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April 15, 2008

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

On June 8, 2007, the University of Missouri – Columbia Research Reactor submitted a request to amend the Technical Specifications appended to Facility License R-103. Enclosed is our response to the U.S. Nuclear Regulatory Commission's request for additional information regarding the proposed amendment, dated March 19, 2008.

If you have any questions, please contact Leslie P. Foyto, the facility Reactor Manager, at (573) 882-5276.

Sincerely,

Walter Meyer for RA Butler

Ralph A. Butler, P.E.
Director

RAB/djr

Enclosures

A020
NRR

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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 Amendment on Fueled Experiment Conditions (TAC No. MD5782)," dated March 19, 2008

By letter dated June 8, 2007, the University of Missouri – Columbia Research Reactor (MURR) submitted a request to the U.S. Nuclear Regulatory Commission (NRC) to amend the Technical Specifications, which are appended to Facility License R-103, in order to perform an 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.

On August 10, 2007, the NRC requested additional information and clarification regarding the proposed amendment in the form of four (4) questions. By letter dated January 10, 2008, the MURR responded to those questions. On March 19, 2008, the NRC requested additional information and clarification regarding the proposed amendment and the responses to the initial request for additional information in the form of one (1) question. That question, and MURR's response to that question, is attached. If there are questions regarding this response, please contact me at (573) 882-5276. I declare under penalty of perjury that the foregoing is true and correct.

Sincerely,

Leslie P. Foyto
Reactor Manager

ENDORSEMENT:
Reviewed and Approved,

Walt A Meyer for R. A Butler

Ralph A. Butler, P.E.
Director

xc: Reactor Advisory Committee
Reactor Safety Subcommittee
Dr. R. Hall, Interim 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
Montgomery County
Commission #08511436

1. *Parts of your reply to questions 1 and 3 dated January 10, 2008, to our request for additional information refer to your application renewal. The NRC has not taken any action on your license renewal application. Please submit complete answers to these questions that incorporates information that you want us to consider in the evaluation of your request.*

The answers to the questions mentioned above (Questions 1 and 3 from the NRC's "request for additional information," dated August 10, 2007) have been revised, additional information to assist in the evaluation process has been added and references to MURR's license renewal application have been removed. Dose to an individual in the containment building from airborne radioiodines was recalculated using the more correct application of 50,000 mrem/yr limit for the target organ (thyroid) instead of the 5,000 mrem/yr whole body limit that was previously used. Additionally, dose to an individual in the unrestricted area increased slightly due to a re-evaluation of the underlying assumptions that are contained in the derivation of the unrestricted release limits for radionuclides, particularly iodines. This re-evaluation does not, however, affect the noble gas doses.

Please provide a detailed analysis with justification of assumptions of the radiological impact to persons in the reactor containment and members of the public from a failure of the fueled experiment irradiation container. While your proposed technical specification (TS) is based on certain iodine and strontium isotopes, please base your calculations on all isotopes that are likely to be released from the failed irradiation container.

The likelihood of a failure of a 5-gram LEU target irradiation container is remote given that its mass was designed to have a large margin of safety from the heat flux limit established for in-pool experiments irradiated in the graphite reflector region. The proposed experiment is calculated to produce a heat flux of approximately 19.5 watts/cm^2; about a factor of two below our current administrative heat flux limit of 38 watts/cm^2 for graphite reflector experiments.

We will employ our normal quality control checks, including leak testing of the seal-welded container, prior to irradiation. Additionally, we will develop an experimental plan that will document each step of the experiment beginning with the manufacture of the 5-gram LEU target itself. One element of the experimental plan will include temperature monitoring of the target, thus assuring that the requirements of TS 3.6.h and TS 3.6.n for reactor experiments are satisfied. TS 3.6.h states that "*Cooling shall be provided to prevent the surface temperature of a submerged irradiated experiment from exceeding the saturation temperature of the cooling medium,*" whereas TS 3.6.n states "*The maximum temperature of a fueled experiment shall be restricted to at least a factor of two below the melting temperature of any material in the experiment. First-of-a-kind fueled experiments shall be instrumented to measure temperature.*"

However, despite the unlikelihood that a failure of the irradiation container would occur, the following analysis provides the radiological impact to individuals in the reactor containment building and the unrestricted environment.

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 x 10^{+13} n/cm^2-sec will produce the following radioiodine, krypton and xenon activities (additionally, approximately 1.40 x 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 the reasons stated above. 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:

$$
\begin{aligned}
&= {}^{131}\text{I concentration in pool water x 40 gal x } 1/225{,}000 \text{ ft}^3 \text{ x } 35.3147 \text{ ft}^3/\text{m}^3 \\
&= 338\ \mu\text{Ci/gal x } (6.28 \text{ x } 10^{-03} \text{ gal/m}^3) \\
&= 2.12\ \mu\text{Ci/m}^3
\end{aligned}
$$

