UNIVERSITY *of* MISSOURI

RESEARCH REACTOR CENTER

July 10, 2013

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

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

SUBJECT: Written communication as required by University of Missouri Research Reactor Technical Specification 6.1.h(2) regarding a deviation from Technical Specifications 3.2.b

The attached document provides the University of Missouri-Columbia Research Reactor (MURR) Licensee Event Report (LER) for an event that occurred on June 11, 2013 that resulted in a deviation from MURR Technical Specification 3.2.b.

If you have any questions regarding this report, please contact John L. Fruits, the facility Reactor Manager, at (573) 882-5319.

Sincerely,

for Ralph Butler

Ralph A. Butler, P.E.
Director

NOTARY PUBLIC NOTARY SEAL STATE OF MISSOURI

MARGEE P. STOUT
My Commission Expires
March 24, 2016
Montgomery County
Commission #12511436

Margee P Stout

RAB:djr

Enclosure
xc: Reactor Advisory Committee
Reactor Safety Subcommittee
Dr. Robert V. Duncan, Vice Provost of Research
Mr. Alexander Adams, Jr., U.S. NRC
Mr. Johnny Eads, U.S. NRC

IE22
NRR

Licensee Event Report No. 13-03 – June 11, 2013
University of Missouri Research Reactor

Introduction

On June 11, 2013, at approximately 09:02 while conducting a "hot reactor startup," the Reactor Operator noted, immediately after stabilizing reactor power level at 5 MW, that the heights of shim control blades 'B' and 'D' were at 21.90 and 23.10 inches withdrawn, respectively. This 1.20 inch difference in shim blade height created a deviation from Technical Specification (TS) 3.2.b, which states, "*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.*" The basis for this Specification is to provide "*...a restriction on the maximum neutron flux tilting that can occur in the core to insure the validity of the power peaking factors described in Section 3.3 of Add. 3 to HSR.*" Section 3.3 of Addendum 3 to the Hazards Summary Report (HSR) provides an evaluation of the power peaking factors in the University of Missouri Research Reactor (MURR) 6.2 kilogram core. Based on the nuclear peaking factors provided in Addendum 3 to the HSR, Appendix F of Addendum 4 to the HSR provides the safety limit (SL) analysis and curves for Mode I and II operation (5 and 10 MW operation) for primary coolant pressurizer pressures of 60 and 75 psia whereas Section 6.0 of Addendum 5 to the HSR extended the original analysis to include a third SL curve for a pressurizer pressure of 85 psia, i.e. the nominal operating pressure.

Although a deviation from TS 3.2.b did occur, the discussion in the Safety Analysis section of this LER shows that the hot spot peaking factor calculated for the core configuration and conditions at the time of this event is well below the highest hot spot peaking factor used in the MURR SL Analysis; therefore, the unbalanced shim blade heights did not create a safety hazard to the reactor.

Description of the Control Blade and Rod Control System

The reactivity of the reactor is controlled by five neutron-absorbing control blades. Each control blade is coupled to a control blade drive mechanism by means of a support and guide extension (offset mechanism). Four of the control blades, referred to as the shim blades, are used for coarse adjustments to the neutron density within the reactor core. The shim control blades are constructed of formed BORAL® plate which is, by weight, 52% ±2% boron carbide and 48% ±2% aluminum. The boron carbide-aluminum mixture is clad with 0.0375 inches of aluminum-alloy 1100 for a nominal blade thickness of 0.175 ±0.007 inches. The minimum weight of boron-10 per unit cross sectional area is 1.0418 gm/cm^2. The active length of the neutron absorbing material is 34 inches. Each shim blade occupies approximately 72° of a circular arc around the outer reactor pressure vessel. The fifth control blade is a regulating blade. The low reactivity worth of this blade allows for very fine adjustments in the neutron density in order to maintain the reactor at the desired power level.

The four shim blades are actuated by electro mechanical control blade drive mechanisms that position, hold, and scram each shim blade. Each control blade drive mechanism consists of a 0.02-HP, 115-volt, one-amp, single-phase, 60-cycle motor connected to a lead screw assembly through a reduction gearbox and overload clutch. The reactivity worth and speed of travel for the control blades are sufficient to allow complete control of the reactor system from a shutdown condition to full power operation. The insertion rate of the control blades is adequate to ensure prompt shutdown of the reactor in the event a scram signal is received. The nominal speed of the shim blades in one inch per minute in the outward direction and two inches per minute in the inward direction.

The shim and regulating blades are withdrawn or inserted manually by three-position ("In-Normal-Out") switches located on the reactor control console. The switches are spring return to the mid-position ("Normal") when released. A five-position ("A-B-C-D-Gang") selector switch enables the reactor operator to select the shim blades individually or as a group.

Detailed Event Description

On June 11, 2013, at 07:28 with the reactor operating at 10 MW in the automatic control mode, an unscheduled power reduction occurred when a "Rod Not In Contact With Magnet" rod run-in was automatically initiated as a result of shim control blade 'A' separating from its electro-magnet during a routine shimming evolution. The reactor was subsequently shutdown. Rod run-ins of this nature occasionally occur because of slight misalignments between the offset mechanism pull rod and housing. The misalignment was corrected and the control blade was satisfactorily withdrawn to the full out position as part of the retest by performing compliance procedure CP-10, "Rod Drop Times." The day prior (June 10), the reactor had reached 10 MW at 20:13 following the completion of maintenance activities that were conducted during the normally scheduled weekly shutdown. Because the reactor had only operated for 11 hours and 15 minutes at 10 MW, a decision was made by the Reactor Manager to conduct a "hot reactor startup" instead of performing a complete core refueling with fuel elements containing no xenon poison. A hot reactor startup, as defined by administrative procedure AP-RO-110, "Conduct of Operations," is "*A startup in which restart capability is in doubt.*" This is based on the ability to override xenon during a reactor restart following an unplanned/unscheduled power reduction. Hot reactor startups are conducted in accordance with operating procedure OP-RO-211, "Reactor Startup – Hot" (Attachment 1), and are very seldom performed because restart capability typically no longer exists about 24 to 36 hours after a normal reactor startup following core refueling.

On June 11, 2013 at 08:38, one hour and 10 minutes after the unscheduled power reduction, a hot reactor startup was commenced in accordance with OP-RO-211 by a Reactor Operator under the direct supervision of a Senior Reactor Operator [who was also the Lead Senior Reactor Operator (LSRO)]. The Assistant Reactor Manager-Operations was also in the Control Room during the startup. While approaching a power level of 5 MW, the shim control blades were repeatedly inserted in "gang" control in small increments to stabilize reactor power and to overcome the positive reactivity that was added by turning on secondary coolant system circulation pumps and cooling tower fans in order to control primary and pool coolant temperatures. During each "gang" insertion, the difference between the heights of the shim control blades increased. This condition went unnoticed until reactor power level was stabilized at 5 MW at which time (09:02) the Reactor Operator noted that the height of shim control blade 'B' was 21.90 inches whereas the height of shim control blade 'D' was 23.10 inches. This 1.20 inch difference in control blade height created a deviation from TS 3.2.b, which states, "*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.*" The LSRO was promptly informed of the shim blade height difference. At that point, the LSRO immediately directed the Reactor Operator to adjust the height of the shim control blades so that no greater than a 0.9 inch difference existed between the highest and lowest shim blades while also maintaining reactor power level at approximately 5 MW. Note: Although the TS allows the reactor to be operated above 100 kW with a maximum distance between the highest and lowest shim control blade at one inch, MURR further limits this distance administratively to no greater than 0.9 inches. This limitation is listed in the "Precautions and Limitations" section of OP-RO-211 as well as a caution box before Step 5.2.7, which allows continuation to 5 MW after the reactor achieves criticality. The reactor startup was paused and all reactor parameters were verified to be normal for 5 MW operation. After verifying all parameters and discussing the event, permission to continue with the hot reactor startup was obtained from the Assistant Reactor Manager-Operations. The reactor reached 10 MW at 09:17.

A thorough visual inspection of all the eight fuel elements associated with core 13-31 was performed by the Reactor Manager after removal from the core. Additionally, the online fission product monitor indicated no increase in coolant activity and the radio-chemical analysis of the weekly scheduled primary coolant sample indicated no abnormalities.

Safety Analysis

As stated above, the basis for TS 3.2.b is to provide "*...a restriction on the maximum neutron flux tilting that can occur in the core to insure the validity of the power peaking factors described in Section 3.3 of Add. 3 to HSR.*" Section 3.3 of Addendum 3 to the HSR provides an evaluation of the power peaking factors in the MURR 6.2 kilogram core. Based on the nuclear peaking factors provided in Addendum 3 to the HSR, Appendix F of Addendum 4 to the HSR provides the safely limit (SL) analysis and curves for Mode I and II operation (5 and 10 MW operation) for primary coolant pressurizer pressures of 60 and 75 psia whereas Section 6.0 of Addendum 5 to the HSR extended the original analysis to include a third SL curve for a pressurizer pressure of 85 psia, i.e. the nominal operating pressure. The SLs are based on a combination of worst-case power peaking factors and associated hot channel enthalpy rise. The safety analysis for natural convective cooling of the core (Mode III operation) is provided in Section 5.5.3 of the HSR.

On August 24, 2011, MURR submitted an application (Attachment 2) to the U.S. Nuclear Regulatory Commission (NRC) to amend Amended Facility License R-103 by revising TS 2.1, "Reactor Core Safety Limit," because of an error that was discovered in the MURR SL Analysis while answering a relicensing Request for Additional Information (RAI) question. (Note: On August 31, 2006, MURR submitted a request to the NRC to renew Amended Facility Operating License R-103.) The current MURR SL curves were developed in 1973 by the NUS Corporation for the 1974 uprate in power from 5 to 10 MW.

As stated in the Amendment application, the NUS Corporation used the Advanced Test Reactor (ATR) preliminary flow tests, which were performed in 1964 and 1966 by Croft and Waters, to show that the flow instability burnout heat flux was 0.6 of the critical heat flux (CHF) predicted by the Bernath CHF Correlation. This supported using a more conservative value of 0.5 of the Bernath Correlation to develop the Mode I and II SLs. If the local value of heat flux anywhere in a fuel element exceeds 50% of the local CHF value, as predicted by the Bernath Correlation, then flow instability is assumed to have occurred. The error that was discovered was a discrepancy between the "diameter of heated surface," known as the variable D_i, as it is defined by Bernath and a more commonly used "heat diameter" definition inadvertently used by the NUS Corporation when developing the SL curves.

The revised SLs are based on new power peaking factors developed by a team of MURR staff working with staff from Argonne National Laboratory (ANL). As described in the August 31, 2010 response to relicensing RAI question 4.17 regarding the NUS

Corporation developed SLs, the power peaking factors used were extremely conservative because they utilized a combination of unrealistic or impossible peaking factors determined by three (3) different 2D diffusion code models, which was the only code method available in the early 1970's. Since 2006, the MURR has been actively collaborating with the Reduced Enrichment for Research and Test Reactor (RERTR) Program on the conversion from highly-enriched uranium (HEU) to low-enriched uranium (LEU) fuel. During this time, the ANL/MURR team has benchmarked the MURR HEU fuel and reactor core design performance. More accurate peaking factors that were used to determine the new SLs have now been developed. Note: As this LER was being finalized, notification was received from the NRC that the SL Amendment (Amendment No. 36) had been approved, dated July 8, 2013.

On May 6, 2010, the NRC requested additional information and clarification regarding the license renewal application in the form of 19 complex questions. One of the complex questions was in regard to unbalanced shim control blade height, specifically the question asked, "*Page 4-40 and Technical Specification (TS) 3.2 b. Discuss how the one inch limit restricts flux tilting to ensure power peaking factors.* (Question 4.14 d.)" Although the basis for the one inch limit is to ensure the validity of the power peaking factors in the SL analysis, nowhere is it described or discussed in any of the design and licensing documentation to how the one inch limit was obtained. On September 8, 2011, MURR responded to that question (Attachment 3). The following is a summary of that response.

The effect of unbalanced control blades on the nuclear peaking factors was checked using the computer code MCNP5. The models used were developed by a combined staff from MURR and ANL. The benchmarking models included the 1971 graphite reflector configuration, which was used for the detailed physics analysis of the original 6.2 Kg ^{235}U aluminide fuel core, and the 2008 graphite reflector, which was used for our mixed core (elements in various stages of burnup) fuel cycles.

The highly detailed equilibrium (MCNP5) models of various core configurations were then used to determine the worst case peaking factors. In particular, the geometry of each modeled core configuration included highly segmented fuel meat regions, where each fuel meat region (in every element) was divided into 24 one-inch axial sections. By tallying the energy deposited from the MCNP5 criticality calculations in each of the axial sections of any given fuel meat region, the nuclear peaking factors were derived as a function of axial fuel length. The largest axial peaking factors were found to be in the inner and outer most fuel meat positions (i.e. in fuel plate-1 and -24, respectively) with fuel plate-1 being the highest. The highest peaking factors can be found in the following two extreme core conditions: "Fresh Core" – eight fresh fuel elements or "Week 58 Core" – a mixed core with two fresh fuel elements (elements 1 and 5) located next to two fuel elements (elements 4 and 8) which will reach a burnup of 150 MWD at the end of the run.

The MCNP5 MURR core configurations used to obtain peaking factors were modeled based on the following five factors, each of which includes two extreme cases:

1. Variation in control blade height – all the same heights (banked) vs. blades 'A' and 'D' having a 4-inch height difference when compared to blades 'B' and 'C';
2. Fuel loading – "Fresh Core (0 MWD) vs. worst case "Week 58 Core" (two elements each with 0, 81, 65 and 142 MWD);
3. Graphite reflector configuration – 1971 vs. 2008;
4. Flux Trap (FT) – all water vs. sample holder loaded with samples inserted into the FT; and
5. Control blade critical position – xenon free vs. equilibrium xenon heights.

The following eight MURR core configurations were modeled based on the above listed five factors. The MCNP5 calculated highest heat flux and enthalpy rise nuclear peaking factors for each configuration are presented in Table 1 below for the hot spot and hot stripe on the hottest fuel plate, which is always fuel plate-1 of the hottest fuel element.

1. Control Blades banked, Fresh Core, 1971 Reflector
2. Control Blades 4-inch difference, Fresh Core, 1971 Reflector
3. Control Blades banked, Week 58 Core, 1971 Reflector
4. Control Blades 4-inch difference, Week 58 Core, 1971 Reflector
5. Control Blades banked, Fresh Core, 2008 Reflector
6. Control Blades 4-inch difference, Fresh Core, 2008 Reflector
7. Control Blades banked, Week 58 Core, 2008 Reflector
8. Control Blades 4-inch difference, Week 58 Core, 2008 Reflector

Table 1 – Summary of the Nuclear Peaking Factors
at the Hotspots of Eight MURR Core Extreme Configurations
(i.e., "Fresh Core" and "Week 58 Core")

Core Configuration	Peaking Factor		Peaking Factor	
	Hot Spot[1]	**With Engr. PF**	**Enthalpy Rise[1]**	**With Engr. PF**
1	3.08	3.65	1.97	2.09
2	3.17	3.75	2.03	2.15
3	3.06	3.62	2.01	2.14
4	3.06	3.62	2.05	2.18
5	3.08	3.64	1.95	2.07
6	3.12	3.70	1.99	2.11
7	3.01	3.57	1.99	2.11
8	3.06	3.62	2.03	2.15
Safety Limits[1]	**3.475**	**4.116**	**2.301**	**2.442**

Note 1: The results presented in Table 1 report the overall nuclear peaking factors for the hot spot heat flux and the hot channel enthalpy rise for the various MURR core configurations. The azimuthal peaking factor across the one-inch sections of fuel plate-1 is 1.07. Therefore the hot spot peaking factor is the highest peaking factor in the MCNP run for that model multiplied by 1.07 (circumferential peaking within plate-1). Then the hot spot peaking factor is multiplied by 1.03 and 1.15 (engineering hot channel factors on flux). The enthalpy rise peaking factor is the highest average plate-1 and -2, which heat coolant channel 2, peaking factors in the MCNP run for that model multiplied by 1.0921 (circumferential peaking within the channel 2). Then the enthalpy rise factor is multiplied by 1.03 and 1.03 (engineering hot channel factors on enthalpy rise).

Considering the power peaking factors for the core configurations with the highest peaking factors, the analysis demonstrates that even a 4-inch difference in shim control blade height would not exceed the HSR peaking factors. However, MURR chose not to revise TS 3.2.b based on this analysis, but to keep the limit at one inch.

Additionally, a detailed simulation of the initial June 10 startup core configuration leading to the June 11 event, with the core configuration and conditions at the time when the 1.20 inch shim blade height difference occurred, was performed using the following modified MONTEBURNS (ORIGEN coupled to MCNP5) and MCNP5 models:

1. Variation in control blade height – Shim blade 'A' at 23.00 inches; shim blade 'B' at 21.90 inches; shim blade 'C' at 23.00 inches; and shim blade 'D' at 23.10 inches with current boron-10 depletion modeled in each control blade;
2. Fuel loading – "Core 13-31 – 598 MWD" (two elements each with 0, 143, 59 and 97 MWD);
3. Graphite reflector configuration – Present (2013);

4. Flux Trap (FT) – sample holder loaded with samples inserted into the FT; and
5. Control blade critical position – 22.75 inches (calculated), non-equilibrium xenon. Note: Xenon concentration is a calculated value based on operating at 10 MWs for 11 hours and 15 minutes, followed by a shutdown of 1 hour and ten minutes, followed by a reactor startup to 5 MW in 24 minutes.

Figure 1 shows a hot spot peaking factor of 3.17 with an enthalpy rise peaking factor of 2.03 for the worst case core configuration used in the MURR SL Analysis: control blades with a 4-inch height difference; fresh core, and 1971 graphite reflector configuration. Note: The azimuthal peaking factor of fuel plate-1 is 1.07. Therefore, the hot spot peaking factor of 2.96 is the highest peaking factor shown in Figure 1 multiplied by 1.07 (2.96 x 1.07 = 3.17).

