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RESEARCH REACTOR CENTER

July 31, 2015

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 No. R-103

Enclosed you will find the University of Missouri-Columbia Research Reactor's responses to the U.S. Nuclear Regulatory Commission's (NRC) request for additional information, dated June 18, 2015, regarding our renewal request for Amended Facility Operating License No. R-103, which was submitted to the NRC on August 31, 2006, as supplemented.

If you have any questions, please contact John L. Fruits, the facility Reactor Manager, at (573) 882-5319 or FruitsJ@missouri.edu.

Sincerely,

Ralph A. Butler, P.E.
Director

RAB/jlb

Enclosures

A020
NRR

1513 Research Park Drive Columbia, MO 65211 Phone: 573-882-4211 Fax: 573-882-6360 Web: www.murr.missouri.edu

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REFERENCE: Docket 50-186
University of Missouri-Columbia Research Reactor
Amended Facility License No. R-103

SUBJECT: Written communication as specified by 10 CFR 50.4(b)(1) regarding responses to the "University of Missouri at Columbia - Request for Additional Information Regarding the Renewal of Facility Operating License No. R-103 for the University of Missouri at Columbia Research Reactor (TAC No. ME1580)," dated June 18, 2015

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

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

On June 1, 2010, the NRC requested additional information and clarification regarding the renewal request in the form of one hundred and sixty-seven (167) 45-Day Response Questions. By letter dated July 16, 2010, MURR responded to forty-seven (47) of those 45-Day Response Questions.

On July 14, 2010, via electronic mail (email), MURR requested additional time to respond to the remaining one hundred and twenty (120) 45-Day Response Questions. By letter dated August 4, 2010, the NRC granted the request. By letter dated August 31, 2010, MURR responded to fifty-three (53) of the 45-Day Response Questions.

On September 1, 2010, via email, MURR requested additional time to respond to the remaining twelve (12) Complex Questions. By letter dated September 27, 2010, the NRC granted the request.

1513 Research Park Drive Columbia, MO 65211 Phone: 573-882-4211 Fax: 573-882-6360 Web: www.murr.missouri.edu

On September 29, 2010, via email, MURR requested additional time to respond to the remaining sixty-seven (67) 45-Day Response Questions. On September 30, 2010, MURR responded to sixteen (16) of the remaining 45-Day Questions. By letter dated October 13, 2010, the NRC granted the extension request.

By letter dated October 29, 2010, MURR responded to sixteen (16) of the remaining 45-Day Response Questions and two (2) of the remaining Complex Questions.

By letter dated November 30, 2010, MURR responded to twelve (12) of the remaining 45-Day Response Questions.

On December 1, 2010, via email, MURR requested additional time to respond to the remaining 45-Day Response and Complex Questions. By letter dated December 13, 2010, the NRC granted the extension request.

On January 14, 2011, via email, MURR requested additional time to respond to the remaining 45-Day Response and Complex Questions. By letter dated February 1, 2011, the NRC granted the extension request.

By letter dated March 11, 2011, MURR responded to twenty-one (21) of the remaining 45-Day Response Questions.

On May 27, 2011, via email, MURR requested additional time to respond to the remaining the remaining 45-Day Response and Complex Questions. By letter dated July 5, 2011, the NRC granted the request.

By letter dated September 8, 2011, MURR responded to six (6) of the remaining 45-Day Response and Complex Questions.

On September 30, 2011, via email, MURR requested additional time to respond to the remaining the remaining 45-Day Response and Complex Questions. By letter dated November 10, 2011, the NRC granted the request.

By letter dated January 6, 2012, MURR responded to four (4) of the remaining 45-Day Response and Complex Questions. Also submitted was an updated version of the MURR Technical Specifications.

On January 23, 2012, via email, MURR requested additional time to respond to the remaining the remaining 45-Day Response and Complex Questions. By letter dated January 26, 2012, the NRC granted the request.

On April 12, 2012, via email, MURR requested additional time to respond to the remaining the remaining 45-Day Response and Complex Questions.

By letter dated June 28, 2012, MURR responded to the remaining six (6) the remaining 45-Day Response and Complex Questions. With that set of responses, all 45-Day Response and Complex Questions had been addressed.

On December 20, 2012, the NRC requested a copy of the current Physical Security Plan (PSP) and Operator Requalification Program.

By letter dated January 4, 2013, MURR provided the NRC a copy of the current PSP and Operator Requalification Program.

On February 11, 2013, the NRC requested updated financial information in the form of four (4) questions because the information provided by the September 14, 2009 response had become outdated.

By letter dated March 12, 2013, MURR responded to the four (4) questions.

On December 3, 2014, the NRC requested additional information in the form of two (2) questions regarding significant changes to the MURR facility since submittal of the licensing renewal application in August 2006.

By letter dated January 28, 2015, MURR responded to the two (2) questions.

On April 17, 2015, the NRC requested additional information in the form of ten (10) questions.

On May 29, 2015, via email, MURR requested additional time to respond to the ten (10) questions.

On June 18, 2015, the NRC requested additional information in the form of two (2) questions.

Those questions, and MURR's responses to those questions, are attached.

If there are questions regarding this response, please contact me at (573) 882-5319 or FruitsJ@missouri.edu. I declare under penalty of perjury that the foregoing is true and correct.

Sincerely,

FOR JOHN FRUITS

John L. Fruits
Reactor Manager

ENDORSEMENT:
Reviewed and Approved,

Ralph A. Butler, P.E.
Director

JACQUELINE L. BOHM
Notary Public-Notary Seal
STATE OF MISSOURI
Commissioned for Howard County
My Commission Expires: March 26, 2019
Commission # 15634308

xc: Reactor Advisory Committee
Reactor Safety Subcommittee
Dr. Garnett S. Stokes, Provost
Dr. Henry C. Foley, Senior Vice Chancellor for Research
Mr. Alexander Adams, U.S. Nuclear Regulatory Commission
Mr. Geoffrey Wertz, U.S. Nuclear Regulatory Commission
Mr. Johnny Eads, U.S. Nuclear Regulatory Commission

State of Missouri
County of Boone
The forgoing document was acknowledged before me this 31 day of July, 2015
Jacqueline L. Bohm, Notary Public
My Commission Expires: March 26, 2019

Attachments:

1. Hot Cell, Glove Box and Fume Hood Locations – Grade Level
2. Hot Cell, Glove Box and Fume Hood Locations – Basement Level
3. MURR Exhaust Ventilation Loads
4. RL-72, "cGMP Lu-177 Chloride Processing"
5. RL-79, "Analyses of Annular LEU Target Fission Product Release during Encapsulation Opening (Y-12 Project)"
6. Radioactive Liquid Releases to the Sanitary Sewer – Calendar Years 2005 to 2014
7. Stack Effluent Releases – Calendar Years 2005 to 2014

1. *NUREG-1537 provides guidance regarding hot cells, glove boxes, and hoods, in the following sections:*

- *Section 1.4 requests the licensee to describe hot cells, glove boxes, and hoods that are located within confinement structures, or the restricted area to which the SAR applies;*
- *Section 6.2.2 requests the licensee to provide information regarding the connection of ventilation systems;*
- *Section 9.5 requests the license to describe and discuss design bases for exhausts, drains, and shields;*
- *Section 10.2 requests the licensee to describe and discuss in detail all experimental facilities, which would include hot cells;*
- *Section 11.1.1.3 requests the licensee to provide information regarding programmatic consideration of hot cells for consideration regarding radiation protection; and*
- *Appendix 14.1, Table 14.4 requests the licensee to provide information regarding required radiation measuring channels specifically for hot cells, glove boxes, and hoods.*

In addition, since the hot cells, glove boxes, and hoods are licensed within the requirements of Title 10 of the Code of Federal Regulations (10 CFR) Part 50, the requirements of 10 CFR 50.59 are applicable to changes to the facility as described in the SAR.

In our review of your application for the renewal of Facility Operating License No. R-103, dated August 31, 2006, and supporting information, including your most recent RAI response dated January 28, 2015 (redacted version is available in ADAMS, Accession No. ML15034A474), we were unable to find a complete description of the MURR hot cells, glove boxes, and fume hoods. Additionally, we have not been able to determine in any analyses regarding cumulative or integrated radiological effect of the use of hot cells, glove boxes and hoods (some of this equipment which has been added to MURR since the 2006 license renewal application), or if a review or analysis to determine the need for radiological control areas or radiation monitoring, have been performed. As such, the following information is needed:

a. *Provide a comprehensive list of hot cells, glove boxes, and fume hoods under the reactor license. Describe their locations (including a map if necessary), and indicate the appropriate radiological control area boundaries.*

Attachments 1 and 2 provide the locations of all of the hot cells, glove boxes and fume hoods at MURR. Attachment 3 lists all of the loads that are attached to the MURR ventilation exhaust system. The radiological control area boundary for each hot cell, glove box and fume hood is the area that is posted with radiological control signs. Typically that posting is on the room that the hot cell, glove box or fume hood is located in; therefore, the radiologically controlled area is the room itself. Table 1 below lists all of the hot cells, glove boxes and fume hoods in the facility by location, component designation and the typical isotope that is handled within the unit. A large number of the hot cells, glove boxes and fume hoods are for general use, meaning that different isotopes may be handled within the unit but the safety analysis specific to that isotope applies within the hot cell, glove box or fume hood. The safety evaluation process for isotope use (processing) within a hot cell, glove box or fume hood is described below.

Table 1 – Hot Cells, Gloves Boxes and Fume Hoods at MURR

Location	Designation	Typical Isotope
Basement	HC-01	General Use
	HC-02A	Mo-99
	HC-02B	Mo-99
	HC-03	General Use
	HC-04	General Use
	GB-19	General Use
	GB-30	General Use
Room 111	GB-11	Pm-147
	GB-24	General Use
	GB-25	Bi-210
Room 213	Fume Hood	General Use
Room 216	Fume Hood	General Use
Room 218	Fume Hood	General Use
Room 222	Fume Hood	General Use
	SEH-02	Non-Rad
Room 224	Fume Hood	General Use
Room 225	Fume Hood	General Use
Room 227	Fume Hood	General Use
	GB-21	General Use
	SEH-05	NAA
Room 232B	GB-03	General Use
	GB-14	Rh-105
	GB-15	Se-75
	GB-17	Au-198
Room 238	GB-01	General Use
	GB-02	Lu-177
	GB-04	General Use
	GB-05	Ge/As-77
	GB-09	General Use
	GB-12	Sm-153
	GB-13	Gd-159
	GB-16	General Use
	GB-20	Re-186/188
	GB-22	Au-198
	GB-23	General Use
	GB-28	General Use

Location	Designation	Typical Isotope
Room 241	HC-05	P-32/33 S-35
	GB-06	P-32/33 S-35
	GB-07	P-32/33 S-35
	GB-08	P-32/33 S-35
	GB-27	P-32/33 S-35
Room 242A	Fume Hood	General Use
	GB-01	General Use
Room 242C	GB-18	General Use
Room 244	Fume Hood	General Use
Room 245	Fume Hood	General Use
	SEH-01	General Use
Room 247	SEH-04	General Use
Room 251	Fume Hood	General Use
Room 255	Fume Hood	General Use
Room 257	Fume Hood	General Use
Room 259	Fume Hood	General Use
Room 299D	HC-08A	Lu-177
	HC-08B	Lu-177
Room 299M	HC-06	Lu-177
Room 299N	SEH-06	Lu-177
Room 299P	HC-10	Mo-99
	GB-29	Mo-99
Room 299R	HC-07	Mo-99
	HC-09	Mo-99
Room 299T	HC-11A	I-131
	HC-11B	I-131
	HC-11C	I-131
Room 299V	Fume Hood	I-131 (QC)

Acronyms: HC = Hot Cell; GB = Glove Box; SEH = Specialty Exhaust Hood; NAA = Neutron Activation Analysis; QC = Quality Control.

b. *Provide a reference to any analyses that establish inventory limits for the hot cells, glove boxes, and fume hoods, and demonstrates that a postulated accident would not exceed 10 CFR Part 20 limits for occupational workers or public.*

Hot cells, glove boxes and fume hoods (processing units) at MURR are controlled in several ways with regard to the radiological aspects of their use and with respect to occupational and public dose considerations. Generally, processing units are controlled by their user group at MURR in accordance with an approved reactor license project authorization (RL Project). A 10 CFR 50.59 screen or evaluation is also performed in conjunction with the RL Project. This control process ensures appropriate radiologic controls and facilitates the research or production aspects of MURR's mission to provide high quality radiopharmaceutical products to the research and medical communities.

