



ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE
8901 WISCONSIN AVENUE
BETHESDA, MARYLAND 20889-5603



December 15, 2010

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

SUBJECT: Request for Additional Information RE: License Amendment, Separation
of Byproduct Material (TAC NO. ME4117)

Sir:

By letter dated August 4, 2010, the NRC requested additional information concerning our license amendment request of June 16, 2010 (License R-84, Docket 50-170). Our responses to the six NRC questions are enclosed along with supplemental material as Attachments 1-4.

This proposed amendment has been evaluated against the criteria of 10 CFR 50.59(c)(2) and the amendment does not increase either the frequency or consequences of an accident previously evaluated in the FSAR.

The frequency of occurrence of an accidental release of radioactive material is not increased in this license amendment. All separations will be performed in the AFRRI Radiation Chemistry Laboratory which was constructed for the specific purpose of conducting radiochemical separations. The Radiation Chemistry Laboratory contains the pneumatic transfer system, hot cell, two fume hoods, and all necessary support facilities including safety showers, hot and warm drains, and radiation area monitors. The lab is maintained under negative pressure to prevent release of material into the surrounding areas and has pressure alarms to indicate any breach of this requirement. Most importantly, air exhausting from the lab is passed through high-efficiency particulate absorber and high-efficiency gas absorber filter banks which prevent the release of radioiodines from the reactor stack. Given these design features of the Radiation Chemistry Laboratory, laboratory scale separation experiments do not increase the frequency of an accidental release of radioactive material.

The consequences of an accidental release of radioactive materials are not increased in this separation experiment. As outlined in the AFRRI Technical Specifications section 3.6, each fueled experiment must be limited so that the total inventory of iodine isotopes 131 through 135 in the experiment is not greater than 1.3 curies and the maximum

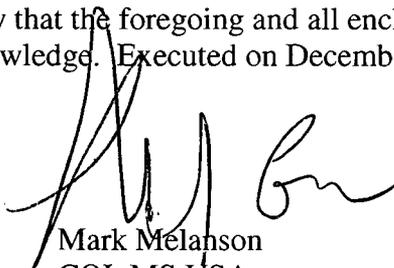
A020
NR/R

strontium-90 inventory is not greater than 5 millicuries. Since AFRRRI is not requesting an increase of these limits, the consequences of an accident remain unchanged.

The language being removed from subparagraph 2.B.(3) is not required by either the Atomic Energy Act or 10 CFR, but was originally inserted in various licenses to ensure that research reactors would not become production facilities. The language proposed for the new subparagraph 2.B.(4) will serve the same purpose and will also permit the vital experiments required to solve the national medical isotope production crisis.

Any questions should be addressed to Mr. Steve Miller at 301-295-1290 or millers@afrrri.usuhs.mil.

I declare under penalty of perjury that the foregoing and all enclosed information is true and correct to the best of my knowledge. Executed on December 15, 2010.



Mark Melanson
COL,MS,USA
Director

Attachments:

1. Dose Rate Calculations in the Radiation Chemistry Laboratory
2. Dose Rate Calculations in the Reactor Room
3. Experimental Protocol and Safety Features of MIPS Experiment
4. Proposed Limiting Condition of Operation and Surveillance Requirement

cc: U.S. Nuclear Regulatory Commission
ATTN: Al Adams, NRR/DPR/PRLB
Mail Stop 12-D3
Washington, DC 20555-0001

1. *Your license condition 2.B.(3) is an old version of the license condition that does not specifically allow for the use of byproduct material produced by operation of the reactor. Please propose wording to clarify that use of byproduct material produced by operation of the reactor is authorized.*

In subparagraph 2.B.(3), delete the words "but not separate" and add the words "and use." The sentence will now read "... and to possess and use such byproduct material as may be produced by operation of the reactor."

2. *Your proposed license condition 2.B.(4) and changes to license condition 2.B.(3) would allow separation on a laboratory scale. The first exception in the third part of the definition of production facility states "laboratory scale facilities designed or used for experimental or analytical purposes." Your proposed license condition does not restrict use to experimental or analytical purposes but states that the separations are needed for the development of a system for the production of molybdenum-99 from low-enriched uranium. Discuss why the research you plan to conduct is for experimental or analytical purposes. Please clarify your proposed license condition as to design or use for experimental or analytical purposes. Please discuss what is considered "laboratory scale" and why your proposed research is laboratory scale.*

Subparagraph 2.B.(4) will read "Byproduct material produced by operation of the reactor may be separated on a laboratory scale and used for experimental or analytical purposes."

