted change to the use of OSL dosimeters.

11.4 *Section 11.1.1.1.3, Argon-41 Release to the Unrestricted Area. Table 11-2 appears to have a typographical error in the title indicating that the doses were calculated at 50 meters when the text of Chapter 11 and Appendix B indicate the doses were calculated at 150 meters. Confirm the appropriate title for the table.*

The title of Table 11-2 should in fact indicate the distance as 150 meters and not 50 meters. The title should read "Maximum Ar-41 Concentrations at 150 Meters North From the MURR Exhaust Stack."

11.5 *Section 11.1.3, ALARA Program. Table 11-11 lists Investigation Levels I and II, but provides no detail regarding the significance of this distinction. Discuss the significance of Level I and II regarding the investigative rigor applied to personnel exposures.*

Investigational levels are described in Section 7, MURR ALARA Program, of MURR Policy Manual POL-3, "Radiation Protection Program," and are excerpted here:

C. Actions for Investigational Levels

1. Personnel dose less than or equal to Investigational Level I

   No action will be taken in those cases where an individual's dose is less than values for Investigational Level I.

2. Personnel dose greater than Investigational Level I but less than or equal to Investigational Level II

   The HP Group will notify the appropriate group's ALARA reviewer of each individual whose dose exceeds Investigational Level I and will report the results of the reviews to MURR Management and actions will be taken if they are deemed appropriate.

3. Personnel dose greater than Investigational Level II

   HP Group will investigate with the appropriate group's ALARA reviewer the cause(s) of all personnel exposures exceeding Investigational Level II and will, if warranted, take action. A report of the investigation will be presented to MURR Management and to the Isotope Use Subcommittee of the Reactor Advisory Committee for information and review of actions.

11.6 *Section 11.1.2, Radiation Protection Program. Provide typical staffing of the Reactor Health Physics Branch.*

The Health Physics Branch normally consists of seven (7) full time staff members:

- Health Physics Manager (currently a CHP)
- 2 Staff Health Physicists
- 4 Health Physics Technicians

11.7 *Section 11.1.2.4, Radiation Safety Officer. Clarify the relationship between the Radiation Safety Officer and the Reactor Health Physics Manager.*

The Radiation Safety Officer fulfills the radiation protection duties required under the Broad Scope Material License. The Reactor Health Physics Manager is responsible for the radiation protection duties required under the Reactor License. These can be the same individual.

11.8 *Section 11.2, Radioactive Waste Management. Please discuss continued access to solid waste disposal sites for the 20-year period of the license renewal. Please discuss planning for material that may not have a disposal path.*

Waste generators located in the Mid-West Radioactive Waste Disposal Compact currently have access to the Energy Solutions' Clive, Utah radioactive waste disposal facility. The majority of the radioactive waste generated at MURR is Class A waste and meets the acceptance criteria of the Clive facility. Most Class A waste is shipped to a licensed waste broker for processing prior to final disposal at that facility. MURR has no reason to believe that access to the Clive Facility will be changed in the foreseeable future.

Restriction placed on access to the Barnwell waste site in 2008 has left MURR with no options for disposal of Class B and C waste. Under normal circumstances MURR makes a shipment of Class B waste approximately every two years. These shipments contain activated metal components produced as a result of normal reactor operations. MURR now maintains these materials in temporary long term storage within the reactor facility until permanent disposal options become available. At the current rate of generation, space for safe and secure Class B waste storage is available for the 20-year period of license renewal.

The Texas Low Level Radioactive Waste Disposal Compact Commission is currently considering rulemaking changes that would allow for the importation of limited quantities of waste from other waste compacts. If these changes are adopted, MURR will explore the possibility of using that facility for permanent disposal.

11.9 *Section 11.2.1, Radioactive Waste Management Program. This section of the SAR states the responsibility for safe disposal of radioactive waste from materials licensed under the MURR Broad Scope Material License but not the reactor license. Please address.*

The third sentence in the second paragraph of Section 11.2.1 should read "The Reactor Health Physics Manager, with the assistance of the Health Physics Branch, is responsible for the safe disposal of radioactive waste generated from materials under the Reactor License."