Prior to a MURR research or production group utilizing a hot cell or glove box, evaluations are conducted by the Reactor Health Physics staff to ensure that the radioactive material used is appropriately shielded by the particular processing unit that will be used. Conversely, if a new isotope is identified for use at MURR, an evaluation occurs to determine if the existing fleet of processing units is sufficient to meet the radiation protection needs of the facility for both occupationally exposed staff and the general public. If no such facility exists, then a review process occurs (either within an existing project or during the creation of a new RL Project) as to what design characteristics are needed to provide the appropriate level of radiation protection to staff and to the general public prior to designing or procuring a new processing unit. Similarly, if higher activities are required for the process, existing hot cells or glove boxes will be evaluated in relationship to the characteristics of the proposed nuclides, including the chemistry and ergonomics of the process, necessary for the safe and effective utilization of the radioisotopes.

Within an RL Project evaluation consideration is given with regard to how occupational exposures will be minimized to the radiation workers based on the quantity of nuclides expected and chemical form to be utilized in the hot cell, glove box or fume hood. The RL Project defines and lists the quantities and chemical forms appropriate for the hot cell or glove box dependent on the specific processing unit shielding and ventilation capabilities. Historically, isotopes irradiated at MURR are metals or metallic compounds that are not subject to volatilization or aerosolization. Any heating during the processing of these isotopes is much less than the melting temperature of the metals or metallic compounds supplied for irradiation. Metallic compounds are usually in the form of nitrates or oxides and are thus more prone to decomposition rather than volatilization. In fact, the heating of these compounds during irradiation is considered during the Reactor Utilization Request (RUR) safety evaluation process prior to placing them in the reactor for irradiation to ensure that adverse heating conditions do not occur due to nuclear heating processes, as high temperatures would destroy the compound being irradiated, thus rendering them useless for processing and further utilization.

Therefore, in RL Projects where there are no or minimal concerns with airborne contamination due to the inherently safe characteristics of the compound or element being irradiated, no language exists within the project documentation explicitly stating that this is not a concern. For example, see RL-72, "cGMP Lu-177 Chloride Processing" (Attachment 4). Conversely, if the RL Project does address the use or generation of nuclides capable of becoming airborne hazards, specific controls are established to ensure workers' and the public's safety is maintained. For example, see RL-79, "Analyses of Annular LEU Target Fission Product Release during Encapsulation Opening (Y-12 Project)" (Attachment 5). Technical Specification 3.8.o applies to this RL Project.

Ultimately, all RL Projects are reviewed by the Isotope Use Subcommittee (IUS) of the Reactor Advisory Committee (RAC) for comprehensive scrutiny and approval, after review by Reactor Health Physics staff and review and approval by the Reactor Health Physics and Reactor Managers. In summary the RL Project process provides a conclusion of any analysis performed during the review process limiting the quantity of radionuclides used within any hot cell, glove box or fume hood with respect to workers' and the public's safety.

c. *Identify services that are required for any hot cells, glove boxes, and fume hoods, based on the assumptions used in the safety analyses or for equipment necessary to mitigate the consequences of any postulated accident (e.g., filters, ventilation, power, emergency power, instrumentation, etc.).*

Services that are available and could possibly be attached to a hot cell, glove box or fume hood include exhaust ventilation, electrical power, filtration, domestic cold water, vacuum and radioactive liquid drains. None of these services are required to mitigate the consequences of any postulated accident in a hot cell, glove box or fume hood. If a hot cell, glove box or fume hood is interfaced with any of the services that are described in the Safety Analysis Report (SAR), then a Modification Record and a 10 CFR 50.59 screen or evaluation is performed. Any filtration that is installed is in keeping with the As Low As Is Reasonably Achievable (ALARA) principles of the MURR Radiation Protection Program to maintain effluent discharges – water and air – as low as possible. Attachments 6 and 7 provide the last 10 years, and average, of radioactive liquid and air releases from the facility. Attachment 6 provides the radioactive liquid releases from the facility to the sanitary sewer in Curies per isotope whereas Attachment 7 provides the stack effluent releases per isotope in percentage of the Technical Specification limit. As you will note, with the exception of argon-41, all other isotopes discharged are less than 0.6% of the release limit. These attachments demonstrate that MURR has an ALARA Program that is both comprehensive and extremely effective.

d. *Identify technical specification changes that are required to ensure that assumptions of postulated accident analyses for these hot cells, glove boxes, and fume hoods, and the associated work areas are maintained.*

There are no postulated accident scenarios that require any of the above hot cell, glove box or fume hood services; therefore, we conclude that no new Technical Specifications are required. Note: A proposed license amendment is currently under review by the NRC regarding the production of iodine-131. The processing facility does have new Technical Specifications associated with the submittal.

2. *In your application supporting License Amendment No. 36, you provided the methodology describing the current MURR steady state operational limits.*

   a. *In Table 3-8, Attachment 10, of your letter dated August 14, 2011 (ADAMS Accession No. ML11237A088), are 15 examples (cases) where critical states were evaluated using the Monte Carlo Neutron Production (MCNP) code. The average deviation provided is approximately 0.64 percent delta k/k (%Δk/k) (640 percent millirho (pcm)) and the maximum is 1.697 %Δk/k (1697 pcm). During discussions with your staff, they were able to demonstrate better agreement using other models and stated that it is possible that the referenced MCNP calculations were performed with control blades not fully represented. Describe the reasons for the differences between the measured and calculated critical eigenvalues for the 15 cited examples (cases).*

Attachment 10 to MURR letter dated August 24 (not August 14), 2011, is MURR internal technical data report TDR-0125, "*Feasibility Analyses for HEU to LEU Fuel Conversion of the University of Missouri Research Reactor*," which was completed in September 2009 for the purpose of documenting the status of the MURR fuel conversion feasibility analyses completed up until that time. Specifically, the report contains the results of reactor design, performance, and steady-state safety analyses. It is essentially a progress/status report of the on-going fuel conversion analysis work jointly being performed by MURR staff and the Reduced Enrichment for Research and Test Reactors (RERTR) Program analysts at Argonne National Laboratory (ANL).

Section 3.3.4 of the referenced report outlines (following Table 3-8) one of the major causes for the higher than desirable difference between the measured critical states and the calculated K_{eff} values, viz., the effect of control blade material [boron-10 (^{10}B)] depletion during long term use. The feasibility analysis at that time had only unirradiated "fresh" control blades modeled. Figure 3.10 of the report gives a graphical representation of this control blade "aging" effect as well.

MURR typically operates with control blades of varying run/core residence times ranging from zero (fresh) to almost ten (10) years. During this time, the ^{10}B material in the bottom few inches of the control blades is observed to deplete significantly, as this portion of the blade always resides in the range of the active fuel length.

Subsequently, MURR performed a detailed control blade modeling and depletion study using MONTEBURNS. MONTEBURNS, which is a coupled MCNP-ORIGEN program, is the primary tool used for fuel and control blade depletion calculations as well as for various other steady-state neutronic calculations at MURR. The results from the control blade depletion study are summarized in Figure 1 below. It shows the axial distribution of ^{10}B in the control blade meat for blades with different operating histories. The results show that towards the control blade end-of-life (EOL), the bottom four (4) inches of the blade has been mostly depleted of any poison material.

Figure 1 – Effect of Core Residence Time on ^{10}B Atom Density in MURR Control Blades

After incorporating the control blade depletion model to the MURR-MCNP model, a series of criticality calculations were performed and the results were compared against the case with non-depleted (fresh) control blades. Table 1 below shows the deviation in % Δk/k from critical state for the two cases. When all-fresh control blades were used in the MCNP simulation (the column labeled "No CB Burnup"), cores with control blades that had the greatest total burnup had the greatest deviation from critical at measured critical control blade heights. However, when the control blades are modeled with control blade depletion compositions (the column labeled "With CB Burnup"), the deviation from critical is much smaller – a reduction from 1.735% to 0.422% for the case with the maximum control blade operating history.

Table 1 – Deviations for the Estimated K_{eff} Values from Critical for the Case with Fresh Control Blades (from feasibility study) and with Boron Depletion Effect Included

CB History (In-cycle Days)	CB Bank Height (inches)	No CB Burnup (% Δk/k)	With CB Burnup (% Δk/k)
287	17.63	-0.260	-0.232
308	18.06	-0.144	-0.139
1040	17.22	-1.301	-0.730
1192	16.72	-1.307	-0.532
1709	16.64	-1.743	-0.390
1835	16.00	-1.735	-0.422

The effect can also be seen in Figure 2 below which provides a plot of the deviation of the calculated K_{eff} values from critical at the measured critical control blade bank height. As seen from the plot, the result of adding control blades with ^{10}B depletion to the MURR core model shows a significant improvement; the average deviation from critical is -0.41% Δk/k when the control blade burnup is modeled, compared with -1.1% Δk /k if the control blades are modeled as fresh.

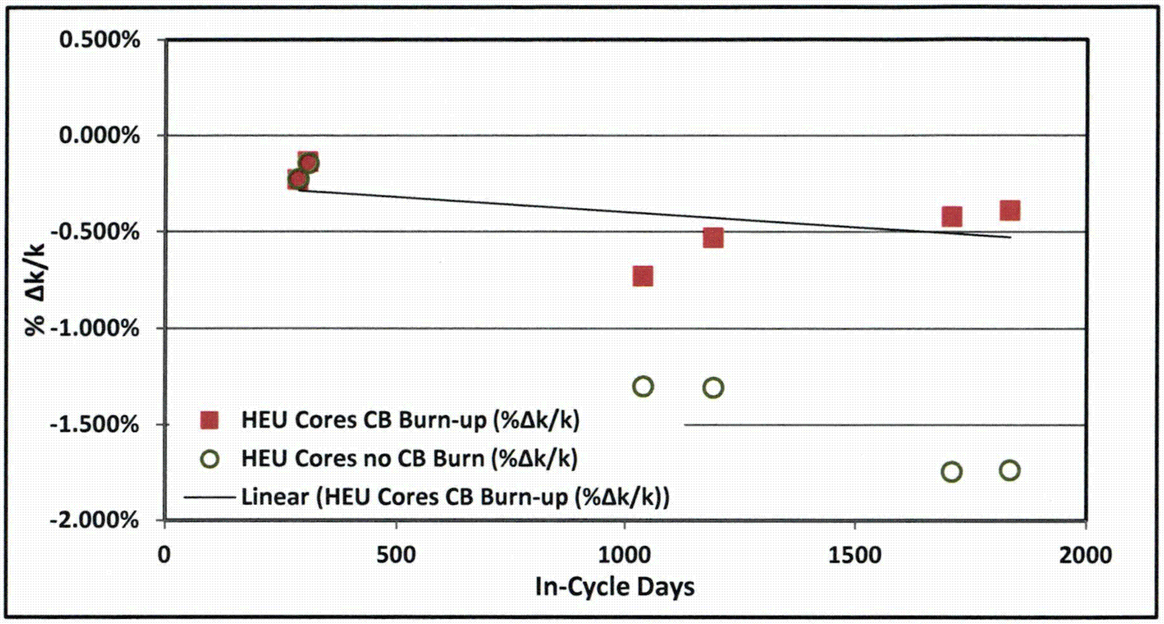


Figure 2 – K_{eff} Deviations from Critical vs. Control Blade Burnup History (for Fresh Control Blades and those with Burnup History)

Since then, MURR has made additional improvements to the MURR-MCNP (neutronics) core model in addition to the control blade burnup effect and have achieved an even better agreement between the calculated and measured values.

Some of the other improvements made to the model include:

- Use of more appropriate (s, α) thermal neutron scattering data; and
- The ageing effect of the beryllium reflector.

With the above model enhancements, the critical control blade height predictions have improved as shown in Table 2 below. In this Table, Estimated Critical Position (ECP) value predictions using the improved MURR-MCNP model for several weekly reactor startups are compared against the actual start-up critical control blade height data. Eight separate cores were selected for the comparison to verify consistency in the model's ability to predict the ECP accurately. Table 2 shows a span of data reported over eight (8) months. Note that reactor startups at MURR require an occasional "strainer" start up, where initial critical data is taken without any samples or sample holder in the central island tube (just a water volume). Two such strainer startups are reported in the Table.