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

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:

$$
\begin{aligned}
&= {}^{85}\text{Kr activity x } 1/225{,}000 \text{ ft}^3 \text{ x } 35.3147 \text{ ft}^3/\text{m}^3 \text{ x } 1{,}000\ \mu\text{Ci/mCi} \\
&= 1.71 \text{ mCi x } (1.57 \text{ x } 10^{-01}\ \mu\text{Ci/mCi-m}^3) \\
&= 2.68 \text{ x } 10^{-01}\ \mu\text{Ci/m}^3
\end{aligned}
$$

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

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 x 10^{-07} µCi/ml	^{133}Xe – 2.97 x 10^{-03} µCi/ml
^{85m}Kr – 1.19 x 10^{-03} µCi/ml	^{135}Xe – 2.14 x 10^{-03} µCi/ml
^{87}Kr – 2.42 x 10^{-03} µCi/ml	^{135m}Xe – 1.06 x 10^{-03} µCi/ml
^{88}Kr – 3.40 x 10^{-03} µCi/ml	^{137}Xe – 5.62 x 10^{-03} µCi/ml
^{89}Kr – 4.35 x 10^{-03} µCi/ml	^{138}Xe – 5.87 x 10^{-03} µCi/ml
^{90}Kr – 4.30 x 10^{-03} µCi/ml	^{139}Xe – 4.82 x 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 x 10^{-02} DAC-hr.

^{131}I concentration in containment	=	2.12 x 10^{-06} µCi/ml
^{131}I DAC (10 CFR 20)	=	2.00 x 10^{-08} µCi/ml
Dose Multiplication Factor	=	(^{131}I concentration) / (^{131}I DAC)
	=	(2.12 x 10^{-06} µCi/ml) / (2.00 x 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 x 10^{-02} DAC-hr)
= 8.75 x 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 (TS 4.2.a). 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:

Radioiodine Concentrations in Air Leaking from Containment
(in microcuries per milliliter)

^{131}I – 5.30 x 10^{-07} μCi/ml	^{133}I – 3.13 x 10^{-06} μCi/ml	^{135}I – 2.95 x 10^{-06} μCi/ml
^{132}I – 1.46 x 10^{-06} μCi/ml	^{134}I – 3.58 x 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 x 10^{-09} μCi/ml	2.98 x 10^{+00} mrem
^{132}I	1.75 x 10^{-08} μCi/ml	8.23 x 10^{-02} mrem
^{133}I	3.74 x 10^{-08} μCi/ml	3.52 x 10^{+00} mrem
^{134}I	4.26 x 10^{-08} μCi/ml	6.68 x 10^{-02} mrem
^{135}I	3.53 x 10^{-08} μCi/ml	5.55 x 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.

Will the target processing be conducted under the reactor license? If so, describe the facilities that will be used. Describe how radioactive material will be controlled during target processing. Describe possible accident scenarios and the potential radiological impact of the scenarios.

Target processing will be conducted under condition 2.B.(3) of Facility License R-103, which allows us "*...to possess, use, but not separate except for byproduct material produced in reactor experiments, such byproduct materials as may be produced by operation of the Facility.*"

Furthermore, the experiment, including both irradiation and processing, will be evaluated under the conditions of 10 CFR 50.59. Additionally, a Reactor Utilization Request (RUR) will be prepared that will describe the experiment in considerable detail, including the activities and isotopes that are produced and the methods of handling the radioactive waste. The most important section of the RUR, and one which is given paramount consideration in its preparation, is the safety analysis. The safety analysis includes all credible accident and transient scenarios to ensure that the experiment does not jeopardize the safe operation of the reactor or constitute a hazard to the safety of the facility staff and general public.

The target processing location will provide two barriers from the release of radioactive material to the environment: the processing equipment itself and a hot cell with appropriate filtration. A processing hot cell with adequate radiation shielding will be used to perform the processing of the target to extract the Mo-99. The process itself is designed to contain the fission products within the process equipment and collect for storage and decay all airborne fission products; therefore, there should be minimal, if any, release of airborne radioactivity to the hot cell. The hot cell will be connected to the facility exhaust ventilation system so that any releases from processing will be through the ventilation system and monitored by the off-gas radiation monitoring system.

As stated above, we will install appropriate hot cell filtration in order to mitigate the release of any fission products from the hot cell to the unrestricted environment should a failure of the processing equipment occur (Note: A failure, or breach, of the hot cell is not considered credible since the fission products will be contained in the processing equipment and the hot cell will be under a negative pressure because it will be connected to the facility ventilation exhaust system.) However, for the purposes of performing worst-case dose calculations to the general public, no filtration was assumed in the following analysis.

