UNIVERSITY *of* MISSOURI

RESEARCH REACTOR CENTER

October 29, 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 the response to the "University of Missouri at Columbia – Request for Additional Information Re: License Renewal, Safety Analysis Report, Complex Questions (TAC No. MD3034)," dated May 6, 2010

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

On May 6, 2010, the NRC requested additional information and clarification regarding the renewal request in the form of nineteen (19) complex questions. By letter dated September 3, 2010, MURR responded to seven (7) of those questions.

On September 1, 2010, via electronic mail (email), MURR requested additional time to respond to the remaining twelve (12) questions. By letter dated September 27, 2010, the NRC granted the request. MURR's responses to two (2) of the remaining questions are attached. Answers to the remainder of the questions will be submitted by November 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,

Ralph A. Butler, P.E.
Director

Margee P. Stout

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

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

Margee P. Stout
2019 Research Reactor Ctr.

13.9 Section 13.2.6, Experiment Malfunction.

a. *The release path for the failure of fueled experiments is different than the MHA. The containment of fission products afforded by the primary system is not present for the failure of a fueled experiment in the reactor pool. Please provide dose calculations for the failure of a fueled experiment in the pool.*

The following fueled experiment failure analysis was performed in support of Amendment No. 34 to Amended Facility License R-103. Amendment No. 34, which was approved by the NRC on October 10, 2008, was required in order to perform a fueled experiment in support of a U.S. Department of Energy (DOE) program to demonstrate the feasibility of producing fission product molybdenum-99 (Mo-99) using low-enriched uranium (LEU) foil targets. The total iodine inventory in the failed experiment is essentially at the Technical Specification limit of 150 curies (TS 3.6.a).

The release of the radioisotopes of krypton, xenon and iodine from a 5-gram LEU target is the major source of radiation exposure to an individual and will, therefore, serve as the basis for the source term for the dose calculations. A 5-gram LEU target irradiated for 150 hours (normal weekly operating cycle) at a thermal neutron flux of $1.5 \times 10^{+13}$ n/cm^2-sec will produce the following radioiodine, krypton and xenon activities (additionally, approximately $1.40 \times 10^{+04}$ μCi of Strontium-90 will be produced):

Radioiodine and Noble Gas Activities in a 5-Gram LEU Target
(in curies)

^{131}I – 6.755 Ci	^{85}Kr – 0.002 Ci	^{133}Xe – 18.925 Ci
^{132}I – 18.635 Ci	^{85m}Kr – 7.580 Ci	^{135}Xe – 13.630 Ci
^{133}I – 39.875 Ci	^{87}Kr – 15.405 Ci	^{135m}Xe – 6.760 Ci
^{134}I – 45.405 Ci	^{88}Kr – 21.660 Ci	^{137}Xe – 35.800 Ci
^{135}I – 37.695 Ci	^{89}Kr – 27.740 Ci	^{138}Xe – 37.380 Ci
	^{90}Kr – 27.410 Ci	^{139}Xe – 30.675 Ci
Total Iodine – 148.365 Ci	**Total Krypton – 99.797 Ci**	**Total Xenon – 143.170 Ci**

A complete failure of the target is unrealistic for many reasons. The worst that can be expected is partial melting; however, in order to present a worst-case dose assessment for an individual that remains in the containment building following target failure, 100% of the total activity of the target is assumed to be released into the reactor pool.

Fission products released into the pool will more than likely be detected by the pool surface and ventilation system exhaust plenum radiation monitors. However, for the purposes of this analysis, it is assumed that a reactor scram and actuation of the containment building isolation system occurs by action of the pool surface radiation monitor. Actuation of the isolation system will prompt Operations personnel to ensure that a total evacuation of the containment building is accomplished within two (2) minutes. The 2-minute evacuation time is used as the basis for the stay time in the dose calculations for personnel that are in containment during target failure.

The radioiodine released into the reactor pool over a 2-minute interval 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 pool over this same time period 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.