Table 2 – Comparison of Estimated Startup Critical Control Blade Height vs. Measured Data

Core Configuration (week of)	Actual Initial Critical Height (inches)	Predicted Initial Critical Height (inches)	K_{eff}	Flux Trap Configuration
1/28/2013	16.79	16.67	0.99993	Strainer
2/04/2013	16.52	16.27	0.99975	FT Samples
4/29/2013	15.98	15.78	1.00017	FT Samples
6/10/2013	15.44	15.42	0.99995	FT Samples
8/05/2013	16.74	16.74	0.99985	Strainer
8/12/2013	15.71	15.61	0.99985	FT Samples
8/19/2013	15.84	15.84	1.00016	FT Samples
8/26/2013	15.64	15.69	1.00029	FT Samples

A negative bias of ~1.5% is seen in the predictions for the early benchmarks. After the additional refinements to the MURR-MCNP model listed earlier were made, the variations in the predictions were within ±0.8% of the actual critical control blade heights (last 5 entries of the Table).

b. *Provide a description of how the MCNP results were used to arrive at the limiting peaking factors detailed in Table F.4 of Attachment 11 (ADAMS Accession No. ML11237A088). Specifically, provide confirmation that the week 58 case was used, indicate the agreement between the case and any measurements obtained, explain how the control blades were modeled, and provide information indicating if the peaking factors values determined were limiting values.*

The week 58 core as shown in Table 3-13 of Reference 10 of Attachment 11 (revised Appendix F, *Safety Limit Analysis for the MURR*) consists of four pairs of fuel elements with the following power histories: fuel elements 1 and 5 [0 MWD per element]; fuel elements 2 and 6 [81MWD per element]; fuel elements 3 and 7 [65 MWD per element]; and fuel elements 4 and 8 [142 MWD per element]. The Safety Limits (SL) are based on the week 58 Core 3B, which corresponds to a xenon-free start up and with an empty island tube or flux trap (no sample holder inserted into the inner pressure vessel, only a water volume as described as a "strainer" startup above). This case results in the highest peaking factors in plate 1 of fuel element 1 and coolant channel 2 between fuel plates 1 and 2. The reactor occasionally operates a few minutes at a critical low power (< 50 kW) with no samples in the inner pressure vessel in order to obtain an initial criticality measurement. This measurement is then used to calculate the reactivity worth of either an empty flux trap sample holder or more typically a flux trap sample holder loaded with samples in a subsequent startup at the beginning of a week of operation.

Results for Core 3B with Equal Angle Stripes (calculated for 2010 Fuel Conversion Feasibility Study Analysis)

Enthalpy Rise in Channel 2			
Nuclear Peaking Factors	Plate1	Plate 2	Average
Fuel Plate 1, 2 and Average	2.2150	1.7536	1.9843
Azimuthal in the Channel			1.0921
Additional Allowable Factor			1.0620
Overall Nuclear Peaking Factor			**2.3014**
Two Additional Engineering Hot Channel Factors			
Fuel Content Variation			1.0300
Fuel Thickness/Width Variation			1.0300
Overall Peaking Factor			**2.4416**

On Heat Flux From Fuel Plate 1 of Fuel Element 1 in Week 58 Core 3B

Power-related Factors	For mesh intervals between the following inches down the fuel plate		
Mesh Interval Number	14 (13-14")	18 (17-18")	19 (18-19")
Nuclear Peaking Factors			
Fuel Plate (Hot Plate Average)	2.2150	2.2150	2.2150
Azimuthal in the Channel	1.070	1.070	1.070
Axial Peaking Factor	1.3805	1.2958	1.2266
Additional Allowable Factor	1.062	1.062	1.062
Overall Nuclear Peaking Factor	**3.4747**	**3.2615**	**3.0873**
Engineering Hot Channel Factors			
Fuel Content Variation	1.030	1.030	1.030
Fuel Thickness/Width Variation	1.150	1.150	1.150
Overall Peaking Factor	**4.116**	**3.863**	**3.657**

The highest Overall Peaking Factor (OPF) for the enthalpy rise in Channel 2, as stated in Table F.4 of Appendix F, is 2.4416. The SL is based on a peaking factor of 3.863 at mesh interval 18 with the worst combination of heat flux and enthalpy rise in the coolant channel. For Core 3B, this occurs four (4) inches further down the coolant channel from the peak heat flux. An Additional Allowable Factor (AAF) of 1.062 was included to account for potential variations in control blade burnup and positioning. Core 3B uses equal angle azimuthal stripes for tallying the heat flux results along the outer edges of the fuel meat. The angle for the outer azimuthal stripes for the fuel meat of plate 1 were the same as for plate 24, which results in the outer stripes of fuel plate 24 being more than 2.4 times as wide as that of fuel plate 1. This could reduce the calculated azimuthal peaking factor in the azimuthally longer fuel plates.

In 2012, explicit modeling of the worst case control blade burnup and height mismatch was completed and included in technical document ANL/RERTR/TM-12-30, "*Technical Basis in Support of the Conversion of the University of Missouri Research Reactor (MURR) Core from Highly-Enriched to Low-Enriched Uranium – Core Neutron Physics*." Section 4.2, "Power

Distribution for Steady-State Safety Margin Evaluations," describes how the various possible core conditions were considered to define 24 different core conditions modeled in MCNP to determine the limiting highly-enriched uranium (HEU) fuel core conditions. The various HEU fuel core conditions are provided in Table 4.2 of ANL/RERTR/TM-12-30 and the key hot-stripe and local peak heat fluxes for each of the 24 different core conditions are given in the associated Table 4.4. All 24 different core conditions include using a standard width 5 mm stripe on the outer axial edges of the fuel meat in all 24 plates as discussed in Section 2.2.5 of the same document.

Note that the detailed results for 16 of the 24 HEU reference cores that modeled control blade burnup and the difference in positioning height were not completed until after the application supporting License Amendment No. 36, which was submitted in May 2011. The 24 HEU reference cores have been searched and Core 3B2 yields the greatest OPF for the hot channel 2 enthalpy rise. The cores with an empty flux trap ("B" designation) have a higher thermal flux at the inner fuel plates than cores with the flux trap sample holder inserted with samples. This increases the peak power densities in the limiting fuel plates 1 and 2 relative to cores with a loaded flux trap sample holder. Therefore, Cores 1A1/1A2 are bounded by Cores 1B1/1B2, and so forth. Also, the cores with equilibrium xenon (e.g., 2B1, 2B2, 4B1, 4B2) are bounded by the cores with no xenon (1B1, 1B2, 3B1, 3B2). Core 3B2 bounds the 3B and 3B1 cores because fuel element X1 is the peak element and Core 3B2 has control blades 'A' and 'D' depleted and one (1) inch higher than blades 'B' and 'C.' The azimuthal centerline of fuel element X1 is located on the centerline between control blades 'A' and 'D,' so the combination of the higher height positioning and burnup increases the peaking on fuel element X1.

Table 4.2 lists the configuration of Core 3B2:

- Four sets of fuel elements with different burnups – Week 58, a typical MURR core;
- Time (Days) 0 – Xenon free;
- Flux Trap empty – only pool water inside the island tube (inner pressure vessel); and
- Control blades 'B' and 'C' have 0 years of burnup ("fresh") and are withdrawn to 16.216 inches whereas control blades 'A' and 'D' have 8 years of burnup and are withdrawn to 17.216 inches.

Core 3B2 is modeled with the more accurate equal width azimuthal stripes on the fuel meat outer edges of all fuel plates. For Core 3B2, the OPF for the channel 2 enthalpy rise is 2.4893 and the highest overall peaking factor is 4.0870 in fuel plate 1 with the AAF maintained at 1.062.

Results for Core 3B2 with Equal Width Stripes (calculated for 2012 Conceptual Design)

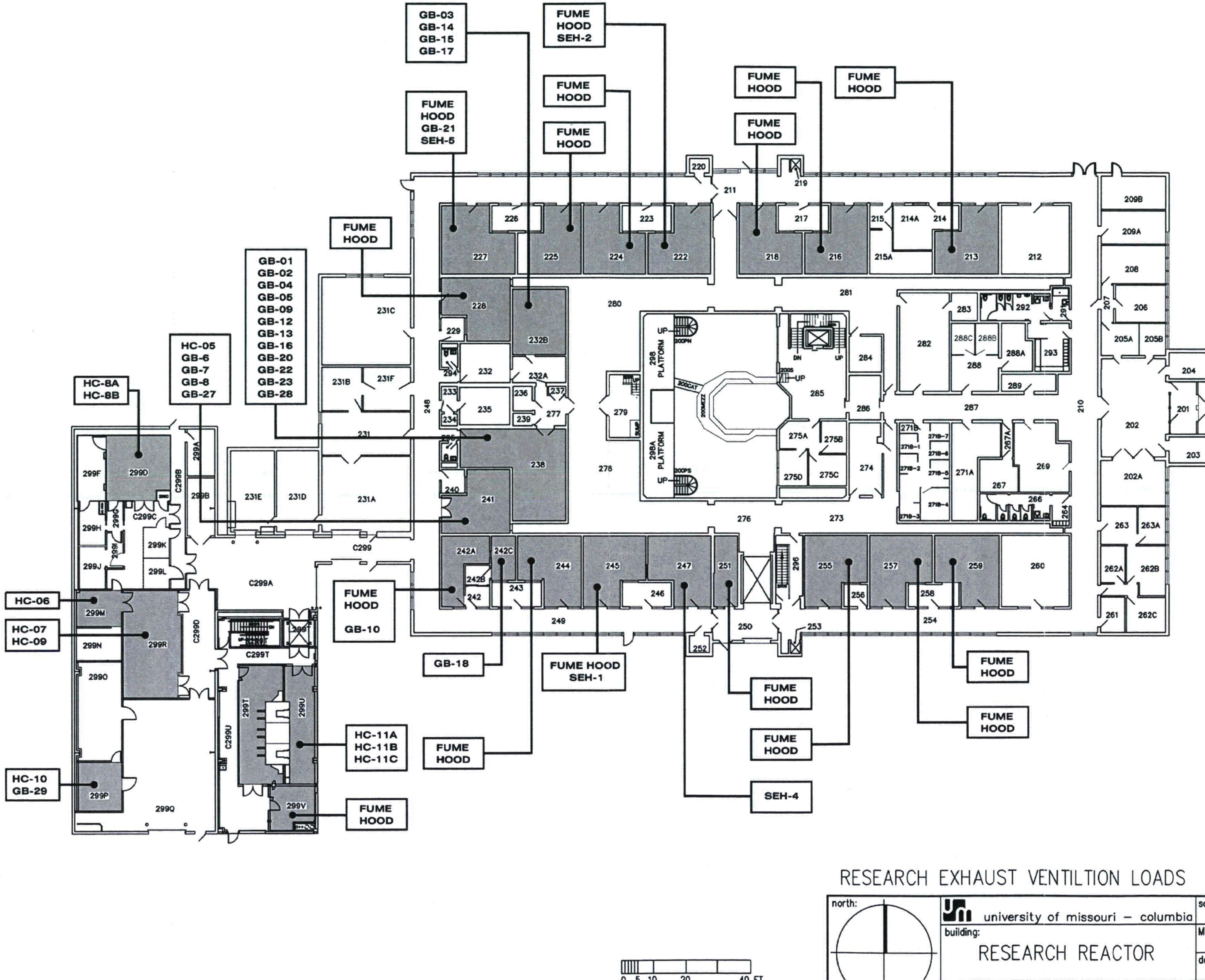
	Plate 1	Plate 2	Average
Enthalpy Rise in Channel 2			
Nuclear Peaking Factors	Plate 1	Plate 2	Average
Fuel Plate 1 and 2 Average	2.2806	1.8144	2.0475
Azimuthal in the Channel	1.0791	1.0791	1.0791
Overall Nuclear Peaking Factor (*with AAF = 1.0*)	**2.4610**	**1.9579**	**2.2095**
Additional Allowable Factor	1.0620	1.0620	1.0620
Overall Nuclear Peaking Factor	**2.6139**	**2.0793**	**2.3464**
Two additional engineering hot channel factors:			
Fuel Content Variation			1.0300
Fuel Thickness/Width Variation			1.0300
Overall Peaking Factor (*with AAF = 1.0*)			**2.3440**
Overall Peaking Factor (*with AAF = 1.062*)			**2.4893**

On Heat Flux from Fuel Plate 1 of Fuel Element 1 in Week 58 Core 3B2 with Worst Case Control Blade Arrangement

Power-related Factors	For mesh intervals between the following inches down the fuel plate			
Mesh Interval Number	14 (13-14")	16 (15-16")	18 (17-18")	19 (18-19")
Nuclear Peaking Factors				
Fuel Plate (Hot Plate Average)	2.2806	2.2806	2.2806	2.2806
Azimuthal in the Channel	1.060	1.060	1.060	1.060
Axial Peaking Factor	1.3440	1.3365	1.2447	1.1890
Additional Allowable Factor	1.062	1.062	1.062	1.062
Overall Nuclear Peaking Factor	**3.4504**	**3.4311**	**3.1955**	**3.0526**
Engineering Hot Channel Factors				
Fuel Content Variation	1.030	1.030	1.030	1.030
Fuel Thickness/Width Variation	1.150	1.150	1.150	1.150
Overall Peaking Factor	**4.0870**	**4.0642**	**3.7850**	**3.6158**

The OPF for the heat flux on fuel plate 1 of fuel element X1 in Core 3B2 is the greatest at axial mesh 14. This value is 4.087, when an AAF = 1.062 is included. With these overall peaking factors for the week 58 Core 3B2, the power SL with the other three variables (pressurizer pressure, core flow rate, and reactor inlet water temperature) at the LSSS values of 75 psia, 3200 gpm, and 155 °F is 14.906 MW. This SL is based on mesh interval 16, which has the worst case combination of enthalpy rise and peak heat flux at this interval. Table F.2 of Appendix F states 14.894 MW as the SL for the week 58 Core 3B with the other three variables at the LSSS values. Therefore, Appendix F is based on limiting values for the peaking factors that include worst case control blade arrangement and burnup history.