The doses were calculated assuming a 20-hour decay of the 5-gram LEU target prior to its removal from the reactor pool and placement in the hot cell. The 20-hour decay will be administratively controlled. The following radioiodine, krypton and xenon activities will exist in the 5-gram LEU target after decay:

Radioiodine and Noble Gas Activities in a 5-Gram LEU Target After 20 Hours of Decay
(in curies)

^{131}I – 6.460 Ci	^{85}Kr – 0.002 Ci	^{133}Xe – 20.060 Ci
^{132}I – 16.000 Ci	^{85m}Kr – 0.348 Ci	^{135}Xe – 12.540 Ci
^{133}I – 21.040 Ci	^{87}Kr – $<10^{-25}$ Ci	^{135m}Xe – 0.740 Ci
^{134}I – $<10^{-04}$ Ci	^{88}Kr – 0.164 Ci	^{137}Xe – $<10^{-25}$ Ci
^{135}I – 4.630 Ci	^{89}Kr – $<10^{-25}$ Ci	^{138}Xe – $<10^{-25}$ Ci
	^{90}Kr – $<10^{-25}$ Ci	^{139}Xe – $<10^{-25}$ Ci
Total Iodine – 48.130 Ci	**Total Krypton – 0.514 Ci**	**Total Xenon – 32.205 Ci**

All noble gases were assumed to be released from the hot cell into the facility exhaust ventilation system whereas only 25% of the radioiodines were assumed to be released due to plating of the radioiodines onto the surfaces within the hot cell, dissolution apparatus and within the ventilation exhaust system itself. As explained in the answer to Question No. 1, a factor which helps to reduce the potential radiation dose in the unrestricted area relates to the behavior of radioiodine, which has been studied extensively 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].

The concentration of the radioiodines and noble gases being released through the facility exhaust stack following a target failure are tabulated below. As mentioned above, a 75% reduction factor was taken into consideration for the radioiodines.

Radioiodine Concentrations Exiting the Facility Exhaust Stack
(in microcuries per milliliter)

Radionuclide	Concentration Exiting the Exhaust Stack
^{131}I	1.30×10^{-06} µCi/ml
^{132}I	3.23×10^{-06} µCi/ml
^{133}I	4.24×10^{-06} µCi/ml
^{134}I	5.44×10^{-12} µCi/ml
^{135}I	9.33×10^{-07} µCi/ml

Noble Gas Concentrations Exiting the Facility Exhaust Stack
(in microcuries per milliliter)

Radionuclide	Concentration Exiting the Exhaust Stack	Radionuclide	Concentration Exiting the Exhaust Stack
^{85}Kr	1.37×10^{-09} µCi/ml	^{133}Xe	1.62×10^{-05} µCi/ml
^{85m}Kr	2.81×10^{-07} µCi/ml	^{135}Xe	1.01×10^{-05} µCi/ml
^{87}Kr	2.31×10^{-10} µCi/ml	^{135m}Xe	5.97×10^{-07} µCi/ml
^{88}Kr	1.32×10^{-07} µCi/ml	^{137}Xe	$< 1.00 \times 10^{-31}$ µCi/ml
^{89}Kr	$< 1.00 \times 10^{-31}$ µCi/ml	^{138}Xe	$< 1.00 \times 10^{-31}$ µCi/ml
^{90}Kr	$< 1.00 \times 10^{-31}$ µCi/ml	^{139}Xe	$< 1.00 \times 10^{-31}$ µCi/ml

All doses were calculated at the point of the nearest residence in relation to the reactor facility, approximately 760 meters due north of the MURR. This is the same point that is used to assess the dose to the general public as presented in Section 5.3.3 of Addendum 3 to the MURR Hazards Summary Report [1]. The 312x dilution, or reduction, factor is the most conservative value obtained when using the Pasquill-Gifford model of atmospheric dispersion for the point of the nearest resident [1]. This dilution factor is the ratio of stack emission concentration at the exit point of the exhaust stack to the concentration at the nearest resident under the most unfavorable atmospheric conditions. In actuality, the 24-hour doses received by this individual would be much lower.

Radioiodine Concentrations and Corresponding Radiation Doses
(24 hours following target failure)

Radioiodine	Concentration	Radiation Dose (thyroid)
^{131}I	4.17×10^{-09} μCi/ml	$2.86 \times 10^{+00}$ mrem
^{132}I	1.03×10^{-08} μCi/ml	7.08×10^{-02} mrem
^{133}I	1.36×10^{-08} μCi/ml	$1.86 \times 10^{+00}$ mrem
^{134}I	1.74×10^{-14} μCi/ml	3.98×10^{-08} mrem
^{135}I	2.99×10^{-09} μCi/ml	6.83×10^{-02} mrem
		Total = 4.86 mrem

Noble Gas Concentrations and Corresponding Radiation Doses
(24 hours following target failure)

Noble Gas	Concentration	Radiation Dose
^{85}Kr	4.39×10^{-12} μCi/ml	8.60×10^{-07} mrem
^{85m}Kr	9.00×10^{-10} μCi/ml	1.23×10^{-03} mrem
^{87}Kr	7.42×10^{-13} μCi/ml	5.08×10^{-06} mrem
^{88}Kr	4.24×10^{-10} μCi/ml	6.45×10^{-03} mrem
^{89}Kr	$< 1.00 \times 10^{-34}$ μCi/ml	$< 1.00 \times 10^{-29}$ mrem
^{90}Kr	$< 1.00 \times 10^{-34}$ μCi/ml	$< 1.00 \times 10^{-29}$ mrem
^{133}Xe	5.19×10^{-08} μCi/ml	1.42×10^{-02} mrem
^{135}Xe	3.24×10^{-08} μCi/ml	6.35×10^{-02} mrem
^{135m}Xe	1.91×10^{-09} μCi/ml	6.55×10^{-03} mrem
^{137}Xe	$< 1.00 \times 10^{-34}$ μCi/ml	$< 1.00 \times 10^{-29}$ mrem
^{138}Xe	$< 1.00 \times 10^{-34}$ μCi/ml	$< 1.00 \times 10^{-29}$ mrem
^{139}Xe	$< 1.00 \times 10^{-34}$ μCi/ml	$< 1.00 \times 10^{-29}$ mrem
		Total = 0.09 mrem