The research involved in the development of a system for the production of Mo-99 from low-enriched uranium is performed on a laboratory scale and used for experimental or analytical purposes only. This means that experiments are performed with the intent of gathering scientific data and do not function to produce commercial quantities of materials. These laboratory scale experiments are used to gain information on a small scale which can be used as proof-of-principle for molybdenum isotope production.

3. *Your proposed license condition 2.B.(4) states that such separation will take place at the Armed Forces Radiobiology Research Institute (AFRRI) facility in Bethesda, MD, will not be in batch mode, and will involve no more than 100 grams of uranium enriched in the isotope uranium-235 per experiment.*

Please verify that the separations proposed in your application will be conducted under the authority of the reactor license and that the Radiation Chemistry Laboratory is within the area of the AFRRI complex under the authority of the reactor license. If the Radiation Chemistry Laboratory is under the reactor license, explain why the proposed license condition states that separations will take place at the AFRRI facility or remove this statement from the proposed license condition. A more specific license condition that reflects your applications is that separations will take place in the Radiation Chemistry Laboratory of the AFRRI facility.

It appears that you are proposing to meet the first exception of part 3 of the definition of production facility. Given that, why does your proposed license condition also refer to parts of

the third exception of part 3 of the definition of production facility? Please justify this wording or remove it from your proposed license condition.

Separations proposed in the application will be conducted under the authority of the reactor license. The Radiation Chemistry Laboratory is within the area of the AFRRRI complex under the authority of the reactor license. Subparagraph 2.B.(4) will read "Such separation will take place in the Radiation Chemistry Laboratory of the AFRRRI facility."

The proposed license amendment meets the first exception of part 3 of the definition of production facility given that the use and processing of irradiated materials containing special nuclear material will be on a laboratory scale used for experimental or analytical purposes. Subparagraph 2.B.(4) will read "Byproduct material produced by operation of the reactor may be separated on a laboratory scale and used for experimental or analytical purposes. Such separation will take place in the Radiation Chemistry Laboratory of the AFRRRI facility." References to additional exceptions of part 3 are removed from the license amendment wording.

4. *Technical Specifications (TS) 3.6b. controls the conduct of fueled experiments to limit doses if an experiment fails. You are proposing no changes to the TS as part of your request to perform separation of byproduct material in fueled experiments. Your applications states that the consequences of an accidental release are not increased as a result of your application to conduct separation experiments. While the inventories of radioactive material will not change as a result of your amendment request, it is not clear if the doses from accidental release of the inventory will remain the same. Please discuss the assumptions for the dose calculations (staff and public) and the results of the calculations that are the bases for TS 3.6b. How do these assumptions and results compare to accidents during separation activities? For example, releases that form the bases for TS 3.6b. and releases from separation activities may have different initiating points (reactor pool vs. laboratory hood), ventilation paths, release points to the environment etc.*

The calculation that forms the basis of TS 3.6b. discusses the "Maximum allowed release of I-131 from AFRRRI stack" and determines the maximum allowed activity of I-131 is 1.34 Ci. The assumption is that the I-131 is released in the reactor room, travels out of the stack, and is inhaled at the ground level.

The ventilation path to the reactor stack resulting from the failure of a separation experiment is less harmful than a failure of an experiment in the reactor room. This is because the charcoal filter is capable of absorbing the radioiodines before they are released from the stack. Once the exhaust air from the Radiation Chemistry Laboratory passes through the filter bank, it joins the reactor room ventilation system and releases through the same stack.

An accident during separation would occur in a fume hood within the Radiation Chemistry Laboratory. As currently configured, the flow rate at the face of the fume hoods with an 18 inch sash height is 125 cfm. Additionally, the Radiation Chemistry Laboratory currently experiences

28 room air changes per hour, with the exhaust air traveling through a HEPA and HEGA filter bank prior to release from the stack. Attachment #1 shows the intake levels from a staff member working in the Radiation Chemistry Laboratory at the time of an experiment failure for the proposed MIPS experiment. The levels are well below the inhalation ALIs, and these calculations do not take into account air exchanges, which would decrease these values of intake by several orders of magnitude. For this separation experiment, the irradiated samples will cool in the reactor pool for a minimum of 24 hours before transfer to the Radiation Chemistry Laboratory. Attachment #2 details the dose associated with fueled experiment failure immediately following irradiation with radioiodine levels at our licensed maximum. It is important to note that the fueled experiment is doubly encapsulated in a mock fuel pin at the bottom of the reactor pool, thus the unlikely failure of both layers of encapsulation immediately following irradiation would yield significantly lower radiation levels to a staff member separated by a minimum of 14 feet of water from the experiment. At this time, separations of fueled experiments with byproduct activities at these high levels are not planned. A description of how these calculations were performed is provided.