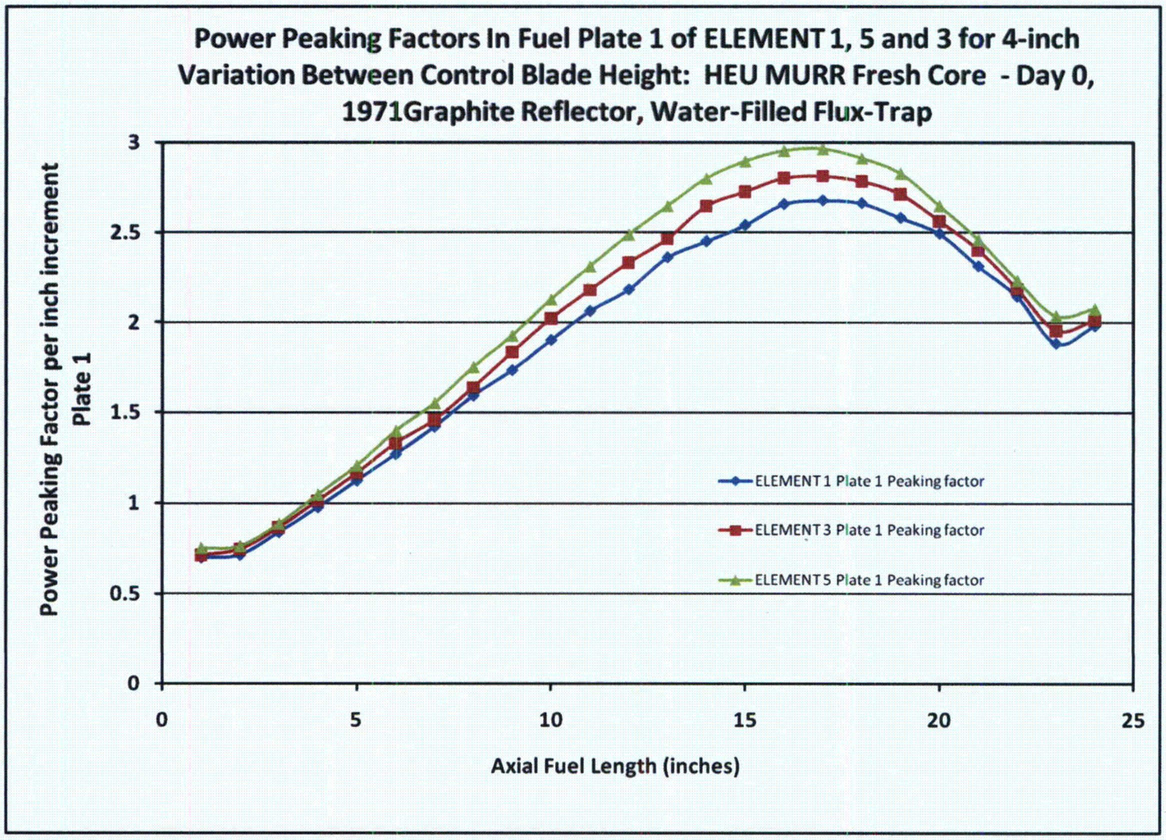

Figure 1
Hot Spot Peaking Factors – Control Blades 4-inch difference, Fresh Core, 1971 Reflector

Figure 2 shows a hot spot peaking factor of 2.37 for the core configuration and conditions (described above) at the time when the 1.20 inch shim blade height difference occurred. At 10 MW, the enthalpy rise peaking factor would have been slightly less than 2.03 (worst case core configuration used in the MURR SL Analysis), but at 5 MW this factor would have been only half as much. Note: The azimuthal peaking factor of fuel plate-1 is 1.06. Therefore, the hot spot peaking factor of 2.24 is the highest peaking factor shown in Figure 2 multiplied by 1.06 (2.24 x 1.06 = 2.37).

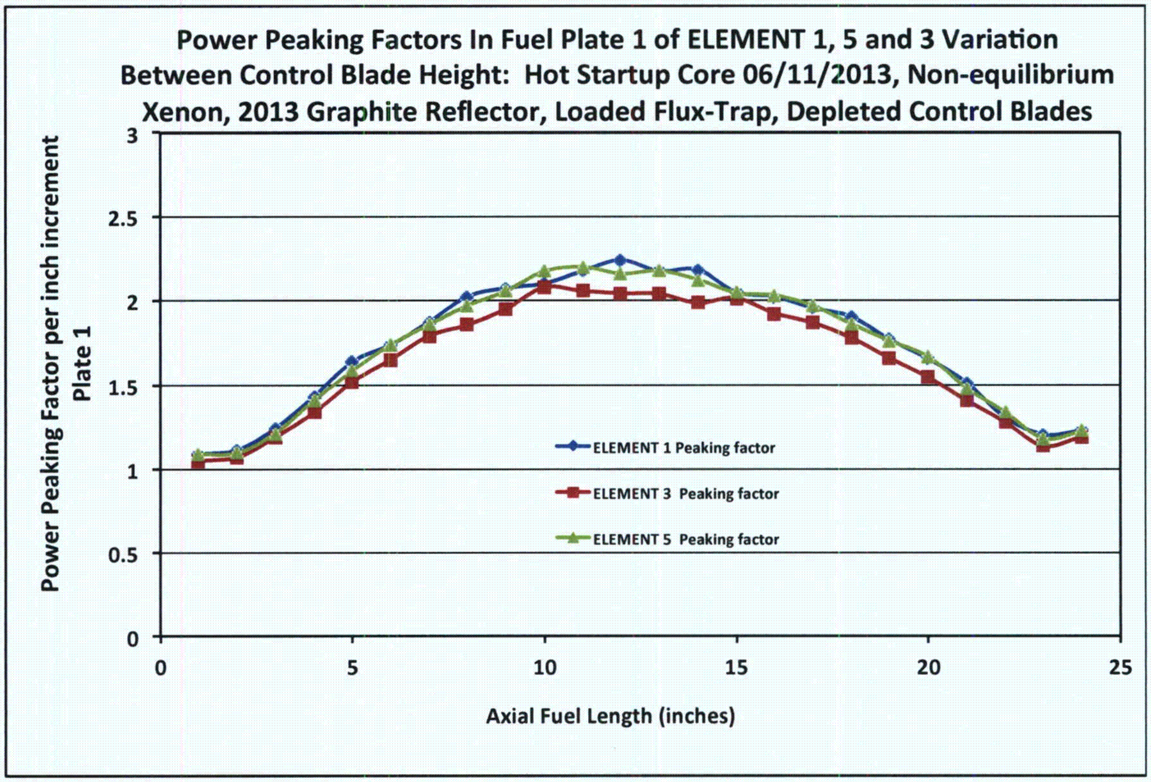

Figure 2
Hot Spot Peaking Factors – Control Blades 1.20-inch difference, Mixed Core 13-31, 2013 Reflector

Hence, the highest hot spot peaking factor of 2.37 that was calculated for the core configuration and conditions at the time of the event is well below the highest hot spot peaking factor of 3.17 used in the MURR SL Analysis with the hot channel enthalpy rise less than half.

Corrective Actions

When it was discovered that the height difference between shim control blades 'B' and 'D' was greater than one inch, the shim control blades were adjusted to restore compliance with TS 3.2.b. The reactor startup was paused at 5 MW to verify all reactor parameters were normal and to obtain permission from the Assistant Reactor Manager-Operations to continue with the hot reactor startup.

The Associate Director of Reactor and Facilities Operations met with the Reactor Manager and the Assistant Reactor Manager-Operations to discuss the event in detail and to consider what short- and long-term corrective actions should be implemented. The Associate Director of Reactor and Facilities Operations, the Reactor Manager and the Assistant Reactor Manager-Operations met with the Operations crew that was on shift when the event occurred to discuss the specifics of the event and to counsel the group on the severity of the incident.

A procedural change has been submitted to revise OP-RO-211, "Reactor Startup – Hot," such that only a Senior Reactor Operator can perform a hot reactor startup.

Senior Management will meet with all of the Reactor Operations Staff to ensure that all individuals are aware of the severity of the event.

Finally, although a deviation from TS 3.2.b did occur, the detailed simulation of the initial June 10 startup core configuration leading to the June 11 event using modified MONTEBURNS (ORIGEN coupled to MCNP5) and MCNP5 models confirmed that the hot spot peaking factor at the time the TS deviation occurred was well below the hot spot peaking factor used in the MURR SL Analysis; therefore, the unbalanced shim blade heights did not create a safety hazard to the reactor.

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

Sincerely,

John L. Fruits
Reactor Manager

ENDORSEMENT:
Reviewed and Approved,

For Ralph Butler

Ralph A. Butler, P.E.
Director

Attachment 1: Operating procedure OP-RO-211, "Reactor Startup – Hot"

Attachment 2: Written communication as specified by 10 CFR 50.4 regarding an application to amend Amended Facility License R-103 by revising the University of Missouri-Columbia Research Reactor Technical Specification 2.1, "Reactor Core Safety Limit," pursuant to 10 CFR 50.90, dated August 24, 2011

NOTARY PUBLIC
NOTARY SEAL
STATE OF MISSOURI

MARGEE P. STOUT
My Commission Expires
March 24, 2016
Montgomery County
Commission #12511436

Margee P. Stout

Attachment 3: 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 – Response dated September 8, 2011

ATTACHMENT 1

MURR

MASTER COPY
ISSUED JUN 2 5 2013

OPERATING PROCEDURE

OP-RO-211

REACTOR STARTUP - HOT

RESPONSIBLE GROUP: Reactor Operations

PROCEDURE OWNER: Rob Hudson

APPROVED BY: John Fruits [signature] Date 5-30-13

This procedure contains the following:

Pages	1	through	9
Attachments	None	through	
Tables	1	through	1
Figures	None	through	
Appendices	None	through	
Check-Off Lists	None	through	

TABLE OF CONTENTS

REACTOR STARTUP - HOT

1.0 PURPOSE

1.1 Provide instructions for performing a hot reactor startup.

2.0 SCOPE

2.1 Contains procedural steps and precautions required to perform a hot reactor startup.

3.0 PRECAUTIONS AND LIMITATIONS

3.1 A Senior Reactor Operator (SRO) or a Reactor Operator under the direct supervision of an SRO must perform this startup procedure.

3.2 A hot reactor startup is performed because the ability to override Xenon is in doubt.

3.3 To preserve maximum positive reactivity, Secondary Coolant Pumps and Cooling Tower Fans should not be started until absolutely necessary.

3.4 It is always acceptable to operate the reactor in manual control as directed by the Lead Senior Reactor Operator (LSRO).

3.5 Above 100 kilowatts (kW) the reactor shall be operated so that the maximum distance between the highest and lowest shim blade shall not exceed 0.90 inches.

3.6 Ganged Control Rods, or any single Control Rod, must not be withdrawn at the same time with the Regulating Blade.

3.7 Gang Rod Control may be used to reduce reactor power.

3.8 The reactor must not be restarted following a scram until the cause of the scram has been determined and appropriate corrective actions implemented.

4.0 PREREQUISITES AND INITIAL CONDITIONS

____ 4.1 The reactor shall not be started unless the Source Range is indicating a neutron count rate of at least 1 cps and the Wide Range monitor is indicating a power level above 1 watt or the Source Range monitor is indicating a neutron count rate of at least 2 cps and is verified just prior to startup by a neutron test source or movement of the monitor that the channel is responding to neutrons. **(TS 3.4.e)**

____ 4.2 An estimated critical position (ECP) has been provided by the LSRO.

____ 4.3 If shutdown was unscheduled and the cause cannot be determined, permission has been obtained from the Reactor Manager to restart the reactor.

5.0 PROCEDURE

5.1 PREPARATION:

NOTE: It is advisable to achieve full power automatic control at a reactor power level that is 4% to 5% less than the desired power and then increase reactor power based on actual power level data.

5.1.1 SET "Power Level Set" to desired initial In-Auto point using Power Schedule Selector Switch 1S9.

5.1.2 SET Intermediate Range Monitor Level Recorder to fast speed (1000 mm/hr) AND RECORD "Fast" (with time and date) on chart.

5.1.3 ENSURE Source Range Monitor Level Recorder AND Scaler are operating.

5.1.4 Place Primary Coolant System Temperature Controller in Manual control AND Close Valve S-1.

5.1.5 Place Pool Coolant System Temperature Controller in Manual control AND Open Valve S-2.

5.1.6 RECORD a set of Startup Nuclear Data.

5.1.7 Obtain permission from LSRO to startup reactor.

5.0 PROCEDURE (CONT.)

5.2 STARTUP:

5.2.1 Announce to facility "Commencing a Hot Reactor Startup."

> NOTE: Significant usable positive reactivity can be obtained by proper control of Primary and Pool Coolant System temperature. The LSRO is responsible for the correct coolant temperature adjustment.

> NOTE: It is preferable to pass through the critical position and develop a positive period before declaring criticality.

TABLE 7.1
ECP Limitations

Prior Use Core Configuration ECP Limits

ECP 11" - 16"	Limit (± 0.75")
ECP 16" - 22"	Limit (± 1.25")
ECP 22" - 26"	Limit (± 2.50")

5.2.2 START withdrawing Control Rods in gang control to criticality within Table 7.1 ECP limits.

5.2.3 RECORD "Commenced a Hot Reactor Startup" in the Console Log Book.

> NOTE: Stopping Control Rod motion is not required to record Startup Nuclear Data.

5.2.4 RECORD a set of Startup Nuclear Data at 5.0 inch increments.

5.2.5 IF reactor is not critical within the table 7.1 ECP limits, THEN insert Control Rods to 2.0 inches below ECP, AND NOTIFY the LSRO.

5.2.6 RECORD "Reactor is Critical" in the Console Log Book.

5.0 PROCEDURE (CONT.)

CAUTION: Above 100 kW the reactor shall be operated so that the maximum distance between the highest and lowest shim blade shall not exceed 0.90 inches.

5.2.7 CONTINUE startup to 5 MW at a Period that is longer than 30 seconds.

5.2.8 MAINTAIN Wide Range Monitor Level indication on scale by switching Range Switch 1S2 upscale.

5.2.9 STOP startup at 5 MW.

5.2.10 RECORD "Reactor Power is at 5 MW" in the Console Log Book.

5.2.11 RECORD "Critical Rod Position" data AND power level information on Startup Nuclear Data Sheet. **(TS 6.1.g.(1))**

NOTE: Reactor power level must remain at 5 MW for greater than five minutes to allow temperatures and nuclear instrumentation to stabilize. The LSRO will make the determination for any additional stabilization time.

NOTE: An automatic power calculator called a "Reactor Power Calculator" is provided. If the "Reactor Power Calculator" is operating accurately, this device can be used for all power calculations. Manual power calculation is always acceptable.

5.2.12 After greater than 5 minutes:

a) VERIFY proper response of "Reactor Power Calculator." IF "Reactor Power Calculator" is inaccurate, THEN perform heat balance calculations manually.

b) VERIFY nuclear instrumentation is in close agreement with power level indication.

c) VERIFY process instrumentation AND Area Radiation Monitoring System (ARMS) are within acceptable limits to support a power increase to 10 MW.

d) RECORD indicated power level in the Console Log Book.

5.0 PROCEDURE (CONT.)

5.2.13 CONTINUE reactor startup to 7.5 MW.

a) At a period longer than 100 seconds, withdraw the regulating Blade until "Reg Blade 60% Withdrawn" Annunciator Window is lit.

b) START withdrawal of Control Rods one-at-a-time.

5.2.14 RECORD "Proceeding to 7.5 MW" in the Console Log Book.

5.2.15 RECORD Power level information on Startup Nuclear Data Sheet. **(TS 6.1.g(1))**

5.2.16 STOP startup at 7.5 MW.

5.2.17 RECORD "Reactor Power is at 7.5 MW" in the Console Log Book.

5.2.18 RECORD power level information on Startup Nuclear Data Sheet. **(TS 6.1.g(1))**

NOTE:	Reactor power level must remain at 7.5 MW for greater than five minutes to allow temperatures and nuclear instrumentation to stabilize. The LSRO will make the determination for any additional stabilization time.

NOTE:	If the Reactor Power Calculator is operating and indicating accurately, it can be used for all power calculations. Manual power calculation is always acceptable.

5.2.19 After greater than 5 minutes:

a) VERIFY proper response of Reactor Power Calculator. IF Reactor Power Calculator is inaccurate, THEN perform heat balance calculations manually.

b) VERIFY nuclear instrumentation is in close agreement with power level indication.

c) VERIFY process instrumentation AND Area Radiation Monitoring System (ARMS) are within acceptable limits to support a power increase to 10 MW.

d) RECORD indicated power level in the Console Log Book.

5.0 PROCEDURE (CONT.)

5.2.20 CONTINUE reactor startup to 10 MW.

a) START withdrawal of Control Rods one-at-a-time.

5.2.21 RECORD "Proceeding to 10 MW" in the Console Log Book.

5.2.22 RECORD Power level information on Startup Nuclear Data Sheet. **(TS 6.1.g(1))**

NOTE:	The following are automatic reactor control permissives: • Intermediate Range Level-2 and Intermediate Range Level-3 Period longer than 35 seconds • Range Switch 1S2 on 5 KW red or greater scale • Wide Range Monitor Level Recorder indicating greater than the auto prohibit set point • "REG BLADE 60% WITHDRAWN" Annunciator Window lit

NOTE:	Automatic reactor control will be terminated by any of the following: • Depressing the Rod Control Mode "MAN" Switch S1-2 • Moving the Regulating Blade Operate Switch 1S5 • Activating any Scram or Rod Run-In • Lowering power less than the Wide Range Monitor Level Recorder auto prohibit set point

NOTE:	Good operating practice is to have the first automatic movement of the Regulating Blade to be in the inward direction. This permits rod motion in a conservative direction while verifying proper automatic rod control circuitry operation.