To finalize the maximum calculated dose to an individual in the unrestricted environment within a 24-hour period after target failure, the doses from the radioiodines and noble gases must be summed together, and result in the following values:

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

Committed Effective Dose Equivalent (Thyroid)	**4.86 mrem**
Committed Effective Dose Equivalent (Noble Gases)	**0.09 mrem**
Total Effective Dose Equivalent (Whole Body)	**4.95 mrem**

As indicated above, the potential dose received from a target failure is well within the 10 CFR 20 dose limits for individual members of the public.

References

[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.

Attachments

1. Section 5.3.3, "Evaluation of Environmental Impact of Increased Stack Release Flow Rate," Hazards Summary Report, Addendum 3, University of Missouri Research Reactor Facility

EVALUATION OF ENVIRONMENTAL IMPACT

OF INCREASED STACK RELEASE FLOW RATE

Stack release limits set for MURR in Technical Specification[1] Number 3.7: "Facility Gaseous and Particulate Radioactive Release" are based on activity concentrations. An increase in stack flow rate affects the total allowable release of activity, and thus this evaluation is made to assess the environmental impact the increase will have on the nearest resident and on the population surrounding the MURR. The change in stack height and exhaust exit path is also considered. The safety significance of the impact is discussed in relation to background radiation and in relation to a previous environmental impact appraisal made by NRC.[2]

Data and Assumptions

The data and calculations in Table 1 describe the physical information of the stack release point. Argon-41 is the principal isotope released in gaseous effluents from MURR. The Technical Specification limit for Ar-41 release is 350 times the MPC listed in Appendix B, Table II, Column I of 10CFR20, or:

$$\begin{aligned} Q &= 350 \times \text{MPC} \times \text{flowrate} \\ &= (350)(4\times10^{-8}\mu\text{Ci/ml})(36500\ \text{ft}^3/\text{min})(2.831 \times 10^4 \text{ml/ft}^3) \\ &= (1.4 \times 10^4\ \mu\text{Ci/min})(1\times10^{-6}\ \text{Ci}/\mu\text{Ci})(1\ \text{min}/60\text{sec}) \\ &= 2.4 \times 10^{-4}\ \text{Ci/sec} \end{aligned}$$

In the previous environmental assessment,[2] the NRC used meteorological data collected at the Callaway Plant, located near Fulton. These data were collected between May 5, 1973 and May 4, 1975, and were judged by the NRC to be "reasonably representative of long-term conditions expected at the MURR site." This current assessment utilizes meteorological data gathered in Columbia, MO from 1960 to 1969.[3] The Columbia data was judged to be more appropriate for use in assessing airborne releases from MURR because of the longer data period and the proximity of the data site to MURR. Table 2 lists wind data (stability, class, speed and frequency) for each of the sixteen campus points.

Table 1

Physical Information for Stack Release Point

Elevation above sea level = 687 feet

Diameter = 40 inches

New Max flowrate = 36500 ft^3/min

Area Cross Section = πr^2

$$= \pi \left(\frac{40 \text{ inches}}{2 \cdot 12 \text{ inches/ft}} \right)^2$$

$$= 8.73 \text{ ft}^2$$

$$\text{Air Velocity (v)} = \frac{36500 \text{ ft}^3/\text{min}}{8.73 \text{ ft}^2} \cdot 0.304 \text{ m/ft} \cdot \frac{1 \text{ min}}{60 \text{ sec}}$$

$$= 21.2 \text{ m/sec}$$

TABLE 2

Meteorological Data--Columbia, MO (1960-1969)[3]

Stability class information NNE

Class[a]	% Class[b]	Wind speed[c] (m/sec)	% NNE[d] wind	%'s[e] comb.
A	0.4	2.3	3.4	1.4e-04
B	4.7	2.8	2.7	1.3e-03
C	11.5	4.0	3.5	4.0e-03
D	53.6	5.7	4.2	2.3e-02
E	17.6	3.8	3.2	5.6e-03
F	12.2	2.4	4.9	6.0e-03

Stability class information NE

Class[a]	% Class[b]	Wind speed[c] (m/sec)	% NE[d] wind	%'s[e] comb.
A	0.4	2.1	1.7	6.8e-05
B	4.7	2.7	2.6	1.2e-03
C	11.5	3.7	2.7	3.1e-03
D	53.6	5.2	3.9	2.1e-02
E	17.6	3.6	2.8	4.9e-03
F	12.2	2.5	4.9	6.0e-03

