UNIVERSITY *of* MISSOURI

RESEARCH REACTOR CENTER

January 6, 2012

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

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, Complex Questions (TAC No. MD3034)," dated May 6, 2010, and 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 May 6, 2010, the NRC requested additional information and clarification regarding the renewal request in the form of nineteen (19) Complex Questions. By letter dated September 3, 2010, MURR responded to seven (7) of those Complex Questions.

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) 45-Day Response Questions. By letter dated July 16, 2010, MURR responded to forty-seven (47) of those 45-Day Response Questions.

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

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

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

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

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

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

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

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

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

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

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

Attached are MURR's responses to four (4) of the remaining 45-Day Response and Complex Questions. Also attached is an updated version of the MURR Technical Specifications. Incorporated into the revised Technical Specifications include the following:

1. Responses to relicensing RAIs 12.2, 12.4, 12.5, 12.7, 12.8, 12.914.1, A.3, A.5, A.6, A.7, A.9, A.13, A.14, A.21, A.31, A.34, A.38, A.48, A.51, A.52 and A.53.
2. Amendment No. 34 to Amended Facility License R-103, "License Amendment on Fueled Experiment Conditions (TAC No. MD5782)," approved October 10, 2008.
3. Request to amend Amended Facility License R-103, "License Amendment, Center Test Hole (TAC No. ME1876)," dated August 6, 2009.
4. Request to amend Amended Facility License R-103 by revising the University of Missouri-Columbia Research Reactor Technical Specification 2.1, "Reactor Core Safety Limit," dated August 24, 2011 (ML 1123A088).

With these responses, the following 45-Day Response and Complex Questions remain unanswered: 4.7, 6.2, 13.4.b, 13.7, C.1 and C.3. With the exception of 4.7, the remainder of the RAIs are thermal-hydraulic questions that still require significant RELAP work. MURR continues to work with the University of Missouri Mechanical Engineering Department to answer RAI 4.7, which involves evaluating control blade thermal distortion. Significant modeling has been done to answer this question and we believe we are nearly finished and should have an answer in a week or two. Additionally, a significant section of Chapter 4 is being revised based on the recent request to amend the MURR Safety Limits (ML 1123A088).

Because of the newly revised Safety Limits (ML 1123A088) and because additional RELAP work is required to answer the thermal-hydraulic questions, MURR is requesting additional time to answer the remaining 45-Day Response and Complex Questions. The extension request will be discussed with our NRC Senior Project Manager.

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

Sincerely,

John L. Fruits
Reactor Manager

ENDORSEMENT:
Reviewed and Approved,

Ralph A. Butler, P.E.
Director

NOTARY PUBLIC STATE OF MISSOURI NOTARY SEAL

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

Enclosed:

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

Attachment 2: Reference 1 for RAI 14.15, Earl E. Feldman, *Implementation of the Flow Instability Model for the University of Missouri Reactor (MURR) That is Based on the Bernath Critical Heat Flux Correlation*, ANL-RERTR/TM-11-28, July 2011.

Attachment 3: Reference 2 for RAI 14.15, ANL Intra-Laboratory Memo from Earl E. Feldman to John G. Stevens, "Thermal-Hydraulic Effects of Reducing the Assumed Minimum Channel Thickness of the University of Missouri Research Reactor HEU Core by 10 Mils to 62 Mils," June 15, 2011.

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

Chapter 4

4.15 Section 4.6, Thermal Hydraulic Design, Page 4-43.

It is unclear what the thermal design limit and the operating limit are. Provide further clarification for the temperature limit of the fuel cladding. Discuss the effect of fuel oxide layer build-up on the results of the analyses.

The temperature limit of fuel cladding is derived by testing; however, calculations show that the actual temperature during normal operations is much lower, and that the "worst-case" is actually a fresh fuel element, rather than a high-burnup element exhibiting an oxide layer.

One of the quality control steps in the MURR fuel element fabrication process is the fuel plate blister test. As required by the fuel fabrication specifications ("Specification TRTR-4 for University of Missouri-Columbia Fuel Elements Assembled for University of Missouri Research Reactor," Revision 4, June 8, 1994), every fuel plate that is assembled into a MURR fuel element is heated to 900 °F to verify that no blisters form; therefore, this temperature is used as the basis for the thermal design and operating limits. However, the actual fuel plate temperature does not even remotely approach this value during normal reactor operations. The following analysis calculates the fuel temperatures during operation and the effect of oxide layer build-up on the surface of the fuel element that occurs while it is being irradiated.

Question 16.1 of the NRC relicensing review raised a related issue. There the concern is the effect on thermal performance due to a reduction in a coolant channel gap as a result of fuel meat swelling from burnup and the buildup of oxide on the cladding surface that occurs while the fuel is being irradiated. Although fresh fuel elements do not have the reductions in channel gap that occurs with burnup and do not have the added thermal resistance on the fuel plate surfaces as a result of the buildup of oxide layers, they do have the highest power peaking factors. Thus, calculation of peak fuel temperature addressed both the most limiting fresh fuel element and the most limiting high-burnup fuel element for upper bounding steady-state operation parameters of the MURR.

Steady-state thermal analysis of the most limiting fresh fuel element is provided in considerable detail in Reference 1. Steady-state thermal analysis of the most limiting high-burnup fuel element is provided in Reference 2. The thermal analyses in these two references, which focused on Safety Limits (SL) for flow instability and critical heat flux (CHF), provide the tool to calculate the heat fluxes and bulk coolant temperatures of the limiting coolant channel at 1-inch intervals over the last 13 inches of the 24-inch fuel meat length. This portion of the fuel length is of particular interest because for both fresh fuel and fuel with high burnup it includes the highest heat fluxes and highest bulk coolant temperatures of the element. This is due to the control rods being at the top outer edge of the core and the downward core coolant flow through the channels. These analyses, which already include all appropriate uncertainty, or hot channel factors, can be easily extended to provide the peak fuel temperatures for the most limiting steady-state operating conditions that are being sought.

Reference 1 provides a very detailed sample problem solution to promote clarity of the analytical model. The sample problem solution determined the SL for reactor power with a reactor inlet water temperature of 155 °F, a pressurizer pressure of 75 psia, and a total primary coolant flow rate of 3200 gpm. This combination of temperature, pressure, and flow corresponds to the Limiting Safety System Setting (LSSS) conditions as defined in the MURR Technical Specifications. The coolant channel chosen for analysis, channel 2 of fuel element 1, is the innermost channel bounded by two fuel plates and is the most limiting channel in the reactor. This sample problem with relatively minor input changes to account for the narrowest channel in the high-burnup fuel element was used in Reference 2 where the most limiting channel was found to be channel 2 of fuel element 8 – the most limiting high-burnup element in the reactor. The analytical model was implemented via modifying the computer spreadsheet, which is shown in Table 8 of Reference 1.

The purpose of this analysis is to show what the upper bounding steady-state temperatures associated with fuel elements are for normal operating conditions.

The computer spreadsheets used in References 1 and 2 were extended for use in the current analysis. The following operating parameters are listed with assumed and normal values.

Parameter	Assumed Value	Normal Value
Power Level	10.5 MW	9.9 ±0.1 MW
Reactor Inlet Water Temperature	130 °F	122 ±2 °F
Primary Coolant Flow	3700 gpm	3800 ±50 gpm
Pressurizer Pressure	80 psia	84 ±2 psia
Pressurizer Level	-6 inches	+3 to -6 inches

These assumed values will provide fuel temperatures that are higher than those actually reached in the fuel elements during normal operation, due to the slightly conservative parameter conditions assumed. Also, Reference 1 and 2 spreadsheets assume an additional 1.258 factor on the hot spot heat flux peaking factors and an additional 1.127 factor on the coolant channel enthalpy rise. Changing the additional 1.258 factor to 1.0 reduces the peak fuel meat centerline temperature from 311.3 °F to 278.6 °F. Reference 1 assumes a coolant channel gap of 72 mils instead of the nominal 80 mils which results in a hot channel flow rate reduction factor of 0.8197. Reference 2 assumes a coolant channel gap of 62 mils instead of the nominal 80 mils which results in a hot channel flow rate reduction factor of 0.6363. Equations were added to both spreadsheets to enable the maximum fuel meat temperature at each of levels 12 through 24 to be determined. At each level the film (or boundary layer) temperature rise was determined and a one-dimensional conduction model was used to calculate the temperature rise from the fuel plate outer surface to the vertical center plane of the fuel plate. Since the ratio of the inner surface radius of the fuel plate to that of the outer is very close to 1.00, the analysis of the curvature of the fuel plate is ignored and the plate is treated as a flat plate.

At each axial level, the temperature rise, or ΔT, through each material – the oxide layer, the aluminum cladding, and from the fuel meat edge to the vertical center plane of the fuel meat – is determined using the basic formula: $\Delta T = q'' \times R$, where q'' is the heat flux, or power per unit of heat transfer area at the surface of the fuel plate, and R is the thermal resistance.

For the oxide layer and the aluminum cladding, the thermal resistance is the material thickness divided by the material thermal conductivity. For the fuel meat, the thermal resistance is half of the distance from the edge to the vertical center plane of the fuel meat divided by the thermal conductivity of fuel meat. Tables 1 and 2 summarize the values for thickness and thermal conductivity used in the analyses and the formulae and values of thermal resistance. The fuel meat formula has a divisor of 4 rather than 2 because the entire fuel meat thickness is used rather than the distance from the edge to the fuel meat centerline.

The differences between Table 1, which is for fresh fuel, and Table 2, which is for high-burnup fuel, are shown in red. The value of thermal conductivity for fresh fuel meat was obtained from

Reference 3. In Table 4 on page 22 of this reference, the UAl_x volume percentage of 35.4 was added to the percent porosity of 6 to obtain a total value of 41.4. Then in Figure 8 on page 27 of the reference, at 41.4 volume percent (fuel + voids), the open square symbols indicate that the thermal conductivity is 40 to 45 W/m-K; therefore, 40 W/m-K was selected. Burnup tends to reduce the thermal conductivity of the fuel meat. A lower bound for the value of the high-burnup MURR HEU fuel of 30 W/m-K is based on the judgment of Reference 4. As the results will show, the peak temperature rise for the fuel meat for both fresh and high-burnup fuel meat is less than 14 °F.

Oxide layer build-up was factored into the analyses as follows.

In examining the fresh fuel element case, reflected in Table 1, the oxide layer thickness and thermal resistance are zero because an oxide layer has not yet formed. In the case of the high-burnup element in Table 2, the oxide layer thickness is assumed to be 0.00127 inches. The values for oxide layer thickness and thermal conductivity were obtained from Reference 5. As described in Reference 5, oxide thickness of fuel plate-24 was measured on nineteen MURR fuel elements in 1987 using the Advanced Test Reactor's (ATR) oxide measurement equipment. Based on these measurements, 0.61 mils was determined to be the maximum oxide thickness of the average depleted fuel element and 1.27 mils would represent the "worst case" oxide thickness, which would occur on fuel plate-1.