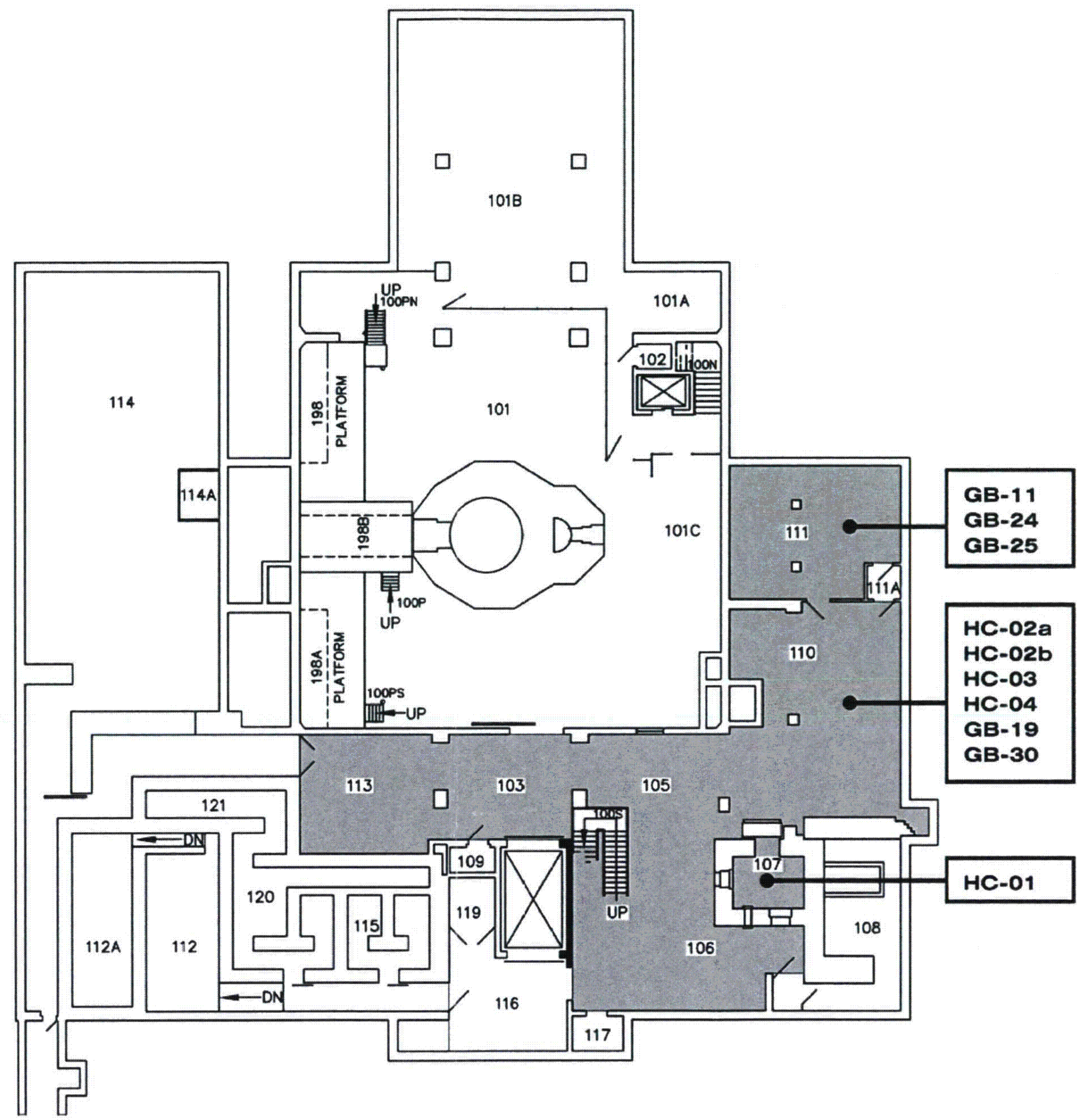
ATTACHMENT 1



RESEARCH EXHAUST VENTILTION LOADS

north:	university of missouri – columbia	scale: N.T.S.
	building: RESEARCH REACTOR	MURR drawing no.:
		date: 7/17/15
	level: GRADE LEVEL	sheet: 2 of 2

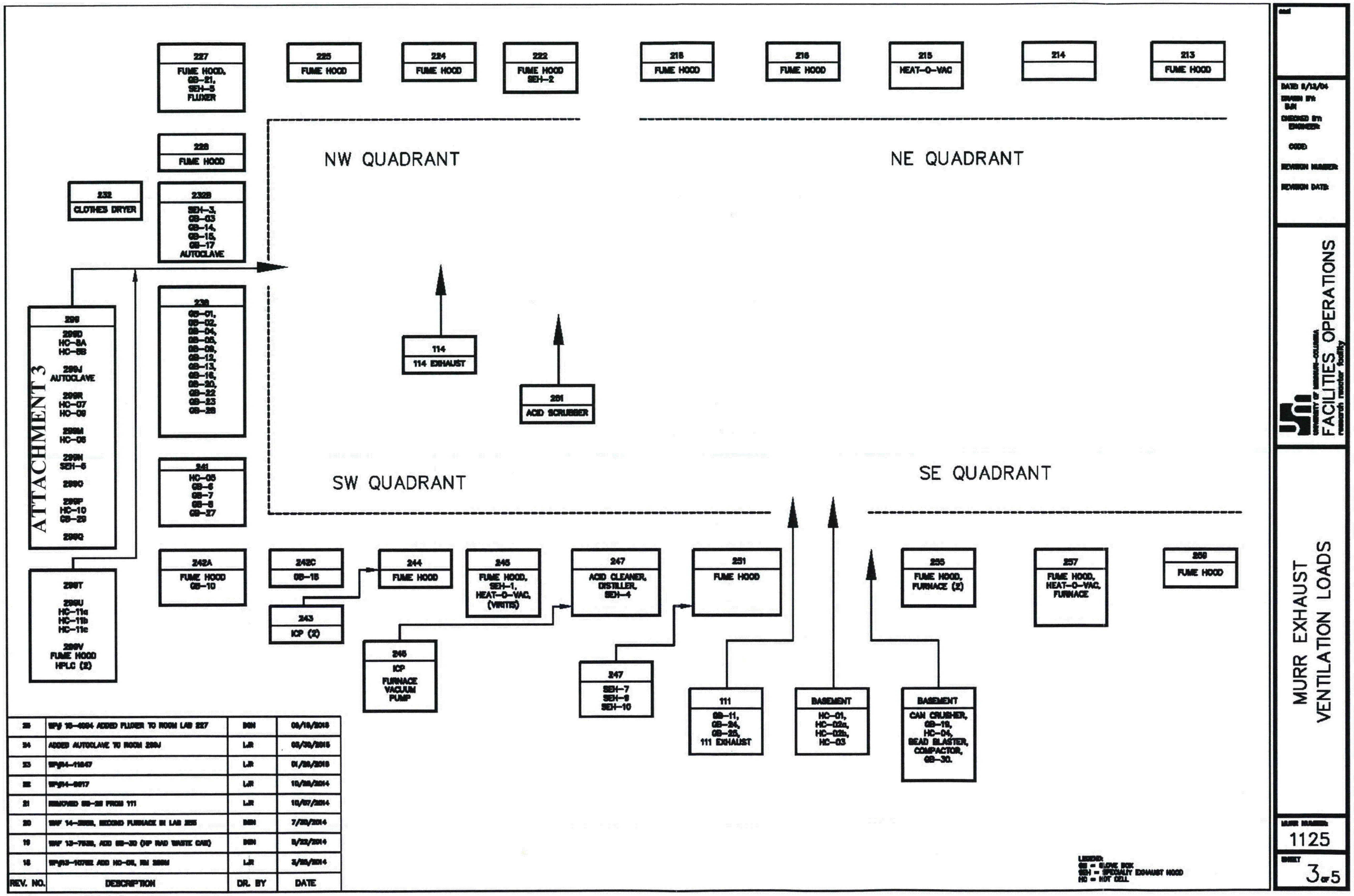
ATTACHMENT 2



0 5 10 20 40 FT.

RESEARCH EXHAUST VENTILTION LOADS

north:	university of missouri – columbia	scale: N.T.S.
	building: RESEARCH REACTOR	MURR No.
		date: 7/17/2015
	level: BASEMENT LEVEL	sheet: 1 OF 2



REV. NO.	DESCRIPTION	DR. BY	DATE
25	WP# 15-4004 ADDED FLUXER TO ROOM LAB 227	DGN	08/18/2015
24	ADDED AUTOCLAVE TO ROOM 299J	LJR	03/30/2015
23	WP#14-11647	LJR	01/26/2015
22	WP#14-9617	LJR	10/28/2014
21	REMOVED GB-28 FROM 111	LJR	10/07/2014
20	WAF 14-2558, SECOND FURNACE IN LAB 255	DGN	7/30/2014
19	WAF 13-7538, ADD GB-30 (HP RAD WASTE CAN)	DGN	5/22/2014
18	WP#13-10762 ADD HC-06, RM 299M	LJR	3/20/2014

ATTACHMENT 4

Project RL-72
Revision 7

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UNIVERSITY OF MISSOURI-COLUMBIA
MISSOURI UNIVERSITY RESEARCH REACTOR

PROJECT APPLICATION and HEALTH PHYSICS SAFETY EVALUATION for

UTILIZATION OF RADIOACTIVE MATERIAL/RADIATION UNDER MURR REACTOR LICENSE

1. **Project Leader:** Noah Garland

   Project Co-Leader: Mike Wilder

2. **MURR Affiliation:** PSO

3. **Project Name:** cGMP Lu-177 Chloride Processing

4. **Description of Radioactive Material/Radiation:**

Principal Radioisotope	Form	Quantity per Process	RUR	Note
Lu-177	Solid, liquid	100,000 mCi	a,b,c	Remote processing unit
Lu-177	Solid, Liquid	100,000 mCi	a,b,c	autoclave

In addition to the desired radioisotope, additional isotopes will be present in sample encapsulations due to activation during irradiation. High purity encapsulation materials are utilized to minimize those activation products. The dominant isotope produced in quartz vials is Si-31 (half-life 2.6 hrs). Dominant isotopes in aluminum encapsulation materials include Na-24, Co-60, Zn-65, and Cr-51. Activity levels at the time of handling will vary dependent upon the decay time between end of irradiation and processing.

RURs Controlling Irradiation:

a.	#295	Lanthanide Nitrates
b.	#278A	Lutetium Nitrate
c.	#278	Lutetium Oxide

5. **Location Requested:**

Work involving any quantity of uncontained material will be coordinated with Health Physics to ensure appropriate control. Work with greater than 100 μCi, but less than 475 mCi of uncontained material will be done in a fume hood. The current locations for processing of Quality Control samples include Rooms 244 and 246. Work with greater than 475 mCi of uncontained material must be performed in a glove box or approved dedicated autoclave.

Isotopes listed above are assigned to a specific glove box or processing unit in an area designated as a radiation area at MURR. HP notification and review are required before moving processes involving greater than 475 mCi.

Locations are adequate for this project and are Level I areas.

6. Purpose and Brief Description of Project:

MURR produces Lutetium-177 Chloride as an active pharmaceutical ingredient (API) suitable for use by researchers involved in early stage clinical trials. This product will be compliant with the applicable regulations in 21 CFR 210 and 211, collectively known as current good manufacturing practices (cGMP), and accompanying guidance in International Conference on Harmonisation (ICH) Q7A, "Good Manufacturing Practice Guidance for Active Pharmaceutical Ingredients." Lu-177 Chloride solution is produced at an activity concentration of 3 curies/mL or 5 curies/mL. The product is processed in a dedicated processing unit, and terminally sterilized in a dedicated autoclave. Approved procedures and HP supervision during process are required for activities greater than or equal to 100 mCi.

