UNIVERSITY *of* MISSOURI

May 23, 2012

RESEARCH REACTOR CENTER

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 the response to the "University of Missouri-Columbia – Request for Additional Information, Re: License Amendment, Safety Limits (TAC No. ME7018)," dated April 12, 2012

By letter dated August 24, 2011, the University of Missouri-Columbia Research Reactor (MURR) submitted a request to the U.S. Nuclear Regulatory Commission (NRC) to amend the Technical Specifications (TSs), which are appended to Facility License R-103, because of an error that was discovered in the MURR Safety Limit (SL) Analysis while answering a relicensing Request for Additional Information (RAI) question.

On April 12, 2012, the NRC requested additional information and clarification regarding the proposed Amendment in the form of five (5) questions. Those questions, and MURR's responses to those questions, are attached. If there are any questions regarding the attached responses 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,

Ralph A. Butler, P.E.
Director

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

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MARGEE P. STOUT
My Commission Expires
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1. *The axial peaking factor was changed from 1.432 in the previous analysis to a new value of 1.2958 in the new analysis. Please provide a justification for this change and a basis for the revised axial peaking factor.*

Due to the limitations of using 2D diffusion code models, as discussed in Section F.4, "Method of Analysis," (of the revised Appendix F of Addendum 4 to the MURR Hazards Summary Report), the previously derived power peaking factors that were used to calculate the 1973 MURR Safety Limits (SL) were extremely conservative and that the new, more accurate peaking factors were obtained using a modern 3D MCNP MURR model. The 1973 axial peaking factor of 1.432 was derived from a 2D R-Z model using a critical rod height from an all fresh fuel element core (no xenon or fuel burnup). The BORAL® control blades were modeled approximately 12 inches withdrawn such that the bottom edge of the blade was only 10 inches above the bottom edge of the fuel meat. Because the control blades are external to the outer pressure vessel, and hence the core, and are positioned such that power is pushed down and inward in the core, the power density in fuel plate-1 is increased which results in higher axial and radial peaking factors in the plate. These higher radial and axial peaking factors were combined with a non-uniform burnup peaking factor that was obtained from a 2D R-θ model of a core consisting of fuel elements with no burnup history mixed with fuel elements having high burnup. These two conditions cannot occur simultaneously because the mixed burnup core will have a higher critical rod height, which will then result in lower axial and radial peaking factors.

The new analysis, as discussed in Sections F.4, "Method of Analysis," and F.5, "Determination of Worst-Case Safety Limit Peaking Factors," more accurately calculates the 3D peaking factors for the various MURR core configurations. The most limiting reactor core configuration was selected from MURR Technical Data Report TDR-0125, "Feasibility Analyses for HEU to LEU Fuel Conversion of the University of Missouri Research Reactor (MURR)" (Attachment 10 to the original Amendment application and Reference 10 to Appendix F) and the corresponding peaking factors were used in the SL analysis, including the new axial peaking factor of 1.2958. As noted from Table F.4, "Summary of MURR Hot Channel Factors," the 1.2958 axial peaking factor is not the highest axial peaking factor but corresponds to mesh interval 18 down the hot fuel plate which is the limiting mess interval for all of the 180 SL data points except for six. Five are limited by the saturation temperature at the coolant channel exit while the other one is based on mesh interval 19, which has a lower axial peaking factor than interval 18 but is one of the lowest core flow rate points with the highest inlet water temperature.

Below are Sections F.4 and F.5 from the revised Appendix F of Addendum 4 to the MURR Hazards Summary Report (Attachment 11 of the original Amendment application). Words and numbers that are in red provide the justification and basis for the revised axial peaking factor.