Radioiodine Concentrations in the Pool Water
(in microcuries per gallon)

^{131}I – 338 µCi/gal	^{133}I – 1,995 µCi/gal	^{135}I – 1,885 µCi/gal
^{132}I – 930 µCi/gal	^{134}I – 2,270 µCi/gal	

When the reactor is at 10 MW with the containment building ventilation system in operation, the evaporation rate from the reactor pool is approximately 80 gallons (303 l) of water per day. However, for the purposes of this analysis, the assumption is that a total of 40 gallons (151 l) of pool water containing the previously listed radioiodine concentrations evaporates over 2 minutes into the isolated containment structure. In fact only about 0.11 gallons (0.42 l) of pool water would evaporate during this time period; therefore, the assumption that 40 gallons will evaporate results in greater than three hundred and sixty (360) times more airborne radioiodine activity in the containment building air than would actually be present at the end of 2 minutes of evaporation. It is also conservatively assumed that all of the iodine activity in the 40 gallons (151 l) of pool water, which was assumed to evaporate over 2 minutes, is released into containment and instantaneously forms a uniform concentration in the containment building air. When distributed into the containment structure, this would result in the following radioiodine concentrations in the 225,000-ft^3 air volume:

Example calculation of ^{131}I released into containment air:

= ^{131}I concentration in pool water x 40 gal x 1/225,000 ft^3 x 35.3147 ft^3/m^3
= 338 µCi/gal x (6.28 x 10^{-03} gal/m^3)
= 2.12 µCi/m^3

$$(2.12\ \mu Ci/m^3) \times (1\ m^3/10^6\ ml) = 2.12 \times 10^{-06}\ \mu Ci/ml$$

Note: Same calculation is used for the other isotopes listed below.

Radioiodine Concentrations in the Containment Building Air After 2 Minutes
(in microcuries per milliliter)

^{131}I – 2.12 x 10^{-06} µCi/ml	^{133}I – 1.25 x 10^{-05} µCi/ml	^{135}I – 1.18 x 10^{-05} µCi/ml
^{132}I – 5.85 x 10^{-06} µCi/ml	^{134}I – 1.43 x 10^{-05} µCi/ml	

As noted previously, the krypton and xenon noble gases released into the reactor pool from the 5-gram LEU target during the 2-minute interval following failure, are assumed to pass immediately through the pool water and enter the containment building air volume where they instantaneously form a uniform concentration in the isolated structure. Based on the 225,000-ft^3 volume of containment building air, and the previously listed curie quantities of these gases released into the reactor pool, the maximum noble gas concentrations in the containment structure at the end of 2 minutes would be as follows:

Example calculation of ^{85}Kr released into containment air:

= ^{85}Kr activity x 1/225,000 ft^3 x 35.3147 ft^3/m^3 x 1,000 µCi/mCi
= 1.71 mCi x (1.57 x 10^{-01} µCi/mCi-m^3)

$= 2.68 \times 10^{-01}\ \mu Ci/m^3$

$$(2.68 \times 10^{-01}\ \mu Ci/m^3) \times (1\ m^3/10^6\ ml) = 2.67 \times 10^{-07}\ \mu Ci/ml$$

Note: Same calculation is used for the other isotopes listed below.

Noble Gas Concentrations in the Containment Building Air after 2 Minutes
(in microcuries per milliliter)

^{85}Kr – 2.67×10^{-07} µCi/ml	^{133}Xe – 2.97×10^{-03} µCi/ml
^{85m}Kr – 1.19×10^{-03} µCi/ml	^{135}Xe – 2.14×10^{-03} µCi/ml
^{87}Kr – 2.42×10^{-03} µCi/ml	^{135m}Xe – 1.06×10^{-03} µCi/ml
^{88}Kr – 3.40×10^{-03} µCi/ml	^{137}Xe – 5.62×10^{-03} µCi/ml
^{89}Kr – 4.35×10^{-03} µCi/ml	^{138}Xe – 5.87×10^{-03} µCi/ml
^{90}Kr – 4.30×10^{-03} µCi/ml	^{139}Xe – 4.82×10^{-03} µCi/ml

The objective of this calculation is to present a worst-case dose assessment for an individual who remains in the containment building for 2 minutes following target failure. Therefore, as noted previously, the radioactivity in the evaporated pool water is assumed to be instantaneously and uniformly distributed into the building once released into the air.

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 2 minutes. Note: Because the release rate of fission products from the failed target over a 2-minute period is difficult to establish, the maximum concentrations stated above would probably not occur until the end of the 2-minute interval, if not later. However, for the purposes of the dose calculations, the above stated maximum concentrations are conservatively used.

