50-186

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April 12, 2001

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Subject: Response to Request Dated 03/01/01 for Additional Information Regarding Amendment Extending License Expiration Date (TAC No. MB0850)

Dear Sir or Madame:

On 12/27/00, the University of Missouri-Columbia Research Reactor submitted a request that Facility Operating License No. R-103 be extended to October 11, 2006. On 03/01/01, the U.S. Nuclear Regulatory Commission requested additional information concerning that proposed license extension. Enclosed is our response.

Sincerely,

Edward A. Deutsch, Ph.D.
Director

LORALYNN SULLIVAN
NOTARY PUBLIC STATE OF MISSOURI
BOONE COUNTY
MY COMMISSION EXP. JUNE 13,2004

EAD:dcp

Enclosure

xc: Mr. Al Adams, USNRC
Mr. Craig Bassett, USNRC

A020

Request for Additional Information
University of Missouri-Columbia Research Reactor
Docket No. 50-186

April 12, 2001

Request for Additional Information
University of Missouri-Columbia Research Reactor
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1. *Please summarize the radiological impact that operation of the facility has had over the past ten years summarizing radioactive liquid (primary isotope curies and total curies) and gaseous effluents (primary isotope curies and total curies), solid waste excluding fuel elements (cubic feet and total curies), and off-site dose monitoring (average and maximum dose measured in the environment). Please provide a summary of the results of your calculations showing that you met the dose constraint in 10 CFR 20.1101(d). Please discuss your estimate of the radiological impact during the time period of construction permit recapture.*

The MURR has operated effectively and within regulatory limits for over 30 years with regards to radiological impacts of environmental releases. Releases to the environment including but not limited to air effluent, sanitary sewer effluent and external radiation have consistently been below regulatory limits. Additional water, soil and vegetation samples confirm the negligible contribution to the environment of the operation of the MURR. Shown below are supporting data for the past 10 years. Information provided in the Public Dose Table shows MURR remains consistently below the 100 mrem/yr dose limit in 10 CFR 20.20.1301. As no substantial changes to the operational parameters of the MURR are anticipated, there should be negligible impact to the environment during the period of construction permit recapture.

Radionuclide Releases 1991-2000

	Air Exhaust		**Sanitary Sewer**	
	(Ar-41 Ci)	Total Ci	(H-3 Ci)	Total Ci
2000	975	982	1.199E-01	1.420E-01
1999	1130	1137	1.670E-01	1.740E-01
1998	1130	1134	5.901E-01	5.980E-01
1997	861	870	1.460E-01	1.510E-01
1996	728	739	1.487E-01	1.560E-01
1995	878	888	8.181E-02	9.000E-02
1994	370	385	1.089E-01	1.270E-01
1993	409	425	2.574E-01	3.160E-01
1992	470	475	1.711E-01	2.150E-01
1991	440	441	2.094E-01	2.580E-01

Radionuclide Releases Notes:
1991 and 1992 data is based on July through June data.
1993 began January through December reporting periods.
Radionuclide release were taken from MURR Operations Annual Reports submitted annually to the NRC.

Some variability in AR-41 production is expected based on the overall use of the pneumatic tube irradiation facility where most of the activity is produced. The step change in activity reported in 1994 to that reported in 1995 is the result of a change in the sampling method used to measure the Ar-41 concentration in the stack exhaust. Previous to 1995, the activity released was calculated using data obtained from a daily grab sample collected with an evacuated marinelli beaker. Subsequent years release activity was calculated from data points recorded by the gas channel of the exhaust stack radioactivity monitor which is in operation 24 hours a day. In both methods, calibrated equipment was used to analyze the sample.

MURR Waste Shipments and Inventory

Shipment Year	Waste Container Type	Volume	Activity
• 2000	2 B-25 Containers	200 ft^3	
	43 LLW Barrels	322.5 ft^3	
	Total 2000*	**522.5 ft^3**	**249 mCi**
• 1999	1 B-25 Containers	100 ft^3	
	62 LLW Barrels	465 ft^3	
	Total 1999	**565 ft^3**	**281 mCi**
• 1998	4 B-25 Containers	400 ft^3	
	28 LLW Barrels	210 ft^3	
	Total 1998*	**610 ft^3**	**53 mCi**
• 1997	**56 LLW Barrels**	**420 ft^3**	**404 mCi**
• 1996	**45 LLW Barrels**	**337.5 ft^3**	**1409 mCi**
• 1995	**LSA Waste**	**None**	
• 1994	**LSA Waste**	**460 ft^3**	**1228mCi**
• 1993	**LSA Waste**	**375 ft^3**	**605 mCi**
	Shielded Cask	**17 ft^3**	**59,500 mCi**
• 1992	**LSA Waste**	**679 ft^3**	**1924 mCi**
• 1991	**LSA Waste**	**772.5 ft^3**	**1146 mCi**

***Notes:** 2000 Waste shipments also included two Surface Contaminated (SCO) Water Storage Tanks-685 ft^3 Total.

1998 Waste shipments also included two Surface Contaminated (SCO) Pool Heat Exchangers- $\approx$300 ft^3 total.

Current MURR Waste inventory contains approximately 80 Barrels and 2 B-25 equivalents of LLW on site.