The isotope is delivered from Hot Cell HC-01 as a very small quantity of powder inside a small, sealed quartz vial. After washing in hot cell HC-04 to remove potentially harmful contaminants, this vial is placed inside a processing unit and opened. The powdered material is dissolved by pipetting in an acid and dispensed into a stock vial and diluted per an established procedure. The resulting stock solution is then withdrawn and dispensed into one or more delivery vials, and terminally sterilized in a dedicated autoclave.

Additional samples are analyzed in the quality control laboratories 244, 246, and microbiological facilities in quantities suitable for fume hood or benchtop handling.

Experience in this type of processing is extensive. Any eventual medical use of the radiochemicals from these processes will be conducted under separate license (transferee's license) which establishes the 10 CFR 35 requirements for its ultimate use in humans.

7. Special Facilities/Utilities/Equipment Required:

Remote Processing Unit:	Hot cells HC-08 A/B inside room 299D, GB-18 in 242C, or an equivalent facility may be used for quantities of Lu-177 up to 100 Ci.
Fume Hood:	Available in laboratories for low-activity dissolutions, analytical methods, etc.
Hot drain:	Available in Rooms 242, 244, 246.
Shielding:	Lead and/or Plexiglass bricks, containers, and shields inside the hot cell, and Plexiglass shields on tongs, pipettes, other tools as needed.
Heat Sources:	A hot plate or heating block is needed for some analyses. Such heat sources are isolated from combustibles, and power is turned off at the receptacle when no procedure is in progress.
Autoclave:	Dedicated autoclave in 299J or an equivalent facility may be used for terminal sterilization of quantities of Lu-177 up to 100 mCi.

Ionization monitors are used for dose rate measurements and these monitors are maintained under the MURR HP Instrument Program.

8. Handling Procedures for Radiation Safety Purposes:

Written documents related to handling of radioactive materials:

Procedures which are approved by the Project Leader and Health Physics Manager (or a draft procedure which is used under the control of an RWP) are required for all handling involving greater than 475 mCi. Batch records or checksheets and Health Physics checklists are completed during each process, providing documentation of activities, dose measurements, dosimeter readings, and other information relevant to the process.

Current procedures related to this project:

GMP-BR-205	Lutetium Chloride Batch Record
GMP-BR-213	HC-08 A/B Lutetium Chloride Batch Record
GMP-BR-217	Lutetium Chloride Batch Record for HC-08 A/B with Terminal Sterilization (number assigned)
GMP-BR-218	Terminal Sterilization of Lutetium Chloride (number assigned)
GMP-QC-252	Determination of Radiochemical Purity of Lu-177
GMP-QC-253	Lu-177 Identification and Determination of Radionuclidic Purity
GMP-QC-201	Determination of the Metal Content and Specific Activity of Lu-177 Chloride Radiochemical Solution
GMP-QC-021	pH Determination of Lu-177 Radiochemical Solution by Microcombination pH Probe
GMP-PRC-201	Transfer of cGMP Lu-177 Chloride Product to Shipping

Other procedures defining related facility and instrument maintenance, specifications for release of product, and other operations for the production of human-use radioisotopes require approval of Project Leader and Quality Assurance Manager.

Procedures and checklists associated with Lutetium are controlled documents and are available from Document Control, in designated controlled manuals, or on the MURR intranet.

Protective Clothing:

Protective clothing is enumerated in a dedicated SOP for gowning, GMP-QC-011, "Gowning Requirements for the MURR-Controlled Cleanrooms," and GMP-QC-010, "Aseptic Gowning Qualification."

Cleanroon Gowning will be worn:	As per above noted procedure.
Lab coat will be worn:	When in laboratory areas unless otherwise specified.
Gloves will be worn:	When handling radioactive samples or potentially contaminated tools and equipment.
Safety glasses will be worn:	When observing or performing work in the laboratory.

Monitoring requirements:

The radiation survey equipment in use at MURR are maintained by MURR Health Physics personnel.

Dosimetry is assigned to each person on a monthly or as needed basis. At a minimum, this will include a whole body badge and rings. Self-reading dosimeters will be worn whenever handling radioactive materials covered by this project.

An HP is present when transferring material into or out of the remote processing unit or autoclave. Procedures identify specific monitoring requirements for each process, including a dose rate approximately 6" above the open sample container taken before processing and general dose readings on the remote processing unit.

Processing personnel self-monitor using portal monitors or hand-held friskers after handling radioactive materials. Lead containers are swipe-checked for contamination before being transferred to the shipping department, and any area suspected of potential contamination is also checked in a timely manner.

Handling procedures are adequate. No additional emergency planning should be required beyond that described for general emergency response to a contamination event.

9. Administrative Controls and Training Requirements:

Class I training is required for authorized supervisors. Additionally, authorized supervisors must have comprehensive project-related knowledge and experience sufficient to enable them to conduct, direct, or train workers to perform project procedures.

Class II training is required for approved workers under this project. Additional task-specific training by an authorized supervisor is conducted in accordance with approved procedures.

10. Anticipated Radiation Doses and Contamination Levels:

Processing personnel doses from external radiation are typically less than 10 mRem whole body per process.

Anticipated contamination levels:

Minimal contamination is expected outside the remote processing unit from routine processing, since the samples are opened only within the remote processing unit. Lead containers are set on clean paper inside the airlock to reduce the likelihood of contamination. They are swiped in preparation for shipping, so any contamination should be detected and addressed before it spreads beyond the processing lab.

Contamination levels in using the established procedures have been minimal. Procedures address the requirements for monitoring and minimizing spread of contamination.

Quality Control processing of lower activity samples in hoods and benchtops has a higher potential for contamination but is also conducted under procedures containing additional precautions.

11. Transfer, Waste Production, and Disposal Requirements:

Shipment:

Typically all activity recovered in process (50-100% of activity produced) is shipped by MURR's shipping department, except for small aliquots (less than 100 mCi per sample for Lu-177) retained for quality control and retention samples.

Waste: Solid and Liquid

Volume: Solid: This amounts to approximately 12 rad waste bags of 1 ft^3 each per year from the entire project.

Liquid: Approximately 325 mL per process.

Description:

Solid: Gloves, pipette tips, plastic syringes, plastic needles, filters, tubing, caps, septa, and medicine cups. Glass/quartz waste is collected and handled separately

Liquid: Approximately 200 mL low-level contamination wash cup fluids, approximately 100 mL Lu-177 contaminated filter integrity test water and approximately 5 mL residual Lutetium stock solution.

Activity Involved: Solid waste represents the activity that is not recovered during process as well as contaminated laboratory materials as noted above.

Liquid waste activity varies from millicurie amounts to curie amounts.

Long-lived impurities which may affect waste disposal are:

Lu-177m (theoretical max ~ 250 μCi per Ci Lu-177 produced) 161 days

Special Handling procedures:

Liquid wastes are typically decayed to background, then disposed in the hot sink. If long-lived impurities are present at levels significantly above background, Health Physics direction is obtained and alternative disposal methods may be utilized.

Glass/quartz or contaminated needles/sharps are collected and discarded separately from the plastic and paper waste. All radioactive waste is sealed in plastic bags or airtight containers; labeled with the isotope, surface dose rate, date, and room number; and taken to the appropriate Radioactive Waste area designated by Health Physics.

Unprocessed irradiated targets are kept in lead containers for at least 10 half-lives of the major isotope. The person disposing of these decayed samples opens each container carefully and checks the dose rate above the container before removing and discarding the sample as described in the paragraph above. Longer decay and/or HP direction is indicated for dose rates over 25 mrad/hr.

Transfer of samples to other MURR areas requires that the sample be placed in a shielded container with a secured lid, as described in RP-HP-105, "Transfer of Radioactive Materials – In Facility" or an approved equivalent. If samples are being produced for eventual radiopharmaceutical use in patients, backup samples may be irradiated to increase assurance of availability of the radiochemical. If backup sample is not needed, then it will typically be allowed to decay prior to disposal as waste.

12. **Other Approvals/Authorizations/Interfaces Required:**

MURR's trained shipping technicians will verify that each recipient is licensed by NRC to receive the activity shipped and ensure that DOT regulations are followed in packaging, labeling, and shipping.

Procedures, equipment and/or specifications may require approval of recipient if an FDA product. Their requirements are typically based on FDA guidelines if the material is used as a component of a drug or medical device.

13. **Revision Analysis:**

This revision switches project leader and project co-leader.

14. I have read the MURR Radiation Worker Procedures and recognize its application to my requested project in the utilization of radioactive material/radiation under the MURR Reactor License. I recognize my responsibility as a project leader to inform and provide a safe work environment for individuals at MURR in accordance with University and NRC requirements. I recognize my responsibility to maintain proper and current documentation in regard to utilization of radioactive material/radiation under this project authorization.

Evaluation Conducted and Submitted By:

Health Physics: [signature] Date: 5-4-15

Project Leader [signature] Date: 5 May 15

Project Co-Leader: [signature] Date: 07 May 15

15. **Approvals:**

Reactor Manager: [signature] Date: 5-14-15

Health Physics Manager: [signature] Date: 5-12-15

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UNIVERSITY OF MISSOURI-COLUMBIA
MISSOURI UNIVERSITY RESEARCH REACTOR

PROJECT APPLICATION and HEALTH PHYSICS SAFETY EVALUATION for

UTILIZATION OF RADIOACTIVE MATERIAL/RADIATION UNDER THE MURR REACTOR LICENSE

1. **Project Leader:** David Robertson

   Project Co-Leader Leo Manson

2. **MURR Affiliation:** Research and Development

3. **Project Name:** Analyses of Annular LEU Target Fission Product Release during Encapsulation Opening (Y-12 Project)

4. **Radioactive Source: [NOTE: These activities are from an ORIGEN run of Mar 25, 2014 and are similar to the activities reported in RL-71,"99Molybdenum (Mo-99) Production (LEU Fission Product)" Rev.1, Apr 23, 2009]**

Principal Radioisotope	Form	Quantity per Activity
U-235	solid LEU foil	5 gm
Nobel Gases and Iodines	gaseous (fission products)	~85,000 mCi
Mixed fission and activation Products (Other)	Any	~232,000 mCi

Note: Above activities are based on processing with a 20 hr. decay post irradiation. Only a small fraction of the fission gas is expected to be released during the can opening as per TDR-0112. This experiment is an attempt, in part, in quantifying the release due to can opening.

RUR's Controlling Irradiation

#443 - Low Enriched Uranium With Aluminum Recoil Barriers
#441 – Uranium, Low Enriched Uranium Foil

Activities after 365 days decay:

Principal Radioisotope	Form	Quantity per Process
U-235	solid LEU foil	5 gm
Nobel Gases and Iodines	gaseous (fission products)	~2 mCi
Mixed fission and activation Products	Any	~1.0 Ci

5. **Location Requested:**

Hot cell HC-09 in the MURR Industrial Building (C299D) will be used as the remote handling facility for this project. HC-09 is shielded with multi-layer lead brick walls (200 mm thick) with a leaded glass window at the work site. HC-09 is equipped with a pass-through tray and manipulator arms for remote operations within the unit. Exhaust air from HC-09 passes through three activated carbon filters (bed depth 1 3/8 inches each) and two HEPA filters prior to entering the MURR exhaust system.

A side entry pass-through port will be used for introducing the LEU target to the cell, and any entries or removal of materials or analysis samples after the LEU target has be introduced to the cell will be conducted through this shielded port. A larger unshielded port may be used at Health Physics discretion prior to the introduction of the LEU material for setup and equipment placement. Specialized equipment has been designed specifically to open the irradiation can within a sealed volume to capture fission gasses released during opening of the irradiation can by use of a cold trap. Work conducted outside the hot cell involving 25 mCi of open or unshielded material will be coordinated with HP's to ensure appropriate control.

To achieve project ALARA personnel and effluent release target goals, the remote process in the hot cell is the necessary method for handling the irradiated LEU target, and its by-products. Personnel training and approval requirements will be reviewed and approved by appropriate supervisors and health physics personnel. The remote processing hot cell (HC-09) located in the MIB (C299D) will be dedicated and controlled as a primary facility for this project.

