UNIVERSITY OF MISSOURI – COLUMBIA

RESEARCH REACTOR

LICENSE No. R-103

DOCKET No. 50-186

RESPONSES TO NRC REQUESTS FOR ADDITIONAL INFORMATION RELATING TO LICENSE RENEWAL

REDACTED VERSION*

SECURITY-RELATED INFORMATION REMOVED

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

UNIVERSITY *of* MISSOURI

RESEARCH REACTOR CENTER

August 31, 2010

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

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

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

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

On June 1, 2010, the NRC requested additional information and clarification regarding the renewal request in the form of one hundred and sixty-seven (167) questions. By letter dated July 16, 2010, MURR responded to forty-seven (47) of those questions. On July 14, 2010, via electronic mail (email), MURR requested additional time to respond to the remaining one hundred and twenty (120) questions. By letter dated August 4, 2010, the NRC granted the request. MURR's responses to fifty-three (53) of those questions are attached. Two (2) of the responses are requests to have those questions included in the set of "complex" questions (120-day response). Responses to the remaining sixty-seven (67) questions are due by September 30, 2010.

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

Sincerely,

Leslie P. Foyto
Reactor Manager

ENDORSEMENT:
Reviewed and Approved,

John Ernst for Ralph Butler

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

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MARGEE P. STOUT
My Commission Expires
March 24, 2012
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CHAPTER 4

4.1 *Section 4.2.1.1, Reactor Fuel System, Page 4-5. The conclusive statement in the text "no aluminde fuel element failures have occurred during the past thirteen years" does not appear to be supported by Reference 4.46, dated 1990. Discuss and provide a reference for MURR fuel performance more recent than 1990.*

The statement made on page 4-5 of Section 4.2.1.1, Reactor Fuel System, that "*no aluminde fuel element failures have occurred during the past thirteen years*" is made in reference to the Advanced Test Reactor's (ATR) aluminide fuel experience and not that of MURR's. ATR has had no aluminide fuel failures since 1993 (private communication with A. J. Vinnola, Research Reactor Fuels, Idaho National Laboratory). As for MURR's experience, except for one incident where slightly elevated levels of iodine-131 were detected in the primary coolant system during the use of a single fuel element in 1997 (Ref. 4.47), MURR has not experienced any aluminide fuel failures. This element was retired early after it had been used for 126 MWDs of its planned 150-MWD usage.

4.5 Section 4.2.2.1, Control Blade Description.

a. *Page 4-15. Provide a further description of the BORAL® plate and its design to withstand the hydraulic and radiation environment of the reactor.*

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 (0.9525 mm) of aluminum-alloy 1100 for a nominal blade thickness of 0.175 ± 0.007 inches (4.445 ± 0.178 mm). The minimum weight of boron-10 per unit cross sectional area is 0.0418 gm/cm^2. All four sides of the blades have a 0.25-inch (6.35-mm) thick aluminum edge clad channel welded to the BORAL® plate so that the blade edge thickness is 0.25 inches (6.35 mm). The active length of the neutron absorbing material is 34 inches (86.35 cm), and the overall blade length is approximately 40 inches (101.6 cm). The upper 6 inches (15.25 cm) of the shim blade is a 0.25-inch (6.35-mm) thick aluminum mounting plate that is curved to the shape of the blade. The shim blade has a nominal inner radius of 6.430 inches (16.33 cm). Each shim blade occupies approximately 72° of a circular arc around the pressure vessel.

The radius of the shim control blade provides a certain amount of inherent stiffness that aids the hydraulic stability of the blade in its operating gap between the outer reactor pressure vessel and the beryllium reflector. During normal operation, only a portion of the shim blade operates in this gap, resulting in a minimized downward force due to flow. The flow of pool coolant past the portion of the blade that is in the gap is of a low velocity, resulting in very little downward force and a minimal likelihood of flow instabilities. Based on the current measured flow induced differential pressure across the reflector elements and the control blade gaps, the pool coolant flow velocity in the control blade gaps is approximately 3.5 feet per second (1.1 m/sec).

The BORAL® and aluminum 1100 that comprise the shim blade have an excellent service history, both at MURR and other facilities, for the radiation conditions of the reactor. From 1966 to 1983, one shim blade and its associated offset mechanism were removed and inspected each calendar quarter. The removal and inspection has been performed on a semi-annual basis from 1983 to date. The resulting experience has shown that shim control blades suffer no operational limitations or flow erosion, and minimal dimensional changes due to the hydraulic and radiation environment.

4.5 Section 4.2.2.1, Control Blade Description.

b. *Provide a discussion of the control blade drive and control independence that prevents a malfunction in one blade from affecting the insertion or withdrawal, or both, of any other blade.*

The physical separation of the control blades and the drive mechanisms prevents mechanical failure of one blade from potentionally affecting the insertion or withdrawal of any other control blade. Each blade is located in its own curved annular gap between the beryllium reflector and the outer reactor pressure vessel. The control blade gaps are separated by five (5) long aluminum spacer bars that run the full vertical height of the beryllium reflector. The horizontal cross-sectional view of the spacers is similar to that of a thick "T," with the thickness of the top of the "T" establishing the proper blade gap spacing (approximately 0.55 inches) between the beryllium reflector and the outer pressure vessel. The vertical parts of the spacer bars are set into vertical slots in the inner radius of the beryllium reflector. These vertical slots effectively lock the spacer bars into their proper position and clearly establish properly sized and shaped independent control blade gaps.

Control blade drive controls, including independence with regard to operation, are discussed in detail in Section 7.5, Rod Control System. These include descriptions of general control independence in the manual and automatic control modes, as well as circuit features such as Rod Withdrawal Prohibit, Rod Run-In and Automatic Shim Control. These descriptions can be summarized by stating that there are no automatic circuits which could withdraw one or more control blade drives, and there are two automatic control circuits – Rod-Run-In and Automatic Shim Control – which could insert the control blade drives. All other blade drive operations are performed manually by the operator.

Independence with regard to malfunction is primarily protected against by the eight independent drive relays that provide insertion or withdrawal for each of the four drives. Each drive has its own insertion relay and its own withdrawal relay, requiring power to be supplied from the rod control circuits mentioned above to operate. A failure of any one of these relays can only impact operation of that one drive in one direction. Thus a malfunction in one blade drive would require one or more additional malfunctions in one of the control circuits to affect the insertion or withdrawal of any other control blade. The special case where both insertion and withdrawal relays are energized simultaneously would still be limited to a single drive, but would have additional protection via Rod Control Fuse 2F4.

4.5 Section 4.2.2.1, Control Blade Description.

c. *Provide a discussion of the decision criteria used for replacement of control blades.*

One shim control blade and its associated offset mechanism is removed semi-annually and inspected with the control blade and offset mechanism submerged in the reactor pool to a depth that provides the required radiation shielding. Inspection of the control blade is required by surveillance Technical Specification 4.3.b. A rebuilt offset mechanism (bearings replaced and offset mechanism aligned) is installed in its place using either a new or used control blade that has been inspected and approved. The inspection and approval of the control blade is performed visually and using geometric templates to ensure no swelling, distortion, blistering, rubbing or other abnormality is present prior to placing a control blade on service. This inspection is performed in accordance with preventative maintenance procedure RX-S-1, "Inspection of Control Blade and Offset Mechanism, and Preparing an Offset for Service."

When a control blade and associated offset mechanism has been removed, a visual inspection is immediately performed with regard to the above criteria. The detailed results of these inspections as well as the signed approval by the Reactor Manager or his designee are captured as part of the Preventative Maintenance (PM) and Surveillance programs.

4.8 Section 4.2.3, Neutron Moderator and Reflector, Page 4-19.

a. *Discuss why the impact of radiation damage and thermal expansion on the rigid structure of the reflector materials is acceptable.*

Section 4.2.3, Neutron Moderator and Reflector, discusses the ability to replace the reflector materials on a scheduled or as-needed basis. Solid beryllium metal is used in many aerospace applications due to its strength and thermal stability, as well as light weight. The original beryllium reflector cracked while in service in an area of peak fluence due primarily to pore-gas pressure build-up. To prevent this cracking, the reflector is periodically replaced on a scheduled basis. Graphite reflector elements can be replaced as needed. Some are replaced periodically due to reconfiguration of their respective irradiation positions. The aluminum used to fabricate these reflector elements has a long service history with regard to radiation fields, and is particularly well suited for a neutron rich environment. Thermal expansion in the graphite reflector is a non-issue due to generous design tolerances within and between the reflector elements.

4.8 Section 4.2.3, Neutron Moderator and Reflector, Page 4-19.

c. *No technical specification (TS) is proposed for the 26,000 Megawatt-Day (MWD) period between replacements of the Beryllium reflector. Propose a TS or justify why this is not needed.*

Replacement of the beryllium reflector is an operational decision and not a safety issue; therefore MURR feels that no Technical Specification is required.

The first beryllium reflector cracked while in service in May 1981. After verification that the control rods were free to travel and met the control rod drop time Technical Specification, the reactor operated with the cracked beryllium until it was replaced in October 1981. The beryllium reflector is replaced approximately every 26,000 Megawatt-days (MWD) to avoid an operational problem that could possibly be created if the beryllium reflector were to crack.

4.8 Section 4.2.3, Neutron Moderator and Reflector, Page 4-19.

d. *Discuss Wigner energy in the graphite reflector and its potential hazard, if any, to the safe operation of the reactor.*

The following 1987 reference provides an excellent discussion of graphite Wigner energy in research reactors and the MURR specifically: *Comment Letter from University of Missouri with regard to NRC Docket No. PRM-50-44 – dated February 2, 1987.*

In this reference, the MURR was asked to submit comments regarding the possible hazards associated with graphite stored energy and the credibility of a graphite fire in research reactor applications. The resulting discussion concluded that Wigner energy generation in MURR graphite is small; the MURR approach to canning graphite in a thermally insulated helium gap promotes thermal annealing to reduce the storage of Wigner energy; and the MURR graphite is enclosed in several non-contiguous masses under several feet of water. Thus it can be stated that

there is insufficient energy available to ignite combustion, nor is there adequate air or other oxidizer available to sustain combustion.

Wigner energy in the MURR graphite reflector poses no potential hazard to the safe operation of the reactor.

4.9 *Section 4.2.4, Neutron Startup Source, Page 4-20. No TS for the administrative control of the irradiation of the regenerative neutron source has been proposed. Discuss actions to ensure that the neutron source licensed activity limit will not be exceeded. Propose TS wording or justify why it is not needed.*

The MURR neutron startup source is an antimony-beryllium (Sb-Be) source. The gamma ray responsible for inducing the (γ, n) reaction in beryllium originates from Sb-124 with a half life of 60 days, so the source has to be irradiated periodically to reactivate the antimony target. Very conservative calculations have been performed to estimate the activity resulting from the periodic irradiation and decay of the neutron source. To achieve a "steady state" activity of 100 Curies, the antimony target has to be irradiated for more than 5.0 hours every thirty days.