Off-Site Dose Monitoring

Environmental TLD's (mrem/yr)

	Average	Max Env.	Loading Dock
2000	-1.3	18.6	202.7
1999	13.5	43.5	119.8
1998	3.4	51.9	202.6
1997	9.2	34.8	195.6
1996	9.2	34.9	221.0
1995	14.6	44.2	216.0
1994	20.5	49.7	236.6
1993	18.1	28.2	275.4
1992	6.3	26.7	265.6
1991	4.4	27.3	127.3

Note: **Average** equals average of all TLD's excluding TLD's affected by shipping functions. The 2000 average is slightly negative due to the inadvertent exposure of a control TLD.
Max Env = Highest environmental TLD which is not confounded by shipping activities.
Loading Dock = South side building TLD which is closest to all shipping and receiving activities.

Environmental TLD's consist of 41 deployed TLD's at various distances and directions from the MURR. There are also 4 control TLD's used to monitor background during storage and transit of TLD's

20.1101(d) Dose Calculations
(NESHAPS)

2000	0.8 mrem/year
1999	0.9 mrem/year
1998	0.9 mrem/year
1997	0.7 mrem/year
1996	0.6 mrem/year
1995	0.7 mrem/year
1994	0.5 mrem/year
1993	0.6 mrem/year
1992	0.4 mrem/year
1991	0.4 mrem/year

Note: Doses are based on calculated air emissions using the Comply code. Values are substantially below the 10 CFR 20.1101(d) constraint of 10 mrem/yr.

2. *You state that all non-radioactive hazardous waste is transferred to the University of Missouri Environmental Health and Safety group for processing. Are there any chemicals released from the facility that would have an impact on the environment? For example, are any chemicals used to control secondary cooling system chemistry or performance that are released in cooling tower spray or blow down? If so, please discuss the past environmental impact of these chemicals. Briefly discuss typical materials transferred to the Environmental Health and Safety group and verify that the group's processing of your materials has no significant environmental impact. Please discuss your estimate of the non-radiological impact during the time period of construction permit recapture.*

The University of Missouri, Environmental Health and Safety Department currently maintains a non-radioactive hazardous waste storage facility under a Part B Permit administered by the Missouri Department of Natural Resources (MDNR). Hazardous materials transferred to this facility are disposed of in compliance with regulations specified by the Environmental Protection Agency (EPA) and the MDNR. Using University policies and procedures, MURR transfers unwanted non-radioactive hazardous materials to this permitted facility. There are no chemicals released from the facility that would have a significant environmental impact.

The reactor secondary cooling system water quality is maintained using four commercial products containing hazardous materials that are the same products used in the air conditioning cooling towers of large office buildings. These chemicals include two biocides, a corrosion inhibitor, and sulfuric acid for pH control. Approximately 100 gallons and 5 gallons of the two biocides and 400 gallons of the corrosion inhibitor are stored in plastic tanks with catch basins in the cooling tower located about 50 meters from the laboratory building. About 1000 gallons of sulfuric acid is stored in an above ground, double walled, plastic storage tank outside the cooling tower. The bulk chemicals are stored in tanks that can be easily monitored for leakage. The treated water in the cooling tower enters the environment through evaporation and through the blow down line that is routed to the sanitary sewer. Volumes of water treatment chemicals used annually are about 100 gallons and 5 gallons of the two biocides, 700 gallons of the corrosion inhibitor and 4000 gallons of sulfuric acid. The EPA has approved these water quality chemicals for the use to which they are applied, and accordingly, have no significant environmental impact.

MURR uses small quantities of typical laboratory chemicals such as acids, bases and organic liquids. These chemicals are purchased in one to four liter quantities. Typical transfers of hazardous material to the Environmental Health and Safety waste storage facility consist of one to four liter quantities of hydrochloric acid, nitric acid, aqua regia and isopropyl alcohol. Hazardous materials are used in appropriate laboratory settings and disposed of in compliance with applicable regulations, and accordingly, have no significant environmental impact.

The historical use of non-radioactive hazardous materials has been in compliance with applicable State and Federal Regulations and there has been no significant environmental

impact from these activities. There are no planned changes to the non-radioactive hazardous material uses described above, and accordingly, there will be no significant environmental impact associated with the requested extension to the license.

3. *What is the environmental impact of the "no action" alternative? If your application for this amendment was denied, what would your actions be and what would be the environmental impact of those actions?*

The alternative to the proposed action for the MURR is to deny the application (i.e. *no action* alternative). If this were the case, the Board of Curators at the University of Missouri-Columbia would, time permitting, apply for license renewal and operate under the timely renewal provisions of 10 CFR 2.109 until the Commission renewed or denied the license renewal application. With operation under timely renewal or renewal, the actual conditions of the reactor would not change. If the Commission denied license renewal or if MURR was unable to submit the license renewal application in a timely manner, the MURR research reactor operations would stop and decommissioning would be required.