Because the airborne radioiodine source is composed of five different iodine isotopes, it will be necessary to determine the dose contribution from each individual isotope and to then sum the results. Dose multiplication factors were established using the Derived Air Concentrations (DACs) listed in Appendix B of 10 CFR 20 and the radionuclide concentrations in the containment building.

Example calculation of thyroid dose due to ^{131}I:

The DAC can also be defined as 50,000 mrem (thyroid target organ limit)/2,000 hrs, or 25 mrem/DAC-hr. Additionally, 2 minutes of one DAC-hr is 3.33×10^{-02} DAC-hr.

^{131}I concentration in containment	=	2.12×10^{-06} µCi/ml
^{131}I DAC (10 CFR 20)	=	2.00×10^{-08} µCi/ml
Dose Multiplication Factor	=	(^{131}I concentration) / (^{131}I DAC)
	=	(2.12×10^{-06} µCi/ml) / (2.00×10^{-08} µCi/ml)
	=	106

Therefore, a 2 minute thyroid exposure from ^{131}I is:

= Dose Multiplication Factor x DAC Dose Rate x 2 minutes
= 106 x (25 mrem/DAC-hr) x (3.33×10^{-02} DAC-hr)

$= 8.75 \times 10^{+01}$ mrem

Note: Same calculation is used for the other radioiodines listed below.

Doses from the kryptons and xenons present in the containment building are assessed in much the same manner as the iodines, and the dose contribution from each individual radionuclide must be calculated and then added together to arrive at the final noble gas dose. Because the dose from the noble gases is only an external dose due to submersion, and because the DACs for these radionuclides are based on this type of exposure, the individual noble gas doses for 2 minutes in containment were based on their maximum concentration in the containment air and the corresponding DAC.

Example calculation of whole body dose due to ^{85}Kr:

The DAC can also be defined as 5,000 mrem/2,000 hrs, or 2.5 mrem/DAC-hr. Additionally, 2 minutes of one DAC-hr is 3.33×10^{-02} DAC-hr.

^{85}Kr concentration in containment	=	2.67×10^{-07} µCi/ml
^{85}Kr DAC (10 CFR 20)	=	1.00×10^{-04} µCi/ml
Dose Multiplication Factor	=	(^{85}Kr concentration) / (^{85}Kr DAC)
	=	(2.67×10^{-07} µCi/ml) / (1.00×10^{-04} µCi/ml)
	=	0.00267

Therefore, a 2 minute whole body exposure from ^{85}Kr is:

= Dose Multiplication Factor x DAC Dose Rate x 2 minutes
= 0.00267 x (2.5 mrem/DAC-hr) x (3.33×10^{-02} DAC-hr)
= 2.20×10^{-04} mrem

Note: Same calculation is used for the other noble gases listed below.

The DACs and the 2-minute exposure for each radioiodine and noble gas are tabulated below.

Part 20 Derived Air Concentration Values and Two-Minute Exposures – Radioiodine
(in microcuries per milliliter and millirem)

Radionuclide	Derived Air Concentration	Two-Minute Exposure
^{131}I	2.00×10^{-08} µCi/ml	$8.75 \times 10^{+01}$ mrem
^{132}I	3.00×10^{-06} µCi/ml	$1.61 \times 10^{+00}$ mrem
^{133}I	1.00×10^{-07} µCi/ml	$1.03 \times 10^{+02}$ mrem
^{134}I	2.00×10^{-05} µCi/ml	5.88×10^{-01} mrem
^{135}I	7.00×10^{-07} µCi/ml	$1.39 \times 10^{+01}$ mrem
		Total = 206.92 mrem

Part 20 Derived Air Concentration Values and Two-Minute Exposures – Noble Gases
(in microcuries per milliliter and millirem)

Radionuclide	Derived Air Concentration	Two-Minute Exposure
^{85}Kr	1.00×10^{-04} µCi/ml	2.20×10^{-04} mrem
^{85m}Kr	2.00×10^{-05} µCi/ml	$4.91 \times 10^{+00}$ mrem
^{87}Kr	5.00×10^{-06} µCi/ml	$3.99 \times 10^{+01}$ mrem