F.4 Method of Analysis

In 1974, when MURR, working with the NUS Corporation, developed the previous SLs, the peaking factors used were extremely conservative because peaking factors had to be determined utilizing three (3) different 2D diffusion code models, which was the only code method available in the early 1970's. These peaking factors were provided in the original Table F.2 of Appendix F of

Addendum 4 to the MURR HSR (A detailed description is provided in Section 3.3 of Addendum 3 to the MURR HSR):

On Heat Flux

Power-Related Factors

Nuclear Peaking Factors:

Radial........................	2.220	*[from all fresh fuel R-Z model]*
Local (Circumferential)...	1.040	*[from all fresh fuel R-Θ model]*
Non-uniform Burnup......	1.112	*[from mixed burnup fuel R-Θ model]*
Axial..........................	1.432	*[from all fresh fuel R-Z model]*

The product of the above four (4) nuclear peaking factors is 3.676. With the engineering hot channel factors on flux included, the overall product is 4.35. This overall peaking factor is extremely conservative because peaking factors from two different core arrangements were combined. For a core consisting of eight (8) fresh fuel elements (0 MWD), causing a lower control rod height, higher radial and axial peaking factors are then combined with the smaller local (circumferential) peaking factor caused by no burnup. For a non-uniform burnup core with a higher control rod height, resulting in lower radial and axial peaking factors, the larger azimuthal (non-uniform burnup) factor was obtained. These four (4) factors, which cannot all occur in the same core, were then combined to provide this very conservative, overall peaking factor.

Current 3D nucleonics codes show the axial peaking factors in a mixed burnup core are lower than in an all fresh fuel core because of less excess reactivity and the corresponding higher critical control rod height position. Also, the 3D model provides an accurate average plate peaking factor, instead of the 2D approach of trying to approximate this by using the all fresh fuel radial factor (R-Z) times the non-uniform burnup azimuthal factor (R-Θ). This is a result of combining the high excess reactivity and lower critical rod height of an all fresh core with the MURR design, consisting of all the control blades external to the outer pressure vessel, which tends to push the power down and inward in the core increasing the axial and radial peaking factors for the hot fuel plate (fuel plate-1).

Since 2006, MURR has been actively collaborating with the Department of Energy's 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 Argonne National Laboratory (ANL)/MURR team has benchmarked the MURR HEU fuel and reactor core design performance, which is documented in MURR Technical Data Report TDR-0125, "Feasibility Analyses for HEU to LEU Fuel Conversion of the University of Missouri Research Reactor (MURR)" (Ref. 10). The computer modeling completed in performing the feasibility analyses provides more accurate 3D peaking factors, which were used in determining the revised SLs. In addition to the feasibility analyses worst-case HEU peaking factors, an additional peaking factor of 1.062 is included to define the maximum allowable nuclear peaking factor. The maximum allowable nuclear peaking factors were combined with the engineering peaking factors, which are provided below in Table F.4, to establish the maximum allowable overall peaking factor for enthalpy rise and heat flux used to generate the revised SLs that meet the existing criteria.

F.5 Determination of Worst-Case Safety Limit Peaking Factors

MURR Technical Data Report TDR-0125 documents the neutronic analysis steps taken to develop the code system, modeling and benchmarking for the MURR HEU fuel performance. To compare the performance of the proposed LEU design to the current HEU operation, models were developed for the current reactor reflector configuration with typical experimental loadings. To properly model the current HEU core fuel utilization, shutdown margin and experimental performance, it was necessary to develop a computational shuffling that would accurately model the actual complex fuel cycle used at MURR. This is also described in detail in TDR-0125.

Also in order to perform a comparison between HEU and LEU fuel, it was necessary to define reference cores that could be compared in order to establish feasibility of all major parameters: fuel cycle performance, shutdown margin, thermal-hydraulic steady-state safety margins, and experimental performance. It was decided that these reference cores should be close to limiting in order to provide additional confidence that the safety margin calculations treat the potentially different limiting power shapes from the HEU and LEU fuel elements. The actual limiting peaking factor for the MURR 775-gram ^{235}U aluminide fuel elements is modeled by multiplying the worst-case peaking factors by an additional 1.062 factor to determine the SLs.

