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January 10, 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

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 August 10, 2007

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 (TSs), 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. These questions, and MURR's responses to those questions, are 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,

Walter Meyer for Ralph 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

Christine M. Errante

CHRISTINE M. ERRANTE
Notary Public - Notary Seal
State of Missouri
Commissioned for Boone County
My Commission Expires: April 14, 2011
Commission Number: 07528381

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NRR

1. *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 containment 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 then 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 more 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 (6.28×10^{-03} gal/m^3)
= 2.12 $\mu Ci/m^3$

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

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 – 2.12×10^{-06} µCi/ml	^{133}I – 1.25×10^{-05} µCi/ml	^{135}I – 1.18×10^{-05} µCi/ml
^{132}I – 5.85×10^{-06} µCi/ml	^{134}I – 1.43×10^{-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 (1.57×10^{-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 from the kryptons and xenons present in the containment building is 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 Derived Air Concentrations (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 in Appendix B of 10 CFR 20. However, doses derived in this manner were selectively verified by using dose conversion factors similar to those used for the radioiodines. 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	$8.75 \times 10^{+00}$ mrem
^{132}I	3.00×10^{-06} µCi/ml	1.61×10^{-01} mrem
^{133}I	1.00×10^{-07} µCi/ml	$1.03 \times 10^{+01}$ mrem
^{134}I	2.00×10^{-05} µCi/ml	5.88×10^{-02} mrem
^{135}I	7.00×10^{-07} µCi/ml	$1.39 \times 10^{+00}$ mrem
		Total = 20.69 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)	**20.69 mrem**
Committed Effective Dose Equivalent (Thyroid)	**0.62 mrem**
Committed Effective Dose Equivalent (Noble Gases)	**1319.22 mrem**
Total Effective Dose Equivalent (Whole Body)	**1319.84 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 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 leakage rate is proportional to the square root of the pressure differential between the containment building and outside atmosphere; therefore, the initial leakage rate out of the containment structure would be approximately 10.3 scfm and it would take approximately 16.5 hours for the leak rate to go to zero after an initial pressure differential of 0.33 psig (2.28 kPa above atmosphere). 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 according to the model developed in Appendix B of the recently submitted MURR Safety Analysis Report (SAR) for license renewal, 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.

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. 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)

$^{131}I - 5.30 \times 10^{-07}$ µCi/ml	$^{133}I - 3.13 \times 10^{-06}$ µCi/ml	$^{135}I - 2.95 \times 10^{-06}$ µCi/ml
$^{132}I - 1.46 \times 10^{-06}$ µCi/ml	$^{134}I - 3.58 \times 10^{-06}$ 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 (See Appendix B of the recently submitted MURR SAR for license renewal) 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 Dose Equivalent (Thyroid)	**7.21 mrem**
Committed Effective Dose Equivalent (Thyroid)	**0.22 mrem**
Committed Effective Dose Equivalent (Noble Gases)	**1.32 mrem**
Total Effective Dose Equivalent (Whole Body)	**1.54 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 1.54 mrem, a value far 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.

2. *How would failure of a fueled experiment be detected? What is the sensitivity of the radiation detection equipment? What actions, both automatic and operator initiated would be taken in response to an experiment failure?*

As indicated in the response to Question No. 1, we have limited the mass, and hence the heat generation, of this experiment such that failure during irradiation is highly unlikely. We will instrument the target to monitor temperature and will remove the target from its graphite reflector irradiation position or shutdown the reactor if (1) the surface temperature of the target approaches the saturation temperature of the cooling medium, or (2) the temperature of target approaches half the melting temperature of any material in the experiment.

Because this is a pilot experiment, with only one irradiation planned, we will perform a daily pool water radiochemical analysis for fission product activity by counting a 50 ml coolant sample on a High Purity Germanium (HPGe) Detector. The following table provides the typical Minimum Detectable Activities (MDA) of selected radioiodines and noble gases in the pool coolant:

Typical Minimal Detectable Activities of
Radioiodines and Noble Gases of Interest in the Pool Coolant
(in microcuries per milliliter)