If MURR were to terminate operations and undergo decommissioning the resulting impact on the environment would be minimal.

- <u>Socioeconomic Impacts</u>:

Since the MURR is the sole approved US provider of isotopes used in three cancer therapies, patients will not receive these therapies. As a result, patients probably would die who otherwise would live. Moreover, thousands of cancer patients who depend on our radioisotopes to relieve the debilitating pain cause by their metastatic bone cancer will not receive pain relief. They will suffer intense pain that they would not have to suffer. Nevertheless, significant adverse socioeconomic impact would result. Terminating operations would also prevent vital research and training of nuclear engineers from occurring. Senate Bill S. 242, "Department of Energy University Nuclear Science and Engineering Act", co-sponsored by Messers. Bingaman, Domenici and Crapo, describes the decline of University nuclear programs and reactors within the United States. Currently, there are only 28 operating research and training reactors, over a 50% decline since 1980. With the MURR being the largest of the university research reactors, terminating operations at MURR would adversely impact a situation that is described in S. 242 as a threat to the health of our nation's people and to our national security.

- <u>Land Use Impact</u>:

Neither terminating operations nor decommissioning is expected to have any immediate impacts on land use.

- <u>Air Quality Impact</u>:

During operations, minimal impacts to air quality result from the use of motor vehicles by facility personnel, not facility operations. Following decommissioning, these minor impacts would end.

- Water Resources and Ecology Impacts:

During decommissioning impacts to water resources and the ecology would be approximately the same as during operations, but would cease once decommissioning is completed.

- Radiological Impacts:

Impact to the public from routine existing operations are minimal. Radiological impacts would be reduced to even lower levels by terminating operations and would be eliminated altogether at the completion of decommissioning. Terminating operations would reduce occupational dose. While undergoing decommissioning occupational does would increase due to handling of radioactive materials during dismantlement.

- Waste Management:

Terminating operations eventually would eliminate generation of spent fuel and low-level radioactive waste (LLW). However, decommissioning would require the disposal of 6,000 to 9,000 cubic feet of LLW. Following decommissioning, no further LLW would be generated.

- Aesthetics Resources:

The primary positive aesthetic impact would be elimination of the steam plume from the facility's small mechanical draft cooling tower. Since decommissioning would not necessarily lead to dismantlement, aesthetic impacts associated with facility appearance might not change except where uncontaminated facilities would be removed.

4. *Please discuss any updating that has been performed on the reactor safety system, radiation monitoring system and on engineering safety features.*

Periodic updating of the reactor safety system, radiation monitoring system, and engineering safety features has been performed to take advantage of technological improvements and state-of-the-art developments, while retaining the desirable design characteristics of the previous system. This has increased reliability and enhanced system performance. The following represent updates since the 10 MW license amendment was granted in 1974. The updates or plant modifications are grouped by reactor safety system, radiation monitoring and engineering safety features categories. Within each category, the most recently completed plant modifications are listed first.

Reactor Safety System

Nuclear Instrumentation Upgrade

The original General Electric nuclear instrumentation (six drawers--one source range, two intermediate ranges, two power ranges, and one linear wide range) is systematically being replaced with Gamma-Metrics nuclear instrumentation (four drawers--three wide range channels and one linear wide range). To date, two of the wide range instrument drawers and their associated detectors have been installed, which replaced four of the old General Electric instrumentation drawers. These two drawers provide the following five channels of neutron flux measurement and safety system input: one source range indication, two intermediate range indications and period trips, and two power range indications and high power trips. One General Electric drawer is still providing the third

power range indication and high power trips. The General Electric linear wide range channel is still being used to maintain the reactor power level in automatic control. Projected completion date: December 2002

Replacement of the 980A/B Temperature Transmitters, Meter Relay Units, and RTDs
Temperature Elements (TEs) 980A and 980B provide an indication of primary coolant heat exchanger outlet temperature and the outputs produced by these TEs provide inputs to the reactor safety system. The Resistance Temperature Detectors (RTDs), Alarm Meter Units, and Temperature Transmitters for TE 980A and 980B were replaced. The replacement was necessitated by the unavailability of replacement parts for the transmitters and meter relay units. The former temperature measurement arrangement utilized a 10 to 50 milliamp system; a configuration that is no longer supported by the industry. The new system operates on the industry standard of 4 to 20 milliamps.
Date Completed: March 2001

Replacement of DPS 929, 928A, and 928B Dual Trip Units
The differential pressure sensors (DPSs) that measure the differential pressure across the reactor core (DPS 929) and primary coolant heat exchangers (DPS 928A/B) provide an indication of the primary system flow and the outputs produced by these sensors provide inputs to the reactor safety system via the dual trip units. This modification was the replacement of the original Honeywell dual trip units with new Moore Model FCA trip units. Because of the unavailability of replacement parts, the dual trip units had become obsolete.
Date complete: December 2000