NOTE:	Primary and Pool Coolant Temperature Controllers may be placed in Automatic control at the discretion of the LSRO.

5.2.23 WHEN Wide Range Monitor Level indication is greater than 2% above "Power Level Set," THEN depress Rod Control Mode "Auto" Switch S2-2.

5.0 PROCEDURE (CONT.)

5.2.24 RECORD "Reactor Power is at 10 MW in Auto Control" in the Console Log Book.

5.2.25 RECORD Power level information on Startup Nuclear Data Sheet. **(TS 6.1.g(1))**

5.2.26 Announce to facility "The Reactor is Operating at 10 MW."

5.2.27 STOP Source Range Monitor Recorder AND Scaler, AND RECORD an "Off" notation (with time and date) on chart.

5.2.28 SET Intermediate Range Monitor Level Recorder to slow speed (125 mm/hr) AND RECORD a "Slow" notation (with time and date) on chart.

5.2.29 WHEN temperatures have stabilized, as determined by LSRO, THEN:

a) Place the Primary Coolant System Temperature Controller in Auto,

b) Place the Pool Coolant System Temperature Controller in Auto,

c) RECORD a set of Nuclear and Process data.

6.0 RECORDS

6.1 Console Log Book

6.2 FM-43, "Nuclear and Process Data Sheet"

6.3 FM-55, "Startup Nuclear Data Sheet"

7.0 TABLES

7.1 ECP Limitations

ATTACHMENT 2

UNIVERSITY *of* MISSOURI

RESEARCH REACTOR CENTER

August 24, 2011

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 regarding an application to amend Amended Facility License R-103 by revising the University of Missouri-Columbia Research Reactor Technical Specification 2.1, "Reactor Core Safety Limit," pursuant to 10 CFR 50.90

The University of Missouri Research Reactor (MURR) is requesting approval from the U.S. Nuclear Regulatory Commission (NRC) to revise Technical Specification (TS) 2.1, "Reactor Core Safety Limit," because of an error that was discovered in the MURR Safety Limit (SL) Analysis while answering a relicensing Request for Additional Information (RAI) question.

The current MURR SL curves were developed in 1973 by the NUS Corporation for the 1974 uprate in power from 5 to 10 MW. These curves establish the maximum allowable reactor power limits, the dependent variable, for safe operation under different combinations of three (3) measureable independent operating parameters – primary coolant flow, reactor inlet water temperature, and primary coolant pressurizer pressure. The limits provide the basis for determining the Limiting Safety System Settings (LSSSs) and operating limits for 5 and 10 MW operation (also known as Mode II and Mode I operation, respectively).

Appendix F of Addendum 4 to the MURR Hazards Summary Report (HSR) (Attachment 1) provided the SL Analysis for Mode I and II operation. This analysis generated two (2) SL curves corresponding to primary coolant pressurizer pressures of 60 and 75 psia. Attachment 2 (*Safety Limit Analysis for the MURR Facility*, NUS Corporation, NUS-TM-EC-9, May 1973) was the base document that was used in preparing Appendix F. Section 6.0 of Addendum 5 to the MURR HSR (Attachment 3) extended the original analysis to include a third SL curve for a pressurizer pressure of 85 psia, i.e. the nominal operating pressure. The power peaking factors used in the determination of the MURR SLs for Mode I and II operation are provided in Section 3.3 of Addendum 3 to the MURR HSR (Attachment 4).

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ML11237A088

Appendix H of Addendum 4 to the MURR HSR (Attachment 5) provides the bases for determining the LSSSs for Mode I and Mode II operation. The safety analysis for natural convective cooling of the core (Mode III operation) is provided in Section 5.5.3 of the MURR HSR (Attachment 6).

By letter dated January 17, 2011, a report, as required by MURR TS 6.1.h (2), was submitted to the NRC which detailed the error and the subsequent actions after the error was discovered. As stated in the report, the NUS Corporation used the Advanced Test Reactor (ATR) preliminary flow tests, which were performed in 1964 and 1966 by Croft and Waters (Attachments 7 and 8), to show that the flow instability burnout heat flux was 0.6 of the critical heat flux (CHF) predicted by the Bernath CHF Correlation (Attachment 9). This supported using a more conservative value of 0.5 of the Bernath Correlation to develop the MURR Mode I and II SLs. If the local value of heat flux anywhere in a fuel element exceeds 50% of the local CHF value, as predicted by the Bernath Correlation, then flow instability is assumed to have occurred. The error that was discovered was a discrepancy between the "diameter of heated surface," known as the variable D_i, as it is defined by Bernath and a more commonly used "heat diameter" definition inadvertently used by the NUS Corporation when developing the SL curves.

Revised SL tables and curves for Mode I and II operation have been developed applying the correct Bernath D_i definition, however no changes to the current MURR LSSSs are required in order to maintain approximately the same safety margins. Additionally, the safety analysis for Mode III operation is based on the original 5.2 Kg ^{235}U alloy fuel which assumed a combination of power peaking factors that exceed the combined power peaking factors for the current 6.2 Kg ^{235}U aluminide fuel core. Therefore, the analysis in Section 5.5.3 still conservatively envelopes Mode III operation of the MURR and requires no revision.

The revised SLs are based on new power peaking factors (Attachment 10) developed by a team of MURR staff working with staff from Argonne National Laboratory (ANL). As described in the August 31, 2010 submittal to relicensing RAI 4.17 regarding the NUS Corporation developed SLs, the power peaking factors used were extremely conservative because they utilized a combination of unrealistic or impossible peaking factors determined by three (3) different 2D diffusion code models, which was the only code method available in the early 1970's. Since 2006, the MURR has been actively collaborating with the Reduced Enrichment for Research and Test Reactor (RERTR) Program on the conversion from highly-enriched uranium (HEU) to low-enriched uranium (LEU) fuel. During this time, the ANL/MURR team has benchmarked the MURR HEU fuel and reactor core design performance. We now have more accurate peaking factors that were used to determine the new SLs.

With the revised curves, the MURR has a new SL of 14.894 MW for Mode I operation with all three (3) non-power LSSS variables set at their corresponding limits, i.e. primary coolant pressurizer pressure at 75 psia, total core flow rate at 3200 gpm, and reactor inlet water temperature at 155 °F. This provides a 2.39 MW margin between the reactor power LSSS of 12.5 MW and the SL. This is actually slightly higher than the previous NUS Corporation calculated SL of 14.892 MW. For Mode II operation, with the revised SL curves, the MURR has a new SL of 8.763 MW with all three (3) non-power LSSS variables once again set at their corresponding limits, i.e. primary coolant pressurizer pressure at 75 psia, total core flow rate at 1575 gpm, and reactor inlet water temperature at 155 °F. This provides a 2.51 MW margin between the reactor power LSSS of 6.25 MW and the SL.

Attached are the revised Appendix F (Attachment 11) and Appendix H (Attachment 12) for Addendum 4 to the MURR HSR which supports the change to MURR TS 2.1. The revised Appendix F combines and replaces the current versions of the following three documents:

1. Section 3.3 of Addendum 3 to the MURR HSR (Attachment 4);
2. Appendix F of Addendum 4 to the MURR HSR (Attachment 1); and
3. Section 6.0 of Addendum 5 to the MURR HSR (Attachment 3).

Also attached are the current TS 2.1 (Attachment 13) and the draft TS 2.1 pages (Attachment 14) that will implement the requested change and a document from ANL (Attachment 15), which is referenced in the revised Appendix F, which helped determine the margin to flow instability for any MURR core coolant channel and the outlet saturation temperature.

If there are any questions regarding this application, 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

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

NOTARY PUBLIC NOTARY SEAL STATE OF MISSOURI

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

Margee P. Stout

Attachments:

1. MURR Hazards Summary Report, University of Missouri Research Reactor Facility, Addendum 4, Appendix F, *Safety Limit Analysis for the MURR*, October 1973.

2. Vaughan, F.R., "Safety Limit Analysis for the MURR Facility," NUS Corporation, NUS-TM-EC-9, Appendix A, May 1973.

3. MURR Hazards Summary Report, University of Missouri Research Reactor Facility, Addendum 5, Section 6.0, *Addendum to the Safety Limit Analysis for the MURR*, January 1974.

4. MURR Hazards Summary Report, University of Missouri Research Reactor Facility, Addendum 3, Section 3.3, *Evaluation of Power Peaking Factors in the MURR 6.2 Kg Core*, 1972.

5. MURR Hazards Summary Report, University of Missouri Research Reactor Facility, Addendum 4, Appendix H, *Bases for Limiting Safety System Settings for Modes I and II Operation*, October 1973.

6. MURR Hazards Summary Report, University of Missouri Research Reactor Facility, Section 5.5.3, *Analysis of Natural Convective Cooling of the Core*, July 1965.

7. Croft, M.W., "Advanced Test Reactor Burnout Heat Transfer Tests," USAEC Report IDO-24475, Babcock & Wilcox Co., January 1964.

8. Waters, E.D., "Heat Transfer Experiments for the Advanced Test Reactor," USAEC Report BNWL-216, Battelle-Northwest, May 1966.

9. Louis Bernath, "A Theory of Local-Boiling Burnout and Its Application to Existing Data," *Chemical Engineering Progress Symposium*, Series No. 30, Volume 56, pp. 95-116, 1960.

10. MURR Technical Data Report TDR-0125, "Feasibility Analyses for HEU to LEU Conversion of the University of Missouri Research Reactor (MURR)," University of Missouri-Columbia Research Reactor, Columbia Missouri, September 2009.

11. Revised Appendix F, *Safety Limit Analysis for the MURR*.

12. Revised Appendix H, *Bases for Limiting Safety System Settings for Modes I and II Operation.*

13. Current TS 2.1, "Reactor Core Safety Limit" (pages 1 through 6).

14. Revised TS 2.1, "Reactor Core Safety Limit" (pages 1 through 6).

15. Feldman, E.E., "Implementation of the Flow Instability Model for the University of Missouri Reactor (MURR) that is Based on the Bernath Critical Heat Flux Correlation," Conversion Program Nuclear Engineering Division, Argonne National Laboratory, June 2011.

TECHNICAL SPECIFICATION

UNIVERSITY OF MISSOURI
RESEARCH REACTOR FACILITY

SUBJECT: Reactor Core Safety Limit

Applicability

This specification applies to reactor power and reactor coolant system flow, temperature and pressure.

Objective

The objective is to set forth parameter safety limits which shall prevent damage to the fuel element cladding.

Specification

Reactor power, coolant system flow, temperature and pressure shall not exceed the following limits during reactor operation.

a. Mode I and II (Core Flow Rates ≥ 400 gpm)
The combination of the true values of the reactor power level, core flow rate, and reactor inlet water temperature shall not exceed the limits described by Figures 2.0, 2.1, and 2.2. The limits are considered exceeded if, for flow rates greater than 400 gpm, the point defined by the reactor power level and core flow rate is at any time above the curve corresponding to the true values of the reactor inlet water temperature and primary coolant system pressurizer pressure. To define values of the safety limits for

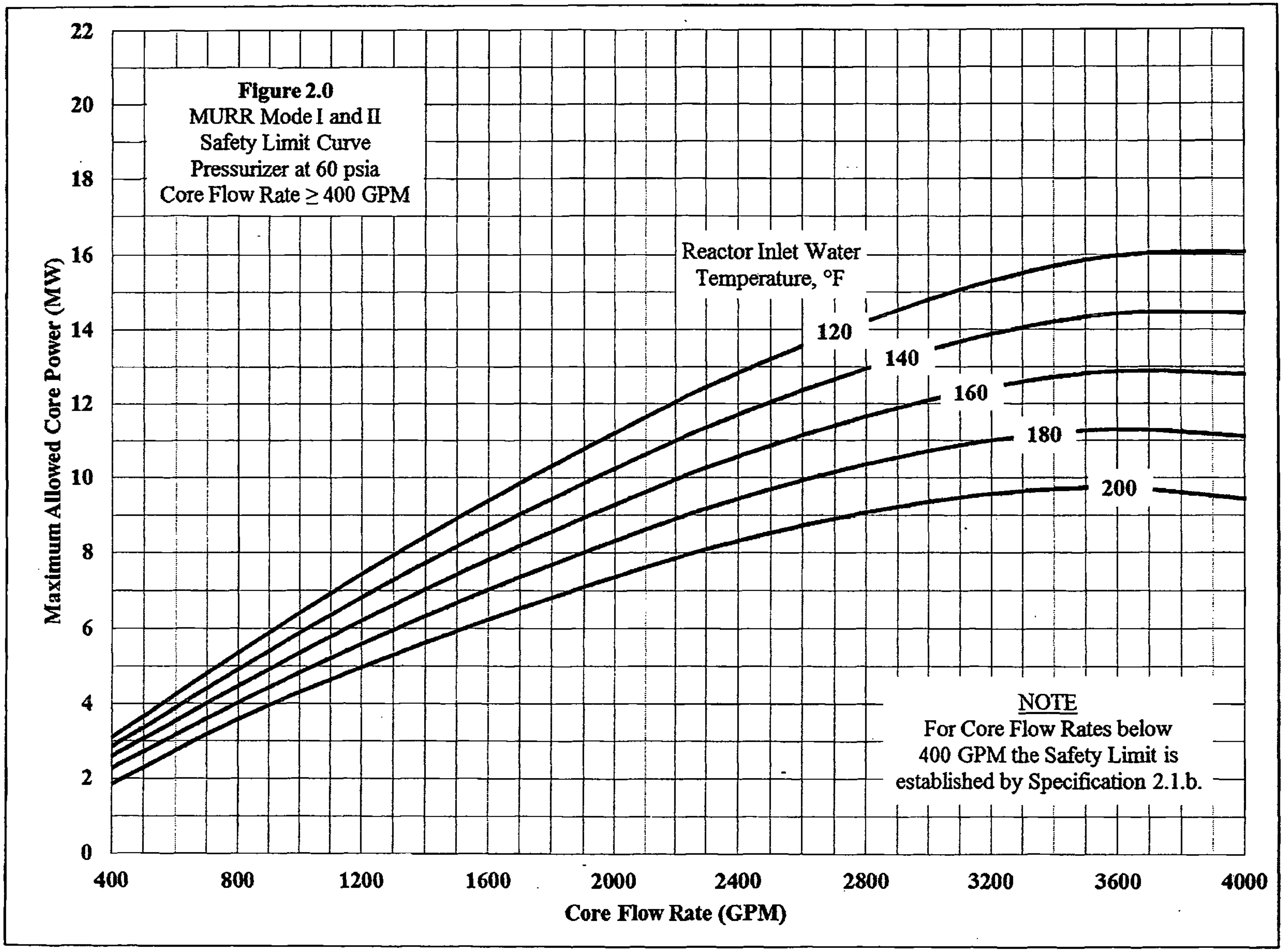

Figure 2.0
MURR Mode I and II
Safety Limit Curve
Pressurizer at 60 psia
Core Flow Rate ≥ 400 GPM

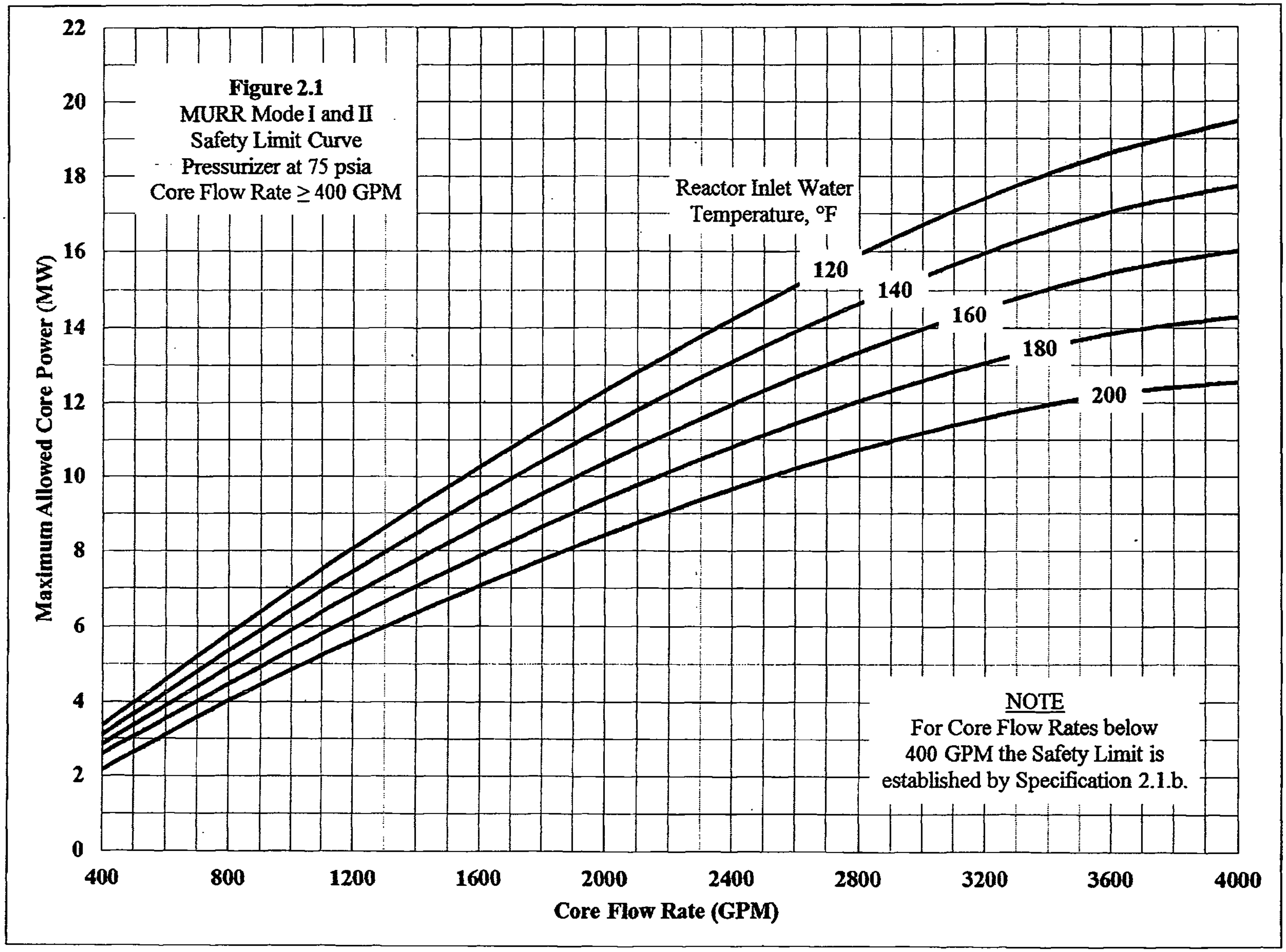

Figure 2.1
MURR Mode I and II
Safety Limit Curve
Pressurizer at 75 psia
Core Flow Rate ≥ 400 GPM