In order to maintain the source strength less than the license limit, the Sb-Be source is irradiated for no greater than two (2) hours every thirty days. The calculations indicate that the activity achieved with this monthly 2.0-hour irradiation at the peak reflector irradiation position is approximately 36 Curies. This periodic irradiation is tracked through the robust Preventive Maintenance (PM) tracking program that MURR utilizes to schedule and record all surveillance requirements and PMs associated with operation of the reactor.

Based on the level of formal control exercised and the conservative nature of the specification, we feel that a Technical Specification requirement regarding the startup source is not necessary to ensure safe operation of the reactor or safety of the general public.

4.10 *Section 4.2.5, Core Support Structure, Page 4-23. Discuss the impact of radiation damage, coolant chemistry and reactivity effects on core support structure materials over the renewal period of the reactor license.*

The greatest change due to radiation effects on core support structure materials occurs in the inner reactor pressure vessel just below core centerline where fast and thermal fluxes are at their peaks. The effect is a slight strengthening of the aluminum 6061-T6 inner pressure vessel in this area with a slight reduction in ductility. This is primarily due to the transmutation of some of the aluminum into silicon. This change is detailed in Section 16.1.2, Primary Coolant System Pressure Boundary.

Conductivity and pH are monitored and maintained in an appropriate range to minimize corrosion and degradation of core support structure materials. Both the primary and pool coolant systems have a clean-up system for demineralization and corrosion control. These ion-exchange systems maintain a low conductivity and maintain the pH in a range between 5 to 6. The slightly acidic pH maintained in the primary and pool coolant systems maximizes aluminum aqueous corrosion resistance. In addition to routine monitoring, water samples are taken weekly from the primary and pool coolant systems and are analyzed for pH, conductivity, and contained radioisotopes.

The small negative reactivity of the aluminum 6061-T6 core support structure materials is slightly reduced with increased fluence. This is due to the transmutation of the aluminum and the partial burnout of the trace alloy elements and impurities in the aluminum.

4.11 Section 4.3, Reactor Pool, Page 4-24.

b. *Provide details of plans to assess irradiation or chemical damage of the pool liner over the term of the license renewal, or justify why this is not needed.*

Irradiation damage to the pool liner is discussed in question 4.11.a. An independent inspection of the pool liner was conducted in 2001 by the engineering firm Sargent and Lundy^{LLC} (Ref. 16.5). The inspection determined the pool liner to be in excellent condition after greater than 30 years of service. The concluding paragraph of the Executive Summary of this report states:

> *Based on this inspection of the liner condition after 34 years of reactor operation, an additional 34 years of good performance by the aluminum pool liner is expected with the operating conditions and operating procedures continuing as they have been.*

Conductivity and pH are monitored and maintained in an appropriate range to minimize corrosion and degradation of the pool liner. Chemical damage to the pool liner is minimized by the pool coolant demineralizer loop, described in Section 5.5, Reactor Coolant Cleanup System. This loop ensures conductivity and pH are maintained at optimum levels to prevent corrosion of the aluminum pool liner. This ion-exchange system maintains a low conductivity and a pH in a range of approximately 5 to 6. The slightly acidic pH maintained in this system maximizes aluminum aqueous corrosion resistance. In addition to the online conductivity monitoring system, water samples are taken weekly from the pool coolant system and are analyzed for pH, conductivity, and contained radioisotopes.

Additionally, MURR re-examines the reactor pool liner during the periodic replacement of the beryllium reflector. Replacement of the beryllium reflector is performed approximately every 26,000 MWD, which corresponds to eight years of operation at our current operating schedule of approximately 150 hours per week. During replacement, the reactor pool is lowered to a depth of about 10 feet (9.1 m), which allows such periodic evaluations to be much more effective.

4.11 Section 4.3, Reactor Pool, Page 4-24.

c. *What is the minimum detectable amount of pool leakage and the length of time a leak could exist before detection? In the event of a pool leak, describe the probable path of the pool water, the potential for the pool water to enter the environment, and potential for radiological impact on members of the public.*

A leak in the pool coolant system could occur either in the containment structure at the reactor pool (pool proper), or in the mechanical equipment room (Room 114), and be caused by a mechanical piping or instrument failure, or by corrosion. In the event of a leak at the reactor pool, the entire biological shield structure that surrounds the pool is effectively separated from its surroundings and surrounded by two drain trenches in the containment basement (Beamport floor). Either of these trenches accommodates approximately 30 gallons (113.6 l) prior to initiating an audible and visual level alarm in the Control Room. In the event of a leak in Room 114, water would be visible during the visual inspection of this room performed every four hours. If leaking into an area that is not visually accessible, a maximum volume of roughly 70 gallons (265 l) could accumulate in the sump prior to initiating a level alarm in the Control Room. In the event of a leak through in-concrete piping, the water would accumulate in the tell-tale annulus and be highly unlikely to breach the outer annulus boundary. This leak would be detected within one week, if not made self-evident by pool level changes.

[redacted]

Finally, in the event of a pool leak, whether the leak is in the pool proper or the pool coolant system (see SAR Section 13.2.9.1), all probable leakage paths provide both retention and detection as described above. Therefore the possibility for pool water to enter the environment is essentially eliminated and the potential for radiological impact on members of the public is minimal.

4.13 Section 4.5, Nuclear Design

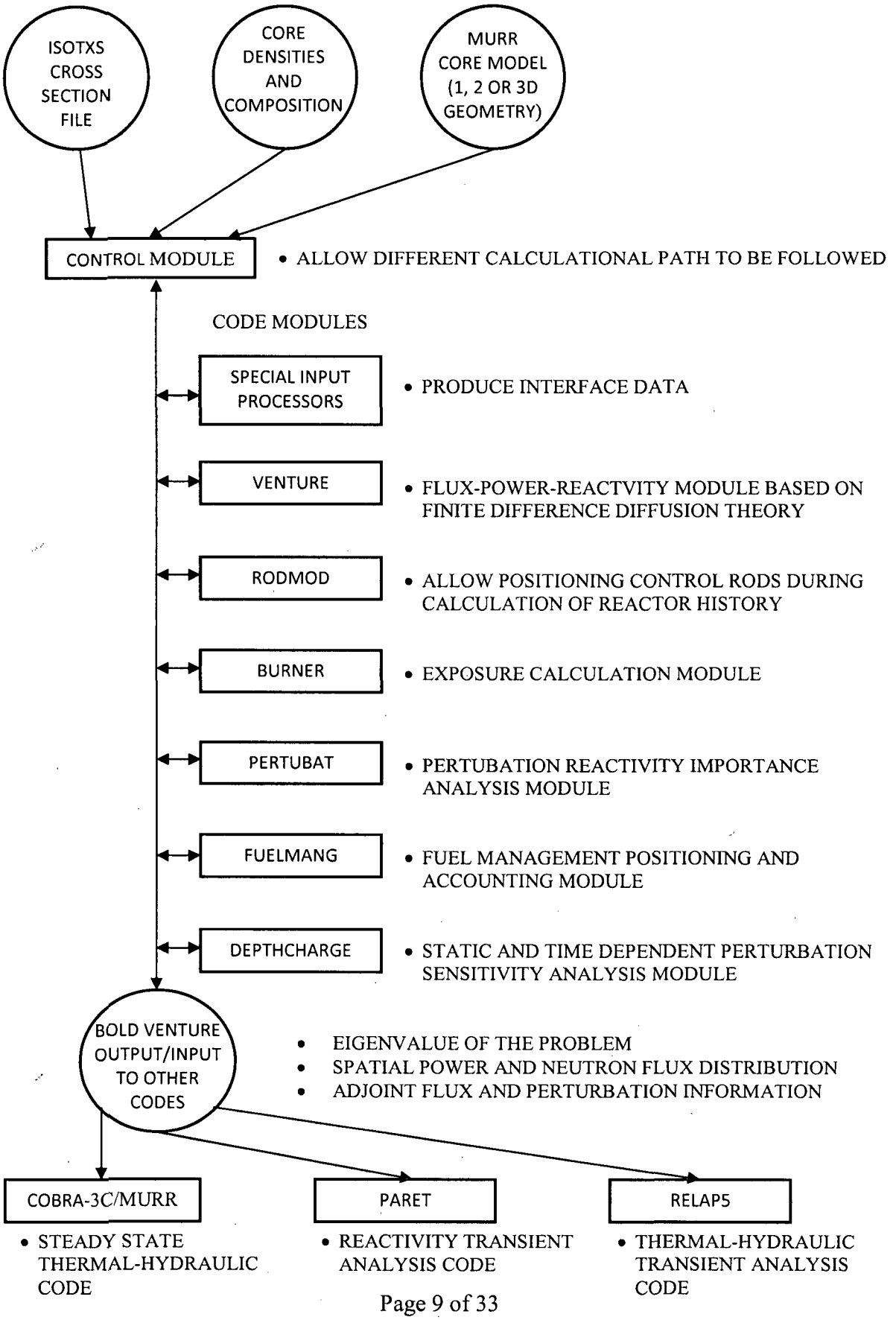
Note: As described in the first six paragraphs (pages 4-26 to 4-29) of Section 4.5, Nuclear Design, the MURR steady-state safety limits are based on nucleonics analyses performed prior to July 1974 using the EXTERMINATOR-II code. Starting with the 4th paragraph on page 4-29, the SAR discusses how the BOLD VENTURE-IV code system was used in the mid-1980s to perform 3D nucleonics analyses, including depletion, using diffusion theory methods. This information up to Section 4.5.1, Normal Operating Conditions, was included to show that using this more sophisticated code system confirmed that the original analyses using the EXTERMINATOR-II code provided sufficiently conservative steady-state safety limits. For example, based on the BOLD VENTURE 3D model, the highest nuclear peaking factor calculated for the MURR 775-gram mixed burnup cores was 3.146, which is well below the safety limit assumed peaking factor of 3.676 obtained from combining the peaking factors from three separate EXTERMINATOR-II code 2D models, viz., R-Z (for radial and axial peaking), Θ-R all fresh fuel (for azimuthal within a single fuel element) and Θ-R mixed burnup (for azimuthal between fuel elements with different burnups).

The information included in Section 4.5 starting with the 2nd paragraph on page 4-36 and up to, but not including, the 1st paragraph on page 4-38 are details from work (Ref. 4.21) performed in 1986 in support of the submittal to the NRC for approval of Amendment 20 to Facility License R-103. Amendment 20, which was issued on August 1, 1990, authorized the MURR to use Extended Life Aluminide Fuel (ELAF) in the reactor core (Ref. 4.46). This amendment allowed the MURR to use a new fuel element design which would have significantly reduced the fuel cycle cost and reduced the amount of uranium-235 needed per MWD of energy produced. Nonetheless, as stated on page 4-29, the ELAF elements have not been fabricated on a production scale, and hence, were never used in MURR cores to date. This information is not required to support the steady-state safety limits analysis approved in 1974 and, in retrospect, should probably not have been included in the application for renewal of Facility License R-103. However, we have answered the questions as requested.