Table 1 – Fuel Plate Thermal Resistance for Fresh Fuel

	Fuel Meat	Cladding	Oxide Layer
Thickness [t]			
Inches	0.020	0.015	0
Millimeters	0.508	0.381	0
Thermal Conductivity (W/cm^2) [k]	40	157	2.25
Thermal Resistance [R]			
Formula	t/(4 k)	t/k	t/k
Value (m^2-C/W)	3.18×10^{-6}	2.43×10^{-6}	0

Table 2 – Fuel Plate Thermal Resistance for High-Burnup Fuel

	Fuel Meat	Cladding	Oxide Layer
Thickness [t]			
Inches	0.020	0.015	0.00127
Millimeters	0.508	0.381	0.0323
Thermal Conductivity (W/cm^2) [k]	30	157	2.25
Thermal Resistance [R]			
Formula	t/(4 k)	t/k	t/k
Value (m^2-C/W)	4.23×10^{-6}	2.43×10^{-6}	1.43×10^{-5}

Figures 1 and 2 provide the axial distributions of bulk coolant temperature and fuel plate surface heat flux for levels 12 through 24 (the exit) of the limiting fresh and high-burnup elements, respectively. The highest heat flux in the fresh fuel case occurred at axial level 14, where the heat flux of the higher power of the two plates bounding the channel was 4.116 times the core average heat flux. The axial level with the highest fuel meat temperature was found to be level 16. The heat fluxes at levels 15 and 16 of this plate are each 4.114 times the core average. The bulk coolant temperature rises 19.7 °F from level 16, where the peak fuel temperature occurs, to the exit. However, because this is more than offset by the effect of the decrease in heat flux over this length, the peak fuel temperature is 40.8 °F lower at the last fuel axial level than at the 16th, where the peak fuel meat temperature occurs. A 6.3% increase in film coefficient over this length also helps in this regard. This increase is largely the result of a decrease in the value of viscosity due to the rising bulk coolant temperature. However, for the high-burnup fuel, the highest heat flux is at the same level of the peak fuel temperature, level 17, and is 3.970 times the core average heat flux. The bulk coolant temperature is 15.6 °F higher at the exit than at level 17, but the peak fuel temperature is 36.4 °F lower at the exit then at level 17.

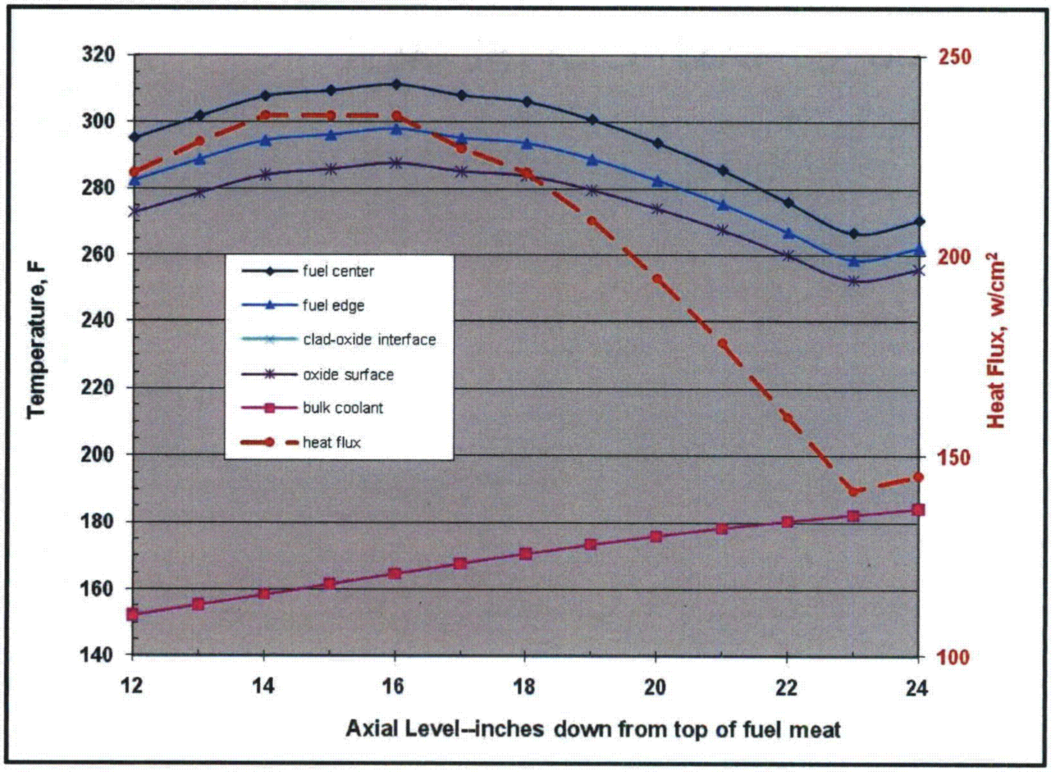

Figure 1 – Temperatures and Heat Fluxes for Fresh Fuel

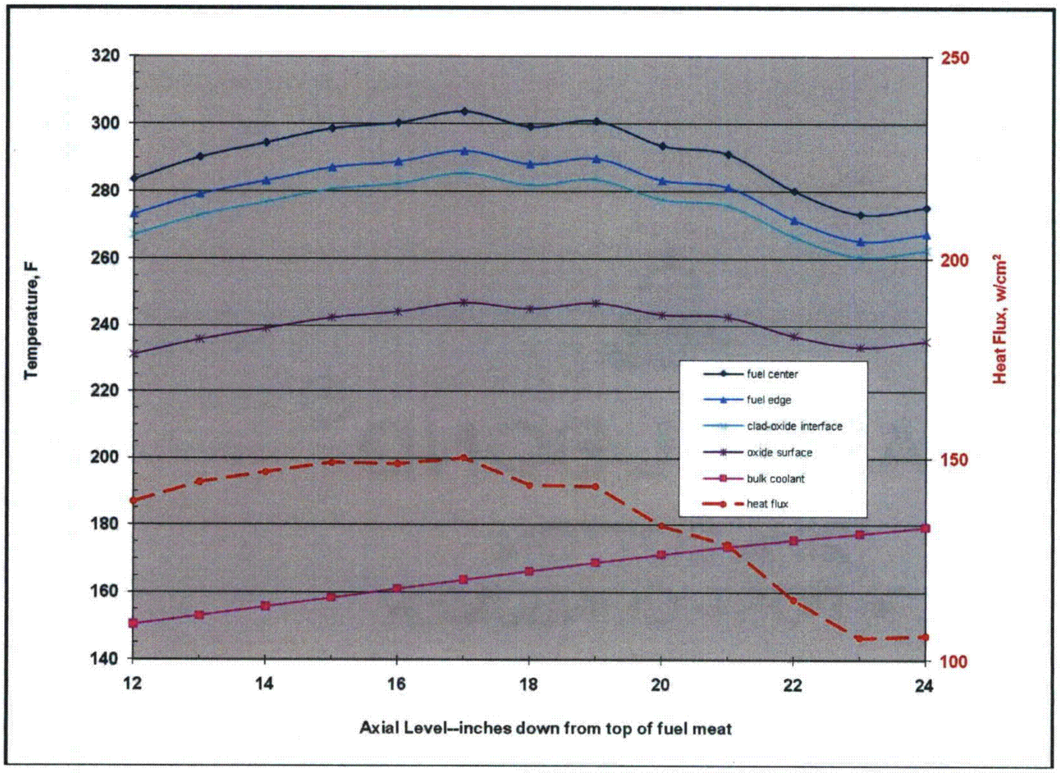

Figure 2 – Temperatures and Heat Fluxes for High-Burnup Fuel

Table 3 summarizes key results for level 16 of the fresh fuel and level 17 of the high-burnup fuel, where the peak fuel temperatures occurred. The upper section of the table shows results taken from References 1 and 2 spreadsheets with the reactor operating conditions as listed above with the power at 10.5 MW. The lower section of the table shows key results that were obtained by extending the earlier analysis so that the film coefficient at each axial level could be determined via the well-known Dittus-Boelter relationship (Nusselt number = $0.023 \times [\text{Reynolds number}]^{0.8} \times [\text{Prandtl number}]^{0.4}$). In the analysis, a hot channel, or uncertainty, factor of 1.2 was applied by dividing the values of film coefficient obtained via the Dittus-Boelter relationship by 1.2. In Table 3, temperatures and temperature rises are shown in red.

Table 3 – Calculation of Peak Fuel Temperature at Limiting Axial Level

	Fresh Fuel	High-Burnup Fuel
Limiting Axial Level	16	17
Hot Channel Heat Flux (W/cm^2)	234.8	150.1
Bulk coolant temperature (°F)	164.7	163.7
Density (kg/m^3)	975.3	975.6
Hot Channel Local Velocity (ft/s)	22.19	20.02
Hydraulic Diameter (inches) (mm)	0.13876 (3.525)	0.1201 (3.051)
Viscosity (micro Pa-s)	384.4	386.8
Reynolds Number	60499	46898
Prandtl Number	2.40	2.41
Nusselt Number, Dittus-Boelter	218.3	178.5
Thermal Conductivity of Coolant (W/m-K)	0.666	0.666
Film Coefficient (W/m^2-K)	41249	38960
Hot Channel Factor on Film Coefficient (HCF)	1.2	1.2
Film Coefficient with HCF (W/m^2-K)	34374	32467
Film Temperature Rise with HCF (°F)	123.0 ΔT	83.1 ΔT
Oxide Surface Temperature (°F)	287.6	246.9
Oxide Temperature Rise (°F)	0.0 ΔT	38.7 ΔT
Temperature at Oxide-Cladding Interface (°F)	287.6	285.6
Aluminum Cladding Temperature Rise (°F)	10.3 ΔT	6.6 ΔT
Temperature at Cladding-Fuel-Meat Interface (°F)	297.9	292.1
Fuel Meat Temperature Rise (°F)	13.4 ΔT	11.4 ΔT
Temperature at Fuel Center (°F)	**311.3**	**303.6**

Figures 1 and 2 provide the axial distributions of bulk coolant, oxide surface, clad-oxide interface, fuel edge and fuel center temperatures of the levels 12 through 24 (the exit) of the limiting fresh and high-burnup elements, respectively. In Figure 1 the oxide surface temperature and clad-oxide interface temperature are coincident since the oxide layer, in effect, has a thickness of 0.0 inches. The red dashed curve in each of these figures provides the heat flux distribution, which is indicated in W/cm^2 along the right axis. As these figures and the bold red values in Table 3 show, the peak fuel temperatures for the fresh and high burnup fuel are 311.3 and 303.6 °F, respectively.

As Table 3 shows, at the limiting axial level the temperature rise from the bulk coolant to the fuel plate centerline is 311.3 – 164.7, or 146.6 °F for the fresh fuel case and 303.6 – 163.7, or 139.9 °F for the high-burnup case. In both instances the clad and fuel meat temperature rises are relatively small and most of the temperature rise is due to the film temperature rise, the temperature rise from the bulk coolant to the surface of the oxide layer or the surface of the fuel plate when there is no oxide layer. For the high-burnup case the oxide temperature rise is 38.7 °F. This value would need to increase by 7.7 °F, or 20%, for the peak fuel temperature of the high-burnup case to achieve the peak fuel temperature of fresh fuel case. Thus, the high-burnup fuel with its oxide layer has lower peak fuel temperatures than does the oxide-layer-free fresh fuel. Also, the blister temperature for aluminide fuel is over 900 °F (482 °C).

In conclusion, the predicted peak fuel temperature is 311.3 °F. It occurs in the fresh fuel rather than in the high-burnup fuel and is far from the cladding blister temperature.