^{88}Kr	2.00×10^{-06} µCi/ml	$1.40 \times 10^{+02}$ mrem
^{89}Kr	2.00×10^{-06} µCi/ml	$1.80 \times 10^{+02}$ mrem
^{90}Kr	3.00×10^{-06} µCi/ml	$1.18 \times 10^{+02}$ mrem
^{133}Xe	1.00×10^{-04} µCi/ml	$2.45 \times 10^{+00}$ mrem
^{135}Xe	1.00×10^{-05} µCi/ml	$1.77 \times 10^{+01}$ mrem
^{135m}Xe	9.00×10^{-06} µCi/ml	$9.73 \times 10^{+00}$ mrem
^{137}Xe	2.00×10^{-05} µCi/ml	$2.32 \times 10^{+01}$ mrem
^{138}Xe	4.00×10^{-06} µCi/ml	$1.21 \times 10^{+02}$ mrem
^{139}Xe	6.00×10^{-07} µCi/ml	$6.62 \times 10^{+02}$ mrem
		Total = 1319.22 mrem

To finalize the occupational dose in terms of Total Effective Dose Equivalent (TEDE) for a 2-minute exposure in the containment building after target failure, the doses from the radioiodines and noble gases must be added together, and result in the following values:

Two-Minute Dose from Radioidines and Noble Gases in the Containment Building
(in millirem)

Committed Dose Equivalent (Thyroid)	**206.92 mrem**
Committed Effective Dose Equivalent (Thyroid)	**6.21 mrem**
Committed Effective Dose Equivalent (Noble Gases)	**1319.22 mrem**
Total Effective Dose Equivalent (Whole Body)	**1325.43 mrem**

Note: The addition of Strontium-90 (^{90}Sr) will increase the above stated TEDE (whole body) by 9.15 mrem (<1%).

By comparison of the maximum TEDE and Committed Dose Equivalent (CDE) for those occupationally-exposed during target failure to applicable NRC dose limits in 10 CFR 20, the final values are shown to be well within the published regulatory limits and, in fact, lower than 30 % of any occupational limit.

As noted earlier in this analysis, the containment building ventilation system will shut down and the building itself will be isolated from the surrounding areas. Target failure will not cause an increase in pressure inside the reactor containment structure; therefore, any air leakage from the building will occur as a result of normal changes in atmospheric pressure and pressure equilibrium between the inside of the containment structure and the outside atmosphere. It is highly probable that there will be no pressure differential between the inside of the containment building and the outside atmosphere, and consequently there will be no air leakage from the building and no radiation dose to members of the public in the unrestricted area. However, to develop what would clearly be a worst-case scenario, this analysis assumes that a barometric pressure change had occurred in conjunction with the target failure. A reasonable assumption would be a pressure change on the order of 0.7 inches of Hg (25.4 mm of Hg at 60 ℃), which would then create a pressure differential of about 0.33 psig (2.28 kPa above atmosphere) between the inside of the isolated containment building and the inside of the adjacent laboratory building, which surrounds most of the containment structure. Making the conservative assumption that the containment building will leak at the TS leakage rate limit [10% of the contained volume over a 24-hour period from an initial overpressure of 2.0 psig (13.8 kPa above atmosphere)], the air leakage from the containment structure in standard cubic feet per minute (scfm) as a function of containment pressure can be expressed by the following equation:

$$LR = 17.85 \times (CP-14.7)^{1/2};$$

where:

LR = leakage rate from containment (scfm); and
CP = containment pressure (psia).

The minimum free volume of the containment building is 225,000-ft^3 at standard temperature and pressure. At an initial overpressure of 2.0 psig (13.8 kPa above atmosphere), the containment structure would hold approximately 255,612 standard cubic feet (scf) of air. A loss of 10%, from this initial overpressure condition, would result in a decrease in air volume to 230,051 scf. The above equation describes the leakage rate that results in this drop of contained air volume over 1,440 minutes (24 hrs).

When applying the TS leakage rate equation to the assumed initial overpressure condition of 0.33 psig (2.28 kPa above atmosphere), it would take approximately 16.5 hours for the leak rate to decrease to zero from an initial leakage rate of approximately 10.3 scfm, which would occur at the start of the event. The average leakage rate over the 16.5-hour period would be about 5.2 scfm.