Additionally, in answering these questions, it was noted that in the fourth line of the 1st paragraph on page 4-38 the maximum power density should be 953 watts/cc and not 953 watts/cm^2 as stated.

a. *Page 4-30. Figure 4.8 appears to show details of AMPX-II and not the computational system BOLD VENTURE IV as stated in the text. Provide a figure similar to Figure 4.8 which illustrates components of BOLD VENTURE.*

Attached is a figure showing the various computational modules of BOLD VENTURE.


ISOTXS CROSS SECTION FILE
CORE DENSITIES AND COMPOSITION
MURR CORE MODEL (1, 2 OR 3D GEOMETRY)
CONTROL MODULE
• ALLOW DIFFERENT CALCULATIONAL PATH TO BE FOLLOWED
CODE MODULES
SPECIAL INPUT PROCESSORS
• PRODUCE INTERFACE DATA
VENTURE
• FLUX-POWER-REACTVITY MODULE BASED ON FINITE DIFFERENCE DIFFUSION THEORY
RODMOD
• ALLOW POSITIONING CONTROL RODS DURING CALCULATION OF REACTOR HISTORY
BURNER
• EXPOSURE CALCULATION MODULE
PERTUBAT
• PERTUBATION REACTIVITY IMPORTANCE ANALYSIS MODULE
FUELMANG
• FUEL MANAGEMENT POSITIONING AND ACCOUNTING MODULE
DEPTHCHARGE
• STATIC AND TIME DEPENDENT PERTURBATION SENSITIVITY ANALYSIS MODULE
BOLD VENTURE OUTPUT/INPUT TO OTHER CODES
• EIGENVALUE OF THE PROBLEM
• SPATIAL POWER AND NEUTRON FLUX DISTRIBUTION
• ADJOINT FLUX AND PERTURBATION INFORMATION
COBRA-3C/MURR
• STEADY STATE THERMAL-HYDRAULIC CODE
PARET
• REACTIVITY TRANSIENT ANALYSIS CODE
RELAP5
• THERMAL-HYDRAULIC TRANSIENT ANALYSIS CODE

4.13 Section 4.5, Nuclear Design

b. *Page 4-35, Figure 4.11. Figure 4.11 does not contain error bars or a legend, and the burnup step being compared is not clear. In light of this, discuss the conclusion that the R-Z MURR model agrees with the observed measurements from fuel element 775-F3.*

As indicated on page 4-30, the only "experimental" data that was available for verifying the validity of the MURR neutronics models that were developed using the BOLD VENTURE code system were the results from the destructive analysis of a single MURR fuel element [redacted] that had reached a burnup of 82.5 MWDs. This element was shipped to Idaho National Laboratory in order to obtain the burnup information of the individual fuel plates of this element.

Figure 4.11 is comparing the results obtained for the peak individual fuel plate burnup values with those calculated using the BOLD VENTURE code. The reasonable agreement between the computational results and the measured values in terms of the general shape of the peak burnup values as well as their comparable magnitude was used to infer the general validity of the BOLD VENTURE neutronics model used for the MURR core analysis.

Also, as indicated on page 4-29, 4th paragraph, the analyses reported here using the BOLD VENTURE suite of codes for the 6.2 Kg MURR core were done primarily to verify and reconfirm many of the reactor physics parameters used in the MURR Hazards Summary Report and its addenda. The results obtained for the current core during the Extended Life Aluminide Fuel (ELAF) analyses were never used to set any safety limits or provide additional safety basis for the current MURR core.

4.13 Section 4.5, Nuclear Design.

c. *Page 4-37, Figure 4.9. Provide further detail of the methodology used to generate Table 4.9 and how reactivity due to burnup of fuel was calculated when control rod position varies at every MWD step.*

It was in error to report the reactivity effect of fuel burnup based on the values provided in Table 4-9 (page 4-37). With the control rods moving simultaneously, only the combined effect of rod motion plus fuel burnup can be derived from the information provided. The only possible way to estimate the fuel burnup effect is to assume the reactivity effect of fuel burnup is linear and estimate the effect from the last two burnup steps indicated in the Table. For the last two burnup steps, the control blades are kept at a constant height and for that 330 MWD burnup step the change in reactivity can be calculated and the value for 1300 MWD burnup can be estimated assuming the reactivity change is linear with fuel burnup. If that is done, the effect of fuel burnup will be approximately 3.3% instead of the 4.6% reported.

The three columns of values listed in Table 4-9 were taken from Table 7 of a MURR internal report (Ref. 4.21) generated for the Extended Life Aluminide Fuel (ELAF) amendment submittal. Table 7 compared the reactivity with and without boron included in the proposed ELAF core design to the MURR 6.2 Kg core. The BOLD VENTURE calculations were performed with the control blades at various heights and fuel compositions readjusted at discrete burnup steps based on the isotope burner module included in the BOLD VENTURE suite of codes for this comparison.

4.13 Section 4.5, Nuclear Design.

d. *Provide details of the Theta-R 2D calculations and how they are incorporated into the 3D MURR model.*

The neutronics module from the BOLD-VENTURE code allows both 2D and 3D calculations. Because of the excessive computational time needed to perform full 3D calculations, several initial scoping calculations were performed using only 2D models, viz., R-Z or Θ-R calculations. Of these, the R-Z model was used extensively since it allows for modeling the axial asymmetry present in the MURR core and for modeling the control blade motion during fuel depletion studies.

The Θ-R model was used only to provide a method to identify the cases with the highest possible peaking factors due to the azimuthal power variation when using mixed fuel core loadings. Using this model, several combinations of mixed fuel core loadings were analyzed to obtain the worse cases in terms of azimuthal power peaking values. These cases were then further analyzed using the more detailed Θ-R-Z 3D calculations to determine the worst case power peaking values.

4.13 Section 4.5, Nuclear Design.

e. *Page 4-36. An error larger than the 0.43% error listed in the text can be calculated in the same manner as the 0.43% error between the "blades-all-out" 2D and 3D model discussion in the text. Discuss the error in k_{eff} for "all blades full-in" and how this impacts reactor control.*

When comparing the k_{eff} values for the "Control Blades Full-In" case for both the R-Z and the Θ-R-Z models, the error in the 2D result is approximately 2.3%. This is higher than the "Control Blades Full-Out" case but still relatively small considering the inherent approximations that are present in the 2D model. For example, as mentioned on page 4-36, the control blade modeling in the R-Z model assumes full 360-degree coverage around the core as opposed to the actual 288-degree coverage offered by the four shim control blades. As expected, the difference in the k_{eff} values between the 2D and 3D cases will be lower in the "Control Blades Full-Out" case since the effect of the control blades on the core is virtually eliminated.

As far as the safety implications regarding reactor operation and control are concerned, the critical safety decisions are based on measured control rod worths and the Technical Specification requirements. During the Extended Life Aluminide Fuel (ELAF) design evaluation performed in 1986-1990 from which these values are taken, the critical decisions were always based on the more detailed Θ-R-Z 3D model and not on the simpler 2D models. 2D models were primarily used for scoping studies in order to save computational time. After the scoping studies were completed, selected cases were further analyzed using the detailed 3D model to verify that cores loaded with only ELAF and cores loaded with both types of fuel (ELAF and [redacted]) would operate within the safety limits established and approved in 1974.

4.13 Section 4.5, Nuclear Design.

f. *Page 4-37. The results k_{eff}=1.109 appears to be calculated from the R-Z model (1.1124 – 0.0034), not from the 3D MURR model as suggested in the text. Provide a clarification for the comparison of the 3D MURR model to the measured value of k_{eff}=1.095.*

That is correct. An error was made when the measured k_{eff} value was improperly compared to the value calculated by the BOLD VENTURE R-Z model instead of the Θ-R-Z model. The 3D model computed k_{eff} value should have been reported as 1.1139 (1.11725 – 0.0034). Also, while reviewing this section, it was noted that the measured k_{eff} value was erroneously reported on page 4-37 as 1.095. The correct measured k_{eff} value is 1.0795 as listed in Table 4-12 (page 4-41). Based on this measured value, the difference in the BOLD VENTURE calculated value becomes 3.2% instead of the reported 1.3%.

4.13 Section 4.5, Nuclear Design.

g. *Page 4-38, Table 4-11. The R-Z model total blade worth differs from what is stated in the text, 0.1809 on pg 4-36. Please explain this difference.*

The 2D R-Z model total blade worth listed in Table 4-11 and stated in the text on page 4-36 do differ. Both values were obtained from the 1986 Benchmark Study (Ref. 4.21). Table 4-11 simply restates the "6.2 Kg Core" portion of Table 9 (page 52) from Reference 4.21. Table 9 lists 0.1813 as the value for the R-Z model total blade worth. The R-Z model total blade worth statement on page 4-36 was taken from the first full paragraph on page 48 of Reference 4.21. There is no explanation given in the study for the slight difference between these two values.

The difference between the k_{eff} for control blades full-out (1.1124) and control blades full-in (0.9315) listed in Table 6 (page 47) of Reference 4.21 is 0.1809. The correct control blade worth for the R-Z model should be equal to 0.1746 and not the difference. This same error appears in the total worth based on the 3D Θ-R-Z model in Table 4-11, which was also taken from Table 9 (page 52) of Reference 4.21. Instead of being the difference between the two k_{eff} values, it should have been reported as 0.1538 ΔK/K in both Table 9 of Reference 4.21 and Table 4-11 of the SAR.

4.13 Section 4.5, Nuclear Design.

h. *Page 4-36. Neither the "flux trap model" nor the "benchmark study" are well described in the text of this section. Provide additional clarification.*

As indicated earlier, when verifying the validity of the MURR neutronics model that was developed using the BOLD VENTURE code system, the only "experimental" data available for benchmarking were the results from the destructive analysis of a single MURR fuel element [redacted] that had reached a burnup of 82.5 MWDs. The fuel element had received this amount of burnup during normal operations with a loaded center test hole canister inserted.

In order to compare the fuel plate burnup values from the BOLD VENTURE neutronics model against those from the destructive analysis of the [redacted] element, the BOLD VENTURE neutronics model had as much detail as possible of the core and the surrounding regions. Since the experimental samples and the center test hole canister in the center test hole (flux trap region) can impact the flux (and power) as seen by the inner fuel plates, the three-tube test hole canister and the samples generally present in the central flux trap region were also included in the BOLD

VENTURE neutronics model. The three-tube test hole canister shown in Figure 10.1 was modeled using cylindrical rings to closely approximate the material densities and locations within the center test hole. This detailed model is referred to as the "Flux Trap" model or the "Benchmark Study" model in the SAR.