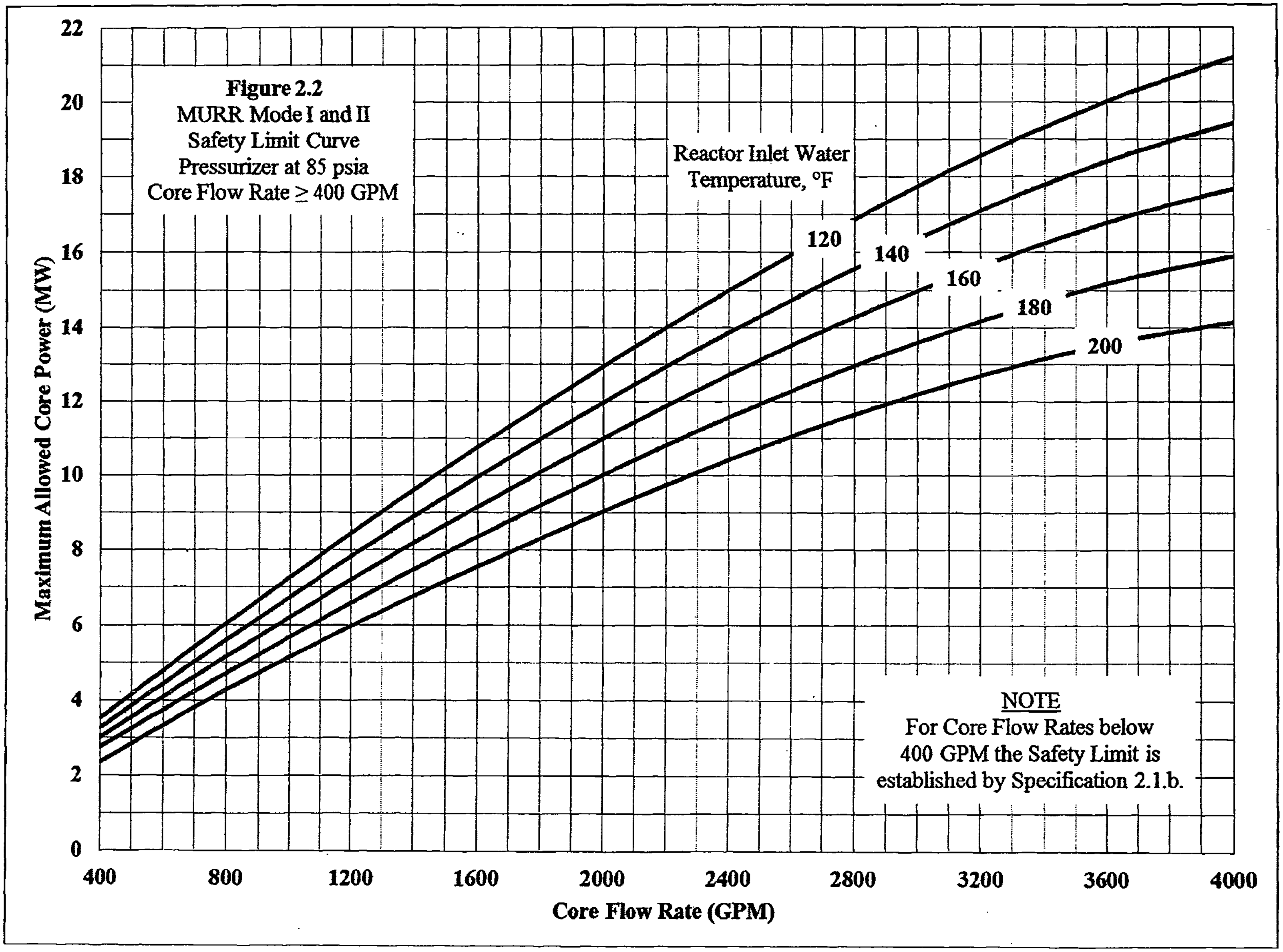
Figure 2.2
MURR Mode I and II
Safety Limit Curve
Pressurizer at 85 psia
Core Flow Rate ≥ 400 GPM
Reactor Inlet Water
Temperature, °F
120
140
160
180
200
NOTE
For Core Flow Rates below
400 GPM the Safety Limit is
established by Specification 2.1.b.
Maximum Allowed Core Power (MW)
0
2
4
6
8
10
12
14
16
18
20
22
Core Flow Rate (GPM)
400
800
1200
1600
2000
2400
2800
3200
3600
4000

SUBJECT: Reactor Core Safety Limit (continued)

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 (Core Flow Rates < 400 gpm)

Steady state power operations in Modes I and II are not authorized for a core flow rate < 400 gpm. Reactor operations with core flow below 400 gpm will occur only after a normal reactor shutdown when the primary coolant pumps are secured or following a loss of flow transient. Under the above conditions the maximum fuel cladding temperature shall not exceed 366 °F.

c. Mode III

Reactor Power..150 Kilowatts (maximum)

Bases

a. A complete safety limit analysis for the MURR is presented in Appendix F of Addendum 4 to the Hazards Summary Report (HSR). A family of curves is presented which relate reactor inlet water temperature and core flow rate to the reactor power level corresponding to a Critical Heat Flux (CHF) ratio of 2.0 based on the Bernath CHF Correlation. This also corresponds to a flow instability Departure from Nucleate Boiling Ratio (DNBR) of 1.2 based on the burnout heat flux data experimentally verified for ATR type fuel elements. Curves are presented for pressurizer pressures of 60, 75, and 85 psia. The safety limits were

SUBJECT: Reactor Core Safety Limit (continued)

chosen from the results of this analysis for Mode I and II operation, i.e. forced convection operation above 400 gpm flow.

b. Steady state reactor operation is prohibited for core flow rates below 400 gpm by the low flow scram settings in the safety system. The region below 400 gpm will only be entered following a reactor shutdown when the primary coolant pumps are secured or during a loss of flow transient where the reactor scrams, the flow coasts down to zero, reverses, and natural convection cooling is established. Below 400 gpm core flow the criterion for the safety limit is that fuel plate temperature must be less than 900 °F; the temperature at which fuel cladding failure could occur. The analysis of a loss of flow transient from the ultra-conservative conditions of 11 MW of power, 3000 gpm core flow and 155 °F core inlet temperature indicated a maximum fuel cladding temperature of 327 °F which is well below the cladding DNB temperature of 366 °F.

c. Analysis of natural convection cooling of the core (Mode III operation) is presented in section 5.5.3 of the HSR.

DCC

ATTACHMENT 3

UNIVERSITY *of* MISSOURI

RESEARCH REACTOR CENTER

September 8, 2011

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

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

Enclosed you will find the University of Missouri-Columbia Research Reactor's responses to the U.S. Nuclear Regulatory Commission's (NRC) request for additional information, dated May 6, 2010 (Complex Questions) and June 1, 2010 (45-Day Response Questions) regarding our renewal request for Amended Facility Operating License R-103, which was submitted to the NRC on August 31, 2006, as supplemented.

If you have any questions, please contact John L. Fruits, the facility Reactor Manager, at (573) 882-5319 or FruitsJ@missouri.edu.

Sincerely,

For Ralph Butler

Ralph A. Butler, P.E.
Director

RAB/djr

Enclosures

Christine M Errante

CHRISTINE M. ERRANTE
Notary Public - Notary Seal
State of Missouri
Commissioned for Boone County
My Commission Expires: April 14, 2015
Commission Number: 11528381

ML112550003

1513 Research Park Drive Columbia, MO 65211 Phone: 573-882-4211 Fax: 573-882-6360 Web: www.murr.missouri.edu
Fighting Cancer with Tomorrow's Technology

UNIVERSITY *of* MISSOURI

RESEARCH REACTOR CENTER

September 8, 2011

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.

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

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.

Attached are MURR's responses to six (6) of the remaining 45-Day Response and Complex Questions. With these responses, the following 45-Day Response and Complex Questions remain unanswered: 4.7, 6.2, 4.15, 4.16, 13.4.a, 13.4.b, 13.6, 13.7, C.1 and C.3. Additionally, MURR is currently evaluating which RAIs will need to be resubmitted based on the recent Amendment request to revise the MURR Safety Limits (ML 1123A088).

Because of the newly revised Safety Limits, 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,

for Ralph Butler

Ralph A. Butler, P.E.
Director

Christine M Errante

CHRISTINE M. ERRANTE
Notary Public - Notary Seal
State of Missouri
Commissioned for Boone County
My Commission Expires: April 14, 2015
Commission Number: 11528381

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

September 8, 2011

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

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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.

ENDORSEMENT:

Sincerely,

Reviewed and Approved,

John L. Fruits
Reactor Manager

Ralph A. Butler, P.E.
Director

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

4.14 Section 4.5.3, Operating Limits.

d. *Page 4-40 and Technical Specification (TS) 3.2.b. Discuss how the one inch limit restricts flux tilting to ensure power peaking factors.*

A summary of the MURR Hot Channel Factors is provided in Table F.4, Appendix F, of the August 24, 2011 University of Missouri-Columbia submittal to the U.S. Nuclear Regulatory Commission (NRC) requesting revised reactor Safety Limits (SLs) (Ref. 1). The MURR SL Analysis is based on an enthalpy rise overall peaking factor product of 2.4416 - the nuclear peaking and the engineering hot channel factor components are 2.3014 and 1.061, respectively. The peak heat flux power-related factors have an overall product of 4.116 - the nuclear peaking and the engineering hot channel factor components are 3.475 and 1.184, respectively. The difference in control rod heights could affect the nuclear peaking factor in both the enthalpy rise and the hot spot heat flux.

Therefore, the effect of unbanked control rods on the nuclear peaking factors was checked using the computer code MCNP5. The models used were developed by a combined staff from MURR and Argonne National Laboratory (ANL) for the future low-enriched uranium (LEU) fuel conversion of MURR. The benchmarking models include the MURR 1971 graphite reflector configuration to the detailed physics analysis on the original 6.2 Kg ^{235}U aluminide fuel core. The current graphite reflector configuration was in place in 2008 when the current model for benchmarking to our mixed core (elements in various stages of burnup) fuel cycle was performed. Burnup for the mixed cores was modeled using REBUS/DIF3 to compare the current highly-enriched uranium (HEU) fuel cycle to the proposed LEU fuel cycle.

To answer this question, the highly detailed equilibrium (MCNP5) models of various core configurations were used to determine the worst case peaking factors. In particular, the geometry of each modeled core configuration included highly segmented fuel meat regions, where each fuel meat region (in every element) was divided into 24 one-inch axial sections. By tallying the energy deposited from the MCNP5 criticality calculations in each of the axial sections of any given fuel meat region, the nuclear peaking factors were derived as a function of axial fuel length. The largest axial peaking factors were found to be in the inner and outer most fuel meat positions (i.e. in fuel plate-1 and -24, respectively) with fuel plate-1 being the highest. The highest peaking factors can be found in the following two extreme core conditions: "Fresh Core" – eight fresh fuel elements or "Week 58 Core" – a mixed core with two fresh fuel elements (elements 1 and 5) located next to two fuel elements (elements 4 and 8) which will reach a burnup of 150 MWD at the end of the run.

The MCNP5 MURR core configurations used to obtain peaking factors were modeled based on the following five factors, each of which includes two extreme cases:

1. Variation in control rod height – all the same heights (banked) vs. rods 'A' and 'D' having a 4-inch height difference when compared to rods 'B' and 'C';
2. Fuel loading – "Fresh Core (0 MWD) vs. worst case "Week 58 Core" (two elements each with 0, 81, 65, and 142 MWD);
3. Graphite reflector configuration – 1971 vs. present (2008);
4. Flux Trap (FT) – all water vs. sample holder loaded with samples inserted into the FT; and
5. Control rod critical position – xenon free vs. equilibrium xenon heights.

The modeled core configurations based on these five factors include the extreme power peaking MURR core configurations. To begin with, the worst case heat flux peaking is found in the "Fresh Core," which has not been used at MURR in about 30 years because of license possession limits. This core requires 6.2 Kg of ^{235}U fresh fuel, which exceeds the current 5 Kg unirradiated ^{235}U possession limit. Since 1973, in order to obtain the maximum burnup from the fuel elements, MURR has operated with a mixed core fuel cycle loaded with elements with burnups from low MWD to high MWD. With equilibrium xenon in the core, the control rods are fully withdrawn with a little more than 700 MWD on the core. During the weekly scheduled maintenance day, the MURR fuel cycle requires reloading the core with eight xenon free fuel elements with the average MWD burnup on a fuel element being.around 70 MWD. The typical core has four pairs of fuel elements with a pair of elements having the same MWD burnup and being loaded across from each other, i.e. fuel positions 1 and 5, 2 and 6, etc. After "Fresh Core," the next worst case heat flux peaking occurs in the extreme mixed burnup core loading.

In performing our LEU conversion feasibility study, the MURR fuel cycle had to be modeled for both the current HEU and the proposed LEU fuel element. For the two different types of fuel elements, over a year's worth of refueling or different core loadings were modeled. The worst case mixed burnup for the HEU modeled fuel cycle was labeled the "Week 58 Core." It consists of two fuel elements each with the following power history: 0, 81, 65 and 142 MWD. This worst case peaking occurs in the 0 MWD element loaded next to a 142 MWD element, which results in higher azimuthal peaking between fuel elements due to the difference in ^{235}U loading. These peaking factors will be higher than those of the average MURR core. The mixed burnup fuel cycle was modeled using the 2008 reflector configuration so that it could be benchmarked against xenon free mixed core loadings. The fuel element burnup atom densities obtained for the "Week 58 Core" were used in the week 58 cases for both the 1971 and 2008 graphite reflector configurations.

The graphite reflector configuration has varied over the years. The two reflector configurations selected are the extreme cases regarding their effect on overall reactivity. The original graphite reflector configuration, which provided the highest excess reactivity for a fresh core, was still being used in 1971 when the physics measurements of the first 6.2 Kg ^{235}U aluminide fuel core were performed. The 2008 graphite reflector configuration has the lowest excess reactivity of the various reflector configurations that have been used. This is because there is less graphite in the reflector elements due to the increase in irradiation positions and certain elements containing boron and cadmium. Results from MCNP5 KCODE calculations using a xenon free, mixed core model, with an empty (flooded) flux trap and banked control rods, showed that the difference in reactivity from the 1971 to the current graphite reflector configuration is -0.00638 $\Delta k/k$.

For these cases, the control rod heights describe the core in either xenon free or an equilibrium xenon state. The highest power peaking factors are always in fuel plate-1, therefore the xenon free control rod critical height and the flux trap region only containing pool water and not samples results in the higher peaking factors. The lower critical control rod height pushes the power density in and downward in the core increasing the peaking in fuel plate-1. The water filled flux trap region also increases the thermal flux reaching plate-1. Therefore, the results included in this answer contain only the peaking factors and graphs for the xenon free and water filled flux trap cases.

First, these worst case power peaking cases modeled show that the peaking factors for the heat flux power and enthalpy peaks are within the MURR hot channel factors on which the SLs are based. Then the control rods are modeled with a four-inch difference between them to see if the peaking factors are still within MURR hot channel factors given in Table F.4 of the revised SLs.

For the 1971 reflector configuration, control rods 'B' and 'C' are placed four inches above the height of control rods 'A' and 'D.' This results in the highest peaking in fuel element position 5 of the 1971 reflector configuration. For the current reflector configuration, control rods 'A' and 'D' are placed four inches above the height of control rods 'B' and 'C.' This results in the highest peaking in fuel element position 1 for the current reflector configuration because the two reflector elements containing boron and cadmium have control rods 'B' and 'C' between them and fuel elements 4, 5and 6, so raising them does not cause as much increase in the peaking factor.

Eight MCNP5 MURR core configurations were modeled based on the above listed five factors. The peaking factors of fuel plate-1 and -24 in fuel elements 1, 3 and 5 are displayed in Figures 1 through 8 for the different core configurations. The MCNP5 calculated highest heat flux and enthalpy rise nuclear peaking factors for each configuration are presented in Table 1 for the hot spot and hot stripe on the hottest fuel plate, which is always fuel plate-1 of the hottest fuel element.