CHAPTER 12

12.2 Section 12.1.3, Staffing and TS 6.1, Organization.

a. *The minimum staffing during reactor operation includes an individual knowledgeable of the facility. Please describe the extent of this knowledge and the actions this person should be prepared to take in an emergency.*

A *knowledgeable person* is defined in MURR operating procedure AP-RO-110, "Conduct of Operations," as "A Reactor Operator Trainee who has successfully completed a 50% board." Successful completion of a 50% qualification board requires the trainee to demonstrate

knowledge of the purpose, function, major components, instrumentation, operation and safety features of the following reactor systems:

- Electrical Distribution System
- Emergency Electrical Distribution System
- Secondary Coolant System
- Primary Coolant System
- Pool Coolant System
- Nuclear Instrumentation System
- Area Radiation Monitoring System
- Reactor Isolation and Facility Evacuation System

The Trainee must also have a demonstrated knowledge of the following general safety, administrative, radiation protection and emergency procedures:

- GS-RA-100 Equipment Tag Out
- AP-RO-110 Conduct of Operations
- AP-HP-119 High Radiation Area Access
- RP-HP-135 Room 114 Entry- Self Monitored
- RP-HP-137 Handling Radioactive Material in Reactor Pool
- EP-RO-004 Fire
- EP-RO-005 Medical Emergency
- EP-RO-012 Reactor Isolation
- EP-RO-013 Facility Evacuation

The *knowledgeable person* must be able to assist the on-site licensed operator in performing actions to place the reactor in a safe condition and supporting actions to facilitate a reactor isolation or facility evacuation.

12.2 Section 12.1.3, Staffing and TS 6.1, Organization.

b. *It appears that during operation or when the reactor is not considered secure, that the minimum staffing requirement can be one reactor operator. Please discuss how you will meet the requirements of 10 CFR 50.54(m)(1) for senior reactor operator coverage and actions for which the presence of a senior reactor operator at the facility is required.*

To meet the requirements of 10 CFR 50.54 (m)(1), the following new MURR Technical Specification 6.1.c will be added:

6.1.c A senior reactor operator licensed pursuant to 10 CFR 55 shall be present at the facility or readily available on call at all times during operation, and shall be present at the facility during all start-ups and approaches to power, recovery from an unplanned or unscheduled shutdown or non-emergency power reduction, and refueling activities.

12.3 *Section 12.1.4, Selection and Training of Personnel. The SAR refers to reference 12.2, ANSI/ANS 15.4, "Selection and Training of Personnel for Research Reactors," 1988. This standard was revised in 2007. Please address the use of the revised standard.*

At the time of license renewal submittal, MURR used the approved version of ANSI/ANS-15.4, "Selection and Training of Personnel for Research Reactors," 1988. MURR fully complies with

and will update reference 12.2 to ANSI/ANS 15.4, "Selection and Training of Personnel for Research Reactors," 2007.

CHAPTER 13

13.1 Section 13.2.1.2, Accident Analysis and Consequences.

a. *Page 13-3. The mass of Al used here is 436 g. Explain the difference in mass numbers between this analysis and the containment pressure analysis of section 6.2.2.2. (Chapter 6 states the core contains 29.4 kg of aluminum while Chapter 13 states 33.56 kg).*

The fuel material at the time of initial startup was a uranium-aluminum alloy with each fuel assembly loaded to a maximum of [redacted] of uranium-235. The fuel plates of that core contained 29.4 kg of aluminum and [redacted] uranium-235. These values were used in the original analyses of the pressure loading on the containment structure and the effect of a possible aluminum-water reaction. These analyses are described in detail in Reference 6.2, "Hazards Summary Report, University of Missouri Research Reactor, University of Missouri, Columbia, Missouri, July 1965."