6. **Purpose and General Description of Project:**

The purpose of this project is to perform post irradiation examinations (PIE) of the annular targets with up to four types of recoil barriers; either aluminum and nickel which will be either plated onto or wrapped around the LEU targets. The PIE scope will include evaluating the magnitude of fission gas release during irradiation can opening and determining the contamination level of fission products on each type of recoil barrier. <u>There is no chemical separation process associated with this experiment.</u>

7. **Special Facilities/Utilities/Equipment Required:**

The hot cell filter bank on HC-09 is plumbed to the MURR exhaust stack. The filter bank consists of three activated carbon filters (each with a bed depth of 1 3/8 inches) and two HEPA filters. The flow through HC-09 is adjusted to approximately 40 to 60 cfm and balanced to maintain a negative (<0.5 in H_2O) pressure during process operations. For flow rates below 100 cfm, the design residence time of the charcoal filters (0.125 seconds) will be met or exceeded.

<u>Special Shielding:</u> All handling outside the processing hot cell will be done in shielded and/or ventilated areas approved by health physics. Target transfer from HC-01 (Main hot cell) to HC-09 will be via a DU transfer cask with a transfer procedure developed for both HC-09 and HC-07.

It is unknown at this time what fraction of the radioactive noble gases and iodine will be released from the target when the irradiation can is opened in HC-09. This amount is expected to be a small percentage of the total inventory available for release. This release analysis is presented in Attachment 1 represents the isotopes available for potential release at 20 hours post EOI. While the emission of the noble gases is likely to be slowed down through physisorption on the carbon filters, an instantaneous release of all the radioactive noble gases in the irradiated fission target results in exhaust stack releases that are well below the 24 hour release limit for these isotopes. The radioactive iodine will be captured on the three carbon filters in the exhaust stream of HC-09. Using a retention efficiency of 99.9% per carbon filter, the instantaneous release of all radioiodine from the fission target would result in exhaust stack releases that are well below the 24 hour release limits. For safety margins, the retention efficiency of 99.9% used in the release limit calculations is a factor of 10 lower than the 99.99% retention efficiency reported by the manufacturer for the carbon filters. A similar "worst case" scenario calculation for the instantaneous release of other fission products as particulates when the target is opened is also provided in Attachment 2. Using a retention efficiency of 99.9% per HEPA filter, the instantaneous release of these fission products as particulates would also result in exhaust stack releases that are well below the 24 hour release limits. It should be stressed, however, that the instantaneous release of all the fission products as particulates in the exhaust stream from the target is very unlikely as complete physical failure of the target would result in particulate spread throughout the hot cell.

8. Handling Procedures for Radiation Safety Purposes:

Written documents related to handling of radioactive materials:

Procedures approved by the Project Leader or Co-Project Leader and Health Physics Manager, or draft procedures with an RWP, are required for all activities involving >100 mCi. Corresponding checklists are completed for each performance of the procedure, providing documentation of activities, dose measurements, dosimeter readings, and other information relevant to the process. Completed procedures and/or checklists will be filed with document control in the project file for reactor license project RL-79.

The following activities will be controlled by their inclusion in the Experiment Plan that encompasses all the activities listed below. This Experiment Plan will incorporate Standard Operating Procedures where applicable as required by the Reactor Manager and Health Physics Manager.

- Fabrication and leak testing of LEU target
- Irradiation and temperature monitoring of LEU target
- Post Irradiation Target Handling and Transfer to Main Hot-Cell
- Target Transfer from Main Hot-cell to HC-09
- Target Disassembly
- Post Process Waste/Hot-cell Securing and Monitoring

Protective Clothing: Lab coats will be worn when performing or observing hot cell activities.

Gloves will be worn: When handling radioactive samples or potentially contaminated tools.

Safety Glasses: Will be worn when observing or performing work in the laboratory.

Monitoring requirements: Both TLD and EPD whole body dosimeters will be worn by all personnel conducting and/or observing the process. Ring dosimeters will be worn by personnel conducting the target transfers. Special air monitoring will be set up in C299D during the target transfer and gaseous release evaluation process. Health Physics personnel will be required to be present for all transfers into and out of the hot cell. Additional project specific dosimeters may be issued at Health Physics discretion.

The experiment procedure will be covered under an RWP by personnel from the health physics office. Procedures may identify additional monitoring requirements for specific steps of the experiment, including dose rates expected during target opening and fission product gas measurement as well as for any waste or samples being passed out of the shielded unit. An HP is required to be present whenever any material is introduced into or removed from the hot cell.

Project personnel must check out on frisker or portal monitor after handling radioactive materials before moving on to another activity. Transfer containers are swipe-checked for contamination before being transferred from the gaseous release evaluation process area. Any work area suspected of potential contamination will be swipe checked in a timely manner.

9. Administrative Controls and Training Requirements:

Class I training is required for authorized supervisors. Additionally, authorized supervisors must have comprehensive project-related knowledge and experience sufficient to enable them to supervise, and/or train workers to perform project procedures.

Class II training is required for approved workers under this project. Additional task-specific training by an authorized supervisor is conducted in accordance with an approved SOP.

Personnel doses from external radiation: Pool transfer cask dose rates expected to be < 80 mrem/hr upon removal from the pool [this dose rate was 50 mrem/hr for both previous LEU foil target transfers]. Dose rates on the DU container upon removal from the Hot Cell could be as much as a factor of 100 higher (8000 mR/hr)[these dose rates were 2000 and 3000 mrem/hr for the previous LEU foil target transfers] due to the thinner walls of the DU container compared to the pool cask. Dose rates at the HC-09 window and walls are expected to be approximately 5 mrem/hr respectively.

11. Transfer, Waste Production, and Disposal Requirements:

Waste Volume:	No liquid waste is expected for this experiment. Dry active waste (including wipes used for decontamination) is expected to be less than 1 cubic foot per experiment.
Fission Gasses:	Fission gasses captured via the use of a cold trap will be stored for decay and ultimate disposal. The cold trap is described in attachment.
Description:	Solid: gloves, plastic tubing and valves. "Sharps" and metal waste is collected and handled separately.
Activity involved:	After a 20 hour decay post irradiation the radioiodine and noble gas inventory is expected to be approximately 85 Curies. TDR-0112, "Assessment of Fission Gas Release During LEU Foil Target Irradiation

and Disassembly," indicates that less than 6% of fission gas atoms have the potential to escape the uranium foil. The fraction of the fission gas atoms that escape the uranium foil are captured by and retained by the fission recoil barrier. The fission gas release during the encapsulation opening is expected to be quite small and will provide challenges to detection and quantification. The quantification of the release is one of the activities for which this RL is proposed.

Long-lived: The irradiated LEU foil will be placed in a sealed glass container after collecting the fission gases. Long lived isotopes will be decayed for sufficient time and then sent off for disposal. It is anticipated that a greater than twelve month decay period will be utilized for sufficient decay of materials before introduction into the MURR general waste stream. A six month decay would yield approximately 2.5 Ci of residual activation and fission products while a decay of 1 year would yield approximately 0.90 Ci of residual activity and fission products. After one year decay the waste will have about 16 μCi of actinides. In order to ship as Class A waste, these actinides will have to be distributed within an accepted matrix media to meet that waste classification.

Retained gas samples: Gas analytical samples may be transferred to an appropriate laboratory for analysis if the gas concentrations are so small that the planned measurement equipment has insufficient sensitivity.

12. Other Approvals/Authorizations/Interfaces Required:

Procedural checklist for radioactive material handling will be reviewed and approved by the IUS prior to these activities. Refer to CAPs 08-0068, 13-0033 and 13-0041 for background information regarding improvements incorporated into this project.

13. Revision Analysis: This is a new project.

14. Attachments

Attachment 1: Predicted Fission Product Activities 20 hr

Attachment 2: RL-79 Release Analysis Sheet

15. I have read the MURR Radiation Worker Procedures and recognize its application to my requested project in the utilization of radioactive material/radiation under the MURR Reactor License. I recognize my responsibility as a project leader to inform and provide a safe work environment for individuals at MURR in accordance with University and NRC requirements. I recognize my responsibility to maintain proper and current documentation in regard to utilization of radioactive material/radiation under this project authorization.

Evaluation Conducted and Submitted By:

Health Physics: ________________ Date: 5-14-14

Project Leader: ________________ Date: 5/16/14

Project Co-Leader: Leonard H. Marsutter Date: 18-MAY-14

16. **Approvals:**

Reactor Manager: ________________ Date: 5-28-14

Health Physics Manager: ________________ Date: 5-19-14

Predicted Fission Product Activities (Ci)

Element	Isotope	EOI	20 HR	365 d
GE	77	1.74E-02	5.11E-03	0
AS	77	4.48E-02	3.43E-02	0
BR	83	3.16E+00	1.03E-02	0
KR	83m	3.16E+00	4.55E-02	0
KR	85	1.83E-03	1.90E-03	1.79E-03
KR	85m	7.58E+00	3.48E-01	0
KR	87	1.54E+01	2.87E-04	0
KR	88	2.17E+01	1.64E-01	0
RB	88	2.18E+01	1.83E-01	0
SR	89	2.51E+00	2.49E+00	1.84E-02
SR	90	1.50E-01	1.50E-02	1.46E-02
Y	90	9.01E-03	1.02E-02	1.46E-02
SR	91	3.51E+01	8.16E+00	0
Y	91	2.43E+00	2.60E+00	3.84E-02
Y	91m	2.03E+01	5.19E+00	0
SR	92	3.58E+01	2.15E-01	0
Y	92	3.59E+01	2.34E+00	0
Y	93	3.86E+01	9.90E+00	0
ZR	95	2.66E+00	2.65E+00	5.50E-02
NB	95	1.69E-01	2.10E-01	1.17E-01
NB	95m	8.85E-03	1.03E-02	4.08E-04
NB	96	3.59E-03	1.98E-03	0
ZR	97	3.49E+01	1.54E+01	0
NB	97	3.50E+01	1.54E+01	0
NB	97m	3.31E+01	1.46E+01	0
MO	99	2.93E+01	2.38E+01	0
TC	99m	2.57E+01	2.26E+01	0
RU	103	2.08E+00	2.05E+00	3.73E-03
RH	103m	1.87E+00	1.85E+00	3.36E-03
RU	105	6.06E+00	2.76E-01	0
RH	105	5.36E+00	4.19E+00	0
RH	105m	1.70E+00	7.75E-02	0
RU	106	2.90E-02	2.90E-02	1.48E-02
RH	106	1.96E-01	2.90E-02	1.48E-02
PD	109	1.78E-01	6.41E-02	0
AG	109m	1.78E-01	6.41E-02	1.81E-18
AG	111	5.41E-02	5.03E-02	1.81E-16
PD	112	7.50E-02	3.76E-02	0
AG	112	7.50E-02	4.44E-02	0
AG	113	6.60E-02	4.85E-03	0

Element	Isotope	EOI	20 HR	365 d
CD	115	4.87E-02	3.78E-02	0
IN	115m	4.87E-02	4.10E-02	1.45E-10
SN	121	7.51E-02	4.48E-02	0
SN	125	3.05E-02	2.87E-02	1.97E-13
SB	126	1.64E-03	1.57E-03	6.00E-09
SB	127	5.38E-01	4.73E-01	0
TE	127	4.65E-01	4.40E-01	4.69E-04
TE	127m	1.88E-03	2.25E-03	4.79E-04
SB	128	6.54E-02	1.41E-02	0
SB	129	3.79E+00	1.55E-01	0
TE	129	3.41E+00	2.31E-01	2.91E-05
TE	129m	6.98E-02	7.12E-02	4.46E-05
TE	131	1.51E+01	3.00E-01	0
TE	131m	2.10E+00	1.33E+00	0
I	131	7.11E+00	6.79E+00	2.90E-13
XE	131m	1.45E-02	1.75E-02	1.62E-10
TE	132	1.91E+01	1.60E+01	0
I	132	1.92E+01	1.65E+01	0
I	133	4.00E+01	2.11E+01	0
XE	133	2.00E+01	2.11E+01	7.91E-20
XE	133M	9.40E-01	9.19E-01	0
I	134	4.54E+01	2.70E-05	0
I	135	3.77E+01	4.63E+00	0
XE	135	1.36E+01	1.25E+01	0
XE	135m	6.76E+00	7.42E-01	0
CS	136	9.55E-03	9.14E-03	5.59E-11
BA	136m	1.57E-03	1.51E-03	9.21E-12
CS	137	1.54E-02	1.54E-02	1.51E-02
BA	137m	1.61E-02	1.46E-02	1.43E-02
BA	140	1.13E+01	1.08E+01	4.16E-08
LA	140	7.78E+00	8.72E+00	4.78E-08
LA	141	3.5E+01	1.12E+00	0
CE	141	4.47E+00	4.58E+00	2.24E-03
CE	143	3.41E+01	2.26E+01	0
PR	143	7.52E+00	8.38E+00	1.27E-07
CE	144	5.24E-01	5.23E-01	2.19E-01
PR	144	5.41E-01	5.23E-01	2.19E-01
PR	144m	6.49E-03	6.27E-03	2.62E-03
PR	145	2.34E+01	2.32E+00	0
ND	147	4.61E+00	4.39E+00	8.16E-10
PM	147	1.19E-02	1.46E-02	5.08E-02
PM	149	5.57E+00	4.46E+00	0

PM	151	2.44E+00	1.51E+00	0
SM	153	8.79E-01	6.55E-01	0
SM	156	8.01E-02	1.84E-02	0
EU	156	2.03E-02	2.11E-02	1.77E-09
EU	157	3.84E-02	1.56E-02	0
NP	239	9.52E+00	7.51E+00	2.11E-19

Activities above 1 mCi at end of irradiation (EOI) and decayed activity at 20 hours and 365 days post EOI. These are the activities predicted by ORIGEN2 v 2.2 from the irradiation of a 5 gram LEU foil target at a thermal flux of 1.5E+13 n cm^{-2} s^{-1} for 160 hours.