Stability class information ENE

Class[a]	% Class[b]	Wind speed[c] (m/sec)	% ENE[d] wind	%'s[e] comb.
A	0.4	2.0	7.8	3.1e-04
B	4.7	2.8	5.1	2.4e-03
C	11.5	3.9	4.3	4.9e-03
D	53.6	4.9	4.7	2.5e-02
E	17.6	3.4	4.2	7.4e-03
F	12.2	2.5	6.8	8.3e-03

Stability class information **E**

Class[a]	% Class[b]	Wind speed[c] (m/sec)	% E[d] wind	%'s[e] comb.
A	0.4	2.0	4.3	1.7e-04
B	4.7	2.9	5.3	2.5e-03
C	11.5	3.8	4.4	5.1e-03
D	53.6	4.9	4.4	2.4e-02
E	17.6	3.5	5.0	8.8e-03
F	12.2	2.5	7.9	9.6e-03

Stability class information **ESE**

Class[a]	% Class[b]	Wind speed[c] (m/sec)	% ESE[d] wind	%'s[e] comb.
A	0.4	2.0	3.4	1.4e-04
B	4.7	2.9	4.7	2.2e-03
C	11.5	3.9	4.8	5.5e-03
D	53.6	5.3	6.1	3.3e-02
E	17.6	4.0	6.1	1.1e-02
F	12.2	2.6	4.5	5.5e-03

Stability class information **SE**

Class[a]	% Class[b]	Wind speed[c] (m/sec)	% SE[d] wind	%'s[e] comb.
A	0.4	2.2	2.6	1.0e-04
B	4.7	2.9	4.6	2.2e-03
C	11.5	4.1	6.4	7.4e-03
D	53.6	5.7	7.8	4.2e-02
E	17.6	4.1	8.2	1.4e-02
F	12.2	2.5	4.3	5.2e-03

Stability class information

SSE

Class(a)	% Class(b)	Wind speed(c) (m/sec)	% SSE(d) wind	%'s(e) comb.
A	0.4	2.3	4.3	1.7e-04
B	4.7	3.0	6.5	3.1e-03
C	11.5	4.1	8.7	1.0e-02
D	53.6	5.6	9.3	5.0e-02
E	17.6	4.1	12.0	2.1e-02
F	12.2	2.7	7.2	8.8e-03

Stability class information **S**

Class(a)	% Class(b)	Wind speed(c) (m/sec)	% S(d) wind	%'s(e) comb.
A	0.4	2.1	6.0	2.4e-04
B	4.7	3.0	10.8	5.1e-03
C	11.5	4.2	14.4	1.7e-02
D	53.6	5.6	11.8	6.3e-02
E	17.6	4.0	17.6	3.1e-02
F	12.2	2.6	12.0	1.5e-02

Stability class information **SSW**

Class(a)	% Class(b)	Wind speed(c) (m/sec)	% SSW(d) wind	%'s(e) comb.
A	0.4	2.4	6.0	2.4e-04
B	4.7	3.1	8.6	4.0e-03
C	11.5	4.1	9.7	1.1e-02
D	53.6	5.6	5.5	2.9e-02
E	17.6	3.9	7.4	1.3e-02
F	12.2	2.6	6.3	7.7e-03

Stability class information **SW**

Class (a)	% Class (b)	Wind speed (c) (m/sec)	% SW (d) wind	%'s (e) comb.
A	0.4	1.8	5.2	2.1e-04
B	4.7	3.0	9.2	4.3e-03
C	11.5	4.1	7.5	8.6e-03
D	53.6	5.4	3.5	1.9e-02
E	17.6	3.9	4.3	7.6e-03
F	12.2	2.5	6.0	7.3e-03

Stability class information **WSW**

Class (a)	% Class (b)	Wind speed (c) (m/sec)	% WSW (d) wind	%'s (e) comb.
A	0.4	2.2	6.0	2.4e-04
B	4.7	3.0	10.8	5.1e-03
C	11.5	4.3	9.0	1.0e-02
D	53.6	5.9	4.9	2.6e-02
E	17.6	3.9	5.7	1.0e-02
F	12.2	2.5	5.9	7.2e-03

Stability class information **W**

Class (a)	% Class (b)	Wind speed (c) (m/sec)	% W (d) wind	%'s (e) comb.
A	0.4	1.8	3.4	1.4e-04
B	4.7	2.8	6.7	3.1e-03
C	11.5	3.9	6.2	7.1e-03
D	53.6	6.0	4.7	2.5e-02
E	17.6	3.7	5.3	9.3e-03
F	12.2	2.5	6.1	7.4e-03

Stability class information **WNW**

Class(a)	% Class(b)	Wind speed(c) (m/sec)	% WNW(d) wind	%'s(e) comb.
A	0.4	2.1	4.3	1.7e-04
B	4.7	2.8	5.4	2.5e-03
C	11.5	4.3	5.1	5.9e-03
D	53.6	6.7	7.9	4.2e-02
E	17.6	4.0	5.5	9.7e-03
F	12.2	2.5	5.0	6.1e-03

Stability class information **NW**

Class(a)	% Class(b)	Wind speed(c) (m/sec)	% NW(d) wind	%'s(e) comb.
A	0.4	2.2	4.3	1.7e-04
B	4.7	2.9	4.4	2.1e-03
C	11.5	4.3	4.7	5.4e-03
D	53.6	7.1	8.8	4.7e-02
E	17.6	4.2	5.1	9.0e-03
F	12.2	2.5	3.6	4.4e-03