A conversion was performed in 1971 to switch to a uranium-aluminide dispersion UAl_x fuel material with a maximum loading of [redacted] of uranium-235 per assembly. The UAl_x fuel core (entire fuel assemblies) contains 33.56 kg of aluminum and [redacted] of uranium-235. These current values were used in the Maximum Hypothetical Accident analysis in Chapter 13.

13.5 *Section 13.2.3.2, Accident analysis, Page 13-29. Provide a reference for the 900 degree F fuel blister temperature.*

The reference is "Specification TRTR-4 For University of Missouri-Columbia Fuel Elements Assembled For University of Missouri Research Reactor, Revision 4," EG&G Idaho, Inc., June 8, 1994.
Section 3.4.1(8) of the above reference states:

Internal Defects and Bond Integrity. All fuel plates produced shall be evaluated to the requirements of this section [3.4.1(8)] by visual inspection, blister testing and ultrasonic testing.

The existence of a metallurgical bond shall be verified by blister test, ultrasonic test, and possibly a bending test on a strip sheared from the plate end trimming. At least 50% grain growth across the cladding/core and cladding/frame interface for all plates which are sectioned is required.

The blister test shall be performed after hot rolling, but before cold rolling, by heating each plate to a temperature of 900 ±13° F, holding at that temperature for a period of two (2) hours, removing from furnace, and allowing to air cool.

Any visual or ultrasonic indications of non bonds, voids, blisters, or laminations or other discontinuities larger than 0.060 inch over the fuel core area or 0.120 inch in edge or end clad shall be cause for rejection. A maximum of two (2) indications less than 0.060 inch in diameter or equivalent area are allowed in fuel core area, provided they are more than 0.250 inches apart. A maximum of two (2) indications less than 0.120 inches in diameter are allowed in any edge or end clad area, outside the fuel core area, provided they are not any closer than 0.050 inches to the edge or end of the fuel plate and no closer together than the major dimensions of the largest indication.

13.7 *Section 13.2.4.2, page 13-55. Explain why assuming longer rather than shorter close times for the isolation valves is a conservative assumption for this accident.*

The assumption of longer closure times for the isolation valves was a conservative assumption for the Loss of Coolant Accident (LOCA), but possibly not for the Loss of Flow Accident (LOFA). The LOFA with shorter closure times for the isolation valves will be run and included with the answers to Chapter 13 RELAP analysis questions.

13.8 *Section 13.2.5.2.1, Damage to a Fuel Element Due to Mishandling. Please discuss the steps taken to minimize the possibility of damage to the reactor or fuel while moving the shipping cask into or out of the reactor pool.*

Moving the spent fuel shipping cask into or out of the reactor pool is performed in an area of the pool called the weir floor, which is well away from the reactor structure and above the elevation of all fuel storage areas. Also during this evolution, all potentially adrift or interfering items such as bins, trays, fixtures and tools are removed to further ensure that inadvertent damage cannot occur to the reactor structure and fuel storage areas. Movement of the cask is performed by an experienced crane operator and multiple spotters.

Additionally, as described in Section 9.2.2, Methods of Storage and Transfer, all cask lifting equipment, including the 15-ton capacity crane, is rigorously maintained, including preventive maintenance and magnetic particle testing (MT) as appropriate. A dye penetrant inspection is also performed on the lifting ears of the shipping cask.

APPENDIX A, TECHNICAL SPECIFICATIONS

A.3 *Definition 1.5, Excess Reactivity. Why is the regulating rod not included in the definition of excess reactivity?*

TS 1.5 will be revised to read:

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

Revising the definition from "shim" to "control" will now include the regulating rod.

A.14 *TS 3.1 g., 3.1 j. and 3.1 k., Reactivity Limitations. Please explain why these experiment reactivity values are not the absolute values.*

The experiment reactivity limitation values stated in TS 3.1.g, 3.1.j and 3.1.k should be designated as absolute values. MURR has always treated them as absolute values, but overlooked indicating them as such.