REFERENCES
[1]Earl E. Feldman, *Implementation of the Flow Instability Model for the University of Missouri Reactor (MURR) That is Based on the Bernath Critical Heat Flux Correlation*, ANL-RERTR/TM-11-28, July 2011.
[2]ANL Intra-Laboratory Memo from Earl E. Feldman to John G. Stevens, "Thermal-Hydraulic Effects of Reducing the Assumed Minimum Channel Thickness of the University of Missouri Research Reactor HEU Core by 10 Mils to 62 Mils," June 15, 2011.
[3]J. E. Matos and J. L. Snelgrove, "Selected Thermal Properties and Uranium Density Relations for Alloy, Aluminide, Oxide, and Silicide Fuel," *Research Reactor Core Conversion Guidebook*, IAEA-TECDOC-643, Volume 4: Fuels Appendices I-1.1, IAEA, Vienna, Austria, 1992, pp. 13-29.
[4]Personal communication with Gerard Hofman, ANL, on February 13, 2009.
[5]University of Missouri Research Reactor Letter to U.S. Nuclear Regulatory Commission, "Response to NRC Request for Additional Information," In answer Number 5 on pages 34-35, September 11, 1987.

4.16 Section 4.6.1, Natural Convective Cooling Analysis, TS 2.1.c., Safety Limits, and TS 2.2.c., Limited Safety Systems Settings.

The TS and Safety Analysis Report (SAR) do not include a limitation on inlet or bulk coolant temperature. Explain the effect of coolant temperature on the analysis and justify why a limit on inlet or bulk coolant temperature should not be required.

Hazards Summary Report (Ref. 4.27 of the SAR) Section 5.5.3, *Analysis of Natural Convective Cooling of the Core*, provides a detailed safety analysis of natural convection cooling of the reactor core for initial low power operation. This analysis shows that the reactor can safely be operated up to a power level of 150 kW in the natural convection mode with the inlet coolant to the core being provided from the bulk pool at a temperature of 100 °F. The analysis calculates that at a power level of 150 kW, and with a coolant inlet temperature of 100 °F, the required local saturation temperature at the core hot spot is 227.2 °F. A water pressure of 20 psia has a saturation temperature of 227.96 °F; therefore, it is assumed that 20 psia of water head is required at the top of the fuel meat. For the lowest recorded Barometric pressure in Columbia of 14.11 psia, the required depth of pool water would be 19.99 feet, which would in turn provide 6.62 feet of 100 °F water above the top of the open pressure vessel and 7.08 feet of 112 °F water within the upper pressure vessel above the top of the fuel meat in the core. The 112 °F water temperature is based on a power level of 150 kW, a flow rate of 11.96 lb/sec, and 93% of the energy being deposited in the core and coolant within the pressure vessel.

To specify a limitation on bulk pool water temperature for natural convective cooling of the core, MURR suggests revising Limiting Condition of Operation (LCO) Technical Specification (TS) 3.4.b to include a limit of 100 °F for "Reactor Pool Temperature." In Mode III operation with natural convective cooling, the core inlet coolant temperature can only change very slowly because the source of core coolant is the very large volume of bulk pool water (approximately 20,000 gallons). Therefore, the Mode III Limiting Safety System Setting of 125% on reactor power level with a SCRAM set point at or below 62.5 kW provides more than adequate protection to avoid approaching the Mode III Safety Limit of 150 kW. TS 3.4.b and its bases will be revised as follows:

Specification

b. Sufficient instrumentation shall be provided to assure that the following limits are not exceeded during steady-state operation:

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

(1) Not required for Mode III operation.

(2) Reactor Pool Temperature limit is a maximum of 100 °F when in Mode III operation and (a) below 50 kW with the natural convection flange and reactor pressure cover removed or (b) with the reactor subcritical by a margin of at least 0.015 ΔK.

Bases

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

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

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

CHAPTER 13

13.4 Section 13.2.3, Loss of Primary Coolant.

a. *Explain why initial conditions of the analysis are not TS and license limits or provide an analysis at TS and license limits.*

The conservatively assumed reactor operating conditions, along with the normal operating values, used in the Loss of Coolant Accident (LOCA) analysis are provided below (also in Table 13-1 of the SAR).

Parameter	Conservative Assumption	Normal Condition
Reactor Power	11 MW	10 MW
Coolant Inlet Temperature	155 °F (68 °C)	120 °F (49 °C)
Core Inlet Flow Rate	3,800 gpm (14,385 lpm)	3,800 gpm (14,385 lpm)
Pool Temperature	120 °F (49 °C)	100 °F (38 °C)
Pressurizer Pressure	60 psig (414 kPa)[1]	62 - 66 psig (427 - 455 kPa)[1]
Anti-Siphon Pressure	26 psig (179 kPa)[1]	36 psig (248 kPa)[1]

[1]Pressure above atmosphere.

The following explains why the above stated conservative assumptions were selected for each of the reactor operating parameters used in the LOCA analysis at the start of the transient.

1. **Reactor Power** – Class 104c Amended Facility License R-103, issued by the U.S. Nuclear Regulatory Commission, authorizes the MURR to operate up to a maximum steady-state power level of 10 MW, so a very conservative steady-state power level of 11 MW is assumed at the start of the transient, which produces 10% more decay heat in the LOCA analysis. The high reactor power rod run-in and scram are currently set at 114 and 119%, respectively.

2. **Coolant Inlet Temperature** – MURR Technical Specification 2.2, Limiting Safety System Settings, and Technical Specification 3.3.a, Reactor Safety System, requires instrumentation necessary for two (2) reactor inlet water temperature scrams to be set at a maximum value of 155 °F. Therefore, a value of 155 °F is the most conservative assumption for coolant inlet temperature used in the LOCA analysis. The reactor inlet water temperature scram is currently set at 148 °F while the normal operating temperature is 120 °F.

3. **Core Inlet Flow Rate** – MURR Technical Specification 2.2, Limiting Safety System Settings, and Technical Specification 3.3.a, Reactor Safety System, requires instrumentation necessary for four (4) primary coolant flow scrams – two instruments per loop – to be set at a minimum value of 1625 gpm. Note: Since 50 gpm of the primary coolant flow is diverted to the cleanup (demineralizer) system, the actual total flow rate through the core equates to 3200 gpm. Additionally, a differential pressure across the core scram, which corresponds to a flow value of 3200 gpm, provides a backup to the primary coolant low flow scrams. However, a core inlet flow rate of 3800 gpm is assumed in the LOCA analysis. A higher flow rate is more conservative since it will result in less coolant remaining in the U-shaped section of primary coolant piping exiting the core. The primary coolant flow scrams are currently set at 1725 gpm (each loop) whereas the differential pressure across the core scram is currently set at 3400 gpm. As stated above, the normal core inlet flow rate is approximately 3800 gpm.

4. **Pool Temperature** – MURR Technical Specification 3.4.b, Reactor Instrumentation, requires sufficient instrumentation to assure that reactor pool water is limited to a maximum temperature of 120 °F. Since the pool coolant system serves as the heat sink during the accident, this assumed value is the most conservative temperature for the pool coolant system in the LOCA analysis. Normal operating temperature, as stated above, is 100 °F.

5. **Pressurizer Pressure** – MURR Technical Specification 2.2, Limiting Safety System Settings, and Technical Specification 3.3.a, Reactor Safety System, requires instrumentation necessary for four (4) primary coolant low pressure scrams to be set at a minimum value of 75 psia (which corresponds to the pressurizer pressure Limiting Safety System Setting with normal primary coolant flow). The RELAP model uses 60 psig as the steady-state pressurizer pressure, which is the most conservative value since the worst-case LOCA is a shear of the primary coolant cold leg inlet piping between the primary coolant isolation valve and the reactor pool. Primary coolant system pressure will then become 0 psig on the core side of the cold leg inlet valve immediately after the transient starts. The lower pressurizer pressure reduces the primary coolant system hot leg (core outlet) pressure and with the primary system break occurring before the core in the cold leg, this decreases the pressure differential that helps slow down the primary coolant flowing through the core.

6. **Anti-Siphon Pressure** – MURR Technical Specification 3.4.b, Reactor Instrumentation, requires sufficient instrumentation to assure anti-siphon system pressure is maintained at a value greater than 27 psig; therefore, a conservative pressure of 26 psig is assumed in the LOCA analysis. Since a lower anti-siphon pressure means less air will be injected into the inverted primary coolant loop, this assumption is most conservative.

13.6 Section 13.2.4, Loss of Primary Coolant Flow.

Explain why loss of pressurizer pressure is the most limiting initiating event for this class of event. Please discuss the effect on the analysis of fuel burnup and oxide layer build-up on the cladding.

A loss of flow (LOF) accident for the primary coolant system can be initiated by any one, or a combination, of the following anomalies:

(a) Loss of facility electrical power (or coolant circulation pump power);
(b) Inadvertent closure of coolant loop isolation valve(s);
(c) Inadvertent loss of pressurizer pressure;
(d) Locked rotor in a coolant circulation pump; and
(e) Failure of a coolant circulation pump coupling.

While the five types of LOF accidents listed above were analyzed, only the results of the "worst-case" accident [accident (c)] were discussed in the SAR. An inadvertent loss of pressurizer pressure is the "worst-case" accident because it results in slightly higher fuel peak centerline and coolant channel temperatures during the first 30 seconds of the transient, as shown in the SAR Figures 13.26 and 13.27, as compared to the other scenarios. When pressurizer pressure decreases to approximately 95% of the normal operating range it will cause both the primary coolant circulation pumps P501A and P501B to secure and primary coolant isolation valves V507A and V507B to start closing. This combines the effects of a loss of facility electrical power and the inadvertent closure of the coolant loop isolation valves.

The effects of fuel burnup and oxide layer build-up on the fuel plate cladding can be explained by considering the response to Request for Additional Information (RAI) 4.15. The analysis for RAI 4.15 assumed "worst-case" values for reactor power level, reactor inlet water temperature, primary coolant flow and pressurizer pressure. For the high-burnup, a worst case coolant channel gap of 62 mils is assumed. Based on oxide thickness measurements taken on nineteen MURR fuel elements in 1987, it was determined that 1.27 mils would represent the "worst-case" oxide thickness, which would occur on fuel plate-1; therefore, an oxide thickness of 1.27 mils is assumed on the high-burnup fuel plate. These assumptions result in temperatures for the primary coolant, fuel plate cladding and the fuel plate centerline to be higher than for normal operating conditions. However, the response to RAI 4.15, as presented in Figures 1 and 2 and Table 3 of the response, shows that the highest fuel plate temperature occurs in the hot spot of the low burnup fuel element. The burnup reduces the generated heat flux in the plate enough to reduce the temperature rise across the fuel meat, cladding and coolant film on the plate surface more than the new temperature rise across the oxide layer that is produced because of the long operating history of the fuel element.

ATTACHMENT 1

APPENDIX A

TECHNICAL SPECIFICATIONS

FOR

THE UNIVERSITY OF MISSOURI RESEARCH REACTOR

FACILITY OPERATING LICENSE R-103
DOCKET 50-186

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Introduction

The Technical Specifications represent an agreement between the licensee and the U.S. Nuclear Regulatory Commission (NRC) on administrative controls, equipment availability, operational conditions and limits, and other requirements imposed on reactor facility operation in order to protect the environment and the health and safety of the facility staff and the general public in accordance with Title 10, Chapter I of the Code of Federal Regulations, Part 50.36 (10 CFR 50.36).