Figures 1 through 8 are for the following control rod, reactor core and reflector configurations (all configurations are xenon free and the flux trap full of water):

Figure 1.a - Control Rods banked, Fresh Core, 1971 Reflector (Plate-1)
Figure 1.b - Control Rods banked, Fresh Core, 1971 Reflector (Plate-24)
Figure 2.a - Control Rods 4-inch diff., Fresh Core, 1971 Reflector (Plate-1)
Figure 2.b - Control Rods 4-inch diff., Fresh Core, 1971 Reflector (Plate-24)
Figure 3.a - Control Rods banked, Week 58 Core, 1971 Reflector (Plate-1)
Figure 3.b - Control Rods banked, Week 58 Core, 1971 Reflector (Plate-24)
Figure 4.a - Control Rods 4-inch diff., Week 58 Core, 1971 Reflector (Plate-1)
Figure 4.b - Control Rods 4-inch diff., Week 58 Core, 1971 Reflector (Plate-24)
Figure 5.a - Control Rods banked, Fresh Core, 2008 Reflector (Plate-1)
Figure 5.b - Control Rods banked, Fresh Core, 2008 Reflector (Plate-24)
Figure 6.a - Control Rods 4-inch diff., Fresh Core, 2008 Reflector (Plate-1)
Figure 6.b - Control Rods 4-inch diff., Fresh Core, 2008 Reflector (Plate-24)
Figure 7.a - Control Rods banked, Week 58 Core, 2008 Reflector (Plate-1)
Figure 7.b - Control Rods banked, Week 58 Core, 2008 Reflector (Plate-24)
Figure 8.a - Control Rods 4-inch diff., Week 58 Core, 2008 Reflector (Plate-1)
Figure 8.b - Control Rods 4-inch diff., Week 58 Core, 2008 Reflector (Plate-24)

Power Peaking Factors In Fuel Plate 1 of ELEMENT 1, 3 and 5 for no Variation Between Control Blade Height: HEU MURR Fresh Core - Day 0, 1971 Graphite Reflector, Water-Filled Flux-Trap

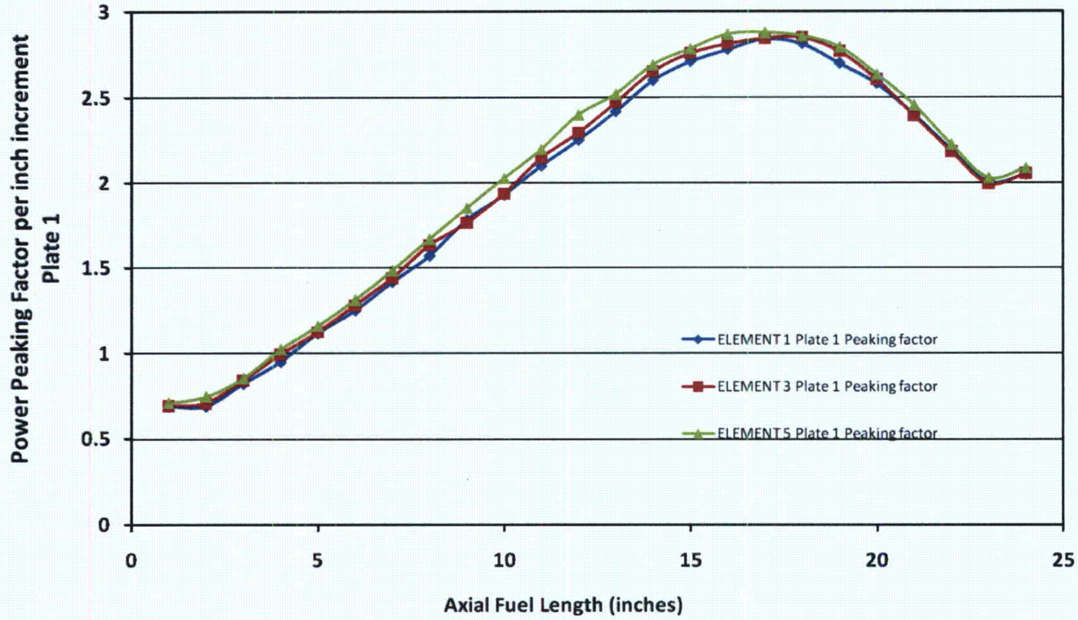

Figure 1.a

Power Peaking Factors In Fuel Plate 24 of ELEMENT 1, 3 and 5 for no Variation Between Control Blade Height: HEU MURR Fresh Core - Day 0, 1971 Graphite Reflector, Water-Filled Flux-Trap

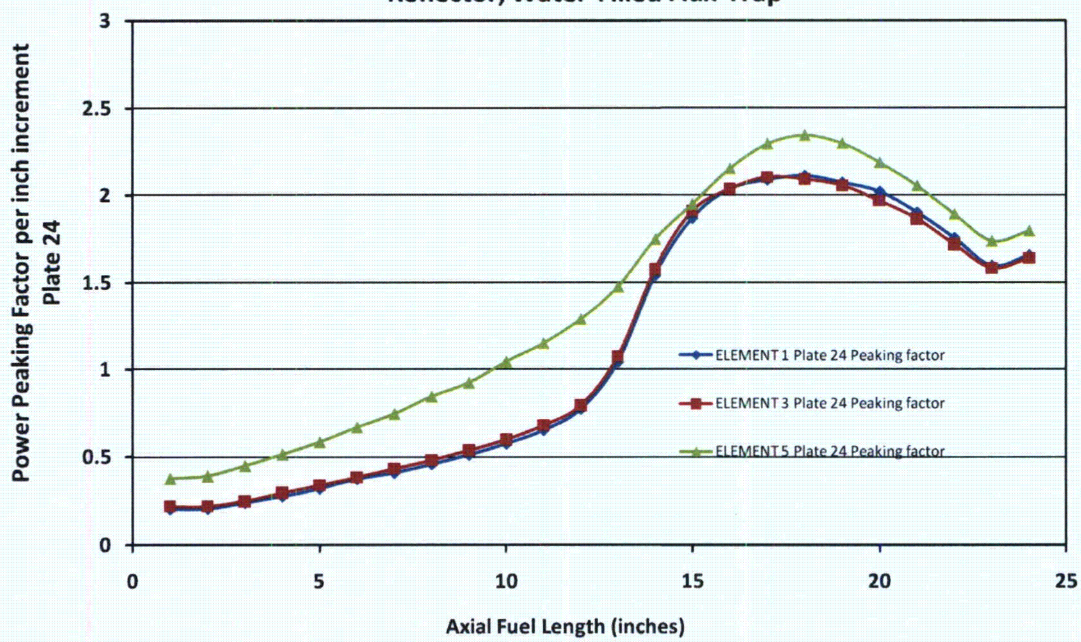

Figure 1.b

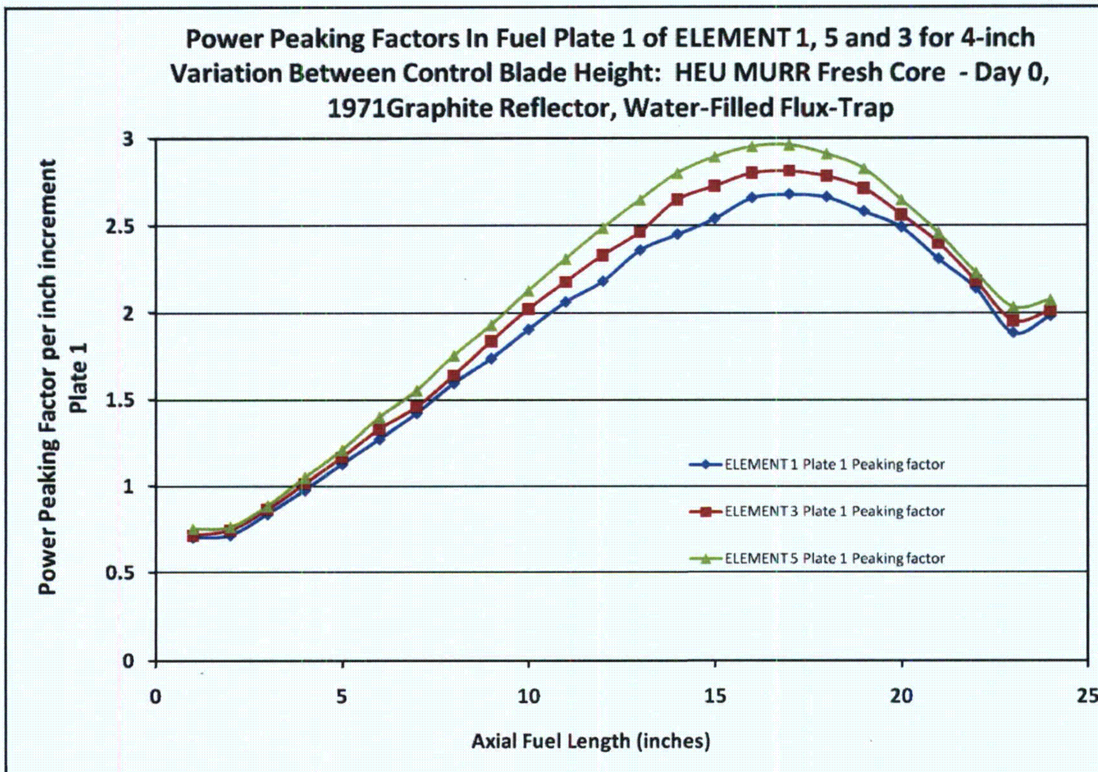

Figure 2.a

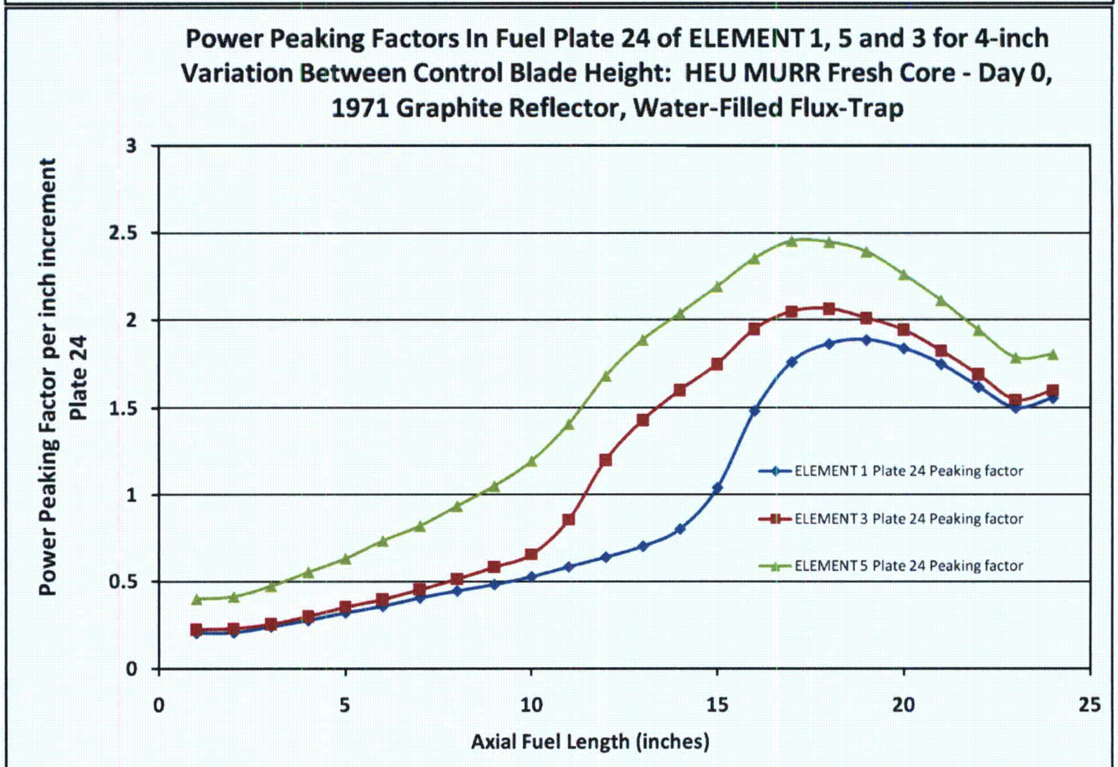

Figure 2.b

Power Peaking Factors In Fuel Plate 1 of ELEMENT 1, 3 and 5 for no Variation Between Control Blade Height: HEU MURR Mixed Core - Week 58 Day 0, 1971 Graphite Reflector, Water-Filled Flux-Trap

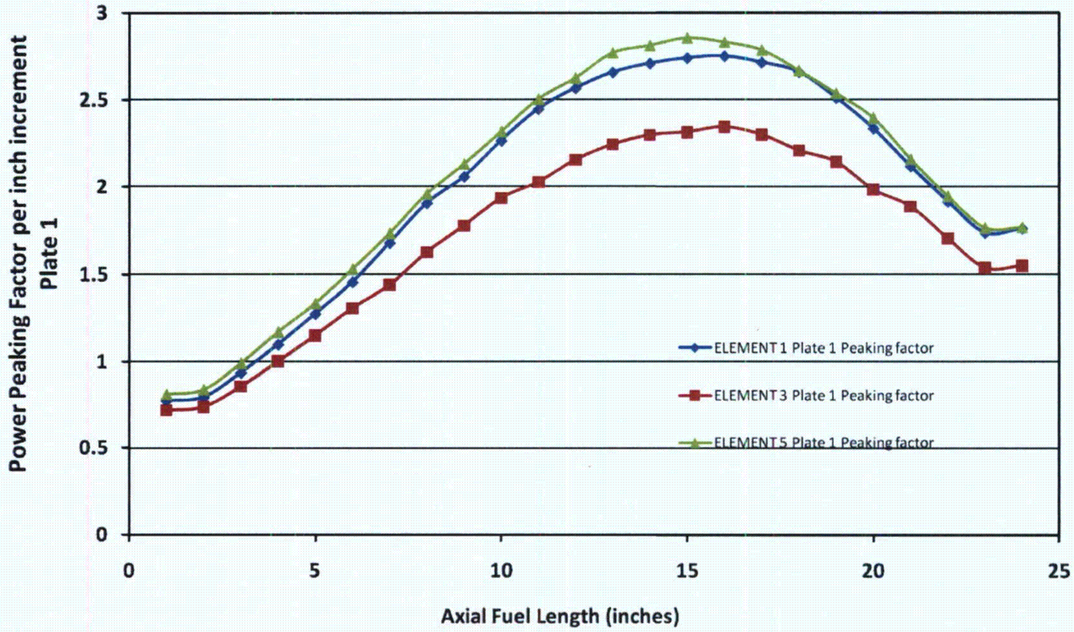

Figure 3.a

Power Peaking Factors In Fuel Plate 24 of ELEMENT 1, 3 and 5 for no Variation Between Control Blade Height: HEU MURR Mixed Core - Week 58 Day 0, 1971 Graphite Reflector, Water-Filled Flux-Trap

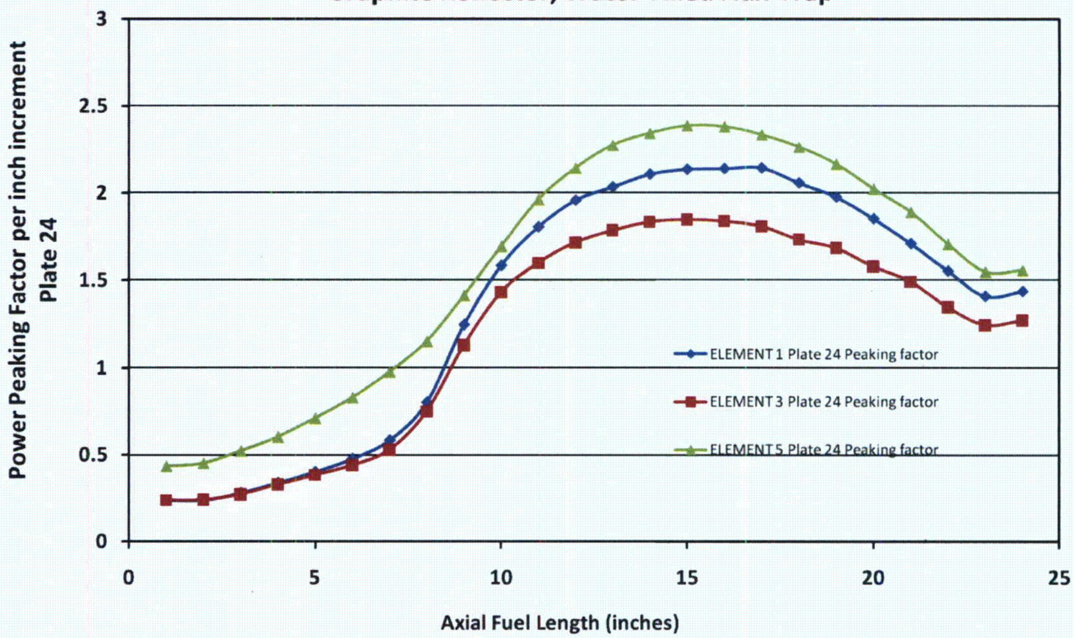

Figure 3.b

Power Peaking Factors In Fuel Plate 1 of ELEMENT 1, 5 and 3 for 4-inch Variation Between Control Blade Height: HEU MURR Mixed Core - Week 58 Day 0, 1971 Graphite Reflector, Water-Filled Flux-Trap