Several factors exist that will mitigate the radiological impact of any air leakage from the containment building following target failure. First of all, most leakage pathways from containment discharge into the reactor laboratory building, which surrounds the containment structure. Since the laboratory building ventilation system continues to operate during target failure, leakage air captured by the ventilation exhaust system is mixed with other building air, and then discharged from the facility through the exhaust stack at a rate of approximately 30,500 cfm. Mixing of containment air leakage with the laboratory building ventilation flow, followed by discharge out the exhaust stack and subsequent atmospheric dispersion, results in extremely low radionuclide concentrations and very small radiation doses in the unrestricted area. A tabulation of these concentrations and doses is given below. These values were calculated following the same methodology stated in Section 5.3.3 of Addendum 3 to the MURR Hazards Summary Report.[1]

A second factor which helps to reduce the potential radiation dose in the unrestricted area relates to the behavior of radioiodine, which has been studied extensively in the containment mockup facility at Oak Ridge National Laboratory (ORNL). From these experiments, it was shown that up to 75% of the iodine released will be deposited in the containment vessel.[2] If, due to this 75% iodine deposition in the containment building, each cubic meter of air released from containment has a radioiodine concentration that is 25% of each cubic meter within containment building air, then the radioiodine concentrations leaking from the containment structure into the laboratory building, in microcuries per milliliter, will be:

<u>Radioiodine Concentrations in Air Leaking from Containment</u>
(in microcuries per milliliter)

^{131}I – 5.30×10^{-07} μCi/ml	^{133}I – 3.13×10^{-06} μCi/ml	^{135}I – 2.95×10^{-06} μCi/ml
^{132}I – 1.46×10^{-06} μCi/ml	^{134}I – 3.58×10^{-06} μCi/ml	

Assuming, as stated earlier, that (1) the average leakage rate from the containment building is 5.2 scfm, (2) the leak continues for about 16.5 hours in order to equalize the containment building pressure with atmospheric pressure, (3) the flow rate through the facility's ventilation exhaust stack is 30,500 scfm, (4) the reduction in concentration from the point of discharge at the exhaust stack to the point of maximum concentration in the unrestricted area is a factor of 312 and (5) there is no decay of any radioiodines or noble gases, then the following average concentrations of radioiodines and noble gases with their corresponding radiation doses will occur in the unrestricted area. The values listed are for the point of maximum concentration in the unrestricted area assuming a uniform, semi-spherical cloud geometry for noble gas submersion and further assuming that the most conservative (worst-case) meteorological conditions exist for the entire 16.5-hour period of containment leakage following target failure. Radiation doses are calculated for the entire 16.5-hour period. Dose values for the unrestricted area were obtained using the same methodology that was used to determine doses inside the containment building, and it was assumed that an individual was present at the point of maximum concentration for the full 16.5 hours that the containment building was leaking.

Average Radioiodine Concentrations at the Point of Maximum Concentration in the Unrestricted Area and Corresponding Radiation Doses
(16.5-hour containment leak following target failure)

Radioiodine	Average Concentration	Radiation Dose
^{131}I	6.33×10^{-09} μCi/ml	$2.98 \times 10^{+00}$ mrem
^{132}I	1.75×10^{-08} μCi/ml	8.23×10^{-02} mrem
^{133}I	3.74×10^{-08} μCi/ml	$3.52 \times 10^{+00}$ mrem
^{134}I	4.26×10^{-08} μCi/ml	6.68×10^{-02} mrem
^{135}I	3.53×10^{-08} μCi/ml	5.55×10^{-01} mrem
		Total = 7.21 mrem

Average Noble Gas Concentrations at the Point of Maximum Concentration in the Unrestricted Area and Corresponding Radiation Doses
(16.5-hour containment leak following target failure)