TS 3.1.g, 3.1.j, and 3.1.k will be revised to read:

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

j. The absolute value of the reactivity worth of each unsecured experiment shall not exceed 0.025 $\Delta k/k$.

k. The absolute value of the reactivity worth of all unsecured experiments which are in the reactor shall not exceed 0.006 Δk/k.

A.18 *TS 3.3, Reactor Safety System. Please explain note 1 to TS 3.3 a. The note and trip set points in the table appear to define forced convection operation which is not consistent with the bases for the safety limits and LSSS for Mode III. Please explain the significance of the natural convection flange and pressure vessel cover being removed. Why is this not stated as a separate limiting condition for operation? Please explain the basis of the reactor being subcritical by 0.015 Δk/k and the need for the channels required by the TS.*

Mode III operation has a maximum power level of 50 kW. The reactor can be operated in this Mode with either natural convection or forced circulation flow.

Mode III operation with forced circulation flow requires the following instrumented channels to be operable: High Power Level, Reactor Period, Primary Coolant Flow, Differential Pressure Across the Core, Primary Coolant Low Pressure, Reactor Inlet Water Temperature, Reactor Outlet Water Temperature, Pressurizer High Pressure, Pressurizer Low Water Level, Pool Low Water Level, Primary Coolant Isolation Valves 507A/B Off Open Position, Power Level Interlock, Facility Evacuation, Reactor Isolation, and Manual Scram.

Note (1) allows operation in Mode III with natural convection flow with the following instrumented channels operable: High Power Level, Reactor Period, Pool Low Water Level, Power Level Interlock, Facility Evacuation, Reactor Isolation, and Manual Scram. The natural convection flow path is described in Section 4.6.1, Natural Convection Cooling Analysis. To establish natural convection flow requires the natural convection flange, which is located in the invert loop, to be removed to allow an inlet path for pool water to enter the primary coolant system. By removing this flange and the reactor pressure vessel head, an open path is provided between the pool and the reactor core thereby allowing natural circulation to take place – warmer primary coolant to rise up out of the core and enter the pool. These two points can be seen on page 4-21/22 – Figure 4.6, "Reactor Assembly and Arrangement." The natural convection flange is the vertical flange located about half way up in the reactor pool and to the right in the figure. It is located at an elevation about midpoint of the finned tubes of the in-pool heat exchanger. The pressure vessel cover is the horizontal flange at the top of the reactor pressure vessels.

Note (1) also allows operation in Mode III without forced circulation or the natural convection flange and pressure vessel cover removed (natural convection flow) with the following instrumented channels operable: High Power Level, Reactor Period, Pool Low Water Level, Power Level Interlock, Facility Evacuation, Reactor Isolation, and Manual Scram. To operate in Mode III without forced circulation or the natural convection flange and pressure vessel cover removed requires the reactor to be subcritical by a margin of at least 0.015 Δk/k. This avoids any significant production of power in the core. The only power increase would be due to subcritical multiplication.

The reactor safety system instrument channels for Mode III operation, as provided in Appendix A of the SAR, Technical Specification 3.3, are the same as was approved in 1974 when the 10 MW operating license was granted.

A.19 *TS 3.3, Reactor Safety System. The pool low water level scram is only required for Mode III operation. The bases state the purpose of the scram is to control radiation levels above the reactor pool with Section 11.1.5.1.1 of the SAR stating that a minimum water depth of 23.6 feet is required over the fuel region at an operating power level of 10 MW. Please explain why this*

scram is not needed for all modes of operation. It is not clear what the baseline of the 23 foot scram set point is. Where is the 23 feet measured from and how does it relate to the minimum water depth discussed in the SAR?

The 23 feet (Min) pool low water level scram for Mode III operation is replaced for Mode I and II operation with Technical Specification 3.4.c, which requires a Rod Run-In Function trip set point at 27 feet (Min). This prevents operating at any mode except Mode III with the pool water level below 27 feet. The pool water level is referenced to the horizontal aluminum pool liner bottom being 0 feet. So the water depth discussed in the SAR is based on the height of water in the reactor pool above the horizontal bottom of the pool liner.