Stability class information **NNW**

Class(a)	% Class(b)	Wind speed(c) (m/sec)	% NNW(d) wind	%'s(e) comb.
A	0.4	2.3	1.7	6.8e-05
B	4.7	2.7	2.9	1.4e-03
C	11.5	4.1	3.0	3.5e-03
D	53.6	6.6	5.8	3.1e-02
E	17.6	4.0	3.6	6.3e-03
F	12.2	2.4	3.0	3.7e-03

Stability class information **N**

Class[a]	% Class[b]	Wind speed[c] (m/sec)	% N[d] wind	%'s[e] comb.
A	0.4	2.4	7.8	3.1e-04
B	4.7	2.7	4.8	2.3e-03
C	11.5	4.0	4.8	5.5e-03
D	53.6	6.0	6.2	3.3e-02
E	17.6	3.8	4.0	7.0e-02
F	12.2	2.5	5.8	7.1e-03

(a) Stability class as defined by Pasquill's Categories.[4]
(b) Annual frequency distribution of stability class for all directions, or the total probability of occurrence for that class.
(c) Average wind speed for stability class and wind direction.
(d) Annual frequency distribution of wind direction for the specific stability class, or the probability of the wind direction given that the stability class exists.
(e) %'s comb. = (% class/100) x (% NNE/100), or the joint probability of the specific stability class and the specific direction occurring at the same time.
Example: A conditional probability is one in which the probability of the events depends upon whether the other event has occurred.[5]
P(A) = probability of Class A conditions = 0.4%.
P(N/A) = probability of wind direction from N given Class A conditions = 7.8%.
P(AN) = probability of having Class A conditions and wind direction from N.
P(AN) = P(A) P(N/A) = 3.1×10^{-4}.

Listed in Table 3 are the equations used to calculate the Ar-41 concentration and dose, along with the associated assumptions used for each case, at a distance, x, downwind from the stack release point. Calculations are based on the Pasquill-Gifford Method of determining stack release concentrations (effective stack height). Data for σ_y and σ_z were obtained from Ref. 4 and the DCF from Ref. 6.

Table 3

Equations and Assumptions

(1) Effective Stack Height[4] (H):

$$H = h + d\left(\frac{v}{\mu}\right)^{1.4}\left(1 + \frac{\Delta T}{T}\right) \qquad \text{(Eq. 1)}$$

where, h = actual height (m)
= difference in elevation from release point to downwind site of dose calculation
d = diameter of release point (m)
μ = average wind speed for specific stability class (m/sec)
v = exit velocity (m/sec)
ΔT = temperature difference between stack air and surrounding air
= assumed to be 0
T = absolute temperature of stack air

Therefore,

$$H = h + d\left(\frac{v}{\mu}\right)^{1.4} \qquad \text{(Eq. 1a)}$$

(2) Concentration Calculation:

$$\frac{\chi}{Q} = \frac{1}{\pi\sigma_y\sigma_z\mu}\exp\left[-\frac{1}{2}\left(\frac{y^2}{\sigma_y^2} + \frac{H^2}{\sigma_z^2}\right)\right] \qquad \text{(Eq. 2)}$$

where, χ = concentration at downwind site of dose calculation (μCi/ml or Ci/m^3)
Q = release rate (Ci/sec)
σ_y = lateral dispersion coefficient at downwind site of dose calculation (m)
σ_z = vertical dispersion coefficient at downwind site of dose calculation for specific stability class (m)
μ = average wind speed for specific stability class (m/sec)
y = distance from plume centerline (m) for maximum concentration, assume to be 0
H = effective stack height (m)

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For maximum concentration:

$$\frac{\chi}{Q} = \frac{1}{\pi\sigma_y\sigma_z\mu} \exp\left[-\frac{1}{2}\left(\frac{H}{\sigma_z}\right)^2\right] \quad . \qquad \text{(Eq. 2a)}$$

Further, for case of ground release (H=0),

$$\frac{\chi}{Q} = \frac{1}{\pi\sigma_y\sigma_z\mu} \quad . \qquad \text{(Eq. 2b)}$$

Considering decay, the equation becomes

$$\frac{\chi}{Q} = \frac{e^{-\lambda t}}{\pi\sigma_y\sigma_z\mu} \qquad \text{(Eq. 2c)}$$

where, λ = decay constant for Ar-41 (sec^{-1})
t = time (sec)
= x/μ

(3) Annual Dose Calculation (D):

$$D = DCF \sum_i \chi_i \ (\%\ comb)_i \qquad \text{(Eq. 3)}$$

where, DCF = dose conversion factor
= 8.84×10^{-3} $\frac{mrem\ m^3}{\rho Ci\text{-}y}$ for Ar-41[6]
i = summation over all stability classes
$(\%\ comb)_i$ = relative frequency for stability class, i, and specific wind direction

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Maximum Individual Dose

To determine the maximum individual dose, the south wind direction was chosen as being the most probable and annual doses determined at maximum release rate for two different distances: 150 m north to the exclusion boundary,[7] and 760 m north to the nearest residence. Elevations for these two sites were estimated from a University of Missouri topographical map (shown in Fig. 1). Data and the maximum calculated dose estimates for these sites are given in Table 4, with an example calculation given in Table 5. The maximum average annual dose at 150 m was calculated as ~ 2 mrem/y and ~ 18 mrem/y and at 760 m. The difference in relative plume height at these sites is what leads to this difference in dose rates.