Note of interest: This irradiation produces 5.83E-06 curies Pu-239 at 365 days post EOI.

Y-12 Potential Release (20 hour decay)

Element	Isotope	Half-Life	EOI (Ci)	20 HR Decay (µCi)	AEC (µCi/mL)	TS Factor	Adjusted AEC (µCi/mL)	Release Limit (µCi)	Ratio (unfiltered)
VOLATILES									
I	130	12.4 h	2.06E-03	6.74E+02	3.00E-09	3500	1.05E-05	1.30E+07	5.18E-05
I	131	8 d	7.11E+00	6.79E+06	2.00E-10	1	2.00E-10	2.48E+02	2.40E+04
I	132	2.3 h	1.92E+01	1.65E+07	2.00E-08	3500	7.00E-05	8.68E+07	1.90E-01
I	133	21 h	4.00E+01	2.11E+07	1.00E-09	3500	3.50E-06	4.34E+06	4.86E+00
I	134	53 m	4.54E+01	2.70E+01	6.00E-08	3500	2.10E-04	2.60E+08	1.04E-07
I	135	6.6 h	3.77E+01	4.63E+06	6.00E-09	3500	2.10E-05	2.60E+07	1.78E-01
KR	83M	1.9 h	3.16E+00	4.55E+04	5.00E-05	3500	1.75E-01	2.17E+11	2.10E-07
KR	85M	4.5 h	7.58E+00	3.48E+05	1.00E-07	3500	3.50E-04	4.34E+08	8.02E-04
KR	85	10.7 y	1.83E-03	1.90E+03	7.00E-07	1	7.00E-07	8.68E+05	2.19E-03
KR	87	1.3 h	1.54E+01	2.87E+02	2.00E-08	3500	7.00E-05	8.68E+07	3.31E-06
KR	88	2.8 h	2.17E+01	1.64E+05	9.00E-09	3500	3.15E-05	3.91E+07	4.19E-03
XE	131M	11.9 d	1.45E-02	1.75E+04	2.00E-06	1	2.00E-06	2.48E+06	7.06E-03
XE	133	5.2d	2.00E+01	2.11E+07	5.00E-07	3500	1.75E-03	2.17E+09	9.72E-03
XE	133M	2.2 d	9.40E-01	9.19E+05	6.00E-07	3500	2.10E-03	2.60E+09	3.53E-04
XE	135	9.1 h	1.36E+01	1.25E+07	7.00E-08	3500	2.45E-04	3.04E+08	4.11E-02
XE	135M	15.4 m	6.76E+00	7.42E+05	4.00E-08	3500	1.40E-04	1.74E+08	4.26E-03
XE	138	14.1 m	3.74E+01	0	2.00E-08	3500	7.00E-05	8.68E+07	0
PARTICULATES									
SR	90	29 y	1.50E-02	1.50E+04	3.00E-11	1	3.00E-11	3.72E+01	4.03E+02
Y	91	58 d	2.43E+00	2.60E+06	2.00E-10	1	2.00E-10	2.48E+02	1.05E+04
ZR	95	64 d	2.66E+00	2.65E+06	4.00E-10	1	4.00E-10	4.96E+02	5.34E+03
CS	137	30 y	1.54E-02	1.54E+04	2.00E-10	1	2.00E-10	2.48E+02	6.21E+01
BA	140	13 d	1.13E+01	1.08E+07	2.00E-09	1	2.00E-09	2.48E+03	4.35E+03
CE	141	32 d	4.47E+00	4.58E+06	1.00E-09	1	1.00E-09	1.24E+03	3.69E+03
CE	144	285 d	5.24E-01	5.23E+05	4.00E-11	1	4.00E-11	4.96E+01	1.05E+04

NP	239	2.4 d	9.52E+00	7.51E+06	3.00E-09	3500	1.05E-05	1.30E+07	5.78E-01

Exhaust of 1.24 E 12 mL/day. Adjusted AEC = TS Factor X AEC. Release Limit = 1.24E12 X Adjusted AEC.
Ratio = (20 HR Decay Activity)/ (Release Limit0. This is the ratio upstream of the filter bank and this ratio does not include the following decontamination factors of the filters:

1. Ratios for iodine with 3 carbon filters. Assumes 99.9% retention per filter. Manufacturer reports 99.99% iodine retention for product.
2. Ratios for krypton and xenon assumes no retention in carbon filters or HEPA filters.
3. Ratios for particulate with 2 HEPA filters. Assumes 99.9 % removal efficiency per HEPA filter.

Iodine and Noble Gases

Iodine — SOR (unfiltered) = 2.40E+04
SOR (3 charcoal @99.9% eff) = 2.40E-05

Noble gases — SOR (unfiltered) = 6.97E-02

Particulates

Particulates — SOR (unfiltered) = 3.48 E+04
SOR (2 HEPA @ 99.9% eff) = 3.48 E-02

Sum of Ratios (Iodine, Nobles and Particulate with Iodine and Particulate filter assumptions) = 1.05E-01

This is for total inventory which is an extremely unlikely release scenario.

RL 79- Annular LEU Target Transfer Procedure

HC-01 Area

1. Place DU transfer container in HC-01 for transfer of the Annular LEU target from main hot cell cask.
2. After removal of target from pool in pool transfer cask, place pool transfer cask into HC-01 and then close HC-01 inner and outer doors.
3. Transfer target from pool transfer cask to DU transfer cask.
4. After transfer is complete, open HC-01 doors with HP monitoring the HC door and the DU cask area.
5. Use forklift to carefully remove DU cask from hot cell HC-01.
6. Check dose rate on DU transfer cask and if acceptable, swipe and decontaminate if necessary, the DU transfer cask.
7. After placing yellow bag around DU container and lifting device, lower transfer cask onto cart and move transfer cart to HC-09 using the material lift to move cart upstairs.
8. Check floor area in front of HC-01 for contamination and decontaminate if necessary.

HC-09 Area

1. When at HC-09, position cart near hot cell access door (east side of HC-09) so that cask can be lifted by forklift onto tray that has been extended out from the hot cell access port area.
2. Carefully raise DU transfer container to level of extended door tray.
3. Carefully remove bag and move forklift and DU container into position to place onto tray.
4. Carefully place DU container onto tray.
5. Slowly back out forklift from access door area.
6. Remove lifting device from cask and carefully push tray into access port area.
7. Place lifting device in bag and place bag on cart.
8. Close outer door and check floor area for contamination.
9. Decontaminate if necessary.

ATTACHMENT 5

Y-12 Annular Target Opening
Gas Release Analysis

Objective:
Safely open annular target and determine fission gas release. Significant preparation has resulted in an enclosed remotely controlled target cutter, purged with helium, flowing to a carbon condenser maintained within liquid nitrogen (LN2) bath. The condenser is maintained in a LN2 Dewar constructed within the cell and Capintec chamber allowing for continuous assay collection while the gas is being purged from the cuter enclosure to the LN2 cooled condenser.

Experimental Protocol: (in conjunction with HP check sheet)

1. Position the **Annular Target** in the target cutter aligning it in the center of the two end cutter blades.
2. Close the door to the enclosure box, and drive can holder down to pinch can in position without touching can with the cutter blades.
3. Adjust Helium (He) purge gas to begin purging box through condenser.
4. Turn on heat tape to warm top of condenser tubing on inlet side.
5. Begin flow of LN2 gas while monitoring temperature at TC-1 (thermal couple at top of LN2 level), and TC-2 (thermal couple at bottom section of heat tape).
6. With Variac set at ~35 to warm heat tape TC-2 should read ~-50°C when TC-1 reaches and is maintained at ~-170°C. Very fine adjustments to the flow of LN2 will maintain temperature.
7. After the system has been stabilized and ~30 minutes of He gas has purged through the system, notify the MURR Control room that the target will be cut open.
8. Using the blade down control lower the cutting wheel blades until they just touch the can.
9. Turn on the rotation to rotate the can (both directions work – no preference).
10. Begin assaying the condenser collection tube with the HPGe detector by continuously counting ~ every 6 min. using 5 min. counts, and saving data.
11. After a couple of rotations of the can slightly slower the blades and continue this action until both ends of the can are cut off.
12. When both ends of the can are cut off discontinue can rotation.
13. Carefully align can longitudinal cut mark with bottom longitudinal cutter by jogging the can rotation button.
14. With proper positioning begin transverse cut of the can using the control for side to side motion of the bottom cutter.

   NOTE: BE CAREFUL NOT TO HIT THE CAN ROTATION CONTROL AS THIS WILL BREAK THE LONGITUDINAL CUTTING BLADE.
15. Once the bottom of the can is cut, release the can by releasing the pressure on the can pincher using the up command on the end blade/pincher control.
16. Continue the He purge, and Condenser HPGe assay counts until the activity on the condenser has stabilized.

17. After the count data appears to have stabilized turn off the LN2 and continue count until the activity peaks and begins to reduce.

NOTE: Most of the fission gas released to the hot cell and hot cell exhaust will occur as the condenser warms.

18. When the experiment is complete, open cutter enclosure box and move target to sealed glass bottle.
19. Close cutter enclosure box.
20. Allow He purge gas to continue at ~100mL/min. for the next 20 – 24 hours to clear the carbon condenser of any remaining fission gas.

ATTACHMENT 5

Health Physics Check List for
Analyses of Annular LEU Target Fission Product Release during Encapsulation Opening (Y-12 Project)

Date and time: ________________________ MURR ID number: ________________________

At least 5 minutes before removing the target from HC-01:

___ Turn on ambient and in-line AMS at HC-09.

___ Verify the HC-09 exhaust valves have been set to include charcoal filtration.

___ Record names and EPD readings for :

	Name	Reading
Processor	______________________________	EPD ____________________mR
Observer	______________________________	EPD ____________________mR
Observer	______________________________	EPD ____________________mR

___ All dosimeters are worn in their proper positions. (Whole body, Ring, EPD)

Immediately before transferring the target to HC-09:

___ Record stabilized ambient AMS reading: ________ cpm ________ LPM __________ time

___ Record stabilized in-line AMS reading: ________ cpm ________ LPM __________ time

___ Measure and record dose rates at HC-09 DU pass through __________________ mR/hr

Dose rate meter model and serial number: ________________________

___ Ensure all processing materials are prepared and in HC-09 and that side doors are closed and latched.

___ Record Magnehelic® differential pressure gauge values for hot cell HC-09 and filters.

Hot Cell DP, HC-09 #1: ______________ Charcoal Train, HC-09 #7: __________

Final HEPA, HC-09 #8: ____________

___ Record the dose rate at the window of HC-09 ___________mR/hr

As Transfer DU is removed from HC-01:

___ Record ~1 meter and contact readings on the DU.

~1 meter dose rate:____________ mR/hr Contact dose rate: ___________ mR/hr

___ Swipe DU and determine contamination levels on alpha capable instrument _____ cpm

___ Observe as DU is placed on cart within a plastic bag.

Health Physics Check List (Cont.)