For the remainder of the reactor physics calculations (such as depletion studies), the flux trap region within the inner pressure vessel was filled with water. No experimental samples nor the test hole canister were included in order to simplify the model for time consuming depletion calculations. As mentioned on page 4-36, this will result in conservative flux in the inner fuel plates, which have the highest power density in the core. Without any experimental samples or the center test hole canister in the flux trap region there is a higher thermal neutron flux in the center test hole. Therefore, more thermal neutrons can reach the fuel plates as opposed to being absorbed in the experimental samples. This "simplified" model was used for most of the reactor physics calculations.

4.13 Section 4.5, Nuclear Design.

i. *Page 4-37, Table 4-10. Provide a clarification of the results presented in the table. How is the peak power density for the "control blades full-in" case calculated from the 3D model?*

The values in Table 4-10 were taken from Table 8 of Reference 4.21 which compared the differences in k_{eff} and peak power density between the [redacted] core and the ELAF [redacted] core with the control blades in the full-out and full-in conditions. In the actual reactor there would obviously be no power density with the control blades full-in (shutdown). However, the BOLD VENTURE output provides peak power densities even for models that are subcritical, which were included in Table 8 of Reference 4.21 to show how power peaking had significantly decreased in the [redacted] core with the reduced fuel loading in the outer fuel plates. For the [redacted] core, the control blades full-out creates the highest power peaking in plate number-23 near the outer reactor pressure vessel. Control blades full-in creates the highest power peaking on plate number-4 near the inner pressure vessel. For the [redacted] core the highest power peaking always occurs in plate number-1. Table 4-10 should have only included the k_{eff} values.

4.14 Section 4.5.3, Operating Limits.

b. *Page 4-37. The cold clean core k_{eff} in Table 4-12 does not agree with pg 4-37 first paragraph. Discuss this difference.*

The cold clean core k_{eff} value listed in Table 4-12 is taken from a document titled "Low Power Testing Program For The Missouri University Research Reactor [redacted] Core," an internal report dated October 20, 1971. The report documents how the total control blade worths were determined, including the cold clean critical rod heights for the first [redacted] uranium-235 core. Based on these values, the report gives the cold clean core k_{eff} as 1.0795. The measured value in the 1st paragraph on page 4-37 should be corrected to 1.0795.

4.17 Section 4.6.2, Steady State Forced Cooling.

a. *It appears that the SAR uses two sets of peaking factors for two different thermal analyses. Provide clarification as to the reason for reporting both analysis results. Are the inputs used bounding for current and all fuel cycles?*

In Section 4.6.2, Steady-State Forced Cooling Analysis, the peaking factors listed in Table 4-13 appear to be different than those provided in Table 4-14 but are actually based on the peaking factors in Table 4-14. These peaking factors were derived as described in the 2nd and 3rd paragraphs of Section 4.5, Nuclear Design, in three (3) different 2D EXTERMINATOR-II models as indicated after the below peaking factors.

Table 4-14, "Summary of MURR Hot Channel Factors," lists the following Nuclear Peaking Factors:

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

The ratio of the highest heat flux to the core average heat flux is the product of these four (4) peaking factors.

- Hot Spot Heat Flux/Ave. Core Heat Flux = 2.220 x 1.112 x 1.040 x 1.432 = 3.676

Table 4-13, "Summary of Heat Transfer Data," lists:

- Max. Power Density in Hot Channel/Ave. Power Density in Hot Channel = 1.489
- Ave. Power Density in Hot Channel/Ave. Core Power Density = 2.469

The Max. Power Density in Hot Channel/Ave. Power Density in Hot Channel equals the Axial times the Local (Circumferential) peaking factor [1.489 = 1.432 x 1.040].

The Ave. Power Density in Hot Channel/Ave. Core Power Density equals the Radial times the Non-uniform Burnup peaking factor [2.469 = 2.220 x 1.112].

The product of these two factors is the ratio of the Max. Power Density in Hot Channel to the Ave. Core Power Density, which with the MURR uniform fuel meat thickness equals the ratio of the highest heat flux to the core average heat flux.

- Max. Power Density in Hot Channel/Ave. Core Power Density = 1.489 x 2.469 = 3.676
- Hot Spot Heat Flux/Ave. Core Heat Flux = 2.220 x 1.112 x 1.040 x 1.432 = 3.676

This combination of peaking factors provides an overall peaking factor that is conservative, because it combines the higher overall radial, axial and local (circumferential) peaking factors of a core consisting of all fresh fuel elements with the azimuthal peaking factor of a mixed burnup core. A 3D code would show the axial and radial peaking factors in a mixed burnup core are lower than in a fresh fuel core due to the higher critical control rod position. This is due to the lower critical rod height of an all fresh core with the MURR control blades external to the outer pressure vessel, which pushes power down and inward in the core increasing the axial and radial

peaking factors. Therefore, the steady-state safety analysis peaking factors listed in Table 4-14 bounds all MURR fuel cycles with the currently approved 775-gram HEU fuel elements.

4.18 Section 4.6.3, Safety Limit Analysis.

d. *Present the results of the calculation from the equation at the top of page 4-60 or explain why the results are not given.*

The equation on the top of page 4-60 was left there inadvertently and is not relevant to the discussion of MURR's Safety Limits or LSSS presented in that section. The equation will be removed from that section.

CHAPTER 5

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

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

5.3 *Section 5.2.4, Heat Exchangers. Provide a discussion of pressures in the primary, pool and secondary coolant system during operation and shutdown conditions and the potential for primary and pool coolant to enter the secondary system. Discuss the sensitivity of the secondary coolant monitoring system to detect radionuclides in the secondary coolant. In the event of leakage into the secondary system, what is the minimum detectable leak rate, the length of time a leak could go undetected, and the radiological impact to the unrestricted environment?*

The design of the plate-type heat exchangers makes a primary-to-secondary or a pool-to-secondary leak extremely unlikely. The plate design is such that the most likely leak path is past a plate-to-plate gasket and into the mechanical equipment room (Room 114). To leak into the adjacent system, a path would have to be created through one of the plates, which in all likelihood is minimized by a design that has no flow stagnation points. In the event of this leak path in the pool coolant heat exchanger, pressures are higher on the secondary coolant side than on the pool side under all operating conditions. This would result in a secondary-to-pool leak. In the event of this leak path in one of the primary coolant heat exchangers, pressures vary on the secondary coolant side such that the primary coolant could conceivably leak into the secondary coolant system. Note: The two (2) tube-type, water-to-shell primary coolant heat exchangers described in Section 5.2.4, Heat Exchangers, were replaced with two (2) plate-type heat exchangers in December 2006.

The secondary coolant monitor is a sodium iodide (NaI) scintillation detector located in the coolant leg leaving the heat exchangers. It is operational and responsive for a wide range of energies which allows it to detect any activity present in the secondary coolant. The detector is roughly 2.5% efficient over the range of energies that would be expected during a pool/primary to secondary leak. The detector is situated in the coolant line such that it views an effective sample volume of 7.5 liters moving past the detector.

To detect an increase in secondary coolant monitor levels would require these levels to increase by roughly 7000 cpm to reach the alarm set point. A maximum interval of two hours is possible between log readings if the alarm set point is not reached prior to the log readings.

Using the total volume of the secondary coolant system and the average total activity concentration of the primary coolant, a 7000 cpm rise in two hours would occur at a leak rate of approximately 2.2 liters per hour. A higher leak rate would result in reaching the alarm set point sooner. A lower leak rate may require one or more additional log intervals to detect, but would make itself readily apparent over time.

At the above leak rate, an increase of 1000 cpm over the baseline would be reached in approximately 17 minutes and an increase of 2500 cpm would be reached in 43 minutes. Thus it is possible that the increases would be noted by the control room staff at some interval between the two hour log data window.

With regards to environmental consequences to this hypothetical leak at the above noted leak rate, the activity concentration for the three most significant isotopes associated with primary coolant intruding into the secondary system would be less than 50% of the Table 2 Effluent Limits in 10 CFR Part 20 Appendix B at the time of the longest interval between logs. Thus, upon the discovery of the leak with the secondary coolant alarm, the activity concentrations of the individual constituents of the contaminated secondary water would be near the uncontrolled release limits for those isotopes and would not pose a dose threat to the general public.

5.4 Section 5.3, Pool Coolant System.

a. *Propose TS wording for surveillance of gamma-emitting isotopes in the reactor pool water or justify why it is not needed.*

The pool coolant system is sampled weekly as required by Preventative Maintenance (PM) procedure P5-W1, "Pool Water Analysis." This surveillance is tracked though the robust PM tracking program that MURR utilizes to schedule and record all surveillance requirements and maintenance associated with operation of the reactor. The pool coolant sample is then analyzed in accordance with MURR operating procedure OP-RO-420, "Primary and Pool Water Analysis," for pH, conductivity and gamma-emitting isotopes.

MURR feels that this administrative control through the PM program is more than sufficient to ensure that the pool coolant system is sampled weekly and that no Technical Specification is required.

5.6 *Section 5.7.1, Nitrogen-16 Control System, Introduction. The statement is made that the design of the nitrogen-16 control system provides reasonable assurance that the system will not interfere with reactor cooling, cause an uncontrolled loss or release of primary coolant, to prevent safe reactor shutdown. Explain the bases for reaching this conclusion.*

As described in Section 5.7, Nitrogen-16 Control System, the system consists of only two components – a hold-up tank in both the primary coolant demineralizer loop and the pool coolant system. The only system failure that can be postulated is a leak in either hold-up tank, which is essentially no different than a leak anywhere else in the primary or pool coolant systems as analyzed in Chapter 13.

A leak in the pool coolant system is described in Section 13.2.9.1. A small leak from the pool coolant hold-up tank would easily be detected before the water level in the reactor pool had lowered significantly. The leak would also be visible during observation of the mechanical equipment room (Room 114) during one of the periodic routine patrols. Any leakage would be collected in sumps that would activate switches that would automatically pump the water to the Liquid Waste Disposal System (as described in Section 9.11.4 of the SAR). Activation of these switches will also cause an audible and visual alarm in the control room. With awareness of the leak, sufficient time would be available for the reactor operator to shutdown the reactor, and secure the pool coolant circulation pumps and close isolation valve V509 before a significant loss of pool water could occur. A large leak could potentially cause a major loss of water from the pool. In the event of a large leak in the pool coolant hold-up tank, an automatic reactor scram would occur from either a reduction in coolant flow, pressure, or pool water level. Regardless, a large or small leak in the pool coolant hold-up tank would not interfere with reactor cooling, cause an uncontrolled loss or release of primary coolant, or prevent a safe reactor shutdown.