Table 4

Maximum Average Annual Individual Dose

Location at 150 m Directly North

Elevation at man height: 636 ft.

Class	Eff height (m)	σ_y (m)	σ_z (m)	χ/Q (s/m^3)	χ ($\mu Ci/ml$) (Ci/m^3)	Dose w/%'s (mrem/y)
A	42	35	23	3.6e-05	8.6e-09	0.0
B	31	25	15	3.3e-05	7.9e-09	0.4
C	25	19	11	2.5e-05	6.0e-09	0.9
D	22	12	7	4.5e-06	1.1e-09	0.6
E	26	9	5	1.9e-09	4.6e-13	0.0
F	35	6.6	3.2	2.7e-28	6.5e-32	0.0
					TOTAL	1.9 mrem/y

Location at 760 m. directly North.
Elevation at man height: 700 ft.

Class	Eff height (m)	σ_y (m)	σ_z (m)	χ/Q (s/m^3)	χ ($\mu Ci/ml$) (Ci/m^3)	Dose w/%'s (mrem/y)
A	23	160	300	3.2e-06	7.8e-10	0.0
B	12	110	90	1.1e-05	2.5e-09	0.1
C	6	81	50	1.9e-05	4.5e-09	0.7
D	3	54	25	4.2e-05	1.0e-08	5.7
E	7	41	18	1.0e-04	2.5e-08	6.7
F	15	30	11	1.4e-04	3.5e-08	4.5
					TOTAL	17.7 mrem/y

Maximum Population Dose Estimate

Population dose estimates were made assuming ground release conditions. Population density data was generated[8] using 1980 census data, 1985 update data, and growth projections provided by City of Columbia officials. Estimates for population doses were based on the projected 1990 population densities (See Table 6).

The maximum average annual dose was determined at the center of each population zone, except for the 16 zones at 0-1 miles. Because residences are no closer than 760 m, the midpoint was chosen at 0.75 miles (1200 m) from MURR. In addition, radioactive decay was considered in these calculations due to the significant amount of time required for the plume to move to these distances. Otherwise, calculations were made as were the individual dose estimate calculations. Data for σ_y and σ_z is given in Table 7, the summary of annual doses in Table 8, and the population dose estimate in Table 9. For the population out to 10 miles, the maximum annual popula-tion dose is estimated to be 145 person-rem.

Table 6

Projected 1990 Population Densities
(Number of People)

	Midpoint Distances (m)					
Wind Direction	1200	2400	4000	5600	7200	12000
NNE	238	437	368	315	262	206
NE	101	845	469	105	210	204
ENE	132	534	449	440	76	305
E	94	1189	1270	3905	220	315
ESE	186	1138	2025	849	51	3850
SE	406	2096	1664	1021	474	402
SSE	354	2747	1676	542	428	920
S	364	2649	2293	644	157	5750
SSW	1131	3163	1843	1404	1513	6135
SW	2699	5137	2491	2387	1571	2877
WSW	1997	4803	1067	1146	1055	5610
W	49	1446	525	385	153	234
WNW	52	592	1182	325	364	316
NW	36	126	644	222	103	315
NNW	288	665	229	30	154	210
N	339	851	974	432	210	255

TABLE 7

σ_y's (top) and σ_z's (bottom) for

population distances and stability

Stability Class	Midpoint Distances (m) 1200	2400	4000	5600	7200	12000
A	220	400	620	900	1050	1800
	800	5000	9700	14000	19000	33000
B	170	310	480	690	820	1300
	150	470	1100	2200	3300	6600
C	130	220	340	480	600	900
	75	130	200	270	320	500
D	80	140	220	300	400	610
	34	53	72	91	100	140
E	60	110	170	220	300	460
	23	40	50	60	70	84
F	42	80	120	160	200	300
	14	22	30	33	40	48

TABLE 8

A Summary of dose rate estimates (mrem/y)

based on wind direction & distance

Wind Direction	Midpoint Distances (m) 1200	2400	4000	5600	7200	12000
NNE	4.5	1.4	0.7	0.4	0.3	0.1
NE	4.3	1.4	0.6	0.4	0.2	0.1
ENE	6.1	2.0	0.9	0.6	0.3	0.2
E	6.7	2.1	1.0	0.6	0.4	0.2
ESE	5.2	1.7	0.8	0.5	0.3	0.1
SE	5.9	1.9	0.9	0.5	0.3	0.2
SSE	8.4	2.7	1.3	0.8	0.5	0.2
S	13.0	4.2	1.9	1.2	0.7	0.3
SSW	6.3	2.0	0.9	0.6	0.4	0.2
SW	5.1	1.7	0.8	0.5	0.3	0.1
WSW	5.7	1.8	0.8	0.5	0.3	0.1
W	5.6	1.8	0.8	0.5	0.3	0.1
WNW	5.5	1.8	0.8	0.5	0.3	0.1
NW	4.7	1.5	0.7	0.4	0.3	0.1
NNW	3.7	1.2	0.6	0.3	0.2	0.1
N	5.5	1.8	0.8	0.5	0.3	0.1