Replace PT 944B Meter Relay Unit with a New Simpson Meter
Primary coolant system pressure is measured at the reactor core outlet by Pressure Transmitter (PT) 944B. The LFE Series 195 meter relay unit for PT 944B was replaced with a new Simpson 3324 meter relay unit. The output signal produced by this transmitter provides an input to the reactor safety system. Because of the unavailability of LFE meter relay replacement parts, the unit had become obsolete.
Date completed: February 1999

Reactor Control Power Upgrade
The Elgar Line Conditioner was replaced with an Uninterruptible Power Supply (UPS) to provide reactor instrumentation and control power. It protects against line transients, line noises, and loss of power during the transitional period between a complete loss of facility electrical power and when the Emergency Diesel Generator assumes the emergency electrical loads. A line conditioner regulates small monitoring voltage fluctuations, whereas a UPS regulates the output for even a complete loss of input voltage.
Date completed: April 1990

Reactor and Pool 100-ohm RTDs
The temperatures in the primary and pool coolant systems hot and cold legs are measured by Resistance Temperature Detectors (RTDs). The original 10-ohm RTDs were replaced

with 100-ohm RTDs. The 100-ohm RTDs provide a 0.227 ohm/°F change over the temperature range of 0 °F to 200 °F, where the 10-ohm RTDs provide a 0.0218 ohm/°F change over this range. The increased ohm/°F obtained by using the 100-ohm RTDs makes the temperature indications less susceptible to variations due to small resistance changes, i.e. terminal board connections, wire connectors, etc. in the temperature indicating current loop. The primary coolant system hot leg temperature monitor also provides an input to the safety system.
Date completed: April 1987

Nuclear Instrumentation Power Supply Modification
A fused parallel power supply was installed so that the nuclear instrumentation system, part of the reactor safety system, and control rod indication were no longer powered through a common fuse (2F1). This eliminated the situation where a single fuse failure would secure power to both primary indications of a reactor shutdown.
Date completed: May 1979

Replacement of FS-928A/B and 929 with DPS 928A/B and 929
Paddle-type flow switches were replaced with differential pressure sensors (DPSs) to monitor differential pressure across the reactor core (DPS 929) and primary coolant heat exchangers (DPS 928A/B), which provide an indication of system flow and input to the reactor safety system. The flow-type paddle switches proved to be unreliable, causing spurious scrams and were difficult to maintain.
Date completed: November 1973

Radiation Monitoring System
Replacement of the Audible Alarm Units in the ARMS
The continuous tone audible alarm units in the Area Radiation Monitoring System (ARMS) remote indicators were replaced with intermittent tone audible alarm units. The intermittent tone provides a discernable sound on the beamport floor and reactor bridge such that background noise does not mask the alarm. This ensures that personnel at these locations will promptly respond as required to a high radiation level alarm.
Date completed: October 2000

Relocation of Nuclepore ARMS to the East Wall
An Area Radiation Monitoring System (ARMS) station was relocated from the west side of the biological shield to the east wall of the reactor containment building beamport floor. This station previously monitored an experiment that was abandoned. Since the experiment was no longer operational, this channel of the ARMS was not required at this location. The beamport floor east wall location was selected since the other three areas of the beamport floor – north, south, and west – have radiation monitors installed. This provided better utilization of the current ARMS stations.
Date completed: April 2000

Installation of Eberline Model PING-1A Stack Monitor
An Eberline Model PING-1A Stack Monitor was installed as a second monitor. The addition of a second stack monitor provides greater operational flexibility and reliability.
Date completed: April 2000

Stack Monitor Replacement
The MURR stack monitor, a Nuclear Measurements Corporation Model AM-221F installed in May 1973, was replaced with a Nuclear Measurements Corporation Model RAK-22ABIB-PB6 stack monitor. Its capabilities allow for quicker activity and concentration assessments during routine or emergency stack releases. This includes providing readouts as a function of time, average concentrations, and total releases for specified time periods.
Date completed: November 1999

Area Radiation Monitoring System (ARMS) Replacement
The original Tracer Lab Area Radiation Monitoring System (ARMS) was replaced with a new Eberline Radiation Monitoring System (RMSII) System. This system is also part of the reactor safety system due to four of the ARMS modules provide SCRAM trips and can initiate a reactor isolation. This modification also replaced the original General Electric Secondary Coolant Monitor (SCM) and Fission Product Monitor (FPM) instrumentation drawers and detectors with Eberline ARMS detectors and control room readout modules.
Date completed: February 1991

Engineering Safety Features
Replace Containment Back-Up Door Control Solenoid Valves
The reactor containment building ventilation system has two primary isolation doors and two back-up isolation doors that close on a reactor isolation. The four (4) 3-way solenoid poppet valves, which control the supply air to the back-up isolation doors, were replaced with equivalent Schrader-Bellows Model NC-N355-41-04853 solenoid valves. Because of the unavailability of replacement valves or spare parts, the solenoid poppet valves had become obsolete.
Date completed: March 2000