Core configurations with potentially high power peaking were identified from the fuel cycle simulation in order to analyze for thermal-hydraulic behavior. The fuel cycle simulations provided a collection of more than 325 HEU Core States (4 state points per week x 82 weeks or different core combinations). Each of these State points from the REBUS-DIF3D model was examined and sorted to rank the cases with the highest peak heat flux in the core. As expected, it was found that the highest heat flux always occurs in cases where a fresh element is loaded next to an element that is near its discharge burnup limit.

Based on the selection criteria, the core conditions defined as week 58 (out of 82) of the HEU fuel cycle simulation was chosen for more detailed evaluation. Depleted fuel material compositions were extracted from the REBUS-DIF3D results for each plate and axial depletion zone in the model; this consisted of 2,304 fuel compositions (8 elements x 24 plates x 12 axial zones) for each of the HEU cores. This data was utilized as material composition data for a detailed MCNP model for calculating power distributions and estimated critical positions.

The week 58 HEU core has fuel elements with the following power history: two elements each with 0, 65, 81 and 142 MWD. Of the four different states for the week 58 core, the beginning of the week xenon-free core with no samples or sample holder loaded in the center flux trap region produces the highest power peaking factors. This is due to the highest peaking factors are always on the inner fuel plate, fuel plate-1, and initial critical rod height combined with the 100% water filled flux trap produces even higher peaking factors in fuel plate-1.

The detailed week 58 core MCNP run for the 3B Core State was used to extract the 3D power peaking factors. From these fuel plate heat flux peaking factors, the peaking factors for the hottest coolant channel (channel 2) were determined. The enthalpy rise in channel 2 is calculated based on the average heat flux for fuel plate-1 and -2. This is conservative because both fuel plate-1 and -2 transfers a little less than half their total heat to coolant channel 2 since the coolant channels on the opposite side of each plate operates at a lower temperature than does channel 2.

The peaking factors for channel 2 are provided in Table F.4. Fuel plate-1 and -2 average and the azimuthal in the channel are calculated from the MCNP results. To establish the maximum allowable overall peaking factor, an additional 1.062 factor is included in both the enthalpy rise and the heat flux. The Engineering Hot Channel Factors are based on the MURR aluminide fuel fabrication specifications and are as used in NUS-TM-EC-9 (Ref. 6). The overall flow-related factor of 0.8197 is the product of the 0.9 (72-mils/80-mils) narrow channel thickness factor times the 0.9108 reduction of coolant velocity in the narrow channel.

2. *The new Safety Limit Analysis for MURR (application Attachment 11), Section F.4 assumes that the center test hole contains only moderator. It appears that this analysis does not address the effect of irradiation samples and experiment test tube canisters in the center test hole on the core heat flux. Please provide a summary of the effect of irradiation samples and canisters in the center test hole on the heat flux profile used for the hot channel factors of Table F.4.*

Section F.4, "Method of Analysis," (of the revised Appendix F of Addendum 4 to the MURR Hazards Summary Report) utilizes peaking factors that were obtained from modeling the center test hole containing only moderator. Since 2006, MURR has been actively collaborating with the Department of Energy's Reduced Enrichment for Research and Test Reactors (DOE-RERTR) Program on the conversion of MURR from highly-enriched uranium (HEU) fuel to low-enriched uranium (LEU). MURR Technical Data Report TDR-0125, "Feasibility Analyses for HEU to LEU Fuel Conversion of the University of Missouri Research Reactor (MURR)" (Attachment 10 to the original Amendment application and Reference 10 to Appendix F), documents the work performed by the Argonne National Laboratory (ANL)/MURR team in benchmarking the MURR HEU fuel and reactor core design characteristics. Computer modeling that was performed as part of the feasibility analyses provided more accurate 3D peaking factors, which were then used to determine the new SLs. The modeling fuel cycle simulations presented a collection of 82 different core cases from which to compare peaking factors. Peaking factors were then obtained for the following four different states for each of the 82 core cases: 1) only moderator in the center test hole at startup with no xenon, 2) only moderator in the center test hole after operating two days with equilibrium xenon, 3) sample holder loaded with samples in the center test hole at startup with no xenon, and 4) sample holder loaded with samples in the center test hole after operating two days with equilibrium xenon.