A leak, or rupture, in the primary coolant system is described in Section 13.2.3. A small leak from the primary coolant demineralizer loop hold-up tank would easily be detected by a lowering of pressurizer liquid level or by observation in Room 114 during one of the periodic routine patrols as mentioned above. Any leakage would be collected in sumps that would activate switches that would automatically pump the water to the Liquid Waste Disposal System. Activation of these switches will also cause an audible and visual alarm in the control room. A large leak would cause an automatic reactor scram to occur from a loss of system pressure as sensed by any one of four (4) pressure transmitters. In addition to causing a scram, the primary coolant circulation pumps would stop and the primary coolant isolation valves V507A and V507B would close. This would isolate the leak from the reactor core, which would prevent the core from being uncovered. Additionally, two (2) 2-inch remote air-operated-to-open, spring-to-close isolation valves, designated V527E and V527F, are installed on the inlet and outlet of the primary coolant demineralizer loop. These valves provide the control room operators with the ability to isolate the primary coolant demineralizer loop, including the hold-up tank, from the rest of the primary coolant system. Therefore, a large or small leak in the primary coolant demineralizer loop hold-up tank would not interfere with reactor cooling, cause an uncontrolled loss or release of primary coolant, or prevent a safe reactor shutdown.

CHAPTER 6

6.2 *Section 6.3.8, Anti-Siphon System, Design Analysis. The analysis of the system using RELAP5 assumes the anti-siphon valves will open 85 msec after the primary coolant rupture occurs. Discuss the basis for this assumption. Provide additional TS wording on anti-siphon valve performance or justify why this is not needed.*

We are requesting that this question be included in the set of "complex" questions (120-day response). This would allow combined review of the description of the benchmarking for Section C.1 with the benchmarking discussion for Chapter 13 analyses.

CHAPTER 7

7.1 *Section 7.2, Design of Instrumentation and Control System. In Section 3.1.2, "Overall Requirements. For Criterion 3: Fire Protection", the SAR states that fire protection is not required to accomplish a safe shut down of the reactor. Discuss the basis for this statement.*

Safe shutdown of the reactor is not dependent on the availability of forced cooling, decay heat removal (see SAR Section 13.2.9.3), coolant charging systems, or emergency electrical power (see SAR Section 13.2.7). Power availability and drive train thermal delay protection and/or fire suppression for these systems is thus not required from the standpoint of fire protection.

All physical components involved in shutting down the reactor are below the pool water line. They are not vulnerable to the typical weaknesses, embrittlement, and deformation failures associated with a fire. Thus, thermal delay protection and/or fire suppression for these components is not required from the standpoint of fire protection.

All remaining components involved in safely shutting down the reactor are essentially electrical in nature, and are utilized in a non-coincidence logic system. Under fire conditions, these control components, circuits and instrument channels may fail, but will fail in either a grounded or an open circuit state. Either and any combination of these failure states in any of the circuits significant to safety will still result in the desired loss of power, and a resultant safe shutdown of the reactor. The failure state where ungrounded continuity could be maintained due to fire conditions would have to occur in multiple locations simultaneously with no grounding and no open circuit failures. Thus, thermal delay protection and/or fire suppression for these instrument channels, circuits and control components is not required from the standpoint of fire protection.

7.2 *Section 7.7, Table 7-8, Reactor Scrams, and TS 3.3, Reactor Safety System. The scrams listed in the table do not appear to match the supporting TS. Discuss TSs for scrams initiated by building plenum and bridge high activity and low primary heat exchanger (HX) differential pressure or justify why they are not needed.*

Table 7-8, "Reactor Scrams," is a general description of the scrams associated with the reactor safety system and the terminology designated on the control room Annunciator.

Each scram is itemized in TS 3.3 with the exception of Building Plenum & Bridge High Activity. Building Plenum & Bridge High Activity is included in TS 3.4.a as required reactor instrumentation, but it is also an input into the reactor isolation system, and hence, a reactor isolation scram.

The low primary heat exchanger differential pressure scrams (DPS 928A and DPS 928B) serve as inputs for primary coolant flow scrams as described in section 7.6.4.1, Primary Coolant System. These detector channels are included in TS 3.3 as indicated below as two of the four primary coolant flow channels required for Mode I operation.

Table 7-8, Reactor Scrams	Associated TS 3.3
Manual	Manual Scram
Channel 2 & 3 Short Period	Reactor Period
Reactor Loop Low Flow	Primary Coolant Flow
Reactor Loop High Temp	Reactor Out Water Temperature
Pressurizer Low Level	Pressurizer Low Water Level
Reactor Pool Below Refuel Level	Pool Low Water Level

Channel 4, 5, & 6 High Power	High Power Level
Power Level Interlock	Power Level Interlock
Pressurizer High Pressure	Pressurizer High Pressure
Reflector Hi-Low Diff. Pressure	Differential Pressure Across the Reflector
Pool Valve 509 Off Open	Pool Coolant Isolation Valve 509 Off Open Position
Evacuation or Isolation	Reactor Isolation and Facility Evacuation
Reactor Loop Low Pressure	Primary Coolant Low Pressure
Low Primary HX Diff. Pressure	Primary Coolant Flow
Pool Loop Low Flow	Pool Coolant Flow

Table 7-8, Reactor Scrams	Associated TS 3.4
Bldg Plenum & Bridge High Activity	Reactor Containment Building Exhaust Plenum Radiation Monitor

7.3 *Section 7.9.3.3, Surveillance. Propose TS wording to support the SAR statement that the Fuel Element Failure Monitoring System's high radiation annunciator alarm is tested before reactor startup or justify why this is not needed.*

The Fuel Element Failure Monitoring System's high radiation annunciator alarm is tested prior to each reactor startup in accordance with the "Process Radiation Monitor" portion of the Long Form Startup Checksheet (FM-57). This checksheet implements MURR operating procedure OP-RO-710, "Radiation Monitoring – Area Monitors" Section 4.3, where the "Reactor Coolant Loop Hi Activity" annunciator and associated circuitry is tested and the alarm set point is verified.

MURR feels that this is sufficient to ensure operability of the system and that no Technical Specification is required.

CHAPTER 9

9.4 *Section 9.1.3, Surveillance. Propose TS wording to support the SAR statement, Fan Failure Alarm Panel is periodically tested for operability, or justify why this is not needed.*

The Fan Failure Alarm Panel is tested weekly prior to each reactor startup in accordance with the "Building Tower Checks" portion of the Long Form Startup Checksheet (FM-57). This checksheet refers to operating aid OA-1, "Facility Exhaust Fans EF-13 and EF-14, EF-13 Running," or OA-2, "Facility Exhaust Fans EF-13 and EF-14, EF-14 Running." These checks include the testing of the standby fan feature and verification of the actuation of the audible low flow alarm.

No Technical Specification is required for the surveillance of the Fan Failure Alarm Panel. No Limiting Condition for Operation exists which is associated with the Fan Failure Alarm Panel. The actuation of the alarm in the reactor control room does not initiate any protective action. The Fan Failure Alarm Panel serves as an operator aid designed to alert the operator that an abnormal condition may exist in the facility ventilation exhaust system and prompt the on duty reactor operator to investigate the cause of the alarm.

MURR feels that this is sufficient to ensure operability of the system and that no Technical Specification is required.

CHAPTER 10

10.5 Section 10.3.1, Center Test Hole. Related to license amendment request of August 6, 2009.

a. *Describe any occurrences where the center test hole canister failed to properly latch or when a canister was removed from the reactor when not intended.*

Each time the center test hole canister is installed, proper latching is verified by two members of the Operations staff, one being a licensed senior reactor operator, both visually and physically (by pulling upward on the canister rope). Instances in which the center test hole canister can not be verified latched result in reinstallation or repair followed by verification. The hold-down fixture within the canister is also verified latched prior to installation. With the hold-down fixture and the existing latching means being verified at each installation, no instances of improper latching or unintended canister removal have occurred.

10.5 Section 10.3.1, Center Test Hole. Related to license amendment request of August 6, 2009.

b. *Describe any changes to procedures for sample handling or irradiation in the center test hole necessitated by the installation of the flux-trap irradiations reactivity safety trip (FIRST) system.*

The existing procedures for sample handling in the center test hole will remain essentially unchanged, with the following exception: with the ability to operate the FIRST system in BYPASS, the appropriate administrative reactivity limit must be selected by the loading preparer, and reviewed and approved by the Assistant Reactor Manager-Physics. If the FIRST system is in BYPASS, the reactivity worth of the center test hole canister must be included in the calculation of the absolute value of the reactivity worth of all experiments in the center test hole. BYPASS operation will require Reactor Manager's approval to perform.

10.5 Section 10.3.1, Center Test Hole. Related to license amendment request of August 6, 2009.

c. *Describe the possible failure modes of the switches used on the FIRST system.*

In December, 2008, a temporary, prototype FIRST device and a mock instrument channel were installed to test operation and reliability of the FIRST system, including the switches. This test has continued to date in order to accumulate operating experience in using the device and to validate its long term reliability.

Within the first few months of the test, it was determined that the original center test hole canister wear ring design could potentially prevent proper contact with the switch plungers, thus preventing the switches from closing and 'making-up' the safety system input. The wear ring was redesigned to ensure full contact with the switch plungers and no additional problems regarding proper contact have occurred.

The failure mode mentioned above – where a switch failed to close – is conservative in that the safety system cannot be reset. While the failure mode where a switch cannot open the contact has not occurred, the failure of one switch would not prevent the remaining redundant switch from performing its intended safety function.

The mode where contact make or break could change with respect to plunger travel or maladjustment have not been observed but would result in one of the above two failures – not closing or not opening. However, neither is a reactor safety system failure.

Periodic surveillance and maintenance inspections will also reveal maladjustments in switch positioning or potential switch failures that do not make themselves apparent through nuisance trips.

10.5 Section 10.3.1, Center Test Hole. Related to license amendment request of August 6, 2009.

d. *The SAR states that the cable jacket will display good radiation resistance for about 10 years of operation. Discuss the consequences of any potential failure of the FIRST cabling.*

Degradation of the FIRST cabling is specifically noted during periodic inspections, and cabling is replaced if signs of degradation are present. Failure modes for the cabling will result in either open circuit, a conservative failure where the safety system cannot be reset, or closed circuit, where the condition would not prevent the remaining circuit from performing its safety function.

Periodic surveillance and maintenance inspections are intended to bring to light any cabling degradation prior to failure, and failure modes can be shown to be conservative.

10.5 Section 10.3.1, Center Test Hole. Related to license amendment request of August 6, 2009.

e. *Discuss the proposed methodology that will be used to determine sample reactivity when the FIRST system is in use.*

The current methodology that is used to determine the reactivity present in the flux trap region or center test hole involves several independent measurements as described below. The same methodology will be used to determine sample reactivity values when the FIRST system is in use.

The reactivity neutral condition of the reactor, as far as the central flux trap region is concerned, assumes a center test hole filled with just water (at 110 °F) and no individual experimental samples or the center test hole canister (also known as the removable experiment sample canister as defined in TS 1.3) installed.