Noble Gas	Average Concentration	Radiation Dose
^{85}Kr	6.37×10^{-12} μCi/ml	8.58×10^{-07} mrem
^{85m}Kr	2.84×10^{-08} μCi/ml	2.68×10^{-02} mrem
^{87}Kr	5.77×10^{-08} μCi/ml	2.72×10^{-01} mrem
^{88}Kr	8.12×10^{-08} μCi/ml	8.50×10^{-01} mrem
^{89}Kr	1.04×10^{-07} μCi/ml	4.90×10^{-03} mrem
^{90}Kr	1.03×10^{-07} μCi/ml	3.23×10^{-03} mrem
^{133}Xe	7.09×10^{-08} μCi/ml	1.34×10^{-02} mrem
^{135}Xe	5.11×10^{-08} μCi/ml	6.88×10^{-02} mrem
^{135m}Xe	2.53×10^{-08} μCi/ml	5.97×10^{-02} mrem
^{137}Xe	1.34×10^{-07} μCi/ml	6.32×10^{-04} mrem
^{138}Xe	1.40×10^{-07} μCi/ml	3.30×10^{-03} mrem
^{139}Xe	1.15×10^{-07} μCi/ml	1.81×10^{-02} mrem
		Total = 1.32 mrem

Doses in the Unrestricted Area Due to Radioiodine and Noble Gases
(in millirem)

Committed Effective Dose Equivalent (Thyroid)	**7.21 mrem**
Committed Effective Dose Equivalent (Noble Gases)	**1.32 mrem**
Total Effective Dose Equivalent (Whole Body)	**8.53 mrem**

Summing the doses from the noble gases and the radioiodines simply substantiates earlier statements regarding the very low levels in the unrestricted area should a target failure occur, and should the containment building leak following such an event. Because the dose values are so low the overall TEDE is still only 8.53 mrem, a value below the applicable 10 CFR 20 regulatory limit for the unrestricted area. Additionally, leakage in mechanical equipment room 114 from such items as valve packing, flange gaskets, pump mechanical seals, etc. was also considered in the target failure analysis. A realistic leakage rate of 15 milliliters within the 2-minute time interval was used - after 2 minutes the pool coolant system would be shutdown and isolated as part of the control room operator's actions. The additional contaminated water vapor and associated isotopes added to the facility ventilation exhaust system made a minimal (<1%) contribution to the total dose of an individual located in the facility. Therefore, the dose contribution to the unrestricted area would be expected to be approaching zero.

REFRENCES:
[1] Hazards Summary Report, Addendum 3, Section 5.3.3, University of Missouri Research Reactor Facility, August 1972 (as revised by the 1989-1990 Operations Annual Report).
[2] Hazards Summary Report, Addendum 4, Appendix C, University of Missouri Research Reactor Facility, October 1973.

APPENDIX B

B.1 *Appendix B should provide the methodology and calculation of off-site doses at the location of the nearest residence and at the location of highest public dose. Due to the elevated nature of calculated plumes for stable Pasquill Stability Classes (E and F), the chosen location at 150 meters cannot be assured to be at the location of maximum offsite dose. The dose at the nearest residence (760 meter) is dominated by the dose from Pasquill Classes D, E, and F, but at 150 meters, the hypothetical plume has not reached ground level. Further, the methodology to weight the atmospheric dispersion conditions will not work for the non-linear calculation of dispersion if only a single receptor distance is used. Provide a revised calculation of off-site dose from Ar-41 that determines the maximum offsite dose.*

The intent of the submittal in the SAR was not to indicate the maximum dose that an individual might receive from argon-41 due to reactor operation but to indicate the maximum dose at two locations – the Emergency Planning Zone boundary and the nearest residence in relation to the facility. These two areas can reasonably be expected to be occupied on a permanent or semi-permanent basis. The Emergency Planning Zone boundary (150 meters north) is at a building occupied approximately 40 to 50 hours per week by the employees who work there. The location of the nearest resident is 760 meters due north.

The actual location of the highest calculated dose due to argon-41 emission occurs approximately 350 meters north of the facility in an unoccupied area that is under the control of the University of Missouri (within the site boundary). Near this location there are two maintenance sheds affiliated with the University of Missouri Golf Course. Using a conservative occupancy factor of 24% (40 hrs/week divided by 168 total hours per week) for this location a calculated annual dose due to argon-41 would be approximately 1.1 mrem [4.4 mrem (at 350 meters North) x 0.24]. If one were to assign occupancy factors of 24% to the occupant at the 150 meter point and 100% to the nearest resident located at the 760 meter point, to reflect the likelihood of a person being present at any one of the locations for a given amount of time, the highest likely dose anyone would receive would still be at the point of the nearest residence (760 meters north). Thus, this is where the highest offsite dose would most likely be received by a member of the public.