Table 3-14 of TDR-0125 compares the hot stripe heat fluxes from an all fresh fuel element core to the worst-case mixed burnup cores for both HEU and LEU fuel. For HEU, the hot stripe is always on fuel plate-1, with the highest value from the week 58 mixed core case at startup with no samples or sample holder inserted in the center test hole; i.e. only moderator. The next highest hot stripe is for the all fresh (no burnup) fuel element core at startup with no samples or sample holder inserted in the center test hole. However, the only time MURR operates with no samples or sample holder inserted in the center test hole is for startups with a xenon free core up to a low power level of about 50 kW for short periods of time to obtain reactivity reference points for determining the total reactivity worth of the sample holder and the contained samples. The all moderator state in the center test hole creates the highest moderation condition in the inner pressure vessel, which produces the highest thermal flux in fuel plate-1 (which is adjacent to the inner pressure vessel). Irradiation samples and the sample holder in the center test hole both displaces the moderator and absorbs some of the thermal flux, which combine to reduce the thermal flux at fuel plate-1 and the corresponding power density.

Table 3-14 from TDR-0125 – Summary of Key Hot Stripe Heat Fluxes Evaluated

Core State that may bound power peaking					Hot Stripe Heat Flux (W/cm^2) Fresh Element in Position X1				Hot Stripe Heat Flux (W/cm^2) Fresh Element in Position X5			
Fuel	Case	Burnup State	Day	Flux Trap	Plate 1	Plate 3	Plate 23	Plate 24	Plate 1	Plate 3	Plate 23	Plate 24
HEU 10MW	1A	Fresh	0	Samples	126.7	91.4	67.3	76.8	128.8	94.0	69.4	80.4
	2A	Fresh	2	Samples	121.6	89.3	74.4	87.3	123.4	89.4	74.8	86.6
	3A	Week 58	0	Samples	131.7	96.6	82.6	96.6	132.3	97.6	79.3	91.8
	4A	Week 58	2	Samples	126.3	92.6	90.4	107.4	125.6	92.6	82.8	97.8
	1B	Fresh	0	Empty	133.2	94.5	66.7	77.2	133.8	96.2	70.0	80.2
	2B	Fresh	2	Empty	127.0	91.3	74.5	87.9	129.3	92.1	74.3	87.1
	3B	**Week 58**	**0**	**Empty**	**138.6**	**99.3**	**83.0**	**97.6**	**138.9**	**99.7**	**78.9**	**92.2**
	4B	**Week 58**	**2**	**Empty**	**132.9**	**94.8**	**90.8**	**109.6**	**132.1**	**93.2**	**82.8**	**97.9**
LEU 12MW	5A	Fresh	0	Samples	116.3	134.4	84.9	100.0	119.4	136.6	90.1	107.0
	6A	Fresh	2	Samples	112.2	129.5	94.6	116.0	113.4	130.4	95.8	117.2
	7A	Week 79	0	Samples	119.0	137.6	103.3	126.6	118.4	137.7	101.3	122.3
	8A	Week 79	2	Samples	114.1	130.4	113.8	142.6	113.3	130.1	105.5	131.1
	5B	Fresh	0	Empty	124.0	139.0	85.0	100.8	125.3	140.9	90.8	108.0
	6B	Fresh	2	Empty	119.1	132.4	95.8	118.0	119.6	133.1	96.4	118.2
	7B	**Week 79**	**0**	**Empty**	**124.9**	**141.0**	**104.7**	**127.6**	**125.1**	**140.8**	**102.0**	**123.2**
	8B	**Week 79**	**2**	**Empty**	**120.3**	**133.9**	**114.3**	**145.4**	**119.4**	**132.8**	**105.7**	**131.3**

Samples indicates a typical loading of samples in all three flux trap tubes
Empty indicates neither samples nor tubes in the flux trap (i.e., "empty island" configuration)

Note that HEU operates at 10 MW, while 12 MW is proposed for LEU operation. Thus a 20% increase in LEU heat flux would be expected if the element was not altered (in design and underlying physics).