^{131}I – 1.90×10^{-07} µCi/ml	^{85m}Kr – 2.80×10^{-07} µCi/ml
^{132}I – 2.80×10^{-07} µCi/ml	^{87}Kr – 6.00×10^{-07} µCi/ml
^{133}I – 2.00×10^{-07} µCi/ml	^{88}Kr – 2.60×10^{-07} µCi/ml
^{134}I – 1.10×10^{-06} µCi/ml	^{135}Xe – 2.00×10^{-07} µCi/ml
^{135}I – 7.00×10^{-07} µCi/ml	^{135m}Xe – 4.40×10^{-06} µCi/ml

Note: The HPGe detector response or efficiency curve is polynomial. The sensitivity is fairly high at low KeVs, reaches a maximum at approximately 100 KeV, and then declines as the KeV increases. A "key line" is assigned for the counting program to use – a specific energy line to calculate from – then the relative abundance of that line is taken into account. Sensitivity, generally speaking, may be as low as 1.0×10^{-15} µCi/ml. However, validating the presence of a specific isotope will be determined from the MDA.

Any elevated radioactivity in the reactor pool coolant as a result of target failure that may become airborne within the containment structure will also be detected by the Off-Gas Radiation Monitoring System; a three-channel radiation detection system designed to measure the airborne concentrations of radioactive particulate, iodine and noble gases in the facility exhaust air.

The following table lists the typical MDAs of selected radioiodines and noble gases in facility exhaust air:

Typical Minimal Detectable Activities of
Radioiodines and Noble Gases of Interest in Facility Exhaust Air
(in microcuries per milliliter)

^{128}I – 1.00×10^{-10} µCi/ml
^{134}I – 5.00×10^{-12} µCi/ml
^{135}I – 2.00×10^{-12} µCi/ml

^{85m}Kr – 1.00×10^{-12} µCi/ml
^{87}Kr – 5.00×10^{-12} µCi/ml
^{88}Kr – 5.00×10^{-12} µCi/ml
^{89}Kr – 1.30×10^{-06} µCi/ml

^{131m}Xe – 1.30×10^{-12} µCi/ml
^{133m}Xe – 6.00×10^{-13} µCi/ml
^{135}Xe – 4.00×10^{-13} µCi/ml
^{135m}Xe – 1.00×10^{-11} µCi/ml
^{137}Xe – 1.00×10^{-08} µCi/ml
^{138}Xe – 2.70×10^{-10} µCi/ml

Note: The sensitivity of the off-gas radiation monitoring system iodine channel for ^{131}I, as determined by the most recent semi-annual calibration is as follows (the discriminator window is established for ^{131}I):

Strip-chart recorder:	Meter:
47,000 CPM/µCi or 2.14×10^{-02} CPM/DPM	45,000 CPM/µCi or 2.05×10^{-02} CPM/DPM

The sensitivity of the iodine channel to detect the other radioidines is as good, if not better, based on their respective energy emissions.

As stated in the response to Question No. 1, the fission products released into the reactor pool from a target failure will more than likely be detected by the installed radiation monitoring equipment; however, it is assumed that a reactor scram and actuation of the containment building isolation system by action of either the pool surface or ventilation system exhaust plenum radiation monitors will occur. Actuation of the isolation system will then prompt Operations personnel to ensure that a total evacuation of the containment building is accomplished within 2 minutes. The Reactor Operator may also manually initiate a reactor isolation prior to automatic actuation should levels reach those addressed by the Facility or Reactor Emergency Procedures.

Additionally, it will be included within the experimental plan that the detection of any elevated levels of radioactivity in either the daily pool water radiochemical analysis or by the off-gas radiation monitoring system will require an immediate shutdown of the reactor while the failed target is being removed and properly contained to minimize the release of radioactivity. Fission products in the pool coolant system can be removed by the reactor coolant cleanup system. This cleanup procedure would be undertaken under closely monitored and controlled conditions.

3. *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.

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 MURR's recent submittal of the Safety Analysis Report (SAR) in support of the application to renew the facility's operating license. 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. 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 Dose Equivalent (Thyroid)	**4.86 mrem**
Committed Effective Dose Equivalent (Thyroid)	**0.15 mrem**
Committed Effective Dose Equivalent (Noble Gases)	**0.09 mrem**
Total Effective Dose Equivalent (Whole Body)	**0.24 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.