The reactivity contribution of the center test hole canister is first determined by performing two back-to-back reactor startups; the first one with the center test hole empty (only water present) and the second one with an empty center test hole canister - either the 3 or 6-barrel as described in Section 10.3.1.2. The difference in the control rod heights between the two startups in conjunction with the known integral rod worth curve is then used to determine the reactivity contribution from the empty test hole canister. Any reactivity contribution due to a change in temperature is accounted for by the primary and pool coolant temperature coefficients of reactivity and the corresponding temperature difference between the two critical measurements.

The same methodology is used to determine the total reactivity contribution from a fully loaded test hole canister. That is, two back-to-back reactor startups are performed; one with just water in the central region and the second one with the loaded test hole canister. The difference in control rod heights is used to determine the reactivity contribution from the loaded center test hole canister.

Individual sample materials are irradiated in the central test hole by placing them inside tightly sealed standard sized aluminum cans. Materials that require multiple encapsulations are first placed inside smaller sealed quartz or aluminum vials and then placed within the welded aluminum cans. Typically, several of these individual sample cans are placed inside the tubes that make up the center test hole canister. The reactivity contribution from these individual sample targets are also measured in a very similar fashion. After first estimating the possible reactivity worth of a new sample material to ensure no reactivity limits will be exceeded, two back-to-back reactor startups are performed. First, a hollow thin-walled aluminum can filled with water (referred to as water can) with the same dimensions as the individual sample can is placed within one of the center test hole canister tubes and a reactor startup is performed. Next the water can is removed and replaced with the sample can of unknown reactivity and a second reactor startup is performed. The difference in the control rod heights is then used to determine the reactivity worth of the unknown sample material.

The current methodology that is used to determine the total reactivity present in the flux trap region, including the reactivity contribution of the center test hole canister itself, will be the same methodology used when the FIRST system is in BYPASS.

The proposed methodology that will be used when the FIRST system is in use does not include the reactivity contribution of the center test hole canister, but continues to sum the reactivity worth of the individual samples. The sum of the individual sample reactivity worth is and will be subject to existing Technical Specification limits, as well as more restrictive administrative limits, and must be approved by the Assistant Reactor Manager and Lead Senior Reactor Operator prior to loading.

The methodology is also described in a letter to the NRC, dated August 6, 2009, requesting to amend the Technical Specifications that would allow implementation of the FIRST system.

10.5 Section 10.3.1, Center Test Hole. Related to license amendment request of August 6, 2009.

f. *Provide a discussion of how the design of the FIRST system meets the requirements of IEEE-279.*

The single failure criterion of IEEE-279 is completely met at the component level. The channel independence criterion of IEEE-279 will in large part be met, but in the case of certain components will not be fully complied with. Detection of the center test hole canister position must occur in the same physical space and from the same physical surface. This is a commonly accepted practice, where the channel independence criterion does not have to be extended to mean independent detection devices. Certain channel components, such as in-pool cabling and the BYPASS switch, are not physically separate. Where these proximities are concerned, no credible failure can be expected to occur that would simultaneously prevent both channels from performing their safety functions. Thus the design of the FIRST system effectively meets the intent of IEEE-279.

10.5 Section 10.3.1, Center Test Hole. Related to license amendment request of August 6, 2009.

g. *There does not appear to be a surveillance requirement to help ensure proper operation of the FIRST system. Propose and justify an appropriate surveillance requirement for the FIRST system or explain why a surveillance is not needed.*

As required by Technical Specification (TS) 4.4.a, the FIRST system will be tested for operability on a semiannual basis. A draft TS surveillance procedure (compliance procedure CP-36, "FIRST Scrams") has been prepared to verify proper operation of the FIRST system with regard to the reactor safety system.

A draft preventative maintenance procedure (RX-Q2, "Inspect FIRST Support Rig") has been prepared to verify mechanical integrity of the FIRST system, and will be performed quarterly.

CHAPTER 13

13.1 Section 13.2.1.2, Accident Analysis and Consequences.

b. *Page 13-4. Provide rationale for not including semi-volatiles such as Cs and Sr in the analysis.*

Cesium and strontium were not included in the initial Maximum Hypothetical Accident (MHA) dose calculations due to their minimal contribution to doses during the accident. The isotopes with the highest probability of contributing to worker and offsite doses during a fuel-related accident are iodines, kryptons and xenons and doses from these isotopes were included in Section 13.2.1.2.

Information concerning the behavior of semi-volatile elements such as cesium and strontium is not as readily available as it is for volatile elements. NUREG-1150, "Severe Accident Risks: An Assessment for Five US Nuclear Power Plants-Final Summary Report," does provide some estimated release fractions for both strontium (3%) and cesium (7%) in accident scenarios. Applying these release fractions to the MHA described in Chapter 13 results in doses from both radioactive strontiums and cesiums that are much less than 1% of the total dose that results from iodines, kryptons and xenons.

13.1 Section 13.2.1.2, Accident Analysis and Consequences.

c. *Page 13-8 and 13-9. Please clarify the units on the Air Submersion Dose conversion. Is it Sv Bq^{-1} m^3 sec^{-1}, not Sv/Bq-m^3 as it is indicated on page 13-8 and Sv/Bq-sec-m^3 on page 13-9?*

The numerical values found on pages 13-8 and 13-9 are all correct, however, there are errors in the units for Air Submersion Dose Equivalent H_T to the Whole Body (Ref. 13.37) found on pages 13-8 and 13-9. The unit used in Ref. 13.37 is Sv per Bq sec m^{-3} which can also be stated as Sv Bq^{-1} m^3 Sec^{-1}.

13.1 Section 13.2.1.2, Accident Analysis and Consequences

d. *Page 13-13. Please clarify how the concentrations are derived. Should they be 25 percent of the average concentrations listed on page 13-8, not 25 percent of those listed on page 13-6?*

The radioiodine concentrations in air leaking from containment listed on page 13-13 are correct; the maximum concentrations listed on page 13-6 should be used in deriving the values given on page 13-13.

The accident analysis and consequences described in Section 13.2 involves two separate analyses and thus two separate sets of nuclide concentrations. The first analysis relates to a person prior to their evacuation from the containment building following the postulated fuel failure and the Total Effective Dose Equivalent (TEDE) that person would receive as a result of this exposure while in containment. This first analysis utilizes an average concentration of iodines and noble gases equal to 50% of the maximum concentrations generated during the hypothetical accident due to the buildup of nuclides in the containment air over the 10-minute scenario. The accident assumes a 10-minute stay time in containment for dose calculation purposes and takes no credit for the deposition of iodines on structures as noted below. This portion of the accident and associated worker dose during the evacuation is described on pages 13-3 through 13-11 of the SAR.

The second portion of the Maximum Hypothetical Accident (MHA) analysis relates to the doses received by individuals downwind of the release from the containment structure due to the escape of fission products generated in the previous postulated accident analysis through an assumed barometric pressure change causing a pressure differential between the containment and laboratory buildings. This accident assumes that the concentrations of nuclides in the containment structure leak out of the containment building until such a time as pressure is equalized in containment with ambient atmospheric conditions. It is during this second portion of the scenario that credit is taken for the 75% deposition (plating out) of the iodines in the containment structure. As a result, 25% of the initial iodine concentration is used in the calculation of offsite doses in this scenario. Thus the correct initial concentrations (prior to taking credit for deposition) to be used in the offsite dose calculations are those listed on page 13-6, the maximum expected concentrations in containment generated by the accident.

13.10 *Section 13.2.9.5, Failure of the Neutron Startup Source. The SAR discusses detection sensitivity for a leaking neutron source. It appears that the measurement is conducted weekly. If the source failed between measurements, what level of antimony could build up in the pool coolant without detection?*

It is highly unlikely that a leaking neutron source could go undetected for more than one day due to the monitoring practices observed by the control room staff as well as the weekly pool coolant sample analysis performed at MURR.

The surface of the reactor pool is monitored by two radiation monitors with remote readouts in the control room. The readouts from these monitors are observed and recorded into the control room logs every two hours by the control room staff. A normal baseline activity and thus surface dose rate exists in the pool from the presence of activation products in the pool water. This baseline varies little from day-to-day except when samples are being handled in the pool. Antimony-124 is not a routine isotope found in the pool water based on the weekly sampling analysis. A sample breach of the antimony-beryllium source with as little as 1% of the source leaking into the pool would cause a dose rate increase of around 33%. This would result in an activity concentration of 1.32E-2 μCi/ml in the pool water, well above the minimum detectable

activity (MDA) for gamma spectral analysis of pool water. An increase of this magnitude over the normal baseline activity in the pool would be readily observable to the control room staff via the above pool radiation monitors and would raise questions among the staff as to the cause of the increase. Once solid samples and other sources in the pool have been ruled out as a cause, the Reactor Manager would request that a pool water sample be taken and analyzed. Once the analysis is completed, the presence of antimony-124 would be confirmed and steps would be taken to mitigate the situation.

CHAPTER 16, OTHER LICENSE CONSIDERATIONS

16.2 *Section 16.1.2, Primary Coolant System Pressure Boundary. Your estimates in this section are based on a license term until 2026. Please update assuming a license term of 2030.*

SAR reference 16.3, "Assessment of Aluminum Structural Materials for Service within the ANS Reactor Vessel," provides 4 x 10^{27} n/m^2 as the highest documented thermal neutron fluence for irradiated aluminum 6061-T6 for which ultimate and yield strength measurements have been performed. Both the ultimate and yield strength measurements indicate that the strength of aluminum 6061-T6 increases with fluence. The SAR statement assumed operation through October 2026. Assuming that the current operating schedule continues through December 2030, the peak thermal fluence on the pressure vessels will be 3.19 x 10^{27} n/m^2, well within the measured envelope for aluminum 6061-T6. The reactor would need to operate an additional 14 years (until December 2044) to reach the maximum thermal fluence of 4 x 10^{27} n/m^2 for aluminum 6061-T6.

With 105,500 MWD of operation, the peak thermal flux area of the aluminum 6061-T6 inner reactor pressure vessel has transmuted to 4.4 wt% silicon. The total swelling is the sum contribution of both void formation and silicon production. Total neutron flux will cause a peak swelling of around 2.7% by the end of 2030.

The maximum cyclic stress for the MURR reactor pressure vessels is the transition between being pressurized and depressurized as part of starting up and shutting down the primary coolant system. If it were assumed that this occurs 200 times per year (more typical is around 70-80 times) there would be 12,850 cycles over a 64 ¼ year operating period. As the 1,160 psi (8 MPa) typical cycle remains well below the infinite fatigue stress of 7,251 psi (50 MPa), these additional cycles pose no limit on the service life of the inner and outer reactor pressure vessels.