Therefore, the effect of irradiation samples and the sample holder inserted in the center test hole is a reduction in the hot channel power peaking factors used in the new MURR SL analysis.

3. *The enthalpy rise and heat flux factors contain a new "additional allowable peaking factor" of 1.062. It appears that this new factor is not described or justified in the application submittal. Please provide additional information justifying the use and the magnitude of the additional peaking factor.*

For a primary coolant pressurizer pressure of 75 psia, a total core flow rate of 3200 gpm and a reactor inlet water temperature of 140 °F, the original Table F.1, from MURR Hazards Summary Report, Addendum 4, Appendix F, October 1973 (Attachment 1 of the original Amendment application), provided a reactor power level safety limit (SL) of 15.967 MW. For a reactor inlet water temperature of 160 °F, with the same pressurizer pressure and core flow rate stated previously, the SL was 14.534 MW. This results in a SL of 14.894 MW with pressurizer pressure, total core flow rate and reactor inlet water temperature at the Limiting Safety System Settings (LSSS) of 75 psia, 3200 gpm and 155 °F, respectively. It was determined that the worst-case 3D peaking factors obtained in performing the work documented in MURR Technical Data Report TDR-0125, "Feasibility Analyses for HEU to LEU Fuel Conversion of the University of Missouri Research Reactor (MURR)" (Attachment 10 to the original Amendment application and Reference 10 to Appendix F), could be increased by a factor of 1.062 for both the local power density and the coolant channel enthalpy rise and still maintain a reactor power SL of 14.894 with pressurizer pressure, total core flow rate and reactor inlet water temperature at the LSSSs. Therefore, this additional allowable peaking factor is included in the analysis to establish the maximum allowable overall peaking factor for enthalpy rise and heat flux used to generate the revised SLs.

4. *There appears to be several changes in your proposed TSs that are not discussed in your amendment application. For example, the word "water" is added to TS 2.1 a. Please describe and justify all requested changes to your TSs.*

   The following is a list of proposed changes to Section 2.1 of the Technical Specifications based on the Amendment application.

   1. Page 1 of 6. In the "Applicability" statement, a comma was added after the word "flow" to make the sentence grammatically correct.

   2. Page 1 of 6. In the "Specification" statement, under 2.1.a, the word "water" was added between the words "inlet" and "temperature" in two places to remove any ambiguity because reactor inlet water temperature is the specific parameter that is being discussed. It is also consistent with the wording in the bases of Specification 2.1.a and with the wording in Specification 2.2, "Limiting Safety System Settings."

   3. Pages 2, 3 and 4 of 6. These pages contain the new safety limit curves, which were rotated 90 degrees because we feel that they are much more functional in this orientation.

   4. Page 5 of 6. In the "Specification" statement, under 2.1.a, a comma was added between the words "psia" and "the" to make the sentence grammatically correct.

   5. Page 5 of 6. In the "Specification" statement, under 2.1.b, the word "is" was replaced by the word "are" to make the sentence grammatically correct.

   6. Page 5 of 6. In the "Bases" statement, under a., the sentence "An extension of this analysis is presented in Section 6.0 of Addendum 5 to the HSR." was deleted. As stated on page 3 of the original Amendment application, dated August 24, 2011, the revised Appendix F, *Safety Limit Analysis for the MURR*, combined and replaced the current versions of the following three documents:

      a. Section 3.3 of Addendum 3 to the MURR Hazards Summary Report (Attachment 4);
      b. Appendix F of Addendum 4 to the MURR Hazards Summary Report (Attachment 1); and
      c. Section 6.0 of Addendum 5 to the MURR Hazards Summary Report (Attachment 3).

The basis for Specification 2.1 is now all contained in the revised Appendix F of Addendum 4 to the MURR Hazards Summary Report.