4. *License condition 2.B.(2) allows you "to posses, but not separate, such special nuclear material as may be produced by the operation of the facility..." Please describe the complete processing that the irradiated uranium targets will undergo. Explain how license condition 2.B.(2) is met during this process.*

The overall LEU-Modified Cintichem process is comprised of the following six major steps: Annular Target Fabrication, Target Irradiation, Target Disassembly, Dissolution of the LEU-Foil, Separation of Mo-99 from Other Fission Products, and Mo-99 Purification. In the dissolution and separation processes, release of fission gases and fission product material is mitigated by (1) capturing the fission gases in a cold-finger assembly, and (2) using the vacuum in evacuated bottles to transfer the dissolved fission products. Each of the six steps is summarized in the following paragraphs.

Annular Target Fabrication

- A 135 μm thick uranium foil is wrapped in a sandwich of a 15 μm thick fission-recoil barrier.
- The foil is then wrapped around the inner target tube. A small "undercut" on the inner tube ensures that the uranium foil is positioned correctly within the target.
- The inner tube with the uranium foil wrapped around it is slid into the outer tube.
- The inner and outer tubes with the uranium foil sandwiched between them are slid into a draw die.
- The inner tube is then expanded by pulling a draw plug through the inside of the inner tube. The diameters of the inner and outer tubes and of the draw plug are chosen such that the inner tube is expanded plastically and the outer tube is expanded elastically. This creates a target in which the uranium foil is held tightly between the two tubes, effecting good heat transfer within and out of the target.
- After removal from the draw die, the two ends of the target are seal-welded.

Target Irradiation

The target will be irradiated in the graphite reflector region for a period of $\leq$ 150 hours (normal weekly operating schedule) at a thermal neutron flux of $\leq 1.5 \times 10^{+13}$ n/cm^2-sec. Following irradiation, the target will be decayed in the reactor pool for a period of $\geq$ 20 hours.

Target Disassembly

After the target is decayed in the reactor pool, it will be placed in a shielded cask and transported to a hot cell where it will be disassembled in order to remove the foil component of the target.

Target Dissolution

A gas-tight dissolver is used to ensure that the radioactive fission gases and volatile iodine species that are liberated from the irradiated foils when the target is dissolved are not released to the hot cell. The irradiated foil (nickel and uranium) will be placed in the dissolver that is then sealed using an O-Ring flange to a gas-tight "T" assembly that contains an inlet port, a pressure gauge, and quick disconnect gas connector with a valve. Nitric acid is then added to the sealed dissolver system through a septum in the inlet port and the dissolver assembly is heated with a heating device to complete the dissolution of the irradiated foils. It takes approximately 20 minutes to dissolve the foil.

After dissolution, the dissolver is connected to a liquid-nitrogen-cooled cold finger through a tube containing a copper gauze filter. When the valve on the "T" is opened, the volatile iodine species are trapped in the copper-gauze filter and the radioactive fission gases (xenon and krypton) are trapped on a molecular sieve in the cold finger. The valve on the cold finger is then closed, it is allowed to warm, and the cold finger is stored in the hot cell for several weeks to allow the fission gases to decay.

Mo-99 Separation

After the fission gases are captured in the cold finger, air is admitted into the dissolver and the solution in the dissolver is pulled through a septum into a process bottle by vacuum in the process bottle. The reagents for the molybdenum precipitation are then added, through a septum, into the process bottle and the solution containing the fission product waste is pulled through a septum into a second evacuated bottle. This "fission waste" bottle is then removed from the processing stream and stored.

Mo-99 Purification

The molybdenum precipitate is redissolved in a mixture of sodium hydroxide and hydrogen peroxide. It is then passed through two columns where the impurities are sorbed. The first column contains a mixture of activated charcoal and silver-coated activated charcoal. The second column contains activated charcoal, silver-coated charcoal, and hydrated zirconium hydroxide.

Condition 2.B.(2) of Facility License R-103 allows the MURR "*...to possess, but not separate, such special nuclear material as may be produced by operation of the facility...*" While there are minimal amounts of Special Nuclear Material (SNM) produced by the proposed experiment – less than 50 micrograms – the experiment process we will use is not capable of separating any SNM from the 5-gram LEU target. During the process, the SNM form will be changed from solid (foil) to liquid to allow the extraction of the Mo-99 for medical research purposes. As part of the waste handling and preparation, this process liquid which contains the SNM will be returned to a solid form suitable for disposal. MURR is working with Argonne National Laboratory to define how to treat this small volume of waste, which is expected to be less than 200 milliliters. The proposed experiment is within the license conditions imposed by 2.B.(2).