This document is divided into the following six sections:

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

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

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

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

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

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

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1.0 DEFINITIONS

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

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

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

c. A reactor safety system component malfunction which renders or could render the reactor safety system incapable of performing its intended safety function unless the malfunction or condition is discovered during maintenance tests or periods of reactor shutdowns;

d. An unanticipated or uncontrolled change in reactivity in excess of 0.006 Δk. Reactor trips resulting from a known cause are excluded;

e. Abnormal and significant degradation in reactor fuel or cladding, or both; primary coolant boundary, or containment boundary (excluding minor leaks), which could result in exceeding prescribed radiation exposure limits of personnel or environment, or both; or

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

1.2 **Calibration or Testing Interval** - A calibration or testing interval is that period of time between normal checks for accuracy or operability of a system or component. To allow for some margin of time for proper scheduling and yet reasonably assure reliability, the calibration or testing interval shall be interpreted as follows:

Interval	Maximum Period Between Checks
Weekly:	9 days
Monthly:	6 weeks
Quarterly:	4 months
Semi-annually or greater:	Interval plus 2 months

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

1.0 **DEFINITIONS** - Continued

1.4 **Cold, Clean, Critical** - The cold, clean, critical condition is a reference to the reactivity state of the core with primary and pool coolant temperatures at 110 °F and negligible reactivity worth of xenon in the 8 fuel elements of the core (> 70 hours of decay after prior use for all 8 fuel elements).

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

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

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

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

1.9 **Instrument Channel** - An instrument channel is an arrangement of sensors, components, and modules as required to provide a single trip or other output signal relating to a reactor or system operating parameter.

1.10 **Instrument Channel Test** - An instrument channel test is the introduction of a simulated input signal to an instrument channel and the observation of proper channel response. When applicable, the test shall include verification of proper safety trip operation.

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

a. Greater than 0.10 megawatt-day;

OR

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

OR

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

1.0 DEFINITIONS - Continued

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

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

1.14 **Operable** - Operable means a system or component is capable of performing its intended function in a normal manner.

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

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

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

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

1.16 **Reactor Containment Building** - The reactor containment building is a reinforced concrete structure within the facility site which houses the reactor core, pool, and irradiated fuel storage facilities.

1.0 **DEFINITIONS** - Continued

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.

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

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

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

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

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

1.0 **DEFINITIONS** - Continued

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

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

OR

(2) Whenever all of the following conditions are met:

a. All four shim rods are fully inserted;

b. One of the two following conditions exits:

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

OR

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

c. No work is in progress involving the transfer of fuel in or out of the reactor core;

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

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

1.24 **Reactor Shutdown** - The reactor shall be considered shutdown when all four of the shim rods are fully inserted and power is unavailable to the shim rod drive mechanism electromagnets.

1.25 **Regulating Blade (Rod)** - The regulating blade (rod) is a low worth control blade (rod) used for very fine adjustments in the neutron density in order to maintain the reactor at the desired power level. The regulating blade (rod) may be controlled by the operator with a manual switch or push button, or by an automatic controller.

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

1.0 **DEFINITIONS** - Continued

1.27 **Safety Limits** - Safety Limits (SL) are limits placed upon important process variables which are found to be necessary to reasonably protect the integrity of certain of the physical barriers which guard against the uncontrolled release of radioactivity.

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

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

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

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

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

1.33 **Unsecured Experiment** - An unsecured experiment is any experiment which is not secured as defined by Definition 1.28, or the moving parts of secured experiments when they are in motion.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limits

Applicability:
This specification applies to the interrelated variables associated with reactor core thermal and hydraulic performance. These measurable operating or process variables include reactor power level, core flow rate, reactor inlet water temperature, and pressurizer pressure.

Objective:
The objective of this specification is to define a four-dimensional safety limit envelope such that operation within this envelope will assure that the integrity of the fuel element cladding is maintained.

Specification:
Reactor power level, core flow rate, reactor inlet water temperature, and pressurizer pressure shall not exceed the following limits during reactor operation:

a. Mode I and II Operation (Core Flow Rate ≥ 400 gpm)
The combination of the true values of reactor power level, reactor core flow rate, and reactor inlet water temperature shall not exceed the limits plotted on Figures 2.0, 2.1 and 2.2. The limits are considered exceeded if, for core flow rates greater than or equal to 400 gpm, the point defined by reactor power level and core flow rate is at any time above the curve corresponding to the true values of reactor inlet water temperature and pressurizer pressure. To define values of the safety limits for temperatures and/or pressures not shown in Figures 2.0, 2.1 and 2.2, interpolation or extrapolation of the data on the curves shall be used. For pressurizer pressures greater than 85 psia, the 85 psia curves (Figure 2.2) shall be used and no pressure extrapolation shall be permitted.

b. Mode I and II Operation (Core Flow Rate < 400 gpm)
Steady-state power operation in Modes I and II is not authorized for a core flow rate less than 400 gpm. Reactor operation with a core flow rate below 400 gpm will occur only after a normal reactor shutdown when the primary coolant circulation pumps are secured or following a loss of flow transient. Under the above conditions, the maximum fuel element cladding temperature shall not approach a temperature that would challenge the integrity of the fuel element cladding.

c. Mode III Operation
Reactor power is limited to a maximum of 150 kilowatts.

2.1 **Safety Limits** - Continued

Bases:

a. A complete safety limit analysis for the MURR is presented in Section 4.6.3 of the SAR. A family of curves is presented which relate reactor inlet water temperature and core flow rate to the reactor power level corresponding to a departure from nucleate boiling ratio (DNBR) of 1.2. This is based on the burnout heat flux data experimentally verified for Advanced Test Reactor (ATR) type fuel elements. Curves are presented for pressurizer pressures of 60, 75 and 85 psia. The safety limits were chosen from the results of this analysis for Mode I and II operation, i.e., forced convection operation with greater than 400 gpm flow.

b. Steady-state reactor operation is prohibited for core flow rates less than 400 gpm by the low flow scram settings in the reactor safety system. The region below 400 gpm will only be entered following a reactor shutdown when the primary coolant circulation pumps are secured or during a loss of flow transient where the reactor scrams, the flow coasts to zero, reverses, and natural convective cooling is established through the decay heat removal system. The analysis of a loss of flow transient presented in Section 13.2.4 of the SAR, from the ultraconservative conditions of 11 MW of power, a core flow rate of 3,800 gpm, and a reactor inlet water temperature of 155 °F, indicated a maximum fuel plate centerline temperature of 280.3 °F and a maximum coolant channel temperature of 237.5 °F, which is well below the saturation temperature of 277 °F.

c. Analysis of natural convective cooling of the core (Mode III Operation) is presented in Section 4.6.1 of the SAR.

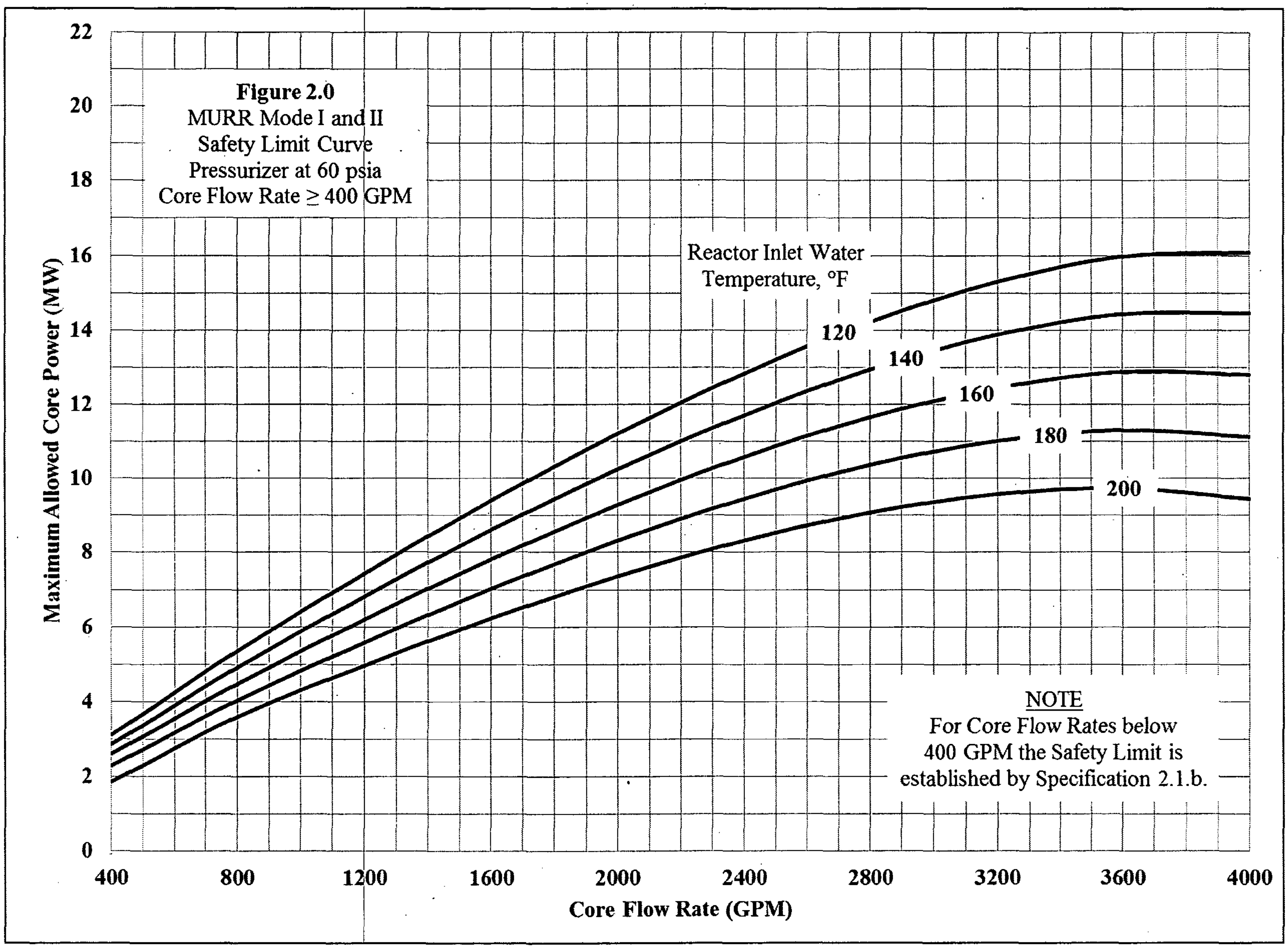
Figure 2.0
MURR Mode I and II
Safety Limit Curve
Pressurizer at 60 psia
Core Flow Rate ≥ 400 GPM
Reactor Inlet Water
Temperature, °F
120
140
160
180
200
NOTE
For Core Flow Rates below
400 GPM the Safety Limit is
established by Specification 2.1.b.
Maximum Allowed Core Power (MW)
0
2
4
6
8
10
12
14
16
18
20
22
400
800
1200
1600
2000
2400
2800
3200
3600
4000
Core Flow Rate (GPM)