A.20 *TS 3.3, Reactor Safety System. The number of instrumentation channels required for primary coolant flow appears to imply a minimum of two instrument channels per pump without specifying a pump must be operating. Propose TS wording to clarify the intent of the TS is to have two instrument channels per operating pump.*

The requirement of two Primary Coolant Flow instrument channels per operating pump is not the intent of TS 3.3. The set points for the Primary Coolant Flow and Differential Pressure Across the Core instrument channels are to ensure that in Mode I operation there is a minimum of 3,200 gpm of flow through the core and in Mode II operation 1,600 gpm of flow. The instrumentation channels required for Primary Coolant Flow are provided in Section 7.6.4, Flow Measurement and Control, in the first paragraph of Subsection 7.6.4.1, Primary Coolant System.

Two (2) Primary Coolant Heat Exchangers are installed for design power operation. Two (2) redundant flow sensors are located in the long run of straight piping in each heat exchanger loop complemented with one detector sensing the total differential pressure across the reactor core. Flow transmitter FT 912A and differential pressure transmitter DPS 928A are redundant sensors monitoring the flow rate through primary coolant heat exchanger HX503A. Flow transmitter FT 912B and differential pressure transmitter DPS 928B are redundant sensors monitoring the flow rate through primary coolant heat exchanger HX503B.

In Mode I operation both heat exchangers are necessary, therefore all four (4) Primary Coolant Flow instrument channels are required. In addition, the Differential Pressure Across the Core instrument channel must trip at a value greater than or equal to 3,200 gpm. The four (4) Primary Coolant Flow measuring channels will ensure that the combined flow rate through the two (2) primary coolant heat exchangers is equal to or greater than 3,250 gpm. This value is greater than the 3,200 gpm flow through the core because primary coolant flow through the heat exchangers combines total core flow rate with the 50 gpm flow through the Primary Coolant Demineralizer Loop.

In Mode II operation only one heat exchanger is necessary, therefore two (2) Primary Coolant Flow instrument channels are required. In addition, the Differential Pressure Across the Core instrument channel must trip at a value greater than or equal to 1,600 gpm. The two (2) Primary Coolant Flow measuring channels will ensure that the combined flow rate through the one heat exchanger is greater than or equal to 1,625 gpm. This value is greater than the 1,600 gpm flow through the core because primary coolant flow through the heat exchanger combines core flow rate with the 25 gpm flow through the Primary Coolant Demineralizer Loop.

The requirement for operating in Mode III is the same as Mode II unless the reactor is in the natural convection flow mode, or the reactor is subcritical by a margin of at least 0.015 $\Delta k/k$.

MURR does not feel that it is necessary to revise this TS.

A.21 *TS 3.3, Reactor Safety System. The set point for the pool coolant flow scram is a minimum flow rate of 850 gpm. However, the bases discuss a flow rate of 425 gpm. Please explain this difference.*

The basis for the pool coolant flow scram is to assure the adequate cooling of the reactor pool, reflectors, control rods, and the flux trap. As stated in Section 5.3.5, Reflector Plenum Natural Convection Valve, to insure compliance with IEEE-279, Single Failure Criterion, the Reflector Plenum Natural Convection Valve (valve V547) is left open. With V547 open, greater than 80% of the total Pool Coolant System flow will flow through the reflectors, control rods, and flux trap regions.

The third paragraph under TS 3.3 Bases: a. will be revised to read:

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

A.25 *TS 3.4 c., Reactor Instrumentation. Please explain note 1 to TS 3.4 c. The note is inconsistent with Mode III operation which is limited to 50 kW(t). Please explain the significance of the natural convection flange and pressure vessel cover being removed. Why is this not stated as a separate limiting condition for operation? Please explain the basis of the reactor being subcritical by 0.015 $\Delta k/k$ and the channel being required by the TS.*

Mode III operation has a maximum power level of 50 kW. The reactor can be operated in this Mode with either natural convection or forced circulation flow.