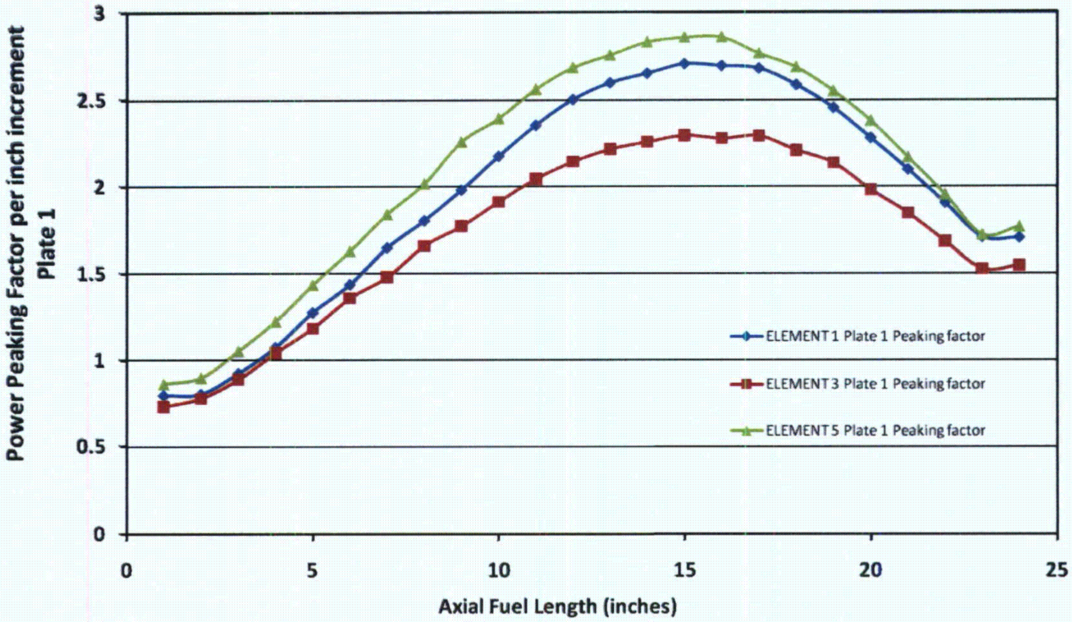

Figure 4.a

Power Peaking Factors In Fuel Plate 24 of ELEMENT 1, 5 and 3 for 4-inch Variation Between Control Blade Height: HEU MURR Mixed Core - Week 58 Day 0, 1971 Graphite Reflector, Water-Filled Flux-Trap

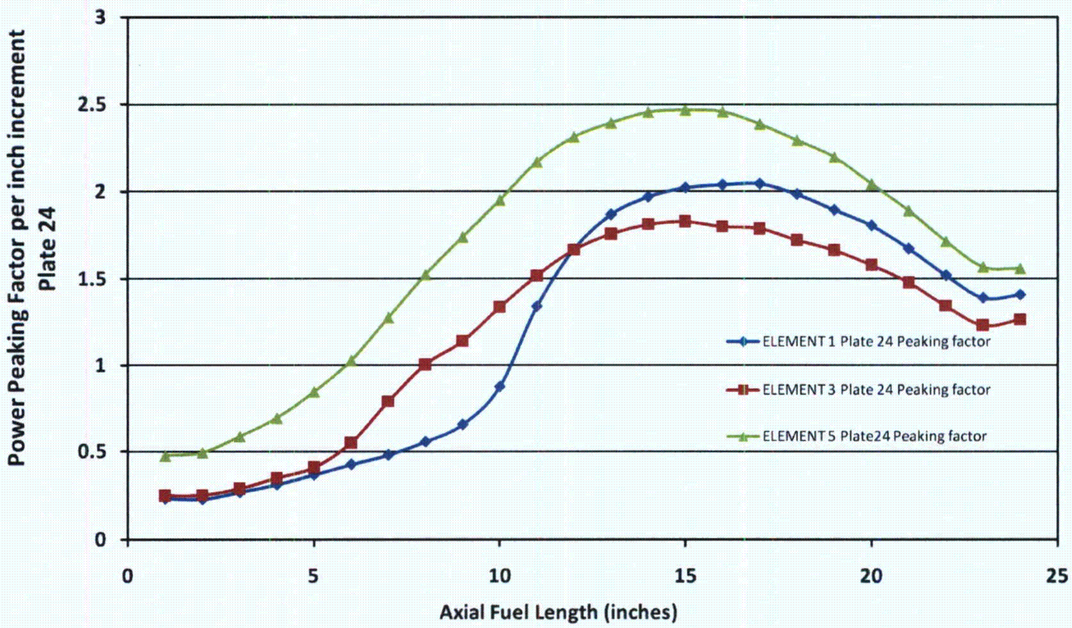

Figure 4.b

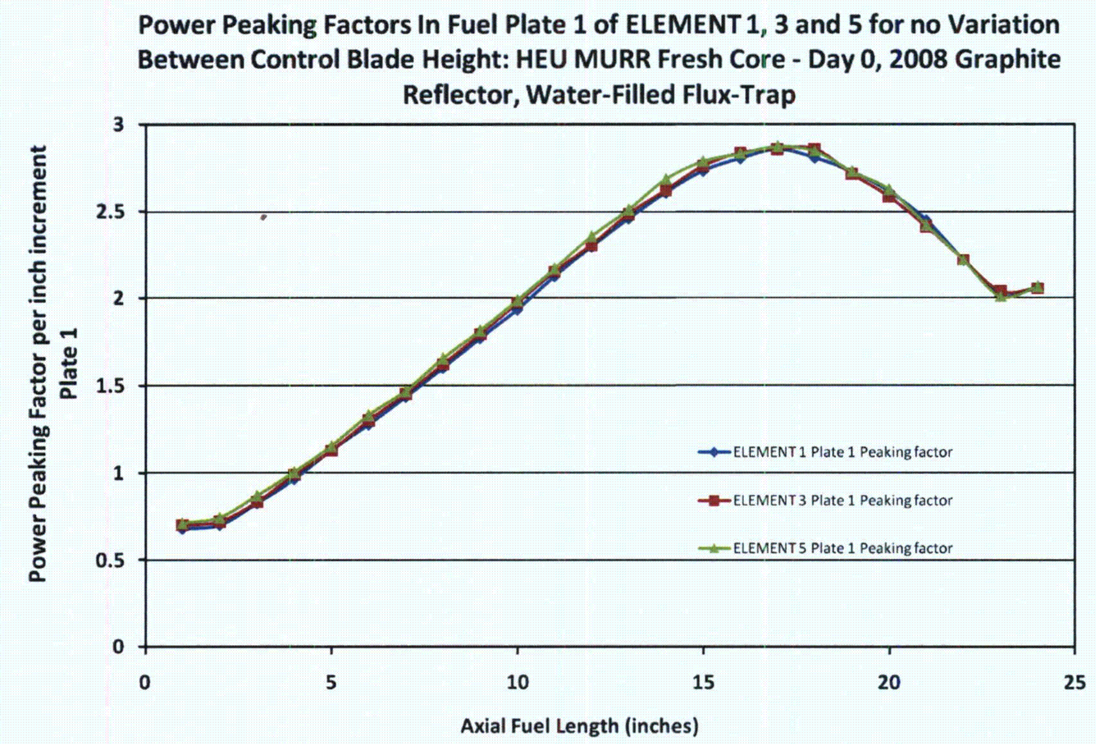

Figure 5.a

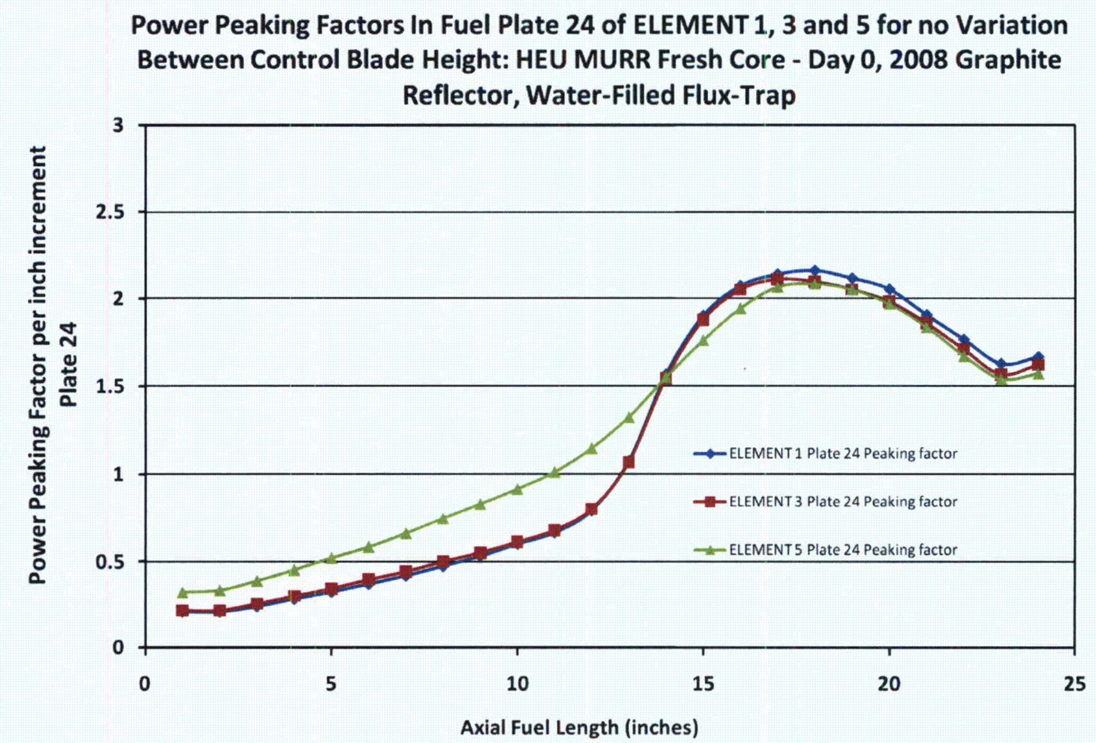

Figure 5.b

Power Peaking Factors In Fuel Plate 24 of ELEMENT 1, 5 and 3 for 4-inch Variation Between Control Blade Height: HEU MURR Fresh Core - Day 0, 2008 Graphite Reflector, Water-Filled Flux-Trap

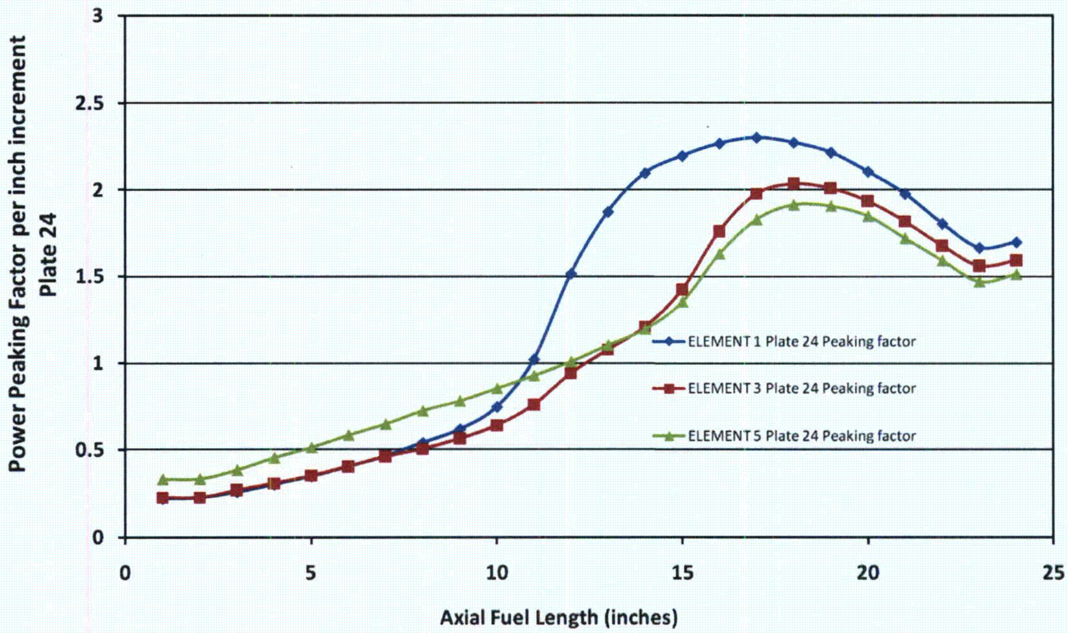

Figure 6.a

Power Peaking Factors In Fuel Plate 1 of ELEMENT 1, 5 and 3 for 4-inch Variation Between Control Blade Height: HEU MURR Fresh Core - Day 0, 2008 Graphite Reflector, Water-Filled Flux-Trap