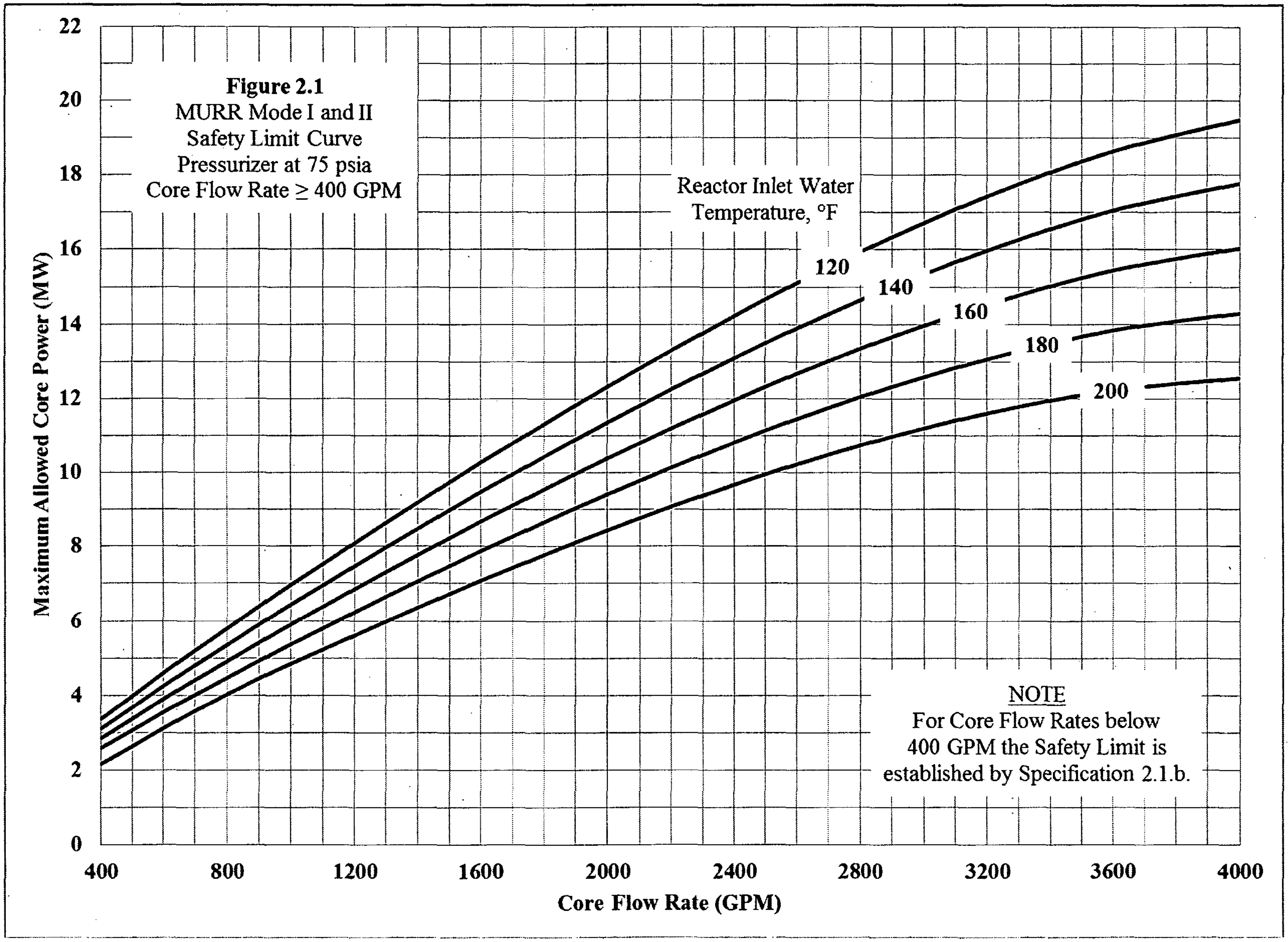

Figure 2.1
MURR Mode I and II
Safety Limit Curve
Pressurizer at 75 psia
Core Flow Rate ≥ 400 GPM

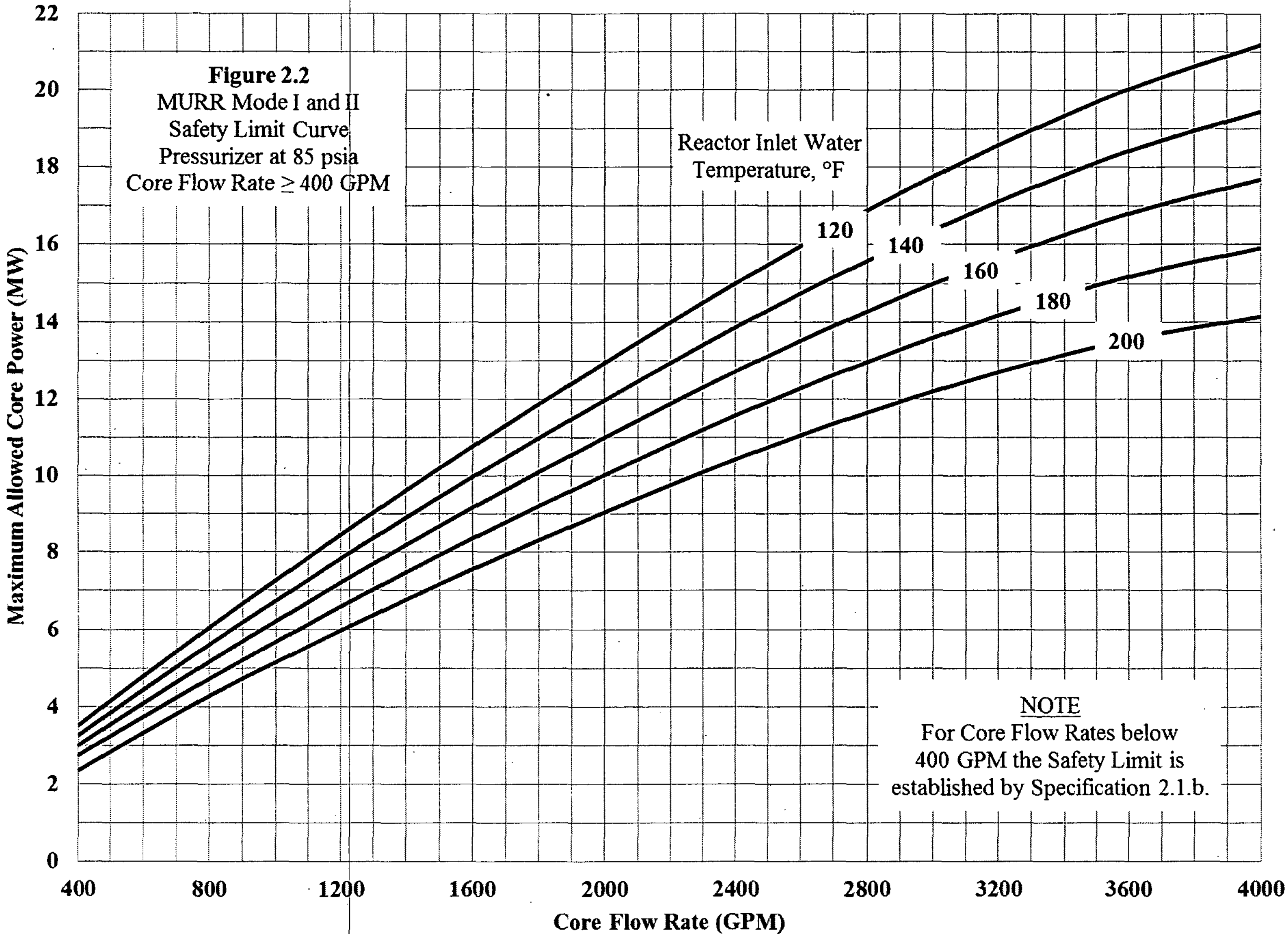

Figure 2.2
MURR Mode I and II
Safety Limit Curve
Pressurizer at 85 psia
Core Flow Rate ≥ 400 GPM

2.2 Limiting Safety System Settings

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

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

Specification:

a. Mode I Operation

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

b. Mode II Operation

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

c. Mode III Operation

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

Bases:

a. - b. The limiting safety system settings (LSSS) are set points which, if exceeded, will cause the reactor safety system to initiate a reactor scram. The LSSS were chosen such that the true value of any of the four safety-related variables, i.e., reactor power level, core flow rate, reactor inlet water temperature and pressurizer pressure will not exceed a safety limit under the most severe anticipated transient. Section 4.6.4 of the SAR presents analyses to show that the LSSS for Mode I and II operation meet this criterion.

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

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 Reactivity Limitations

Applicability:
This specification applies to the reactivity of the reactor core and the reactivity worths of the control blades and experiments.

Objective:
The objective of this specification is to assure that the reactor can be controlled and shut down at all times and that the safety limits will not be exceeded.

Specification:

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

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

c. The regulating blade total reactivity worth shall be a maximum of 6.0×10^{-3} Δk/k and the maximum rate of reactivity insertion shall be 2.5×10^{-4} Δk/k/sec.

d. The maximum rate of reactivity insertion for the four shim blades operating simultaneously shall not exceed 3.0×10^{-4} Δk/k/sec.

e. The reactor shall be subcritical by a margin of at least 0.02 Δk/k with the most reactive shim blade and the regulating blade in the fully withdrawn positions.

f. The reactor core excess reactivity above cold, clean, critical shall not exceed 0.098 Δk/k. Core excess reactivity shall be verified after any changes are made in the reactor core.

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

h. The absolute value of the reactivity worth of all experiments in the center test hole shall be limited to 0.006 Δk/k.

i. Each movable experiment or the movable parts of any individual experiment shall have a maximum absolute reactivity worth of 0.001 Δk/k.

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

3.1 **Reactivity Limitations** - Continued

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

Bases:

a. Specification 3.1.a limits one of the parameters which assures that core damage will not occur following any credible step reactivity insertion as analyzed in Section 13.2.2 of the SAR.

b. The average core void coefficient of reactivity also limits the step reactivity insertion accident as analyzed in Section 13.2.2 of the SAR.

c. The regulating blade total reactivity worth is limited by Specification 3.1.c such that any condition resulting in the step insertion of the maximum worth of 6×10^{-3} $\Delta k/k$ will not result in fuel plate damage. The limit on the rate of reactivity addition provides for reasonable response from operator control.

d. Specification 3.1.d assures that power increases caused by control rod motion will be safely terminated by the reactor safety system. The continuous control rod withdrawal accident is analyzed in Section 13.2.2 of the SAR.

e. Specification 3.1.e assures that a shutdown margin, as defined by Definition 1.31, is maintained.

f. Specification 3.1.f provides additional assurance that Specification 3.1.e is satisfied.

g. Specification 3.1.g provides assurance that any inadvertent insertion/removal or credible malfunction of a secured removable experiment would not introduce positive reactivity whose consequences would lead to radiation exposures in excess of the 10 CFR 20 limits. The step reactivity insertion is analyzed in Section 13.2.2 of the SAR.

h. The reactivity worth of experiments in the center test hole is limited by Specification 3.1.h such that the introduction of the maximum reactivity worth of all experiments would not result in damage to the fuel plates as analyzed in Section 13.2.2 of the SAR.

i. Specification 3.1.i provides assurance that the movement of movable experiments or movable parts of any experiment will not introduce reactivity transients more severe than one that can be controlled without initiating a reactor safety system action as analyzed in Section 13.2.2 of the SAR.

3.1 **Reactivity Limitations** - Continued

j. Specification 3.1.j prevents the installation of an unsecured experiment which could introduce, as a positive step change, sufficient reactivity to place the reactor in a transient that would cause a violation of a safety limit as analyzed in Section 13.2.2 of the SAR.

k. Specification 3.1.k assures that the reactivity worth of all unsecured experiments shall not exceed the maximum value authorized for a single secured removable experiment.

3.2 Control Blades

Applicability:
This specification applies to the operation of the reactor control blades.

Objective:
The objective of this specification is to reasonably assure proper operation of the reactor control system, thus avoiding conditions which could jeopardize the integrity of the fuel element cladding or endanger personnel health and safety.

Specification:

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

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

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

Bases:

a. Specification 3.2.a ensures that the normal method of reactivity control is used during reactor operation.

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

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

3.3 Reactor Safety System

Applicability:
This specification applies to the reactor safety system instrument channels.

Objective:
The objective of this specification is to specify the minimum number of reactor safety system instrument channels that must be operable for safe reactor operation.