7. Page 5 of 6. In the "Bases" statement, under a., the sentence "A family of curves is presented which relate the reactor inlet water temperature and core flow rate to the reactor power level corresponding to a DNB ratio (DNBR) of 1.2 based on the burnout heat flux data experimentally varified for ATR type fuel elements." was changed to the following two sentences "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."

   The change more accurately describes the bases for the safety limit curves, which is presented in detail in Section F.2, "Safety Limit Criteria," of the revised Appendix F of Addendum 4 to the MURR Hazards Summary Report.

8. Page 6 of 6. In the "Bases" statement, under a., the comma was deleted after "i.e." to make the sentence grammatically correct.

9. Page 6 of 6. In the "Bases" statement, under b., the sentence "Below 400 gpm core flow the criterion for the safety limit is that fuel plate temperature must be below that temperature which would result in fuel cladding failure." was changed to the following sentence "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."

   All MURR fuel assemblies undergo a cladding bond integrity test during fabrication. As part of the bond integrity test, a blister test is performed by heating each fuel plate to a temperature of 900 °F (Ref. "Specification TRTR-4 For University of Missouri-Columbia Fuel Elements Assembled For University of Missouri Research Reactor, Revision 4," EG&G Idaho, Inc., June 8, 1994). Fuel plate temperatures above 900 °F may cause blistering which could lead to fuel cladding failure.

   Section 3.4.1(8) of the above reference states:

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

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

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

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

5. *Attachment 12 of your amendment application discusses a Case 1 severe power transient and concludes that ample safety margin exists for safety system reaction time. Please describe this transient in detail which supports your conclusion.*

The Limiting Safety System Settings (LSSSs) for Modes I and II, i.e. 10 MW and 5 MW operation respectively, are as follows. For reactor power level, the LSSS is 125% of full power for both modes, thus the highest powers obtainable before a reactor scram would initiate is 12.5 MW (1.25 x 10 MW) in Mode I and 6.25 MW (1.25 x 5 MW) in Mode II. For both modes, the LSSS on pressure is a minimum of 75 psia in the primary coolant pressurizer and the LSSS on reactor inlet water temperature is a maximum of 155 °F. The LSSS on primary coolant flow for Mode I is a minimum of 1625 gpm in either of the parallel coolant loops. The same LSSS of 1625 gpm applies for a single operating loop in Mode II. Since 50 gpm of the primary coolant flow is diverted to the cleanup (demineralizer) system before the core, the actual total core flow rates at the LSSS would be 3200 gpm and 1575 gpm for Modes I and II, respectively. Note: Total core flow rate for Mode II operation would actually be more conservatively limited to 1600 gpm, instead of 1575 gpm, because of the "Differential Pressure Across the Core" reactor safety system instrument channel set point (Technical Specification 3.3.a).

The Amendment application to revise the *Safety Limit Analysis for the MURR*, presented in Appendix F of Addendum 4 to the MURR Hazards Summary Report, presents parametric curves for the conditions which would lead to reaching the conservatively defined critical heat flux (CHF) which corresponds to 0.5 times the Bernath CHF Correlation. From the analysis, Figure H.1 depicts the CHF conditions for the LSSS on a pressurizer pressure of 75 psia. From the curves one can predict the safety margin for several anticipated transients.

Case 1 postulates a severe power transient with primary coolant flow and pressure reduced to their LSSSs in Mode I operation. Figure H.1 predicts that the reactor inlet water temperature LSSS of 155 °F could not be reached until power has risen to the safety limit (SL) of 14.896 MW, which is 2.396 MW greater than the reactor power scram LSSS set point. With a primary coolant flow rate of 3200 gpm, the loop flow transient time is about 24 seconds, which means an increase in power will take approximately 24 seconds to cause an associated increase in reactor inlet water temperature. Therefore, to exceed the SL, reactor power would have to increase by 2.396 MW or an additional 19% above the reactor power level LSSS of 12.5 MW and have the high reactor power level scram take longer than 24 seconds to actuate. Thus, an ample safety margin exists for safety system reaction time required to prevent reaching the CHF threshold.