Sealing Containment Inside Surfaces
The reactor containment building internal surfaces were sealed with an elastomeric resin-water base copolymer compound (commercial name – Decadex). Addendum 2 of the Hazards Summary Report discusses the application of a sealant to the concrete containment walls to enhance leak proofing. The sealant used in this modification again only serves to enhance the leak proofing of the containment walls.
Date completed: October 1988

Containment Isolation Valve 16B Control Air System
The facility exhaust fan connects to the reactor containment building through a 16-inch exhaust line. The exhaust line has two isolation valves, 16A and 16B, that close on a reactor isolation. The four-way solenoid valve, which controls the air supply to valve 16B, was replaced with redundant three-way solenoid valves. This provided redundancy to the system to ensure that a failure of any one of the solenoid valves will not prevent valve 16B from closing during a reactor isolation.
Date completed: July 1974

Installation of new Containment Isolation 16-inch Valve
A second 16-inch isolation valve was installed in the containment building exhaust line. This provided redundancy for the isolation of the reactor containment building in the event of a potential release of radioactive gases.
Date completed: January 1974

5. *Please discuss the material condition of the primary water system and pool systems. Discuss steps to limit corrosion and degradation to these systems through the recapture period.*

The primary coolant system and pool coolant system are constructed of aluminum 6061-T6 components with the exception of the pool liner, which is aluminum 5052 or 5086, and a few items such as the pumps and heat exchangers, which are stainless steel. The pressure boundary components, i.e., the surfaces in contact with the coolant, are in good material condition.

Sargent & Lundy, an Engineering Firm hired by the University, conducted a visual inspection of the pool liner utilizing video cameras with video recorder in April and June 2000. Since the welds and adjacent areas of the aluminum pool liner are where corrosion is most likely to occur, the inspection of the liner focused on welds and the aluminum plate and components around the welds. No evidence of a number of potential corrosion mechanisms, and forms of linear distress, including cracks, deformations (including bulges), buckling, and tears (at anchorages or attachments) were found on the inspected welds and plates. The conclusion from the inspection was that, based on the condition after 34 years of reactor operations, an additional 34 years of good performance by the aluminum pool liner is expected with the operating conditions and operating procedures continuing as they have been.

Both the primary and pool coolant systems incorporate an ion exchange column for demineralization and conductivity monitors that read out in the reactor control room. The demineralization systems, besides maintaining a low conductivity, maintain the pH in a range around 5-6. Aluminum 6061-T6 aqueous corrosion resistance is high, especially in slightly acidified water in the pH range 4.5-7.0. The conductivity is recorded routinely as part of the process logs and the monitors provide an alarm to the console operator in the event of an above normal conductivity condition. Also, water samples are taken weekly from the primary coolant and pool coolant systems and are analyzed for pH and contained radioisotopes (Health Physics SOP VII "Pool and Primary Water Analysis"). Thus, the pH is monitored to be and maintained in an appropriate range to minimize corrosion and degradation of the aluminum piping and components beyond the recapture period.

6. *Please discuss the material condition of components (e.g., reactor pressure vessel, control rods and reflectors) subject to high neutron fluence including surveillance, scheduled component replacement, and future planned component replacements during the recapture period.*

The MURR reactor is designed so that there are five components/regions that receive a high fluence of neutrons: inner and outer pressure vessels, island tube sample holder, control blades, beryllium reflector, and graphite reflector region wedges. The reactor was designed so all of these components are replaceable. The material condition, surveillance, scheduled component replacement, and future planned replacements during the recapture period are discussed by component.

Pressure Vessels

The inner and outer pressure vessels are in good material condition and are the only structural components in high neutron fluence that have not been replaced nor are planned to be replace. The pressure vessels are capable of lasting for the 60-year planned operating history of the reactor. Both vessels are constructed from aluminum alloy 6061-T6. The vessels are designed with a significant margin between the maximum design stress and the allowed stress limit for aluminum 6061-T6. The pressure vessels have operated in a temperature and neutron environment that have either maintained or increased their material strength. These conclusions are supported by the following paragraphs.

The pressure vessels separate the pressurized primary coolant system from the open pool system. The pressure vessels are located completely inside the reactor pool. A break in either pressure vessel would cause a primary coolant system leak into the pool system and a primary coolant low pressure scram. There are four independent primary coolant low pressure scrams and each one can safely shutdown the reactor. Therefore, a break in the pressure vessel does not cause a reactor safety problem, but only prevents operating the reactor. The failed pressure vessel would have to be replaced before the reactor could be restarted. A spare inner and outer pressure vessels are on hand.