Specification:

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

Safety System or Measuring Channel	Number Required (N) Mode I	Mode II	Mode III	Trip Set Point
High Power Level	3	3	3	125% of full power (Max)
Reactor Period	2	2	2	8 Seconds (Min)
Primary Coolant Flow	4	2	2(1)	1,625 gpm(2) (Min)
Differential Pressure Across the Core	1	0	0	3,200 gpm(3) (Min)
Differential Pressure Across the Core	0	1	1(1)	1,600 gpm(3) (Min)
Primary Coolant Low Pressure	4	4	4(1)	75 psia(5) (Min)
Reactor Inlet Water Temperature	2	1	1(1)	155 °F (Max)
Reactor Outlet Water Temperature	1	1	1(1)	175 °F (Max)
Pool Coolant Flow	2	2	0	850 gpm(4) (Min)
Differential Pressure Across the Reflector	1	0	0	2.52 psi (Min) 8.00 psi (Max)

3.3 **Reactor Safety System** - Continued

Safety System or Measuring Channel	Number Required (N) Mode I	Mode II	Mode III	Trip Set Point
Differential Pressure Across the Reflector	0	1	0	0.63 psi (Min) 2.00 psi (Max)
Pressurizer High Pressure	1	1	1[(1)]	95 psia (Max)
Pressurizer Low Water Level	1	1	1[(1)]	16 inches below centerline (Min)
Pool Low Water Level	0	0	1	23 feet (Min)
Primary Coolant Isolation Valves 507A/B Off Open Position	1	1	1[(1)]	Either valve off open position
Pool Coolant Isolation Valve 509 Off Open Position	1	1	0	Valve 509 off open position
Power Level Interlock	1	1	1	Scram as a result of incorrect selection of operating mode
Facility Evacuation	1	1	1	Scram as a result of actuating the facility evacuation system
Reactor Isolation	1	1	1	Scram as a result of actuating the reactor isolation system
Manual Scram	1	1	1	Push button on Control Console
Center Test Hole	2[(6)]	2[(6)]	2[(6)]	Scram as a result of removing the center test hole removable experiment test tubes or strainer

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

3.3 **Reactor Safety System** - Continued

(2) Flow orifice or heat exchanger ΔP (psi) in each operating heat exchanger leg corresponding to the flow value in the table.
(3) Core ΔP (psi) corresponding to the core flow value in the table.
(4) Flow orifice ΔP (psi) corresponding to the flow value in the table.
(5) Trip pressure is that which corresponds to the pressurizer pressure indicated in the table with normal primary coolant flow.
(6) Not required if reactivity worth of the center test hole removable experiment test tubes and its contents or the strainer is less than the reactivity limit of Specification 3.1.h. This safety function shall only be bypassed with specific authorization from the Reactor Manager.

Bases:

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

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

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

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

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

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

3.3 **Reactor Safety System** - Continued

The reactor scrams caused by the primary and pool coolant isolation valves (507A/B and 509) leaving their full open position provide the first line of protection for a loss of flow accident (in their respective system) initiated by an inadvertent closure of the isolation valve(s).

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

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

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

The center test hole scram provides assurance that the reactor cannot be operated unless the removable experiment test tubes or the strainer is inserted and latched in the center test hole. This is required anytime the reactivity worth of the center test hole removable experiment test tubes and the contained experiments or the strainer exceeds the limit of Specification 3.1.h (Ref. Section 13.2.2 of the SAR). The center test hole scram may be bypassed if the total reactivity worth of the removable experiment test tubes and the contained experiments or the strainer does not exceed the limit of Specification 3.1.h and is authorized by the Reactor Manager.

3.4 Reactor Instrumentation

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

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

Specification:

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

Channel	Minimum Numbers Operable Mode I	Mode II	Mode III
Source Range Nuclear Instrument Channel	1[1]	1[1]	1[1]
Reactor Pool Temperature	1	1	1
Reactor Bridge Radiation Monitor	1[2]	1[2]	1[2]
Reactor Containment Building Exhaust Plenum Radiation Monitor	1	1	1
Off-Gas Radiation Monitor	1[3]	1[3]	1[3]

(1) Required for reactor startup only.
(2) The trip setting may be temporarily set upscale during periods of maintenance and sample handling. During these periods, the radiation monitor indication will be closely observed.
(3) The off-gas radiation monitor may be placed out of service for up to 2 hours for calibration and maintenance. During this out-of-service time, no experimental or maintenance activities will be conducted which could likely result in the release of unknown quantities of airborne radioactivity.

b. Sufficient instrumentation shall be provided to assure that the following limits are not exceeded during steady-state operation:

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

3.4 **Reactor Instrumentation** - Continued

(1) Not required for Mode III operation.
(2) Reactor Pool Temperature limit is a maximum of 100 °F when in Mode III operation and (a) below 50 kW with the natural convection flange and reactor pressure cover removed or (b) with the reactor subcritical by a margin of at least 0.015 ΔK.

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

	Number Required (N)			
Rod Run-In Function	Mode I	Mode II	Mode III	Trip Set Point
High Power Level	3	3	3	115% of full power (Max)
Reactor Period	2	2	2	10 Seconds (Min)
Pool Low Water Level	1	1	0	27 feet (Min)
Vent Tank Low Level	1	1	0	1 foot below centerline (Min)
Rod Not-In-Contact With Magnet	4	4	4	Magnet disengaged from any rod
Anti-Siphon System High Level	1	1	1(1)	6 inches above valves (Max)
Truck Entry	1	1	1	Loss of entry door seal pressure
Regulating Blade Position	2	2(2)	2(2)	< 10% withdrawn and bottomed
Manual Rod Run-In	1	1	1	Push button on Control Console

(1) Not required (a) below 50 kW operation with the natural convection flange and reactor pressure vessel cover removed or (b) in operation with the reactor subcritical by a margin of at least 0.015 Δk.
(2) Not required during calibration measurements of the regulating blade.

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

3.4 **Reactor Instrumentation** - Continued

e. The reactor shall not be started up unless:.

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

OR

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

Bases:

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

The reactor pool temperature instrument is required to ensure that pool temperature does not increase to a level which would jeopardize the ability to cool in-pool components.

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

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

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

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

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

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

3.4 **Reactor Instrumentation** - Continued

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

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

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

The specification on the truck entry door prohibits reactor operation without the door's contribution to containment integrity as required by Definition 1.17.a.

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

d. Specification 3.4.d ensures that, during a startup, the reactor power level is continuously monitored over the entire range.

e. Specification 3.4.e provides for adequate neutron flux level monitoring to ensure that subcritical multiplication and criticality can be observed during a startup.

3.5 Reactor Containment Building

Applicability:
This specification applies to the reactor containment building.

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

Specification:

a. 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.

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.

Bases:

a. Specification 3.5.a assures that the reactor containment building can be isolated at all times except when plant conditions are such that the probability of a release of radioactivity is negligible.

b. Radiation monitors located at the reactor bridge and in the containment building ventilation exhaust plenum supply input signals to meters located in the reactor control room. A containment isolation will occur when radiation levels in these areas exceed a predetermined value. During operations such as the removal of experiments or equipment from the pool, the radiation level at the level of the reactor bridge or in the exhaust plenum can increase significantly for short periods. To prevent inadvertent containment isolations, it may be necessary to raise the set point on the reactor bridge or exhaust plenum monitor. During periods in which the set point is raised to more than one decade above the normal reading, the radiation level in the area of the monitor will be continuously monitored. Thus, should the radiation level increase from unknown causes or from material which could be released to the unrestricted environment, 3.5 **Reactor Containment Building** - Continued

the containment building can be quickly isolated by manually actuating the isolation system.

3.6 Experiments

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

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

Specification:

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

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

c. Where the possibility exists that the failure of an experiment could release radioactive gases or aerosols into the containment building atmosphere, the experiment shall be limited to that amount of material such that the airborne concentration of radioactivity when averaged over a year will not exceed the limits of 10 CFR 20, Appendix B, Table I. Exception: Fueled experiments (See Specification 3.6.a).

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.

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

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

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

3.6 **Experiments** - Continued

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

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

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

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

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

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

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

o. Fueled experiments containing inventories of iodine-131 through iodine-135 greater than 1.5 curies or strontium-90 greater than 5 millicuries shall be in irradiation containers that satisfy the requirements of Specification 3.6.i or be vented to the facility ventilation exhaust stack through high efficiency particulate air (HEPA) and charcoal filters which are continuously monitored for an increase in radiation levels.

<u>Bases</u>:

a. Specification 3.6.a restricts the generation of hazardous materials to levels that can be handled safely and easily. Analysis of fueled experiments containing a greater inventory of fission products has not been completed, and therefore their use is not permitted. (Ref. Section 13.2.6 of the SAR).

3.6 **Experiments** - Continued

b. Specification 3.6.b is intended to reduce the likelihood of accidental voiding in the reactor core or water annulus surrounding the center test hole by restricting materials which could generate or accumulate gases or vapors.

c. The limitation on experiment materials imposed by Specification 3.6.c assures that the limits of 10 CFR 20, Appendix B, are not exceeded in the event of an experiment failure.

d. Specification 3.6.d is intended to reduce the likelihood of damage to reactor or pool components resulting from the detonation of explosive materials (Ref. Section 13.2.6 of the SAR).

e. Specification 3.6.e is intended to limit the experiments that can be moved in the center test hole while the reactor is operating to those that will not introduce reactivity transients more severe than one that can be controlled without initiating safety system action (Ref. Section 13.2.2 of the SAR).

f. - g. Specifications 3.6.f and 3.6.g provide guidance for experiment safety analysis to assure that anticipated transients will not result in radioactivity release and that experiments will not jeopardize the safe operation of the reactor.

h. Specification 3.6.h is intended to reduce the likelihood of reactivity transients due to accidental voiding in the reactor or the failure of an experiment from internal or external heat generation.

i. Specification 3.6.i is intended to reduce the likelihood of damage to the reactor and/or radioactivity releases from experiment failure.

j. Specification 3.6.j provides assurance that no chemical reaction will take place to adversely affect the reactor or its components.

k. Specification 3.6.k provides assurance that the integrity of the beamports will be maintained for all loop-type experiments.

l. Specification 3.6.l assures that corrosive materials which are chemically incompatible with reactor components, highly flammable materials, and toxic materials are adequately controlled and that this information is disseminated to all reactor users.

m. The extremely low temperatures of the cryogenic liquids present structural problems that enhance the potential of an experiment failure. Specification 3.6.m provides for the proper review of proposed experiments containing or using cryogenic materials.

3.6 **Experiments** - Continued

n. Specification 3.6.n is intended to reduce the likelihood of damage to the reactor and/or radioactivity releases from experiment failure.

o. Specification 3.6.o restricts the generation of hazardous materials to levels that can be handled safely and easily. Analysis of fueled experiments containing a greater inventory of fission products has not been completed, and therefore their use is not permitted. (Ref. Section 13.2.6 of the SAR).

3.7 Facility Airborne Effluents

Applicability:
This specification applies to the release of gaseous and particulate activity from the facility ventilation exhaust stack.

Objective:
The objective of this specification is to assure that exposure to the public resulting from the radioactivity released from the reactor facility to the unrestricted environment will not exceed the limits of 10 CFR 20.

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

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

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

Bases:
a. Dispersion calculations based upon standard reference material and experiment data obtained at the reactor show that argon-41 concentrations under average conditions will be 0.008 of the AEC limits in the unrestricted area surrounding the reactor facility. Dilution factors under conservative conditions are in the range of 5×10^4 under both average and stable conditions at ground level from the facility building.