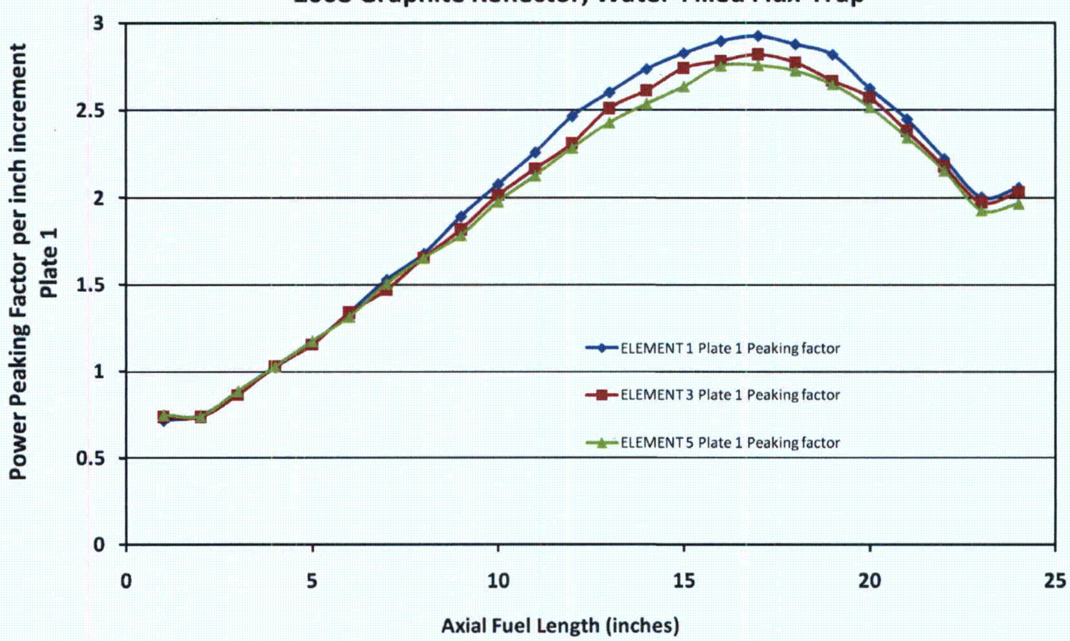

Figure 6.b

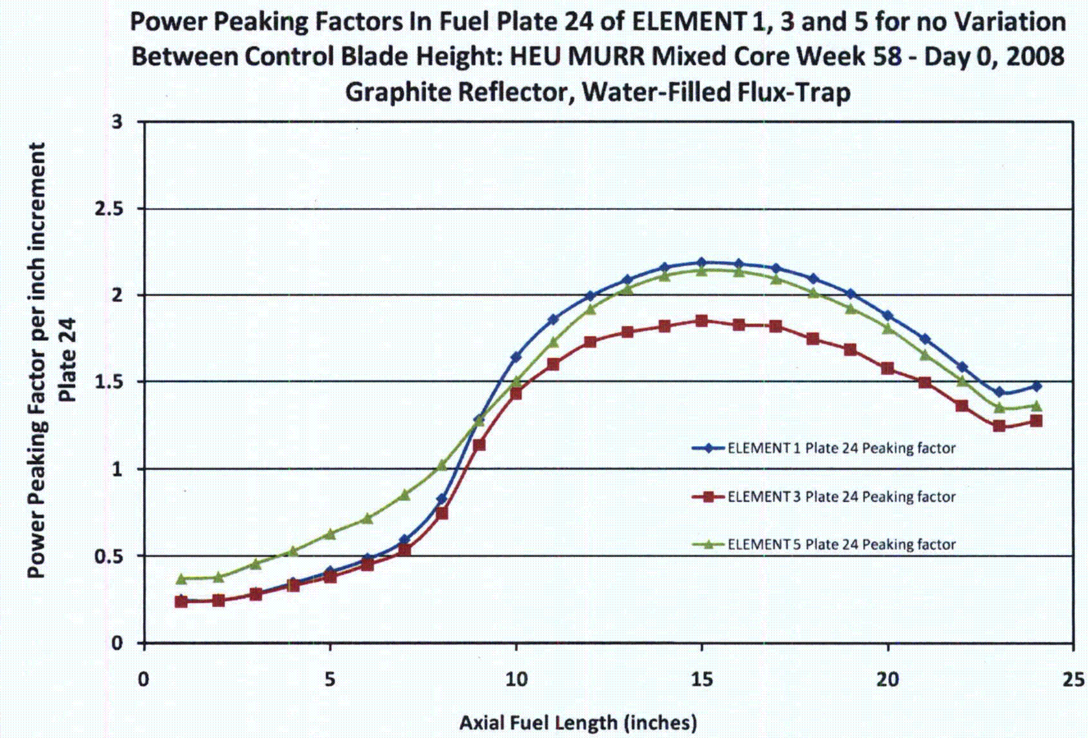

Figure 7.a

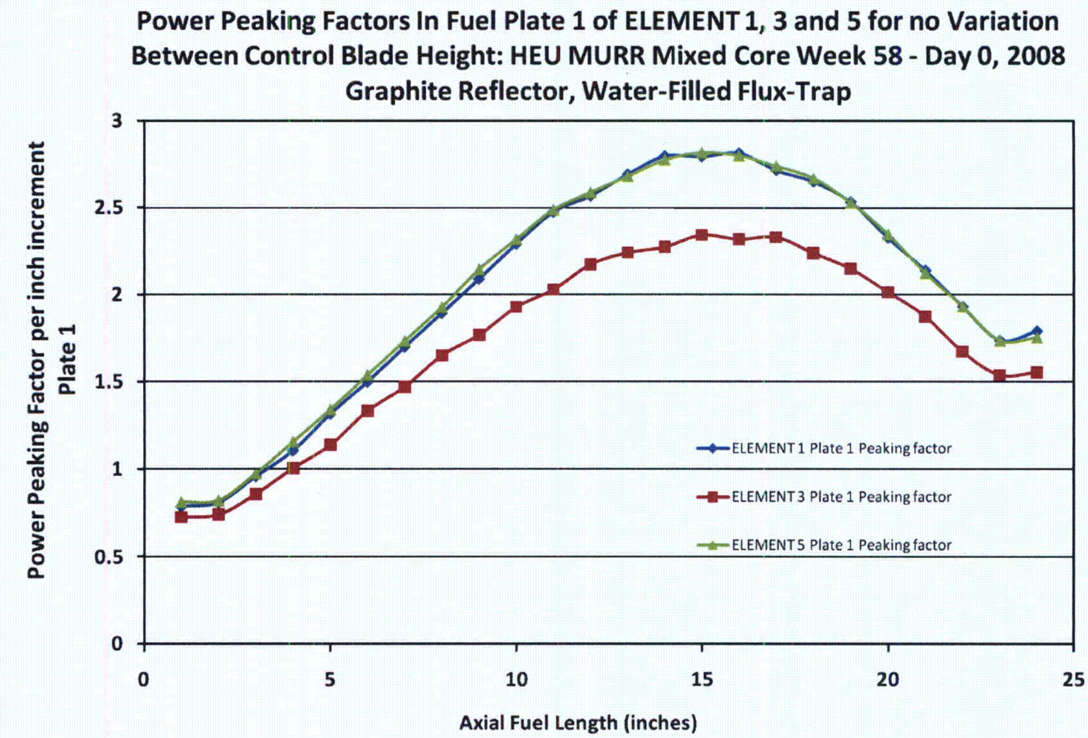

Figure 7.b

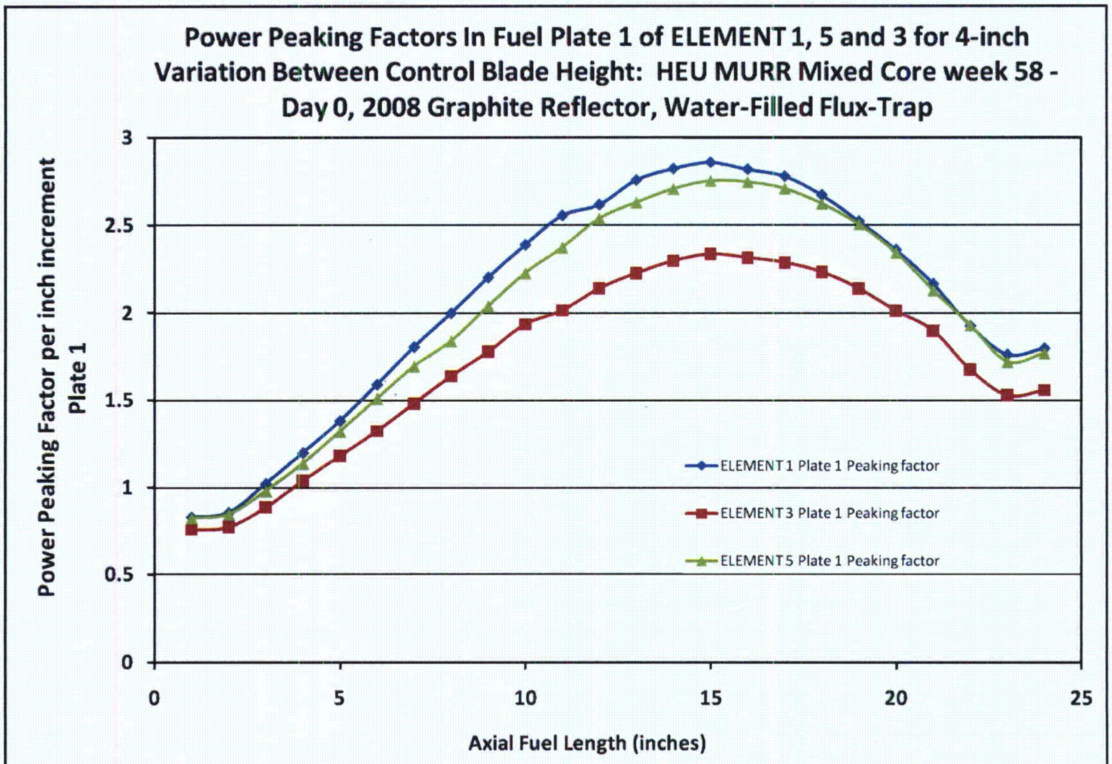

Figure 8.a

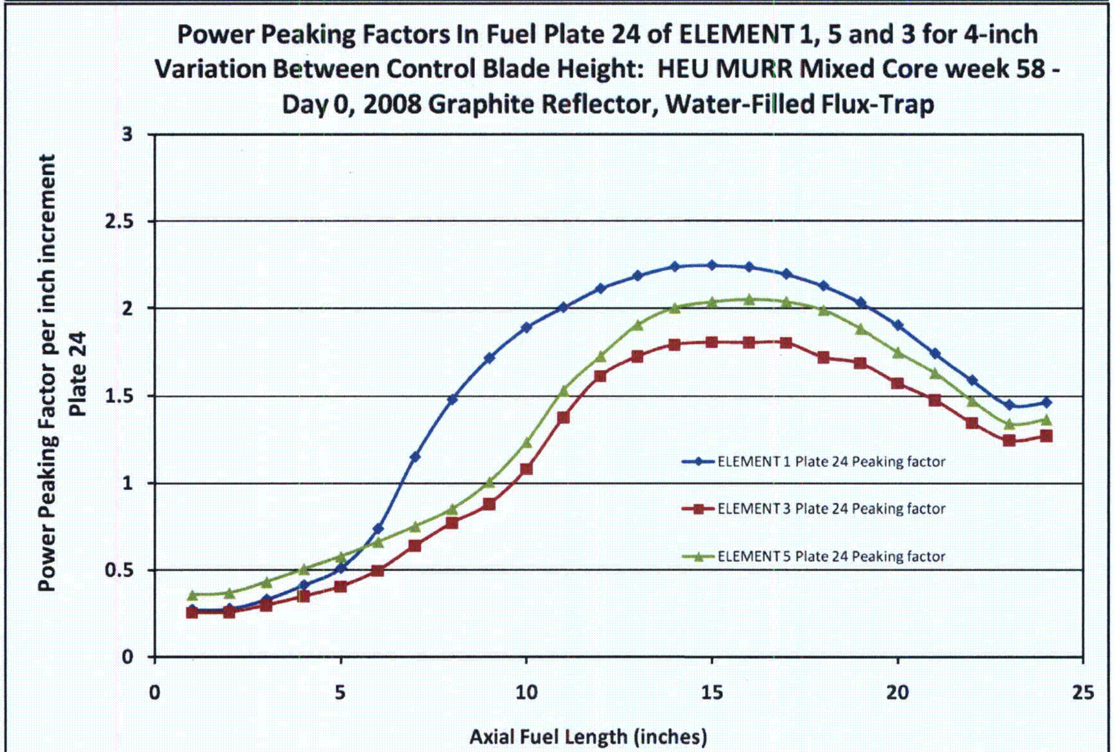

Figure 8.b

The eight control rod, reactor core and reflector configurations for fuel plate-1 nuclear peaking factors listed in Table 1 (all configurations are xenon free and the flux trap full of water):

1. Control Rods banked, Fresh Core, 1971 Reflector
2. Control Rods 4-inch diff., Fresh Core, 1971 Reflector
3. Control Rods banked, Week 58 Core, 1971 Reflector
4. Control Rods 4-inch diff., Week 58 Core, 1971 Reflector
5. Control Rods banked, Fresh Core, 2008 Reflector
6. Control Rods 4-inch diff., Fresh Core, 2008 Reflector
7. Control Rods banked, Week 58 Core, 2008 Reflector
8. Control Rods 4-inch diff., Week 58 Core, 2008 Reflector

Table 1 - Summary of the Nuclear Peaking Factors
at the Hotspots of Eight MURR Core Extreme Configurations
(i.e., "Fresh Core" and "Week 58 Core")

Core Configuration	**Peaking Factor**		**Peaking Factor**	
	Hot Spot[1]	**With Engr. PF**	**Enthalpy Rise**[1]	**With Engr. PF**
1	3.08	3.65	1.97	2.09
2	3.17	3.75	2.03	2.15
3	3.06	3.62	2.01	2.14
4	3.06	3.62	2.05	2.18
5	3.08	3.64	1.95	2.07
6	3.12	3.70	1.99	2.11
7	3.01	3.57	1.99	2.11
8	3.06	3.62	2.03	2.15
Safety Limits[1]	**3.475**	**4.116**	**2.301**	**2.442**

Note 1: The results presented in Table 1 reports the overall nuclear peaking factors for the hot spot heat flux and the hot channel enthalpy rise for the various MURR core configurations. The azimuthal peaking factor across the one-inch sections of fuel plate-1 is 1.07. Therefore the hot spot peaking factor is the highest peaking factor in the MCNP run for that model multiplied by 1.07 (circumferential peaking within plate-1). Then the hot spot peaking factor is multiplied by 1.03 and 1.15 (engineering hot channel factors on flux). The enthalpy rise peaking factor is the highest average plate-1 and -2, which heat coolant channel 2, peaking factors in the MCNP run for that model multiplied by 1.0921 (circumferential peaking within the channel 2). Then the enthalpy rise factor is multiplied by 1.03 and 1.03 (engineering hot channel factors on enthalpy rise).

Note 2: The SL peaking factors are taken from Table F.4, Appendix F, of the August 24, 2011 University of Missouri-Columbia submittal to the U.S. Nuclear Regulatory Commission requesting revised reactor SLs.

Considering the power peaking factors for these highest peaking factor configurations with a four-inch difference in the control rod heights do not exceed the SAR Table 4-14 peaking factors, the one inch limit between the highest and lowest control blades when the reactor is at a power level greater than 100 kW insures that an excess in flux tilting is not created.

REFERENCES:

[1]Letter Request to the U.S. Nuclear Regulatory Commission, Written communication as specified by 10 CFR 50.4 regarding an application to amend Amended Facility License R-103 by revising the University of Missouri-Columbia Research Reactor Technical Specification 2.1, "Reactor Core Safety Limit," pursuant to 10 CFR 50.90, University of Missouri-Columbia Research Reactor, August 2011 (ML 1123A088).

CHAPTER 13

13.2 Section 13.2.2.1, Accident initiating events and scenarios, Page 13-17.

d. Please discuss the effect of build up of an oxide layer on the fuel cladding.

During reactor operation, a thin film of aluminum oxide ($Al_2O_3.H_2O$ or boehmite) forms on the fuel plate surface. Based on fuel element oxide thickness measurements taken in July 1987 and using the Griess Correlation, the average hot spot oxide thickness was calculated to be 0.631 mils on a fully burned up MURR fuel element. The "worst case" hot spot oxide thickness was calculated to be 1.27 mils[1]. Previous studies of the oxide formation on aluminum 6061 have shown that the spallation of the oxide layer does not occur until a thickness in the range of 2 mils has developed[2].

To investigate the effect of this oxide layer on reactivity transients, a 2.0 mil thick oxide layer was added to the MURR PARET reactivity transient analyses model. Typically, the oxide layer gradually builds up on fuel plate surface during reactor operation and, by the time the layer reaches a thickness of 2.0 mils, that fuel element is near its end-of-life and hence will have a lower power load. However, to be conservative in modeling, the maximum power load was applied to the fuel as if it is a fresh fuel element. Additionally, conservative values were selected for the oxide thermal properties in the PARET model.

With these conservative assumptions incorporated, the MURR limiting step reactivity insertion transient of 0.006 $\Delta k/k$ was reanalyzed. The results obtained are discussed below.

The initial peak power burst reached during the transient went from 34.4 MW to 36.8 MW. This change is to be expected since the insulating oxide layer will delay the heat transfer from the fuel to the coolant and hence will delay the inherent feedback mechanisms that help limit the transient. This effect was also evident since the peak power is reached in 0.057 seconds from the start of the transient compared to 0.047 seconds for the no oxide layer case.

The peak centerline temperature of the fuel increased from 418.5 °F (214.7 °C) for the no-oxide layer case to 581.7 °F (305.4 °C) with the 2.0 mil oxide layer. Again, this is consistent since the heat transfer from the fuel to the coolant is inhibited by the oxide layer thus raising the peak centerline temperature attained during the transient. The maximum temperature reached is still well below the melting point or the blister temperature of the fuel.

REFERENCES:

[1]University of Missouri Research Reactor Letter to U.S. Nuclear Regulatory Commission, "Response to NRC Request for Additional Information," In answer Number 1, September 11, 1987.

[2]J.C. Griess, H.X. Savage, and J.L. English, Effect of Heat Flux on the Corrosion of Aluminum by Water, Part IV, ORNL-3541, Union Carbide Corp. Nuclear Division, Oak Ridge National Laboratory, 1964.

13.3 Section 13.2.2.1.2, Continuous Control Blade Withdrawal.

Justify why the regulating blade is not part of this evaluation. Explain if reactor operation in Mode II or III changes the results of the evaluation.

The regulating blade was not initially included as part of the "Continuous Control Blade Withdrawal" evaluations since manual withdrawal of the four shim blades, either individually or in gang, while simultaneously withdrawing the regulating blade is prohibited by administrative procedure AP-RO-110, "Conduct of Operations," and operating procedures OP-RO-211, "Reactor Startup - Hot," and OP-RO-212, "Reactor Startup - Recovery From Temporary Power Reduction." With separate switches for manipulating the shim blades and the regulating blade, inadvertent simultaneous manipulation of the two control mechanisms is highly unlikely.

In order to evaluate the consequences of withdrawing the regulating blade at the same time as the four shim blades during a reactor startup, the "Continuous Control Blade Withdrawal" accident, as discussed in SAR Section 13.2.2.1.2, was reanalyzed using the computer code PARET.

For this analysis, the maximum positive reactivity insertion rate allowed by the Technical Specification (TS) due to the withdrawal of the regulating blade was superimposed on top of the maximum allowed reactivity insertion resulting from the simultaneous withdrawal of the four shim blades. It should be noted that since the total reactivity worth of the regulating blade is limited to 0.006 $\Delta k/k$ (TS 3.1.c), and at a maximum reactivity addition rate of 0.00025 $\Delta k/k$/sec (also TS 3.1.c), the reactivity addition due to withdrawal of the regulating blade occurs only during the first 24 seconds of the "Continuous Control Blade Withdrawal" accident scenario. Beyond that, the only reactivity addition will be due to shim blade motion. It should also be noted that the current total reactivity worth of the regulating blade is less than half of the TS limit. In addition, by applying the total reactivity worth of the regulating blade using the maximum insertion rate allowed by the TS, very conservative results can be expected. In practice, the regulating blade displays typical control blade worth behavior of a cosine shape, i.e. maximum worth is only in the middle section of the worth curve and much less effect towards either end of blade travel.

The results obtained are shown in Figure 1. As expected, reactor power level reached during the transient is higher when compared to the case where only the four shim blades are continuously withdrawn. Even though the reactivity worth of the regulating blade is considerably less than the reactivity worth of the shim blades, the rate of reactivity addition is significant because of its higher rate of travel (40 inches/min for the regulating blade compared to one inch/min for the shim blades). Figure 1 shows reactor power continuously rising and reaching a value of almost 11.0 kW within 140 seconds of the accident initiation. It should be noted that for this analysis no credit is taken for the reactor protective system initiating a rod run-in or reactor scram once the appropriate short period set points are reached. For example, the short period rod run-in set point is 11 seconds and this value is reached approximately 120 seconds after initiation of the accident (compared to 140 seconds for the shim blade only withdrawal case). At this point, reactor power level is only 17.2 watts. If the reactor protective system fails to initiate a rod run-in, the scram circuit will initiate a reactor scram once reactor period decreases to 9 seconds - which happens about 123 seconds after accident initiation (compared to 143 seconds for the shim blades only case). Reactor power level at this point is only 22.8 watts.

Thus, even though the effect of a "Continuous Control Blade Withdrawal" accident appears to be more serious when the regulating blade is withdrawn simultaneously along with all four shim blades, the higher rate of reactivity insertion will cause the reactor protective system to initiate a rod run-in or scram at an earlier time which will terminate the transient before any fuel damage can occur.

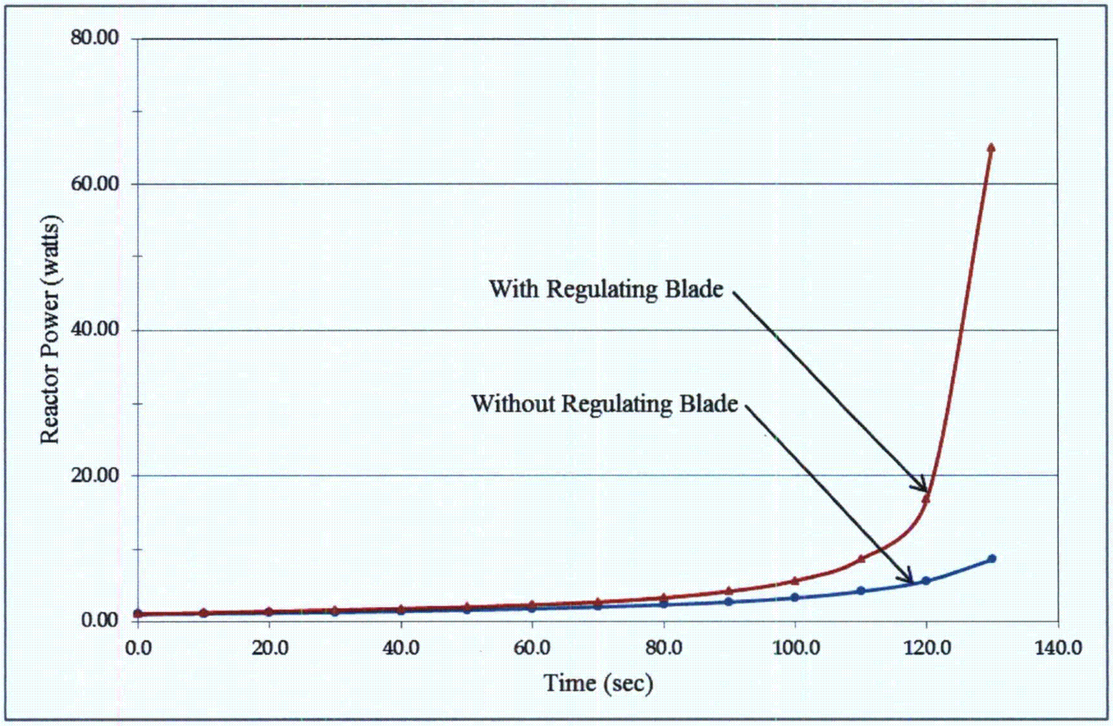

Figure 1 - Reactor Power Versus Time For Continuous Control Blade Withdrawal Accident (With and Without the Regulating Blade)

CHAPTER 14, TECHNICAL SPECIFICATIONS

14.1 *Section 14.2, Format and Content and Introduction section of TSs. The SAR states that Section 5 of the TSs, Design Features, only contains specifications. The regulations in 10 CFR 50.36(a)(1) requires bases for design features TSs. Please amend your proposed design feature TSs to include bases.*

The Technical Specifications will be revised as follows to include the bases for the design features:

5.1 **Site Description**

<u>Bases</u>

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

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 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

Bases

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

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 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**

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.