APPENDIX A, TECHNICAL SPECIFICATIONS

A.2 *Definition 1.2, Calibration or Testing Interval. The intervals given in this definition are for operational flexibility and are not long term frequencies. Please revise the definition to clarify this or justify the use of the proposed frequencies for the long-term.*

MURR feels that Technical Specification 1.2 is consistent with ANSI/ANS-15.1-2007. This definition is for the maximum interval between checks, not the scheduling requirements for surveillance. MURR defines the periodicity of surveillance requirements in specific procedures, for example in IC-HP-349, "Calibration-Lab Impex Stack Monitor-Particulate Channel," semi-annual is defined as *Semi-Annual: calibration performed every six months with a maximum allowed interval of eight months.*

A.10 *Definitions 1.25, Regulating Blade (Rod), and 1.30, Shim Blade (Rod). Please clarify if the control rods have in-run capability referred to in TS 3.4. c.*

As defined by Technical Specification 1.4, a control blade is either a shim blade or the regulating blade. Section 7.5, Rod Control System, provides a detailed explanation on the operation of the control blades. The Rod Run-In System is discussed in Section 7.5.5, Rod Run-In System. In this section, the term control blade refers to the four high worth shim blades and not the regulating blade. All four shim blades will automatically insert upon activation of the Rod Run-In System. Activation of the system also acts to remove the reactor from automatic control causing the regulating blade position to remain constant.

A.17 *TS 3.2.b requires that above 100 kW, the maximum distance between the highest and lowest shim blade is one inch. However, there is no Surveillance Requirement for this LCO. Propose an appropriate TS Surveillance Requirement for TS 3.2.b or justify why one is not needed.*

MURR is requesting that this question be included in the set of "complex" questions (120-day response) since it is very similar to question 4.14.d. This would allow a combined review of both answers.

A.27 *TS 3.5 a. (2), Reactor Containment Building. Please provide a technical basis for not needing containment integrity when handling fuel with a decay time of greater than sixty days.*

Below is our analysis that confirms that containment integrity is not required when handling fuel with a decay time of greater than 60 days. The analysis derives the potential dose from a fuel handling incident, with an element that has decayed for 60 days, and shows that this dose would be less than 1% of the annual limit for a radiation worker.

Many types of accidents have been evaluated in conjunction with the operation of the MURR. The calculations to determine the postulated dose from a potential release from a fuel element handling incident mirror the Maximum Hypothetical Accident (MHA) calculations for personnel exposure due to a release of fission products as presented in Section 13.2.1 of the SAR. The objective of these calculations is to present a worst-case dose assessment for a person who remains in the containment building for one hour following the release from a fuel element.

The two outer fuel plates, number-1 and -24, of a fuel element are the plates most likely to be damaged during fuel handling. The number-1 fuel plate has the peak power density of the core and contains [redacted] grams of uranium-235 (^{235}U). The number-24 fuel plate has a peak power density less than the number-1 plate and contains [redacted] grams of ^{235}U. To be conservative, the analysis assumes that [redacted] grams of ^{235}U is exposed during the fuel handling incident, which corresponds to removing a section of fuel meat from a plate that is one inch square and 5 mils thick. A power peaking factor of 3.0 is also conservatively assumed.

The release of radioisotopes of krypton, xenon and iodine are the major sources of radiation exposure to personnel and will, therefore, serve as the basis for the source term for this dose calculation. For operation at 10 MW for 1,200 MWD in twelve 10-day cycles over a 300-day period with [redacted] (normal operating cycle is 6.5 days with a total of less than 700 MWD on the core), the following radioiodine, krypton and xenon activities per gram of ^{235}U will be present in a fuel element with 60 days of decay. Note: All other short-lived iodines and noble gases used in the MHA analysis have decayed to background levels and thus were not used in this analysis.

Radioiodine and Noble Gas Activities in the Core
(in curies per gram of ^{235}U)

^{129}I – 1.35×10^{-7} Ci/gm ^{85}Kr – 6.99×10^{-2} Ci/gm ^{133}Xe – 2.26×10^{-2} Ci/gm
^{131}I – 1.43×10^{-1} Ci/gm

The radioiodine released into the reactor pool is conservatively assumed to be instantly and uniformly mixed into the 20,000 gallons (75,708 l) of bulk pool water, which then results in the following pool water concentrations for the iodine isotopes. The krypton and xenon noble gases released into the reactor pool are assumed to pass immediately through the pool water and evolve directly into the containment building air volume where they instantaneously form a uniform concentration in the isolated structure.

Calculation of ^{129}I and ^{131}I concentrations in the reactor pool water:

= ^{129}I activity in fuel (Ci/gm of ^{235}U) x (grams of ^{235}U exposed) x Power Peaking Factor x 1/20,000 gal x 1,000 mCi/Ci
= 1.35×10^{-7} Ci/gm x 0.125 gm x 3.0 x 1/20,000 gal x 1,000 mCi/Ci
= 2.54×10^{-9} mCi/gal

= ^{131}I activity in fuel (Ci/gm of ^{235}U) x (grams of ^{235}U exposed) x Power Peaking Factor x 1/20,000 gal x 1,000 mCi/Ci
= 1.43×10^{-1} Ci/gm x 0.125 gm x 3.0 x 1/20,000 gal x 1,000 mCi/Ci
= 2.69×10^{-3} mCi/gal

Calculation of ^{85}Kr and ^{133}Xe activities released into the reactor pool water:

= ^{85}Kr activity in fuel (Ci/gm of ^{235}U) x (grams of ^{235}U exposed) x Power Peaking Factor x 1,000 mCi/Ci
= 6.99×10^{-2} Ci/gm x 0.125 gm x 3.0 x 1,000 mCi/Ci
= 26.2 mCi

= ^{133}Xe activity in fuel (Ci/gm of ^{235}U) x (grams of ^{235}U exposed) x Power Peaking Factor x 1,000 mCi/Ci
= 2.26×10^{-2} Ci/gm x 0.125 gm x 3.0 x 1,000 mCi/Ci
= 8.48 mCi

When the reactor is at 10 MW and the containment building ventilation system is in operation, the evaporation rate from the reactor pool is approximately 80 gallons (303 l) of water per day. For the purposes of this calculation, it is assumed that a total of 40 gallons (151 l) of pool water containing the previously listed radioiodine concentrations evaporates instantaneously into the containment building. Containment air with a temperature of 75 °F (25 °C) and 100% relative humidity contains H_2O vapor equal to 40 gallons (151 l) of water. Since the air in containment is normally at about 50% relative humidity, the assumption that 40 gallons (151 l) of pool water evaporating overestimates by a factor of two the quantity of evaporated pool water in containment air under typical conditions. It is also conservatively assumed that all of the iodine activity in the 40 gallons (151 l) of pool water is released into containment and instantaneously forms a uniform concentration in the containment building air. When distributed into the containment building, this would result in the following radioiodine concentrations in the 225,000 ft^3 air volume:

Calculation of ^{129}I and ^{131}I concentrations in containment air:

= ^{129}I concentration in pool water x 40 gal x 1/225,000 ft^3 x 35.3147 ft^3/ m^3
= 2.54×10^{-9} mCi/gal x 6.28×10^{-3} gal/m^3
= 1.60×10^{-11} mCi/m^3
= 1.60×10^{-8} µCi/m^3

$$(1.60 \times 10^{-8}\ \mu Ci/m^3) \times (1\ m^3/10^6\ ml) = 1.60 \times 10^{-14}\ \mu Ci/ml$$

= ^{131}I concentration in pool water x 40 gal x 1/225,000 ft^3 x 35.3147 ft^3/ m^3
= 2.69×10^{-3} mCi/gal x 6.28×10^{-3} gal/m^3
= 1.69×10^{-5} mCi/m^3
= 1.69×10^{-2} µCi/m^3

$$(1.69 \times 10^{-2}\ \mu Ci/m^3) \times (1\ m^3/10^6\ ml) = 1.69 \times 10^{-8}\ \mu Ci/ml$$

The krypton and xenon noble gases released into the reactor pool are assumed to pass immediately through the pool water and enter the containment building air volume where they instantaneously form a uniform concentration in containment. This assumption is conservative since it ignores the known solubility of krypton and xenon noble gases in the 100 °F pool water, which would reduce their release into the containment building. Based on the 225,000 ft^3 volume of containment building air and the previously listed millicurie quantities of these gases released into the reactor pool, the maximum noble gas concentrations in the containment building would be as follows:

Calculation of ^{85}Kr and ^{133}Xe concentrations in containment air:

= ^{85}Kr activity in the pool x 1/225,000 ft^3 x 35.3147 ft^3/m3 x 1,000 µCi/mCi
= 26.2 mCi x 1.57×10^{-1} µCi/mCi-m^3
= 4.11 µCi /m^3

$$(4.11\ \mu Ci/m^3) \times (1\ m^3/10^6\ ml) = 4.11 \times 10^{-6}\ \mu Ci/ml$$

= ^{133}Xe activity in the pool x 1/225,000 ft^3 x 35.3147 ft^3/m^3 x 1,000 µCi/mCi
= 8.48 mCi x 1.57×10^{-1} µCi/mCi-m^3
= 1.33 µCi /m^3

$$(1.33\ \mu Ci/m^3) \times (1\ m^3/10^6\ ml) = 1.33 \times 10^{-6}\ \mu Ci/ml$$

Based on the source term data provided, it is possible to determine the radiation dose to the thyroid from radioiodine and the dose to the whole body resulting from submersion in the airborne noble gases and radioiodine inside the containment building. As previously noted, the exposure time for this dose assessment is one hour. These values are given below for the radioiodines along with the applicable radioiodine dose conversion factors used to calculate the dose.

Radioiodine Concentrations in the Containment Building Air
During the One Hour Following the Release
(in microcuries per milliliter)

^{129}I - 1.60×10^{-14} µCi/ml ^{131}I - 1.69×10^{-8} µCi/ml

Committed Dose Equivalent Per Unit Intake to the Thyroid (SAR Ref. 13.36)
(in Sieverts per Becquerel)

^{129}I – 1.56×10^{-6} Sv/Bq ^{131}I - 2.92×10^{-7} Sv/Bq

Air Submersion Dose Equivalent H_T to the Whole Body (SAR Ref. 13.37)
(in Sieverts-m^3 per Becquerel-sec)

^{129}I - 3.80×10^{-16} Sv-m^3/Bq-sec ^{131}I - 1.82×10^{-14} Sv-m^3/Bq-sec

Sv	=	Sieverts,
Bq	=	Becquerel,
Curie (Ci)	=	3.7×10^{10} Bq,
Microcurie (μCi)	=	3.7×10^{4} Bq; and
Breathing Rate	=	3.3×10^{-4} m^3/sec (SAR Ref. 13.38)

Since the airborne radioiodine source is composed of two different iodine isotopes, it will be necessary to determine the dose contribution from each individual isotope and to then sum the results. The calculation of the doses from inhalation and submersion for ^{129}I and ^{131}I is shown below. The results of these calculations are then summed to show the total iodine dose to an individual who remains in the containment building for one hour after the fuel handling incident occurs.