The normal short burst releases at the facility are five to ten seconds in duration and occur on an average of ten times per day five days per week. The short bursts affect the concentration by less than 1% when averaged over a one-day period.

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

3.8 Reactor Fuel

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

Objective:
The objective of this specification is to assure that the reactor fuel is operated within acceptable design considerations thus ensuring fuel element integrity is maintained.

Specification:

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

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

c. The reactor core shall consist of eight fuel assemblies.
Exception: The reactor may be operated to 100 watts above shutdown power on less than eight assemblies for the purposes of reactor calibration or multiplication measurement studies.

d. All fuel elements or fueled devices outside the reactor core shall be stored in a geometry such that the calculated K_{eff} is less than 0.9 under all conditions of moderation.

e. Irradiated fuel elements shall be stored in an array which will permit sufficient natural convection cooling such that the fuel element temperature will not exceed its design values.

Bases:

a. Specification 3.8.a restricts the peak fissions per cubic centimeter burnup to values that have been correlated to result in less than 10% swelling of the fuel plates. It has been found that fuel plate swelling of less than 10% has no detrimental effect on fuel plate performance (Ref.: Change No. 4 to Facility License R-103, Change No. 6 to Facility License R-103, and Application dated September 12, 1986 with supplements).

b. Specification 3.8.b assures that fuel elements which have been inspected and found to be defective are no longer used for reactor operation.

c. Operation at a power level greater than 100 watts requires a full core of eight full 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

3.8 **Reactor Fuel** - Continued

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 be able to observe subcritical multiplication. Then one additional fuel element would be added at a time in between approaches to critical. The reactor would be operated in this manner only to perform necessary conservative approaches to criticality.

d. - e. The limits imposed by Specifications 3.8.d and 3.8.e are conservative and assure safe fuel storage.

3.9 Reactor Coolant Systems

Applicability:
This specification applies to the reactor coolant systems.

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

Specification:

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

   (1) Anti-Siphon System;
   (2) Primary Coolant Isolation Valves V507A/B; and
   (3) In-Pool Convective Cooling System.

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

   (1) A continuous primary coolant system fuel element failure monitor is operable,

   OR

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

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

d. The anti-siphon system will be maintained pressurized to a value of 30 to 45 psig. In the event of a system low pressure alarm, immediate action will be taken to add air to obtain the specified pressure. The system pressure will be verified, recorded, and readjusted as required every 4 hours as part of the facility routine patrol. Procedures will be established for manual verification of water level in the anti-siphon system for conditions when system pressure has an unexplained rise of 4 psi or more. If water level is 6 inches or more above the anti-siphon isolation valves or a system leak or other malfunction prevents the maintenance of pressure in the specified range, the reactor will be shut down until the malfunction can be corrected.

Bases:

a. The first line of protection against a loss of core water resulting from a rupture of the primary coolant system is provided by the check valve on the inlet line and by the invert loop and the anti-siphon system on the outlet line. Upon opening, the

3.9 **Reactor Coolant Systems** - Continued

anti-siphon isolation valves will admit a fixed volume of air to the highest point of the invert loop, thus preventing the reactor core from becoming uncovered by breaking any potential siphon which may have been created by the pipe rupture (Ref. Section 6.3 of the SAR).

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

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

b. - c. The primary coolant system with an iodine-131 concentration of 5×10^{-3} μCi/ml would contain a total iodine-131 inventory of 0.038 Ci in the system. Based on the iodine-131 activity in the reactor core provided in Section 13.2.1.2 of the SAR, this iodine-131 concentration would equate to less than 0.000022 % of the total core iodine-131 inventory in the primary coolant. Specifications 3.9.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.

d. Specification 3.9.d ensures that the anti-siphon system will perform its intended function as designed by imposing certain operational limits on the system (Ref. Section 6.3 of the SAR).

3.10 Auxiliary Systems

Applicability:
This specification applies to the reactor auxiliary systems.

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

Specification:

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

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

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

Bases:

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

b. Specification 3.10.b provides for an adequate supply of primary grade water for make-up during all modes of operation.

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

4.0 SURVEILLANCE REQUIREMENTS

4.1 Containment System

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

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

Specification:

a. The reactor containment building leakage rate shall be measured annually, plus or minus four (4) months. No special maintenance shall be performed just prior to the test.

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

Bases:

a. Annual measurement of the containment building leakage rate has proven adequate to ensure that the leakage rate of the structure will remain within the design limits outlined in Specification 5.2.c. No special maintenance will be performed prior to the test so that the results demonstrate the historic integrity of the containment structure.

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

4.2 Reactor Coolant Systems

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

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

Specification:

a. The primary coolant system relief valves shall be tested for operability at two-year intervals, with at least one of the valves tested on an annual basis.

b. The primary coolant isolation valves and the anti-siphon isolation valves shall be tested for operability at monthly intervals except during extended shutdown periods when the valves shall be tested prior to reactor operation.

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

Bases:

a. Satisfactory performance of both relief valves during the testing program over the past 40 years has demonstrated the reliability of the valves and the assurance of operability gained by the testing frequency outlined in Specification 4.2.a.

b. The past 40 years of operation of the primary coolant and anti-siphon isolation valves has shown that monthly testing is adequate to provide assurance of continued operability.

c. The weekly radio-chemical analysis will provide assurance that a fuel element leak will be discovered so that corrective action can be taken to prevent the release of fission products. Specification 4.2.c establishes the frequency of verification of compliance with Specification 3.9.c.

4.3 Control Blades

Applicability:
This specification applies to the surveillance requirements of the reactor control blades.

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

Specification:

a. The drop time of each of the four shim blades shall be measured at quarterly intervals.

b. A different one of the four shim blades shall be inspected each six months so that every blade is inspected every two years. The reactor shall not be operated with a control blade that exhibits abnormal swelling or abnormalities that affect performance.

Bases:

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

b. Periodic inspection of the shim blades provides detection of singular blade abnormalities and any potential generic blade design deficiencies. Specification 4.3.b further assures that the reactor will not be operated using shim blades with suspected generic design deficiencies.

4.4 Reactor Instrumentation

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

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

Specification:

a. All instruments, as required by these specifications, shall be calibrated on a semiannual basis.

b. Radiation monitoring instrumentation, as required by these specifications, shall be checked for operability with a radiation source at monthly intervals.

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

Bases:

a. Semiannual calibration of the reactor instrument channels will assure that long-term drift of the channels will be corrected.

b. Experience has shown that monthly verification of operability of the radiation monitoring instrumentation in conjunction with the semiannual calibration is adequate assurance of proper operation over a long time period.

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

4.5 **Reactor Fuel**

Applicability:
This specification applies to the surveillance requirements of the reactor fuel elements.

Objective:
The objective of this specification is to reasonably assure proper performance of the reactor fuel.

Specification:
a. One out of every eight (8) fuel elements that have reached their end-of-life will be inspected for anomalies.

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

4.6 Auxiliary Systems

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

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

Specification:

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

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

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

Bases:

a. The University of Missouri-Columbia water supply system provides a virtually unlimited source of raw water for the emergency pool fill system. Water supply is maintained at a high pressure by automatically-controlled pumping stations. The above check, in light of the reliability of the emergency pool fill system, provides assurance that Specification 3.10.c is satisfied.

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

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

5.0 DESIGN FEATURES

5.1 Site Description

Applicability:
This specification applies to the site of the MURR facility.

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

Specification:

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

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

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

   For emergency planning purposes, the site boundaries consist of the following: Stadium Boulevard; Providence Road (Route K)[1]; the MU Recreational Trail; and the MKT Nature and Fitness Trail. The area within these boundaries is owned and controlled by MU and may be frequented by people unacquainted with the operation of the reactor. The Reactor Facility Director has authority to initiate emergency actions in this area, if required.

[1]Providence Road crosses MU property separating the University Research Park from another MU-owned tract of land lying to the east. The road runs north and south with the closest point of approach being approximately 400 meters east of the reactor facility. MU has the authority to determine all activities including the exclusion or removal of personnel and property and to temporarily secure the flow of traffic on this road during an emergency.

5.1 **Site Description** - Continued

Bases:

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

5.2 Reactor Containment Building

Applicability:
This specification applies to the building in which the reactor is located.

Objective:
The objective of this specification is to assure adequate restriction to the accidental release of radioactivity to the environment.

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

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

b. Whenever containment integrity, as defined by Definition 1.17, is required, containment building ventilation exhaust shall be discharged at a minimum of 55 feet above containment building grade level.

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

d. The containment building shall have a secured fuel storage room with the key or combination under control of the Reactor Manager.

Bases:

a. No credible accident scenario has been identified which can result in a significant overpressure condition in the reactor containment building. However, Specification 5.2.a assures that a sufficient free volume exists to prevent any pressure buildup in the containment building (Ref. Section 6.2.2.2 of the SAR).

b. Specification 5.2.b assures a sufficient stack height for more than adequate atmospheric dispersion.

c. Specification 5.2.c assures that the containment building will have sufficient integrity to limit the leakage of contained potentially radioactive air in the event

5.2 **Reactor Containment Building** - Continued

of any reactor accident to ensure exposures are maintained below the limits of 10 CFR 20 (Ref. Sections 6.2.10 and 13.2.1 of the SAR).

d. Specification 5.2.d assures safe and secure storage of fresh fuel.

5.3 Reactor Coolant Systems

Applicability:
This specification applies to the reactor coolant systems.

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

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

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

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

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

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

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

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

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

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

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

5.3 **Reactor Coolant Systems** - Continued

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

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

Exceptions:

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

b. Some materials in off-the-shelf commercial components may be excepted from Specification 5.3.e.

Bases:

a. - k. The reactor coolant systems are described and analyzed in Section 5 of the SAR. The reactor can be safely operated at 10 MW with the coolant systems as described.

Specification 5.3.a as excepted, permits reactor operation at 50% of full power in the event of a major component failure in which repairs cannot be accomplished in a reasonable period of time. The reactor was designed and has extensive safe operating history for operation at 50% of 10 MW cooling capacity. In this event, the shutdown system shall be secured in a manner such as to assure system integrity.

Specification 5.3.e assures strength and corrosion resistance of the coolant system components and excepts some components in the instrumentation of the system which are not commercially available in the materials specified. The size of these components is such that a failure would not result in a hazard to safe operation of the reactor.

5.4 Reactor Core and Fuel

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

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

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

a. Each reactor fuel element shall contain 24 fuel-bearing plates with a nominal active length of 24 inches and a plate thickness of 0.050 inches. The nominal distance between the fuel plates shall be 0.080 inches. Plate nominal cladding thickness shall be 0.015 inches.

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

c. Each fuel element shall have a maximum uranium-235 loading of 775 grams.

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

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

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

g. The reactor shall have the following experimental facilities:
   1. Six beam tubes which penetrate the graphite reflector;
   2. A center test hole located in the flux trap;
   3. A portion of the graphite reflector;
   4. A bulk pool consisting of the water region above and outside the graphite reflector; and
   5. A thermal column.

Bases:

a.- c. The MURR fuel elements are one of a configuration (aluminide UAl_X dispersion fuel system) successfully and extensively used for many years in test and research reactors. Specifications 5.4.a, 5.4.b and 5.4.c require fuel content and dimensions

5.4 **Reactor Core and Fuel** - Continued

of the fuel elements to be in accordance with the design and fabrication specifications (Ref. Section 4.2.1 of the SAR).

d. Specification 5.4.d assures that the reactor fuel is properly positioned in the pressure vessel during operation (Ref. Section 4.2.5 of the SAR).

e. Specification 5.4.e assures proper neutron reflection as required by design (Ref. Section 4.2.3 of the SAR).

f. Specification 5.4.f assures reactivity of the reactor is properly controlled as required by design (Ref. Section 4.2.2 of the SAR).

g. Specification 5.4.g assures that the reactor consists of the experimental facilities as required by design (Ref. Chapter 10 of the SAR).