CHAPTER 16

16.1 Section 16.1.1, Fuel and Fuel Cladding, TS 3.8, Reactor Fuel, and TS 4.5, Reactor Fuel.

a. The bases for TS 3.8.b states that the TS assures that fuel elements found to be defective are no longer used for reactor operation. The TS contains a limit on dimensional changes of coolant channel between fuel plates of 10 mils. What is the basis for the 10 mils and what is the impact of this amount of fuel channel dimensional change on thermal-hydraulic and accident analysis?

The allowance for a 10-mil reduction in a coolant channel gap is to account for swelling due to fuel meat expansion caused by fuel burnup and clad thickening from oxidation buildup. Because of the fuel meat void fraction of aluminide fuel, the reduction in channel gap from fuel burnup is delayed. This results in heat fluxes of the reduced channel being substantially less than those that existed when irradiation of the fuel element began. With a reduction in heat flux caused by fuel burnup, the 62-mil (0.062 inches) end-of-life minimum allowed coolant channel gap was calculated to have a Safety Limit (SL) of 19.690 MW for reactor power [7.19 MW greater than the reactor power Limiting Safety System Setting (LSSS) of 12.5 MW]. This is 32% greater than the 14.894 MW reactor power SL for 80-mil nominal and 72-mil worst case wide coolant channels. Thus, the fuel elements with the highest burnups and coolant channels reduced to the minimum width allowed are operating well within the MURR SLs.

The design gap of all coolant channels in the core that are bounded by two fuel plates is 80±8 mils, therefore the narrowest channel gap could potentially be 72 mils (0.072 inches) after fabrication. The assumed additional 10-mil reduction in channel gap due to oxidation and/or fuel swelling causes the minimum channel gap to be decreased to 62 mils. Making the channel thinner reduces both its coolant velocity and its flow area. Both of these effects reduce its flow rate. Section 5 of Reference 1 considered an analogous situation where the limiting channel was assumed to have a gap of 72 mils instead of its nominal width of 80 mils (0.080 inches). The same analysis applies here using 62 mils in place of 72 mils. The equations that were used are:

$$\frac{V_H}{V_N} = \left(\frac{D_H}{D_N}\right)^{\frac{1+\alpha}{2-\alpha}} \tag{1}$$

$$\frac{W_H}{W_N} = \frac{V_H \times A_H}{V_N \times A_N} = \frac{V_H}{V_N} \times \frac{A_H}{A_N} \tag{2}$$

where:

V = velocity;
D = hydraulic diameter;
A = flow area;
α = the exponent in the friction factor versus Reynolds number relationship; and
Subscripts H and N represent the hot and nominal channels, respectively.

The friction, f, versus Reynolds number, Re, relationship is $f \propto 1/Re^{\alpha}$. Based on the Blasius formula for turbulent flow friction factor, $f = 0.316/Re^{0.25}$, α is 0.25.

In Reference 1, the hot channel gap is 72 mils, the nominal channel gap is 80 mils, D_H is 0.13876 inches, and D_N is 0.15828 inches. This value of D_H was obtained as 4 times the flow area divided by the wetted perimeter. The channel flow area is the product of the channel arc length along the average of the inner and outer radii and the channel gap. The channel arc length is one-eighth the circumference of a circle reduced by both the thickness of two side plates (0.150 inches each) and the clearance between adjacent elements (0.04 inches). The nominal radii of the limiting channel analyzed in Reference 1, coolant channel 2 of fuel element 1, are 2.820 and 2.900 inches, corresponding to a channel gap of 0.080 inches. For the hot channel, the inner radius was increased by 0.004 inches and the outer radius was decreased by 0.004 inches to account for the channel gap tolerance of 0.008 inches, which reduced the overall channel gap to 0.072 inches. Thus, for the Reference 1 hot channel, the flow area in square inches is:

$$\{2\pi [(2.820 + 2.900)/2] / 8 - (2 \times 0.150 + 0.04)\} \times 0.072 = 0.13725$$

and the wetted perimeter in inches is:

$$2\pi (2.820 + 2.900) / 8 - 2 \times (2 \times 0.150 + 0.04) + 2 \times 0.072 = 3.9565$$

The corresponding hydraulic diameter, D_e, in inches is $4 \times 0.13725 / 3.9565 = 0.13876$. For the 0.062-inch channel gap, the calculations for the flow area, wetted perimeter, and hydraulic diameter are the same as above except that 0.072 is replaced by 0.062. Thus, the new flow area, wetted perimeter, and hydraulic diameter are 0.11819 in^2, 3.9365 inches, and 0.1201 inches, respectively.

For the current analysis, the hot channel gap is 62 mils instead of 72 mils and the nominal channel gap is unchanged. The hot channel flow area factor, which is the area ratio A_H/A_N, is 62/80 = 0.7750. For the 62-mil channel gap, equation (1) yields:

$$V_H/V_N = (0.1201/0.15828)^{(1+0.25)/(2-0.25)} = 0.8210.$$

V_H/V_N is the engineering hot channel factor for velocity identified in the Reference 1 analysis. Equation (2) yields:

$$W_H/W_N = 0.8210 \times 0.7750 = 0.6363.$$

The Reference 1 methodology also requires the value of the channel heated diameter (D_i), defined as the channel heated perimeter divided by π. D_i is the same for the reduced channel gap as for the nominal channel gap since thinning the channel does not change its heated perimeter.

In summary, changing the limiting channel gaps from 72 to 62 mils reduces A_H/A_N, which is the hot channel flow area factor from 0.90 to 0.775, V_H/V_N, which is the engineering hot channel factor for velocity, from 0.9108 to 0.8210, and the hydraulic diameter (D_e) from 0.13876 inches to 0.1201 inches. Substituting, the three new values, 0.775, 0.8210, and 0.1201, which are the hot channel flow area factor (cell B65), the engineering hot channel factor on velocity (cell B68), and D_e (cell D86), respectively into the Table 8 model of Reference 1 accounts for the heat removal reduction caused by the 62-mil narrow channel. A substantial amount of fuel burnup must occur before there is sufficient fuel swelling and clad surface oxidation to cause a 10-mil reduction in channel gap. This burnup considerably reduces the element power. This is addressed in the next section.

The thermal limiting criteria for the MURR during steady-state operation are based on avoiding both flow instability and critical heat flux (CHF). The August 24, 2011 University of Missouri-Columbia submittal to the U.S. Nuclear Regulatory Commission requesting revised reactor SLs provides a detailed description of the model for defining the safety envelope for steady-state operation of the MURR (Ref. 2). This model, as described in Reference 1, was used to address this question about reduced channel thicknesses. Reference 1 includes a very detailed sample problem solution to promote clarity of the analytical model. Table 8 of Reference 1 provides the computer spreadsheet that was used for the sample problem to predict the allowed reactor power SL of 14.894 MW with all three (3) non-power LSSS variables set at their corresponding limits, i.e. primary coolant pressurizer pressure at 75 psia, total core flow rate at 3200 gpm, and reactor inlet water temperature at 155 °F. The coolant channel chosen for analysis, channel 2 of element 1, is the innermost one bounded by two fuel plates and is the most limiting channel in the reactor. This sample problem with relatively minor changes to its input was used to calculate the limiting power level for the reduced channel thickness.

Table 1 - Limiting Channel Plate Axial Peaking Factors and Coolant Temperature Rise Distribution

Level	Plate Axial Peaking Factor	Fraction of Bulk Coolant Temperature Rise
1	0.461	0.0188
2	0.466	0.0377
3	0.537	0.0597
4	0.632	0.0857
5	0.705	0.1147
6	0.803	0.1478
7	0.872	0.1839
8	0.967	0.2241
9	1.016	0.2667
10	1.105	0.3132
11	1.152	0.3616
12	**1.214**	**0.4130**
13	**1.257**	**0.4658**
14	**1.278**	**0.5195**
15	**1.299**	**0.5743**
16	**1.297**	**0.6287**
17	**1.310***	**0.6833**
18	**1.250**	**0.7355**
19	**1.247**	**0.7875**
20	**1.162**	**0.8359**
21	**1.122**	**0.8821**
22	**1.002**	**0.9235**
23	**0.920**	**0.9614**
24	**0.925**	**1.0000**

*Maximum value

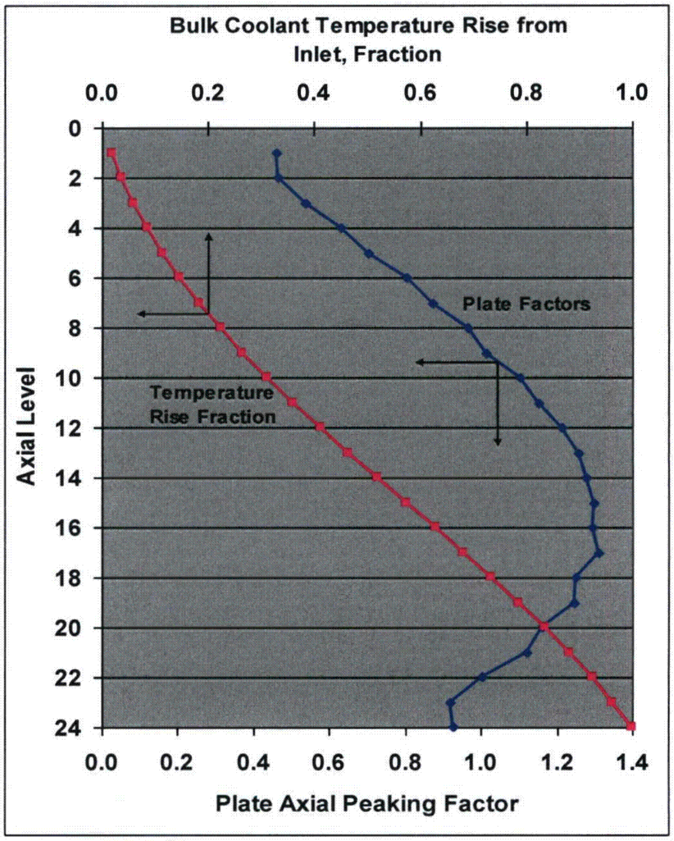

Figure 1- Limiting Channel Plate Axial Peaking Factors and Coolant Temperature Rise Distribution

Table 1 and Figure 1 show the axial heat flux peaking factors for fuel plate-1 and coolant channel 2 in fuel element 8, which is the most limiting high burnup element with 142 MWd of burnup. The bulk coolant temperature rise is based on a weighted average of the heat fluxes of the two fuel plates that bound the channel. The corresponding core configuration is for week 58, xenon free (day 0), with an average control rod height of 17 inches withdrawn. In Table 2, the peaking factors for fuel plate-1 and coolant channel 2 of the high burnup fuel element 8 are compared to plate-1 and channel 2 peaking factors of the 0 MWd fuel element 1 in a week 58 core, which is the basis for the MURR SLs. The power peaking factors were obtained from MCNP modeling of the MURR fuel cycle. The MCNP modeling to determine the power peaking factors only included the nominal coolant channel dimensions. If the hot channel 2 had been modeled as narrower, it would have further reduced the power peaking due to the localized reduction in moderation.

The variation of heat flux along the azimuthal direction of each fuel plate is also taken into account. In the neutronics analysis the heated arc length along the azimuthal direction of each fuel plate is divided into a series of nine equal-radian arc length vertical strips. The average heat flux of each strip is calculated. The ratio of the highest of these nine averages to the plate-average heat flux is called the "azimuthal peaking factor." The plate axial peaking factors provided for each level in Table 1 are based on level-averaged heat fluxes rather than the level azimuthal maximum. The azimuthal peaking factor for plate-1 in fuel element 8 is 1.04, which is a lower value than for fuel element 1 because burnout of the hot stripe is greater than the plate average burnout.

As shown in Table 2 below, the power factor for enthalpy rise for element 1 is 2.4416, whereas for element 8 it is 1.8184, a 25% reduction in enthalpy rise, while the channel flow reduction factor was reduced by 22%. The power factor for heat flux in element 1 at interval 18, which is the limiting point, is 3.863, whereas for element 8 at the limiting interval 19 the power factor is 2.504. This corresponds to a 35% reduction in the heat flux at the limiting point. When all of the new factors due to both the reduction in channel size and fuel burnup are in place in the Reference 1 model and the high burnup axial power shape is represented in lines 78 and 79, the 62-mil channel is predicted to reach its maximum SL power level of 19.69 MW. This is 32% larger than the 14.894 MW reactor power SL of Reference 0, demonstrating that for the high burnup fuel element the additional 10-mil reduction in the coolant channel gap creates no safety problem.

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]Letter Request to the U.S. Nuclear Regulatory Commission, Written communication as specified by 10 CFR 50.4 regarding an application to amend Amended Facility License R-103 by revising the University of Missouri-Columbia Research Reactor Technical Specification 2.1, "Reactor Core Safety Limit," pursuant to 10 CFR 50.90, University of Missouri-Columbia Research Reactor, August 2011 (ML 1123A088).

Table 2 - Peaking Factors for Week 58 Fuel Elements 1 and 8

	Safety Limit Factors	
On Enthalpy Rise In Channel 2	Element 1	Element 8
Power History of Fuel Element	0 MWd	142 MWd
Coolant Channel 2 Gap	72 mils	62 mils
Power-related Factors		
Nuclear Peaking Factors		
Fuel Plate-1 and -2 Average	1.9843	1.5310
Azimuthal in the Channel	1.0921	1.0542
Additional Allowable Factor	1.0620	1.0620
Engineering Hot Channel Factors		
Fuel Content Variation	1.0300	1.0300
Fuel Thickness/Width Variation	1.0300	1.0300
Overall Product	2.4416	1.8184
Flow-related Factors		
Core/Loop Flow Fraction	1.0000	1.0000
Assembly Minimum/Average Flow Fraction	1.0000	1.0000
Channel Minimum/Average Flow Fraction		
Inlet Variation	1.0000	1.0000
Width Variation	1.0000	1.0000
Thickness Variation	0.72/0.80	0.62/0.80
Within Channel Minimum Channel Thickness affect on:		
Velocity Factor	0.9108	0.8210
Overall Factor on Flow Reduction	0.8197	0.6363

On Heat Flux From Fuel Plate-1	For mesh interval between the following inches down the fuel plate	
Power-related Factors		
Mesh Interval Number	18(17-18")	19(18-19")
Nuclear Peaking Factors		
Fuel Plate (Hot Plate Average)	2.215	1.5345
Azimuthal Within Plate	1.070	1.040
Axial Peak	1.2958	1.247
Additional Allowable Factor	1.062	1.062
Engineering Hot Channel Factors		
Fuel Content Variation	1.030	1.030
Fuel Thickness/Width Variation	1.150	1.150
Overall Product	3.863	2.504
Percentage Enthalpy Rise at Hot Spot	74.8%	78.8%
Energy Fraction Generated in Fuel Plate	93%	
Safety Limit at LSSS	**14.894 MW**	**19.69 MW**

APPENDIX A, TECHNICAL SPECIFICATIONS

A.48 *TS 5.1, Site Description. Clearly describe the area under the authority of the reactor license.*

Technical Specification 5.1 has been revised as follows to help clarify the area under the authority of the reactor license.

5.1 **Site Description**

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.

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[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.

Note: Based on the above revised Technical Specification (TS) 5.1, MURR is requesting that TS Definition 1.6, Exclusion Area, be deleted from the submitted TSs. This term is a holdover from the currently approved TSs and is not used in any operating or emergency procedure. Additionally, MURR requests that Section 2.1.2, Operational Boundaries, of Chapter 2 of the SAR, be revised as follows to be consistent with the revised TS 5.1.

2.1.2 Operational Boundaries

There are three areas of concern regarding the normal operation, safety, and emergency actions associated with the reactor facility: the restricted area within the operations boundary; the unrestricted area within the site boundary; and the Emergency Planning Zone (EPZ).

The operations boundary consists of the fencing at the border of the 7.5 acre lot surrounding the MURR facility itself. The area within this boundary is the restricted, or licensed, area where the Reactor Facility Director has direct authority and control over all activities, normal and emergency. The adjacent reactor cooling tower building is also included within the restricted area. A tunnel connects the cooling tower building to the laboratory building basement. There are pre-established evacuation routes and procedures known to personnel frequenting this area. The operations boundary is within the site boundary.

The site boundary consists of the following: Stadium Boulevard; Providence Road (Route K)[1]; the MU Recreational Trail; and the MKT Nature and Fitness Trail. The unrestricted area within these boundaries is owned and controlled by the University of Missouri 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.

In addition, an Emergency Planning Zone (EPZ) has been established for which emergency plans have been developed to ensure that prompt and effective actions can be taken to protect the public in the event of an accident. MURR's EPZ is the area bounded by a 150-meter radius from the reactor facility ventilation exhaust stack and lies completely within the site boundary (Figure 2.3).

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[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.