Mode III operation with forced circulation flow requires the following instrumented channels to be operable: High Power Level, Reactor Period, Rod Not-In-Contact With Magnet, Anti-Siphon System High Level, Truck Entry, Regulating Blade Position, and Manual Rod Run-In.

Note (1) allows operation in Mode III with natural convection flow with the following instrumented channels operable: High Power Level, Reactor Period, Rod Not-In-Contact With Magnet, Truck Entry, Regulating Blade Position, and Manual Rod Run-In. The natural convection flow path is described in Section 4.6.1, Natural Convection Cooling Analysis. To establish natural convection flow requires the natural convection flange, which is located in the invert loop, to be removed to allow an inlet path for pool water to enter the primary coolant system. By removing this flange and the reactor pressure vessel head, an open path is provided between the pool and the core thereby allowing natural circulation to take place – warmer primary coolant to rise up out of the core and enter the pool. These two points can be seen on page 4-21/22 – Figure 4.6, "Reactor Assembly and Arrangement." The natural convection flange is the vertical flange located about half way up in the reactor pool and to the right in the figure. It is located at an elevation about midpoint of the finned tubes of the in-pool heat exchanger. The pressure vessel cover is the horizontal flange at the top of the reactor pressure vessels.

Note (1) also allows operation in Mode III without forced circulation or the natural convection flange and pressure vessel cover removed (natural convection flow) with the following instrumented channels operable: High Power Level, Reactor Period, Rod Not-In-Contact With Magnet, Anti-Siphon System High Level, Truck Entry, Regulating Blade Position, and Manual

Rod Run-In. To operate in Mode III without forced circulation or the natural convection flange and pressure vessel cover removed requires the reactor to be subcritical by a margin of at least 0.015 $\Delta k/k$. This avoids any significant production of power in the core. The only power increase would be due to subcritical multiplication.

A.29 *TS 3.5, Reactor Containment Building. ANS 15.1, Section 3.5 suggests requirements for ventilation system TSs. Please propose TSs for ventilation during reactor operation or justify why such TSs are not needed.*

Technical Specification 3.5.a requires that containment integrity shall be maintained at all times except when: (1) The reactor is secured, and (2) Irradiated fuel with a decay time of less than sixty (60) days is not being handled. The requirements for containment integrity include ventilation system requirements that are given in Technical Specification 1.17 below:

1.17 **Reactor Containment Integrity** - For reactor containment integrity to exist, the following conditions must be satisfied:

a. The truck entry door is closed and sealed;
b. The utility entry seal trench is filled with water to a depth required to maintain a minimum water seal of 4.25 feet;
c. All of the reactor containment building ventilation system's automatically-closing doors and automatically-closing valves are operable or placed in the closed position;
d. The reactor mechanical equipment room ventilation exhaust system, including the particulate and halogen filters, is operable;
e. The personnel airlock is operable (one door shut and sealed); and
f. The most recent reactor containment building leakage rate test was satisfactory.

Reactor Containment Integrity as defined in TS 1.17 contains the following requirements that relate to the ventilation system during reactor operation:

a. The truck entry door being closed and sealed ensures that the largest possible opening into the containment structure is not a leakage path.
b. The utility entry seal trench filled with water to a depth of at least 4.25 feet ensures the utility trench is sealed to withstand a differential pressure up to 2 psi.
c. The containment building ventilation system's automatically-closing doors and automatically-closing valves being operable or placed in the closed position ensures containment integrity can be automatically set.
d. The reactor mechanical equipment room ventilation exhaust system being operable requires the exhaust fan to be operating and the equipment room particulate and halogen filters operable.
e. The personnel airlock being operable provides a way to enter and exit containment while still maintaining containment integrity.
f. The most recent reactor containment building leakage rate test being satisfactory ensures it meets the containment integrity requirements.

A.31 *TS 3.6 d., Experiments. Please define what constitutes an explosive material.*

TS 3.6.d will be revised to read:

d. Explosive materials shall not be irradiated nor shall they be allowed to generate in any experiment in quantities over 25 milligrams of TNT-equivalent explosives.