5.5 Emergency Electrical Power System

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

Objective:
The objective of this specification is to assure adequate emergency electrical power in the event of normal electrical power failure.

Specification:
The following design feature applies to the emergency electrical power system:

a. The MURR shall have an emergency power generator capable of providing emergency electrical power to the emergency lighting system, the facility ventilation exhaust system, reactor instrumentation, and the personnel air lock doors.

Bases:

a. The emergency electrical power system is described in Section 8.2 of the SAR. Specification 5.5.a assures that a system exists to provide the necessary electrical power to monitor the reactor systems and assure personnel safety in the event of a normal power failure to the reactor facility.

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization

a. The organizational structure of the University of Missouri-Columbia (MU) relating to the University of Missouri Research Reactor (MURR) shall be as shown in Figure 6.0.

b. The following positions shall have direct responsibility in implementing the Technical Specifications as designated throughout this document:

(1) Reactor Facility Director: Responsible for establishing the policies that minimize radiation exposure to the public and to radiation workers, and that ensures that the requirements of the license and Technical Specifications are met.

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

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

ii. Oversight of the experiment review process.

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

i. Compliance with Technical Specifications and license requirements regarding radiation safety, byproduct material handling and the shipment of byproduct material; and

ii. Implementation of the Radiation Protection Program.

c. At a minimum during reactor operation, there shall be two facility staff personnel at the facility. One of these individuals shall be a Reactor Operator or a Senior Reactor Operator licensed pursuant to 10 CFR 55. The other individual must be knowledgeable of the facility.

d. A Senior Reactor Operator licensed pursuant to 10 CFR 55 shall be present at the facility or readily available on call at all times during operation, and shall be present at the facility during all startups and approaches to power, recovery from an unplanned or unscheduled shutdown or non-emergency power reduction, and refueling activities.

6.2 Review and Audit

a. A Reactor Advisory Committee (RAC) shall provide independent oversight in matters pertaining to the safe operation of the reactor and with regard to planned research activities and use of the facility building and equipment. The RAC shall review:

(1) Proposed changes to the MURR equipment, systems, or procedures when such changes have safety significance, or involve an amendment to the facility operating license, a change in the Technical Specifications incorporated in the license, or a question pursuant to 10 CFR 50.59. Changes to procedures that do not change their original intent may be made without prior RAC review if approved by the TS designated manager, either the Reactor Health Physics Manager or Reactor Manager, or a designated alternate who is a member of Health Physics or a Senior Reactor Operator, respectively. All such changes to the procedures shall be documented and subsequently reviewed by the RAC;

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

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

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

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

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

The RAC and its subcommittees are to maintain minutes of meetings in which the items considered and the committees' recommendations are recorded. Independent actions of the subcommittees are to be reviewed by the parent committee at the next regular meeting. A quorum of the committee or the subcommittees consisting of at least fifty percent of the appointed members must be present at any meeting to conduct the business of the committee or subcommittee. The RAC shall meet at least once during each calendar quarter.

6.2 **Review and Audit** - Continued

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

c. Any additions, modifications or maintenance to the systems described in these Specifications shall be made and tested in accordance with the specifications to which the system was originally designed and fabricated or to specifications approved by the U.S. Nuclear Regulatory Commission (NRC).

d. Following a favorable review by the NRC, the RAC, or the Reactor Facility Management, as appropriate, and prior to conducting any experiment, the Reactor Manager shall sign an authorizing form which contains the basis for the favorable review.

e. Audits

(1) Audits of the following functions shall be conducted by an individual or group without immediate responsibility in the area to be audited:

i. Facility Operations, for conformance to the Technical Specifications and license conditions, at least annually;
ii. Operator Requalification Program, for compliance with the approved program, at least every two years; and
iii. Corrective Action items associated with reactor safety, at least annually.

(2) Audit findings which affect reactor safety shall be immediately reported to the Reactor Facility Director.

6.3 **Procedures**

a. Written procedures shall be in effect for operation of the reactor, including the following:

(1) Startup, operation, and shutdown of the reactor;

(2) Fuel loading, unloading and movement within the reactor;

(3) Maintenance of major components of systems that could have an effect on reactor safety;

(4) Surveillance checks, calibrations and inspections that may affect reactor safety;

6.3 **Procedures** - Continued

(5) Administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect reactor safety or core reactivity; and

(6) Implementation of the Emergency Plan.

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

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

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

6.4 **Records**

Records of the following activities shall be maintained and retained for the periods specified below. The records may be in the form of logs, data sheets, or other suitable forms or documents. The required information may be contained in single or multiple records, or a combination thereof.

a. Lifetime Records - The following records are to be retained for the lifetime of the reactor facility: (Note: Applicable annual reports, if they contain all of the required information, may be used as records in this section.)

(1) Gaseous and liquid radioactive effluents released to the environs;

(2) Off-site environmental-monitoring surveys required by the Technical Specifications;

(3) Radiation exposure for all monitored personnel; and

(4) Updated drawings of the reactor facility.

6.4 **Records** - Continued

b. Five Year Records - The following records are to be maintained for a period of at least five years or for the life of the component involved, whichever is shorter:

(1) Normal reactor facility operation (but not including supporting documents such as checklists, log sheets, etc. which shall be maintained for a period of at least one year);

(2) Principal maintenance operations;

(3) Reportable occurrences;

(4) Surveillance activities required by the Technical Specifications;

(5) Reactor facility radiation and contamination surveys required by applicable regulations;

(6) Experiments performed with the reactor;

(7) Fuel inventories, receipts and shipments;

(8) Approved changes to operating procedures; and

(9) Records of meetings and audit reports of the review and audit group.

6.5 **Reportable Events and Required Actions**

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

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

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

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

i. Applicable circumstances leading to the violation including, when known, the causes and contributing factors;
ii. Date and approximate time of the occurrence;
iii. Effect of the violation upon the reactor and associated systems;

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

iv. Effect of the violation on the health and safety of the facility staff and general public; and
v. Corrective actions to prevent recurrence.

(4) The follow-up report will be submitted within fourteen (14) days to the NRC Document Control Desk.

b. Release of Radioactivity - Should a release of radioactivity greater than the allowable limits occur from the reactor facility boundary, the following actions shall be taken:

(1) Reactor conditions shall be returned to normal or the reactor shall be shut down;

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

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

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

c. Other Reportable Occurrences - In the event of an Abnormal Occurrence, as defined by Definition 1.1, the following actions shall be taken:

(Note: Where components or systems are provided in addition to those required by these Technical Specifications, the failure of the extra components or systems is not considered reportable provided that the minimum numbers of components or systems specified or required perform their intended reactor safety function.)

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

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

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

(3) The reactor shall be shut down or placed in a safe condition and return to normal reactor operations will not be allowed until authorized by the Reactor Manager.

d. Other Reports - A written report shall be submitted to the NRC Document Control Desk within thirty (30) days of:

(1) Any significant change(s) in the transient or accident analyses as described in the SAR; and

(2) Permanent changes in the facility organization involving the Office of the Provost or the Director's Office.

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

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

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

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

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

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

(6) A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as measured at or prior to the point of such release or discharge;

(7) A description of any environmental surveys performed outside the reactor facility; and

(8) A summary of radiation exposures received by facility staff, experimenters, and visitors, including the dates and time of significant exposure, and a brief summary of the results of radiation and contamination surveys performed within the facility.

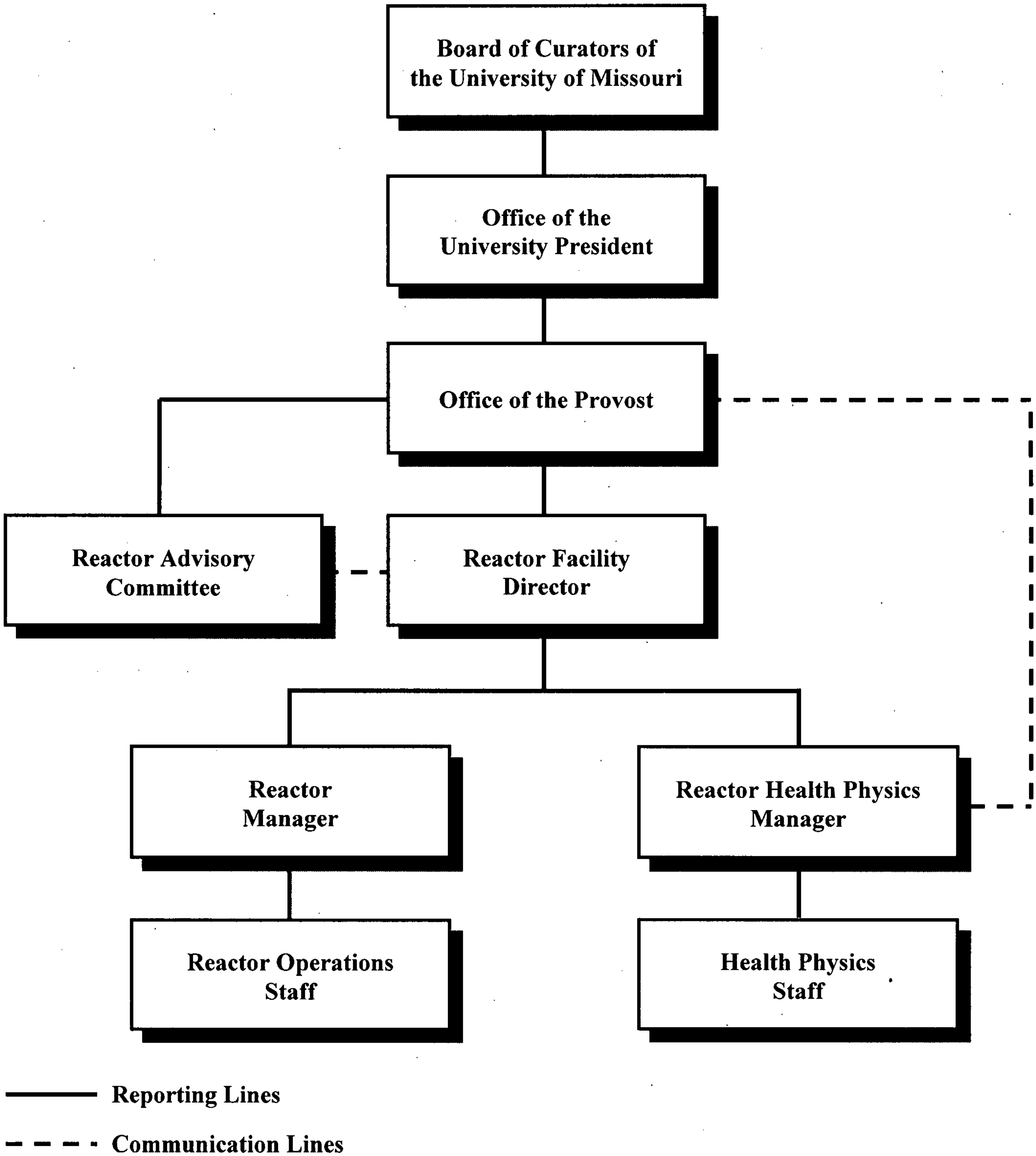

FIGURE 6.0
UNIVERSITY OF MISSOURI RESEARCH REACTOR (MURR)
ORANIZATION