The environment in which the pressure vessels operate can affect their material condition: stress, temperature, neutron fluence, and neutron spectrum. The primary coolant system design pressure is 100 psig or 0.689 MPa. The limiting safety system setting (LSSS) for the primary coolant pressure is 75 psia at the primary pressurizer. The pressurizer connects to the primary coolant system before the primary system piping enters the reactor pool; therefore the pressure vessels are at the pressurizer pressure only when the primary pumps are not operating. During normal reactor operation, there is approximately 3,800 gpm flow in the primary coolant system, which results in a lower pressure in the pressurizer due to the flow caused pressure drop.

The design pressure of 100 psi is used to calculate the pressure vessel stress. The outer pressure vessel has an OD=12.55 inches and an ID=11.925 inches. Therefore the wall thickness is 0.3125 inches. The inner pressure vessel has an OD=5.06 inches [in the

vertical groves--smallest OD] and an ID=4.50 inches. Therefore the wall thickness is 0.280. The stress on the pressure vessels can be calculated from these values.

- The stress on the outer pressure vessel = internal pressure x radius/thickness
  Stress = 0.689 MPa x 5.963"/0.3125" = 13.1 MPa
- The stress on the inner pressure vessel = external pressure x OD/thickness
  Stress = 0.689 MPa x 5.06"/0.280" = 12.5 MPa

The original design calculations gave 8500 psi = 58.6 MPa as the allowed stress limit for not welded 6061-T6. There are no welds on the portion of the pressure vessels located in any significant neutron flux.

Both pressure vessels have the primary coolant system on the high-pressure side and the pool coolant system on the low-pressure side. During normal reactor operations, the temperature range for the primary coolant system falls within 48-60 °C and for the pool coolant system falls within 38-48 °C. Therefore the pressure vessels temperature stays below 100 °C. This maintains the tempered strength of 6061-T6 and as calculated above, provides a significant margin from the allowed stress limit.

The pressure vessels have been in the reactor during the approximately 89,500 MWD of operation. This has resulted in the peak fast fluence [>0.1 Mev] received by the inner pressure of 1.96E27 n/m^2. The peak thermal fluence received by the inner pressure vessel is 1.52E27 n/m^2. The peak fast fluence received by the outer pressure is 1.54E27 n/m^2. The peak thermal fluence received by the outer pressure vessel is 1.35E27 n/m^2. The peak thermal flux has generated approximately 3 wt% Silicon in the inner pressure vessel by transmutations. Silicon is insoluble in aluminum at temperatures below 200 °C. The silicon precipitates are responsible for most of the radiation strengthening in 6061-T6 alloy, discussed later.

The reference for the effect of the radiation environment on material properties of aluminum 6061-T6 is the 1995 report by Dr. Kenneth Farrell of ORNL: "Assessment of Aluminum Structural Materials for Service Within the ANS Reflector Vessel." Figure 4.4 in the report shows how the strength and elongation of 6061-T6 vary as a function of thermal neutron fluence. This indicates that the ultimate strength and yield strength both increase with thermal neutron fluence above 10^{24} to 10^{25} n/m^2. The maximum thermal fluence indicated in Figure 4.4 is 4E27 n/m^2. Assuming our current operating schedule continues through October 2006, the peak thermal fluence on the pressure vesses will be 1.82E27 n/m^2, or less than half of the maximum value in Figure 4.4. Therefore the high neutron fluence does not put the pressure vessels at risk to fail due to stress during the time period of construction permit recapture.

The measured swelling of 6061-T6 as a function of fast neutron fluence is displayed in figure 4.6 of the previously referenced ORNL report. The figure shows the total swelling and the swelling attributable to the Silicon produced by the thermal fluence. The fast fluence causes swelling due to producing microscopic voids. The total swelling includes swelling caused by both the void formation and the silicon production. It indicates that a fast neutron fluence of 1.7E27 n/m^2 with a thermal to fast flux ratio of 2 would cause

approximately 2% swelling. This includes 1% void swelling due to the fast fluence and 1% Silicon swelling due to the 3.4E27 n/m^2 thermal fluence. Assuming our current operating schedule continues through October 2006, the peak thermal fluence on the pressure vessels will be 1.82E27 n/ m^2 and the peak fast flux will be 2.35E27 n/ m^2. This would cause a peak swelling of approximately 1.9%. Therefore the swelling will not cause a significant increase in stress during the time period of construction permit recapture.

Fatigue stress will not be a problem for the inner or outer pressure vessels. The reference used for fatigue stress is the 1993 report by Dr. G. T. Yahr of ORNL: "Fatigue Design Curves for 6061-T6 Aluminum." Based on figure 2 "Design Fatigue Curve is Constructed by Reducing Fit to Data by Factor of 20 on Life and 2 on Stress" from this report, the infinite lifetime stress for fatigue is 50 MPa [$>10^7$ cycles]. The maximum cyclic stress for the MURR pressure vessels is the transitions between being pressurized and depressurized as part of starting up and shutting down the reactor. This results in a pressure change of 60 psi causing a stress of approximately 8 MPa, less than 20% of the infinite lifetime stress limit. If it were assumed that this occurs 200 times per year (more typical is around 70-80), there would be 12,000 cycles over a 60-year operating history.