A.34 *TS 3.8 c., Reactor Fuel. Please provide a technical basis for operation with less than eight fuel elements at power levels up to 100 Watts(t).*

TS 3.8.c basis will be revised to read:

c. Operation at a power level greater than 100 watts requires a full core of eight fuel elements to assure the validity of the safety limit curves (Specification 2.1) and other safety analyses. When it may be important to conservatively determine the actual critical core loading, Specification 3.8.c allows operation with less than eight fuel elements up at a lower level not to exceed 100 watts. This maximum power limit is low enough to ensure no fuel damage will occur. This provides for a conservative approach to criticality with less than eight new fuel elements. Typically, the first approach to critical would be with a number of fuel elements insufficient to achieve criticality but still be able to observe subcritical multiplication. Then one additional fuel element would be added at a time in between approaches to criticality. The reactor would be operated in this manor only to perform necessary conservative approaches to criticality.

A.38 *TS 3.9 c., Reactor Coolant System. Please discuss the basis for limiting the Iodine-131 concentration of the primary coolant to 5 x 10^{-3} μCi/ml.*

Using a primary coolant system volume of 2000 gallons, the following calculation determines the total iodine-131 inventory in the system having a concentration of 5E-3 μCi/ml:

2000 gal * 3.7853E3 ml/gal * 5E-3 μCi/ml = 3.785E4 μCi = 0.03785 Ci

Dividing this activity by the total core iodine-131 activity:

0.03785 Ci/1.7E5 Ci = 2.23E-7 = < 0.000022%

This also corresponds to 0.00018% of the iodine-131 activity in the average MURR fuel element at the end of a one week run. Therefore, it provides early detection of a leaking fuel element.

The basis will be revised to read:

b. - c. The primary coolant system with an iodine-131 concentration of 5 x 10^{-3} μCi/ml would contain a total iodine-131 inventory of 0.038 Ci in the system. Based on the iodine-131 activity in the reactor core provided in Section 13.2.1.2 of the SAR, this I-131 concentration would equate to less than 0.000022% of the total core iodine-131 inventory in the primary coolant. Specifications 3.9.b and 3.9.c provide for the early detection of a leaking fuel element so that corrective action can be taken to prevent the release of fission products. Refer to Specification 4.2.c for surveillance sampling of the primary coolant system.

A.50 *TS 5.3 i., Reactor Coolant Systems. Describe the natural convection flow path consistent with the evaluation in the SAR.*

The natural convection flow path is described in Section 4.6.1, Natural Convection Cooling Analysis, of the SAR.

To establish natural convection flow requires the natural convection flange, which is located in the invert loop, to be removed to allow an inlet path for pool water to enter the primary coolant

system. By removing this flange and the reactor pressure vessel head, an open path is provided between the pool and the core thereby allowing natural circulation to take place – warmer primary coolant to rise up out of the core and enter the pool.

These two points can be seen on page 4-21/22 – Figure 4.6, "Reactor Assembly and Arrangement." The natural convection flange is the vertical flange located about half way up in the reactor pool and to the right in the drawing. It is located at an elevation about midpoint of the finned tubes of the in-pool heat exchanger. The pressure vessel cover is the horizontal flange at the top of the reactor pressure vessels.

A.51 *TS 5.4 b., Reactor Core and Fuel. The term "fully enriched" may not be clear. Provide a nominal enrichment level for the reactor fuel.*

The reactor fuel is enriched to 93.0 ±1.0% in the isotope uranium-235.

In order to make the design feature unambiguous, we propose revising TS 5.4.b to read:

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

A.52 *TS 5.4 d., Reactor Core and Fuel. Given that fuel can also be stored in racks, clarify this TS.*

TS 5.4.d will be revised to read:

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

The in-pool fuel storage locations and fuel storage vault are discussed in Section 9.2, Handling and Storage of Reactor Fuel.