Calculation of thyroid and whole body doses from ^{129}I and ^{131}I:

from inhalation:
Iodine Air Concentration x Time x Breathing Rate
^{129}I (1.60×10^{-8} μCi/m^3) x 3600 sec x (3.3×10^{-4} m^3/sec) x (3.7×10^{4} Bq/μCi) = 7.03×10^{-4} Bq
^{131}I (1.69×10^{-2} μCi/m^3) x 3600 sec x (3.3×10^{-4} m^3/sec) x (3.7×10^{4} Bq/μCi) = 743 Bq

Thyroid Dose -
^{129}I (7.03×10^{-4} Bq) x (1.56×10^{-6} Sv/Bq) = 1.10×10^{-9} Sv = 1.10×10^{-4} mrem
^{131}I (743 Bq) x (2.92×10^{-7} Sv/Bq) = 2.17×10^{-4} Sv = 21.7 mrem

from submersion:
^{129}I (1.60×10^{-8} μCi/m^3) x (3.7×10^{4} Bq/μCi) x 3600 sec = 2.13 Bq-sec/m^3
^{131}I (1.69×10^{-2} μCi/m^3) x (3.7×10^{4} Bq/μCi) x 3600 sec = 2.25×10^{6} Bq-sec/m^3

Whole Body Dose -
^{129}I (2.13 Bq-sec/m^3) x (3.80×10^{-16} Sv-m^3/Bq-sec) = 8.09×10^{-16} Sv = 8.09×10^{-11} mrem
^{131}I (2.25×10^{6} Bq-sec/m^3) x (1.82×10^{-14} Sv-m^3/Bq-sec) = 4.10×10^{-8} Sv = 4.10×10^{-3} mrem

A tabulation of the results of the calculations for the two different iodine isotopes along with a total dose for the thyroid and whole body resulting from inhalation and submersion is shown below. These totals are then expressed as current dose quantities where:

CDE	=	Committed Dose Equivalent
CEDE	=	Committed Effective Dose Equivalent
DDE	=	Deep Dose Equivalent
TEDE	=	Total Effective Dose Equivalent

Dose to the Thyroid from Inhalation
(in mrem)

^{129}I	=	1.10×10^{-4} mrem
^{131}I	=	21.7 mrem
Total Thyroid Dose	=	21.7 mrem

Dose to the Whole Body from Submersion
(in mrem)

^{129}I	=	8.09×10^{-11} mrem
^{131}I	=	4.10×10^{-3} mrem
Total Submersion Dose	=	0.00 mrem

By converting these totals to current dose quantities, where rem = 10^{-2} Sv and millirem = 10^{-5} Sv, the following values can be derived and will represent the dose from radioiodine to an individual remaining inside the MURR containment building for 60 minutes after the release of fission products from a fuel element.

60-Minute Dose from Radioiodines in Containment

CDE (thyroid)	=	21.7 mrem
CEDE (thyroid)	=	0.651 mrem
DDE (whole body)	=	0.00 mrem
TEDE (whole body)	=	0.651 mrem

Dose from the kryptons and xenons that are present in the containment building is assessed in much the same manner as the iodines, and the dose contribution from each individual radionuclide is calculated and then added together to arrive at the final noble gas dose. Since the dose from the noble gases is only an external dose due to submersion, and since the Derived Air Concentrations (DACs) for these radionuclides are based on this type of exposure, the individual noble gas doses for 60 minutes in containment are based on their average concentration in the containment air and the corresponding DAC value in Appendix B of 10 CFR 20. The average noble gas concentrations in the containment building during the 60-minute period following the release of fission products from a fuel element and the corresponding doses for a 60-minute occupancy are given below.

Average noble gas concentrations in the containment building air during the 60-minute period following the fuel handling incident	Noble gas doses for a 60-minute containment occupancy following the fuel handling incident
^{85}Kr – $\frac{4.11 \times 10^{-6}\ \mu Ci/ml \times 1\ hr \times 5000\ mrem}{(1E\text{-}4\ \mu Ci/ml \times 2000\ hr)}$	0.103 mrem
^{133}Xe – $\frac{1.33 \times 10^{-6}\ \mu Ci/ml \times 1\ hr \times 5000\ mrem}{(1E\text{-}4\ \mu Ci/ml \times 2000\ hr)}$	0.033 mrem
Total Deep Dose Equivalent (DDE) Whole Body for 60-minute exposure to Noble Gases in the containment building following the fuel handling incident	0.136 mrem

To finalize the occupational dose in terms of TEDE for a 60-minute exposure in the containment building after the decayed fuel element breach, the doses from the radioiodines and noble gases are added together, and result in the following values:

60-Minute Dose from Radioiodines and Noble Gases in Containment

CDE (thyroid)	=	21.7 mrem
CEDE (thyroid)	=	0.65 mrem
DDE (radioiodines)	=	0.00 mrem
DDE (noble gases)	=	0.14 mrem
TEDE (whole body)	=	0.79 mrem

However, comparison of the maximum TEDE and CDE for those occupationally-exposed during this release to applicable NRC dose limits in 10 CFR 20 shows that the final values are well within the published regulatory limit and, in fact, less than 1% of any annual occupational limit.

A.37 *TS 3.9 b. (1), Reactor Coolant System. Please discuss the action set point for the fuel element failure monitor. Please add the set point to the TSs or justify why the set point is not needed in the TSs. Please discuss the basis for the four-hour interval between analyses.*

The Fuel Element Failure Monitor alarm set point is currently set at 12,000 cpm. The alarm set point corresponds to an iodine-131 concentration in the primary coolant system of approximately 1E-4 μCi/ml. This monitor typically indicates a count rate of around 2,000 cpm with a corresponding primary coolant water analysis of approximately 1.5-1.7E-5 μCi/ml of iodine-131. A primary coolant sample would be taken and analyzed if a significant increase in count rate is detected or if an alarm occurs. The alarm set point corresponds to an iodine-131 activity of approximately 2% of the Technical Specification (TS) limit of 5E-3 μCi/ml (TS 3.9.c). Therefore, as answered in question A.38, TS 3.9.c provides the appropriate TS for the maximum iodine-131 concentration allowed in the primary coolant system. TS 3.9.b requires monitoring the radioiodine level in the primary coolant system to an increasing iodine-131 activity well before reaching the limit. If the Fuel Element Failure Monitor is temporarily inoperable, monitoring the iodine-131 level every four hours by sampling the primary coolant system will provide a sufficient indication of a developing problem well before reaching the limit set by TS 3.9.c. Therefore, with TS 3.9.c establishing the set point and TS 3.9.b requiring monitoring the activity level well before the action level, MURR feels that no additional set point is necessary for this TS.

A.39 *TS 4.0, Surveillance Requirements. There appears to be no requirement for surveillance testing on any TS required system after replacement, repair or modification to declare the system operable and returned to service. Please propose and justify a TS or justify why such a specification is not needed.*

The requirement for surveillance testing of any Technical Specification required system after replacement, repair or modification to declare the system operable and returned to service is included in TS 6.2.c. TS 6.2.c states that "*Any additions, modifications or maintenance to the systems described in these Specifications shall be made and tested in accordance with the specifications to which the system was originally designed and fabricated or to specifications approved by the U.S. Nuclear Regulatory Commission (NRC).*"

A.41 *TS 4.0, Surveillance Requirements. ANS-15.1 suggests a surveillance requirement for thermal power verification. Provide appropriate TS wording to incorporate a surveillance requirement for thermal power verification or justify why it is not needed.*

Thermal power verification is addressed in Section 6.7.2 of MURR administrative procedure AP-RO-110, "Conduct of Operation." Two types of manual heat balances are performed: a secondary system heat balance that is calculated hourly and a primary/pool system heat balance that is calculated daily. After the heat balances are calculated, they are recorded on form FM-43, "Nuclear and Process Data." All inputs into the manual heat balance calculations are calibrated as required by TS 4.4.a and the MURR Preventative Maintenance (PM) program.

MURR feels that this is sufficient for thermal power verification and that no Technical Specification is required.

A.45 *TS 4.0, Surveillance Requirements. TS 3.9.a (3) requires the operability of the incore convection coolant system. However, there is no surveillance requirement for this LCO. Propose an appropriate TS surveillance requirement for TS 3.9.a (3) or justify why no TS is needed.*

The requirement for operability of the In-Pool Convective Cooling System during reactor operation (TS 3.9.a) is verified weekly during shutdown of the primary coolant system. In the course of performing MURR operating procedure OP-RO-410, "Primary Coolant System," step 6.1.5 specifically states to verify operation of the In-Pool Heat Exchanger Automatic Isolation Valves 546A and 546B after the primary coolant circulation pumps are secured. As described in Section 5.8.4, Description of Operation, these valves will automatically open on a loss of primary coolant flow, primary loop isolation or a loss of electrical power. In this case, securing the primary coolant circulation pumps creates a loss of flow condition which causes these valves to automatically open. Successful completion of this procedure is documented in the Reactor Console Log.

Additionally, the In-Pool Heat Exchanger Automatic Isolation Valves are also tested during performance of compliance procedures CP-4A, "Primary Flow Transmitter FT-912A Calibration and Scram Set Point Check of Flow Alarm 920A," and CP-4D, "Primary Flow Transmitter FT-912H Calibration and Scram Set Point Check of Flow Alarm 920G." A simulated test pressure is applied to the pressure transmitter and lowered until the associated scram set point is reached and valves 546A and 546B open. These compliance procedures (Technical Specification surveillance procedures) are completed semi-annually staggered three months apart which achieves an additional quarterly verification of operability in addition to the weekly procedural check.

MURR feels that these checks are sufficient to ensure operability of the system and that no Technical Specification is required.

APPENDIX B

B.2 *Section B.6 compares the calculated offsite dose to the annual average radiation dose of 360 mrem for the US population as reported in NCRP Report 94. This report has been superseded by NCRP Report 160 (2009). Further, the 360 mrem annual dose includes several components that are regional in nature (i.e., radon dose from granitic formations) and may not be appropriate for comparison with MURR conditions. Justify the use of 360 mrem/yr for comparison purposes or provide other comparison data.*

The reference noted above and Reference B.7 in the SAR should be changed to NCRP Report No. 93, "Ionizing Radiation Exposure of the Population of the United States," September, 1987.

The use of 360 mrem as noted in NCRP Report No. 93 is consistent with the timeframe that the SAR was developed and was only used as a comparison to the average background doses to the US population for illustrative purposes. If the dose estimates provided in NCRP Report No. 160 were used, only the percentage of the dose received by the receptors at 150 and 760 meters from the operation of the reactor, in comparison to background doses would change, with the ratio between the two doses actually being lower.

Additionally, the use of the average dose from radon is in fact a more likely representation of central Missouri, as Columbia is not located in an area known for higher than average radon levels.

APPENDIX C

C.1 *Section C.1, Introduction. Please describe any benchmarking that has been performed on the RELAP model and conclusions as to the accuracy of the model's results.*

We are requesting that this question be included in the set of "complex" questions (120-day response). This would combine the description of the benchmarking for Section C.1 with the benchmarking discussion for Chapter 13 analyses.