Island Tube Sample Holder
The island tube sample holder, referred to as the "six-barrel flux trap" at MURR, is also made of aluminum 6061-T6. It is designed to hold aluminum irradiation cans containing samples in the high neutron flux region of the reactor defined in the Technical Specifications as the "Flux Trap." Part of the pool coolant flows down through the island tube sample holder to cool the samples and the holder. As previously stated concerning the pressure vessels, the aluminum 6061-T6 material properties perform well in the high neutron flux/pool coolant system environment. The design and construction of the island tube sample holder have changed over the operation of the reactor to accommodate changes in the irradiation needs. The six-barrel flux trap design island tube sample holder was put into service in December 1999. Therefore it has received significantly less neutron fluence than the pressure vessels and by design has very little stress on the irradiated portion. The island tube sample holder is unloaded and loaded every week and the material condition is observed during this process. If ever the need arises, the island tube sample holder can be and would be replaced.

Control Blades
The reactor has four control blades that travel in the water channel between the outer pressure vessel and inside surface of the beryllium reflector. The material condition of the control blades and their associated components are maintained by removing and replacing one of the four shim blades every six months so that each blade is inspected every two years. This complies with Technical Specification 5.3, which defines the surveillance requirements for the reactor control blade system. The control blade travel and alignment is determined by the offset mechanism to which the blade is attached. A spare offset mechanism is disassembled and reconditioned, e.g., replace bearing, etc., before it is reinstalled for use. The control blades are inspected to validate their proper material condition before they are used or reused. The control blades are typically used

for two to three times, i.e., four to six years of being in the reactor. The offset mechanisms and/or control blades are permanently removed from service if there is a concern about their material condition.

The operational condition is also monitored during routine operations. The drop time of each of the four reactor shim blades are measured at least quarterly to verify they meet the limits in Technical Specifications (Compliance Procedure CP-10 "Rod Drop Times").

These practices that are maintaining the control blades in appropriate material condition will continue through the recapture period.

Beryllium Reflector
The reactor is designed with a cylindrical sleeve beryllium reflector located around the section of the outer pressure vessel that contains the reactor core. The neutron fluence causes tensile stresses due to differential irradiation induced swelling within the beryllium reflector. The radiation heating of the beryllium causes thermal stresses when the reactor is operating. The first beryllium reflector cracked due to the combined stresses before it was replaced in 1981. Based on this experience, the beryllium reflector is replaced approximately every 26,000 MWD of reactor operation before the combined stresses can cause the reflector to crack. Since 1977 the reactor has been operated at 10 MW for 90% of the time each year, so the beryllium reflector has been replaced every eight years. After the 1981 replacement, it was replaced again in 1989 and 1997. The beryllium reflector was inspected when removed and no cracks or poor material condition was observed. A new beryllium reflector is scheduled to be installed in the Fall of 2005.

Graphite Reflector Region
The design also includes a graphite reflector region outside the beryllium reflector. This region is made up of 12 removable reflector elements, which can be reconfigured to provide sample irradiation positions. The elements are designed to accommodate sample holders of various sizes, the irradiation tips for the pneumatic tube system, and the beam ports. With the reactor being an open pool reactor, the reflector elements can be observed and monitored routinely. The irradiation elements are replaced when a different sample holder size or location needs to be accommodated, or when the condition of the element indicates it needs to be replaced. All reflector elements have been replaced at least once. With the routine monitoring, the graphite reflector elements will be maintained in good material condition and replacements will be performed on an as-needed basis. With the frequency of element replacement, additional elements may be replaced during the recapture period.

7. *Please discuss the material condition of the radiation and effluent monitoring system.*

The radiation and effluent monitoring system can be broken down into three components. The first, TLD dosimetry is provided by a NVLAP certified vendor. TLD's are changed out on a quarterly basis (Technical Specification 6.1(h)(4)(g)) and the results are supplied

annually to the NRC via the University of Missouri Research Reactor Operations Annual Report.

Two continuously operating stack monitors capable of detecting radionuclide gases, iodines and particulates provide air effluent monitoring. One system (NMC-RAK) has been operational for approximately ten years while the newest (Eberline PING) has been on-line for about one and one-half years. All systems are checked for operations at least weekly (Reactor Startup Check Sheet SOP/A-1) and calibrated on a semi-annual basis (Health Physics SOP III-3 through III-10). Operational checks are performed to satisfy operations procedures to ensure that the monitoring system performs as designed while semi-annual calibrations are performed to satisfy the requirements of the Technical Specifications. Routine operational checks indicate that the equipment functions properly as designed. Any discrepancies are noted with the HP group for follow up to ensure equipment functionality.

Sanitary sewer effluents are routinely monitored via Liquid Scintillation Counting (Packard 2300 TR) and HPGe detection systems. These systems are checked for operation at least weekly and calibrated on a semi-annual basis (Health Physics SOP III-19 through III-21). The calibration and check frequencies are based on MURR operational requirements. The Packard is approximately six years old and the HPGe systems are of various ages.