As Transfer DU is loaded into pass through of HC-09:

___ Observe as DU is put into position and slid into pass through of HC-09.

___ Record dose rate door of pass through when DU is in pass through and door is closed.

dose rate: ___________ mR/hr

___ Swipe hoist hook and determine contamination levels on alpha capable instrument _____ cpm

___ Clean hook and re-swipe if necessary.

Immediately before opening the inner door and sliding the DU into HC-09:

___ Verify that the outer door of pass through is closed and latched.

As the inner door is opened and DU is moved into HC-09:

___ Observe dose rates at various points outside the hot cell. ________ time.

Location__________ dose rate: _______ mR/hr Location __________ dose rate: _______ mR/hr

Location__________ dose rate: _______ mR/hr Location __________ dose rate: _______ mR/hr

As the DU is opened:

___ Observe dose rates at multiple points outside the hot cell. ________ time.

Location__________ dose rate: _______ mR/hr Location __________ dose rate: _______ mR/hr

Location__________ dose rate: _______ mR/hr Location __________ dose rate: _______ mR/hr

___Record AMS reading: ________ cpm

As the target can is taken from DU and placed into target cutter:

___ Observe dose rates at multiple points outside the hot cell. ________ time.

Location__________ dose rate: _______ mR/hr Location __________ dose rate: _______ mR/hr

Location__________ dose rate: _______ mR/hr Location __________ dose rate: _______ mR/hr

___Record ambient AMS reading: ________ cpm

___Record in-line AMS reading: ________ cpm

Health Physics Check List (Cont.)

As the target can is cut open:

___ Observe dose rates at multiple points outside the hot cell. _______ time.

Location__________ dose rate: ______ mR/hr Location __________ dose rate: ______ mR/hr

Location__________ dose rate: ______ mR/hr Location __________ dose rate: ______ mR/hr

___Record ambient AMS reading: ________ cpm

___Record in-line AMS reading: ________ cpm

As the target is removed from the target can and placed into the collection vessel:

___ Observe dose rates at multiple points outside the hot cell. ______ time.

Location__________ dose rate: ______ mR/hr Location __________ dose rate: ______ mR/hr

Location__________ dose rate: ______ mR/hr Location __________ dose rate: ______ mR/hr

____Record ambient AMS reading: ________ cpm

____Record in-line AMS reading: ________ cpm

During the gas transfer and analysis process (repeat as necessary using additional sheets):

____Observe dose rates at multiple points outside the hot cell. ______ time

Location__________ dose rate: ______ mR/hr Location __________ dose rate: ______ mR/hr

Location__________ dose rate: ______ mR/hr Location __________ dose rate: ______ mR/hr

____Record ambient AMS reading: ________ cpm

____Record in-line AMS reading: ________ cpm

After the gas transfer process is completed and annular target has been placed in to storage bottle:

____Record final HC-09 dose rates. ________ time.

Window: ___________mR/hr AMS _________cpm

Health Physics Check List (Cont.)

____Record Magnehelic® differential pressure gauge values for hot cell HC-09 and filters.
Hot Cell DP, HC-09 #1: ______________ Charcoal Train, HC-09 #7: ____________

Final HEPA, HC-09 #8: _______________

____Record the dose rate at the window of HC-09 ___________mR/hr

____Record names and EPD readings for:

	Name	Reading
Processor	______________________________	EPD _______mR
Observer	______________________________	EPD ______ mR
Observer	______________________________	EPD ______ mR

Comments/Modifications: ____

____

Continue comments on back if necessary

Operator___________________________ H.P.___________________________ Date___________

During the gas transfer and analysis process (repeat as necessary):

____Observe dose rates at multiple points outside the hot cell. ________ time.

Location__________ dose rate: ______ mR/hr Location __________ dose rate: ______ mR/hr

Location__________ dose rate: ______ mR/hr Location __________ dose rate: ______ mR/hr

____Record ambient AMS reading: ________ cpm

____Record in-line AMS reading: ________ cpm

During the gas transfer and analysis process (repeat as necessary using additional sheets):

____Observe dose rates at multiple points outside the hot cell. ________ time.

Location__________ dose rate: ______ mR/hr Location __________ dose rate: ______ mR/hr

Location__________ dose rate: ______ mR/hr Location __________ dose rate: ______ mR/hr

____Record ambient AMS reading: ________ cpm

____Record in-line AMS reading: ________ cpm

Health Physics Check List (Cont.)

During the gas transfer and analysis process (repeat as necessary using additional sheets):

____Observe dose rates at multiple points outside the hot cell. ________ time.

Location__________ dose rate: ______ mR/hr Location __________ dose rate: ______ mR/hr

Location__________ dose rate: ______ mR/hr Location __________ dose rate: ______ mR/hr

____Record ambient AMS reading: ________ cpm

____Record in-line AMS reading: ________ cpm

During the gas transfer and analysis process (repeat as necessary using additional sheets):

____Observe dose rates at multiple points outside the hot cell. ________ time.

Location__________ dose rate: ______ mR/hr Location __________ dose rate: ______ mR/hr

Location__________ dose rate: ______ mR/hr Location __________ dose rate: ______ mR/hr

____Record ambient AMS reading: ________ cpm

____Record in-line AMS reading: ________ cpm

ATTACHMENT 6

Radioactive Liquid Releases to the Sanitary Sewer – Calendar Years 2005 to 2014

Isotope	2005	2006	2007	2008	2009	2010	2011	2012	2103	2014	Average
	(in Curies)										
H-3	9.29E-02	1.17E-01	9.99E-02	1.36E-01	5.12E-02	7.37E-02	9.80E-02	1.62E-01	9.82E-02	2.60E-01	1.19E-01
S-35	6.65E-03	3.37E-03	9.33E-03	6.70E-03	6.45E-03	4.65E-03	2.48E-02	4.84E-03	7.31E-03	1.23E-02	8.64E-03
Lu-177	2.48E-03	3.95E-03	1.85E-03	3.34E-03	4.73E-03	4.62E-03	9.17E-03	2.57E-03	6.62E-03	7.12E-05	3.94E-03
Co-60	1.36E-03	1.15E-03	3.12E-03	4.06E-03	3.26E-03	1.49E-03	2.59E-03	2.72E-03	2.26E-03	6.42E-03	2.84E-03
P-32	3.72E-05	2.78E-04	5.04E-04	6.68E-04	2.87E-03	7.94E-04	5.18E-03	1.16E-03	1.38E-03	1.26E-03	1.41E-03
Ca-45	1.37E-03	4.06E-04	6.22E-04	1.02E-03	2.51E-03	1.13E-03	6.20E-04	8.90E-04	1.56E-03	6.25E-04	1.07E-03
Zn-65	5.85E-04	6.24E-04	5.60E-04	1.64E-03	1.00E-03	4.67E-04	7.67E-04	1.15E-03	6.52E-04	1.93E-03	9.38E-04
Lu-177m	3.15E-04	6.44E-04	2.47E-04	4.91E-04	1.63E-03	6.51E-04		6.51E-05			5.78E-04
Mo-99								1.30E-04	9.86E-04	2.75E-04	4.64E-04
Ho-166				3.84E-04							3.84E-04
As-77	3.57E-04										3.57E-04
Sc-46	1.85E-05	5.26E-05	7.01E-04	3.91E-04	1.92E-04	1.61E-04	1.35E-04	1.32E-04	3.08E-05	1.44E-04	1.96E-04
Tc-99m								1.70E-04	2.66E-05	3.59E-04	1.85E-04
Tl-201	1.67E-04										1.67E-04
Cr-51	1.41E-04	8.83E-05	2.54E-04	3.02E-05	2.58E-04	5.12E-05		1.18E-04	5.56E-05	2.79E-04	1.42E-04
Ag-110m	3.72E-04	7.53E-05	5.00E-05	2.65E-04	1.19E-04	3.20E-05	2.32E-05		3.38E-05	2.89E-04	1.40E-04
W-181	9.18E-05		5.17E-05						3.13E-04	2.19E-05	1.19E-04
Mn-54	7.36E-05	1.00E-05	1.96E-04	4.28E-04	1.99E-04	2.58E-05	3.18E-05	2.99E-05		5.68E-05	1.17E-04
Sb-124		2.07E-05	1.65E-04				6.14E-05			1.19E-04	9.15E-05
Ru-105		2.96E-05			1.30E-04						7.95E-05
Ru-103										7.37E-05	7.37E-05
Rh-105		5.76E-05			6.02E-05						5.89E-05
Gd-159					4.91E-05						4.91E-05

ATTACHMENT 6

Radioactive Liquid Releases to the Sanitary Sewer – Calendar Years 2005 to 2014

Fe-59		3.15E-05	1.04E-04	1.70E-05				5.96E-05		2.42E-05	4.72E-05
Re-188	4.03E-05	3.09E-05					6.63E-05				4.58E-05
Cu-67				6.55E-05	1.28E-05						3.92E-05
Re-186			2.88E-05	3.21E-05							3.05E-05
Rb-86		2.14E-05				3.95E-05					3.04E-05
Sm-153			3.59E-05	1.44E-05							2.51E-05
Na-22					1.47E-05		3.06E-05				2.27E-05
K-42		2.24E-05									2.24E-05
Nb-95				2.06E-05							2.06E-05
In-115m										2.06E-05	2.06E-05
I-131				1.12E-05			3.05E-05		1.59E-05		1.92E-05
Au-198						1.53E-05					1.53E-05
Co-58			1.53E-05								1.53E-05
Gd-153				1.33E-05							1.33E-05
Sb-122				1.27E-05							1.27E-05
Co-57							1.24E-05				1.24E-05
Np-239		1.17E-05									1.17E-05
Cs-134										1.14E-05	1.14E-05
Cs-137										1.11E-05	1.11E-05
As-76									1.05E-05		1.05E-05

ATTACHMENT 7

Stack Effluent Releases – Calendar Years 2005 to 2014

Isotope	**2005**	**2006**	**2007**	**2008**	**2009**	**2010**	**2011**	**2012**	**2103**	**2014**	**Average**
	(% of Technical Specification Limit)										
Ar-41	76.6876	72.8113	78.3592	77.37	70.3004	58.0857	45.14	68.00	78.1054	74.2642	69.91238
C-14	0.777	0.74	0.793	0.7867	0.613	0.58	0.477	0.723	0.0083	0.0079	0.55059
Os-191	0.0011	0.0018	0.0066		4.1739	0.0294	0.0008	0.0003	0.0001	0.0002	0.46824
I-131	0.0921	0.0435	0.0401	0.0782	0.6035	0.0415	0.0506	0.0503	0.0169	0.2201	0.12368
Ce-144			0.1165	0.0852							0.10085
Co-60	0.0853	0.0792	0.3372	0.0784		0.0084		0.0049	0.0054		0.08554
H-3	0.0732	0.0521	0.0485	0.0527	0.0328	0.0353	0.0496	0.0426	0.0633	0.0558	0.05059
Kr-79								0.0482	0.0274		0.0378
Sc-46			0.0263	0.0022							0.01425
K-40			0.0093	0.0164					0.01		0.0119
Cd-109		0.0112									0.0112
I-125	0.0215	0.0041		0.0021	0.0073				0.0037		0.00774
Fe-59			0.0038								0.0038
Se-75	0.0005				0.0057						0.0031
Sb-125								0.0026			0.0026
Zn-65	0.0005	0.001	0.0026				0.0009				0.00125
Hg-203	0.0002	0.001	0.0002		0.0013					0.0033	0.0012
Cs-137	0.0007	0.0013	0.0006	0.0003		0.0004	0.0012				0.00075
Zr-95				0.0005			0.0005				0.0005
I-133	0.0003	0.0001	0.0001	0.0001	0.0003	0.0001	0.0001		0.0001	0.003	0.00047
Sn-113				0.0009					0.0003	0.0001	0.00043
Au-196	0.0005	0.0003	0.0004					0.0003		0.0004	0.00038
Gd-153				0.0003							0.0003

ATTACHMENT 7

Stack Effluent Releases – Calendar Years 2005 to 2014

Cu-67							0.0003				0.0003
Pa-233		0.0002					0.0003				0.00025
S-35		0.0001			0.0001	0.0005	0.0002				0.00023
Hf-181	0.0004	0.0001		0.0001				0.0002			0.0002
Ce-141	0.0003		0.0002	0.0001							0.0002
Xe-133										0.0002	0.0002
Ba-140		0.0003		0.0002	0.0001	0.0002					0.0002
Nb-95			0.0003		0.0001						0.0002
Br-82	0.0002				0.0001						0.00015
Co-58					0.0001	0.0001	0.0002				0.00013
As-77		0.0002			0.0001	0.0001					0.00013
Ce-139	0.0001				0.0001						0.0001
Ru-103	0.0001					0.0001				0.0001	0.0001
Mn-54			0.0001								0.0001
Be-7				0.0001							0.0001
Co-57					0.0001						0.0001
Hf-175						0.0001	0.0001				0.0001
Xe-135m									0.0001		0.0001