TABLE 9

Person-Rem Estimates (person-rem/y)

	Midpoint Distances (m)					
Wind Direction	1200	2400	4000	5600	7200	12000
NNE	1.1	0.6	0.2	0.1	0.1	0.0
NE	0.4	1.2	0.3	0.0	0.1	0.0
ENE	0.8	1.0	0.4	0.2	0.0	0.0
E	0.6	2.5	1.3	2.4	0.1	0.1
ESE	1.0	1.9	1.6	0.4	0.0	0.5
SE	2.4	4.0	1.5	0.6	0.2	0.1
SSE	3.0	7.5	2.1	0.4	0.2	0.2
S	4.7	11.1	4.5	0.8	0.1	1.9
SSW	7.2	6.5	1.7	0.8	0.5	1.0
SW	13.9	8.5	1.9	1.1	0.5	0.4
WSW	11.3	8.7	0.9	0.6	0.3	0.8
W	0.3	2.6	0.4	0.2	0.0	0.0
WNW	0.3	1.1	1.0	0.2	0.1	0.0
NW	0.2	0.2	0.5	0.1	0.0	0.0
NNW	1.1	0.8	0.1	0.0	0.0	0.0
N	1.8	1.5	0.8	0.2	0.1	0.0
Subtotals	50.0	59.8	19.2	8.2	2.4	5.3
					TOTAL	144.8

Consideration of Normal Operational Releases

For the past five years, MURR has released ~ 1000 Ci/y of Ar-41 with a stack flowrate of ~ 16,500 ft^3/min. Production of Ar-41 is expected to remain the same, and so the average Ar-41 concentration is anticipated to be:

$$3.7E-6(\mu Ci/ml) \cdot 16500/36500 = 2E-6\ \mu Ci/ml$$

which is ~ 13% of the Technical Specifications Limit. Because the dose estimates calculated thus far are proportional to the total amount of Ar-41 released, the dose estimates for actual operating conditions are easily calculated using the ratios of the stack release flow rates (given Ar-41 production remains constant). The actual operational dose estimates are:

Individual @ 150 m = 0.2 mrem/y
Individual @ 760 m = 2 mrem/yr
Population to 10 miles = 15 person-rem

Comparison of Risk

In the Safety Evaluation made by the NRC in support of Amendment No. 12,(2) an individual located at the nearest residence was estimated to receive an annual average total body dose of 13 mrem per year based on the 1977/78 release of 1925 Ci/y and 29 mrem/y for the maximum estimate. In the same NRC evaluation, the population dose for implementing Amendment No. 12 was estimated to be 20 person-rem. Although assumptions, data, and conditions for calculation are not fully described in the NRC Amendment No. 12, estimated doses are greater than those predicted by the current assessment, which utilizes more realistic model (effective stack height and stability class weighting) and better site-specific data (meteorological data and updated population densities). The NRC concluded "that there would be no significant environmental impact attributable" to an increase in stack release limit to 350 MPC. With lower doses estimated for the current change in stack height and flowrate, it is also concluded that no significant environmental impact exists. The same conclusion applies to instantaneous release limits.

Another method of assessing risk from the estimated doses is to compare them to natural background dose rates. The average whole body dose to an individual in the US is 360 mrem/y.(9) The estimated doses in terms of % of natural background are:

	Maximum Case	Normal Operation
Individual @ 150 m	0.5%	0.1%
Individual @ 750 m	5%	0.6%
Population	0.4%	<0.1%

Variations of this magnitude can be found in annual dose for populations living in different areas of the US with no observable effects.

Conclusion

The estimated dose rates calculated using improved methods and data were no greater than those calculated from previous appraisals where impact was judged by the NRC to be not significant in environmental impact. Therefore, there is no significant reduction in safety as the result of the changes in the MURR stack release conditions.

References

(1) Appendix A: Technical Specifications for University of Missouri Research Reactor Facility--Facility Operating License No. R-103.

(2) NRC Amendment No. 12 for R-103, July 5, 1979.

(3) Callaway Environmental Report, Operational License Stage, Vol. 1, Tables 2.3-19 and 2.3-20.

(4) Cember, Herman, Introduction to Health Physics, Second Edition, Pergamon Press, 1983, pp. 340-352.

(5) DeGroot, Morris H., Probability and Statistics, Addison-Wesley Publishing Company, Inc., 1975, pp. 49-50.

(6) Regulatory Guide 1.109: "Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluation Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977.

(7) NRC Amendment No. 8 for R-103, February 24, 1978.

(8) Environmental Report for Upgrade of MURR, 1987 (Draft).

(9) NCRP Report No. 94: "Exposure of the Population in the United States and Canada from Natural Background Radiation," December 1987.