The area radiation monitoring system consists of 12 modules located in the control room tied to 12 detectors at various locations throughout the facility. The entire system (Eberline) is approximately ten years old and the components are repaired or replaced as necessary. Calibrations are performed semi-annually on the entire system.

The overall condition of all three components of the environmental monitoring system is sound. Replacements will be made if failure of any single component occurs or is reasonable anticipated. See the equipment descriptions in response to Question #4 for additional detail

8. *Please discuss the results of any inspections of core components that have been carried out (do not include inspections of fuel elements).*

The inspection of core components is discussed by component.

Beryllium Reflector

MURR replaced the first beryllium reflector in 1981 due to it cracking before the planned replacement. Based on this experience, the beryllium reflector is replaced approximately every 26,000 MWD before the neutron fluence can cause tensile stresses combined with thermal stresses sufficient to cause the beryllium to crack. After the 1981 replacement, it was replaced again in 1989 and 1997. The beryllium reflectors were inspected after removal and no cracks or poor material condition were observed. This methodology for maintaining the beryllium reflector material condition has performed well.

Pressure Vessels
There are no required inspections of the pressure vessels. However, since the reactor is refueled at least weekly and the pressure vessel head has to be removed to refuel, a portion of the vessels is inspected weekly. All eight fuel elements are replaced each week. In performing the refueling, an operator is looking closely at the outer surface of the inner pressure vessel and the inner surface of the outer pressure vessel, while he is raising and lowering fuel elements through this region. Additionally, the fuel elements have rollers at the top and bottom of both the inside surface and the outside surface of the fuel element. The rollers on the inside surface, fuel plate 1 side, are concave rollers. The rollers on the outside surface, fuel plate 24 side are convex rollers. The concave rollers roll on the outside surface of the inner pressure vessel. The convex rollers roll on the inside surface of the outer pressure vessel. This provides the reactor operator performing the refueling an excellent feel of the pressure vessel surfaces. Thus the refueling provides a weekly material condition check of the pressure vessels.

The pressure vessels receive a close visual inspection during beryllium reflector change out. With the control blades and beryllium reflector removed, the full length of the outer pressure vessel can be visually inspected. The upper extension of the outer pressure is removed so that the top of the installed outer pressure is a split ring flange. The split ring flange is 64 inches above the core centerline. An inspection is performed, with the observer located just above the split ring flange end of the outer pressure vessel and with their head at or below the upper end of the inner pressure vessel. This provides an excellent visual inspection between the pressure vessels where the reactor core is located and of the outside of the outer pressure vessel where the control blades are located during reactor operations. This inspection has been performed during each beryllium reflector replacement and has found the material condition of the pressure vessel surfaces in good condition. A video recording was made of the pressure vessel surfaces during the 1989 and 1997 beryllium change outs.

Control Blades
Inspection of control blades is discussed in answer 6.

Graphite Reflector Elements
There are no required inspections of the reflector elements, but with the reactor being an open pool type design, the reflector elements can be observed and monitored routinely. The reflector elements are separated from the reactor core, i.e., fuel elements, by the beryllium reflector and the outer pressure vessel. The irradiation elements are replaced when a different sample holder size or location needs to be accommodated, or when the condition of the element indicates it needs to be replaced. All reflector elements have been replaced at least once.

Pool Liner
Sargent & Lundy conducted a visual inspection utilizing video cameras with a video recorder in April and June 2000. Since the welds and adjacent areas of the aluminum reactor pool liner are where corrosion is most likely to occur, the inspection of the aluminum pool liner focused on welds and the aluminum plate and components around

the welds. No evidence of a number of potential corrosion mechanisms, and forms of linear distress, including cracks, deformations (including bulges), buckling, and tears (at anchorages or attachments) were found on the inspected welds and plates. The conclusion from the inspection was that, based on the condition after 34 years of reactor operations, an additional 34 years of good performance by the aluminum pool liner is expected with the operating conditions and operating procedures continuing as they have been.

543 & 546 valves

The in-pool portion of the primary coolant system contains a siphon break system and a reactor convective cooling loop. Each of these systems has automatic controlled redundant parallel valves. The siphon break system valves (543 valves) and the convective cooling loop valves (546 valves) are operationally checked as part of the reactor startup checks (OP-RO-410 "Primary Coolant System"). This check validates that each of the 546 valves are opening and closing. It also validates that the 543 valve actuators are going to the open and close positions, that both valves are closed when called to be, and that at least one valve opens when called to be. A surveillance test is performed monthly to validate that each valve opens when called to be open (Compliance Procedure CP-24 "Anti-Siphon System Valves 543 A/B").

The 543 and 546 valves are butterfly valves with o-rings on the valve actuating shaft and rubber seating surfaces. The valves are rebuilt during each beryllium reflector replacement and whenever the previously discussed operational checks determine it is needed.