5. *The purpose of TS 3.6b. is to help ensure that the consequences of the accidental release of the fission product inventory of a fueled experiment is acceptable. While the consequences are acceptable, it is assumed that the failure of a fueled experiment will be a very rare occurrence. Likewise, the accidental release of the fission product inventory of a sample undergoing experiment or analysis as a separation experiment should also be a very rare occurrence. Your application states that laboratory scale experiments do not increase the frequency of an accidental release of radioactive material. Please discuss the features of separation experiment design that will help ensure that a failure that would release radioactive material from the experiment will be a rare occurrence.*

The experimental design features that ensure a release of radioactive material from an experiment during separation will be a very rare occurrence are described in Attachment #3. A summary of these features is provided below:

Several different irradiations will be performed in support of proof of principle experiments. Solutions of LEU (19.8% U-235) and/or DU (0.22% U-235) as uranyl nitrate will be prepared for use in irradiation tests at AFRRRI using a TRIGA reactor. These solutions will be irradiated in-core to allow various chemical and physical measurements in support of proof of principle tests for MIPS (Medical Isotope Production System) for domestic production and purification of Mo-99 from a LEU uranyl nitrate solution for medical use. The irradiated solutions will be used to examine solution behavior (both physical and chemical) during and after irradiation.

Small volumes (1 – 5 mL) of LEU and/or DU solutions will be irradiated. The solutions will be kept in quartz vials with plastic valves for gas analysis, and two vials will be irradiated at a time. The two quartz vials will be placed in an aluminum sleeve inside an aluminum mock fuel pin, which will serve as the secondary container. The secondary container is sealed at the top end by a viton O-ring on the top plug. The aluminum mock fuel pin will be placed in a specially

fabricated lead-shielded cask for transport to the radiation chemistry laboratory. Dose rate calculations have been performed for the solutions that will be the most and least radioactive. The samples will not be handled until 24 hours after irradiation for all irradiated solutions.

There are several different safety features that are included in the experimental plan to minimize the risk for release of radioactive materials. All radioactive materials will be doubly contained at all times beginning with insertion of the mock fuel pin into the reactor pool. Acidic absorbent paper will be utilized in the hoods and underneath the lead cask during sample removal as an added safety precaution if a spill occurs. Additionally, radioactive gases will be collected in a collection vessel.

Irradiated solutions will be doubly contained at all times. Specially fabricated tungsten shielding containers will serve as secondary containers for the samples when they are in the quartz vials or scintillation vials. When transfers are made out of hoods, irradiated solutions will be kept in shielding containers, trays, and sealed zip lock bags to prevent escape of radioactive materials.

With the quartz vial securely contained in its tungsten shielding container, the valve on the top of the quartz vial will be connected to the vacuum manifold setup, which is connected to the GCMS instrument. The vacuum manifold setup will be utilized to inject gas samples into the GCMS, and it will also be used to pump the gases down in the quartz vial to remove all radioactive gases and fill each quartz vial with helium. The quartz vials containing the irradiated solutions will be pumped down until the chromatogram on the GCMS shows that only He remains in each quartz vial. This will ensure that when the quartz vials are opened in the hood to remove the liquid irradiated solution, no radioactive gases can be released. All gases will be collected in a gas collection vessel and pumped into an empty gas cylinder.

The experimental plan has been designed to minimize the risk for a spill to occur. However, if a spill were to occur, a clean-up plan has been developed. If a spill were to occur in the hood or on the floor, the appropriate safety personnel would be contacted immediately. The person responsible for chemical spills is Barry Bender, and the person responsible for radiological spills is AJ Teachout. Additionally, large quantities of acidic absorbent paper are stored in the radiation chemistry laboratory, and the absorbent paper would be placed over the spill to aid in its cleanup and prevent it from spreading. Acidic and basic spill kits are also present in the lab and will be used for any non-radiological spills. If a radiological spill occurs, AJ Teachout (RSO) would be contacted and AFRRI's procedure for cleaning up radioactive spills will be followed.

The materials selected for the design of the irradiation vials and their ability to withstand the desired level of radiation are described below:

Gas sampling valve design and materials

The sample vial is equipped with gas sampling valve that seal sample during irradiation and allow drawing of the head space gases for analysis. Design of the valve is based on the standard

vacuum valve design. It uses a double o-ring seal on the stem and direct polymer to glass seal for the isolation of the gas sample.

Valve construction

The sample vial with valve consists of four parts: glass vial with thread, valve stem, nut, and set of o-rings. The principle design of the valve is depicted on the figure 1. The nut has two threads: left and right handed. When nut rotates, it pushes or pulls the stem. The stem slides on the set of two o-rings. The tip of the stem is coned to match saddle in the quartz vial body.

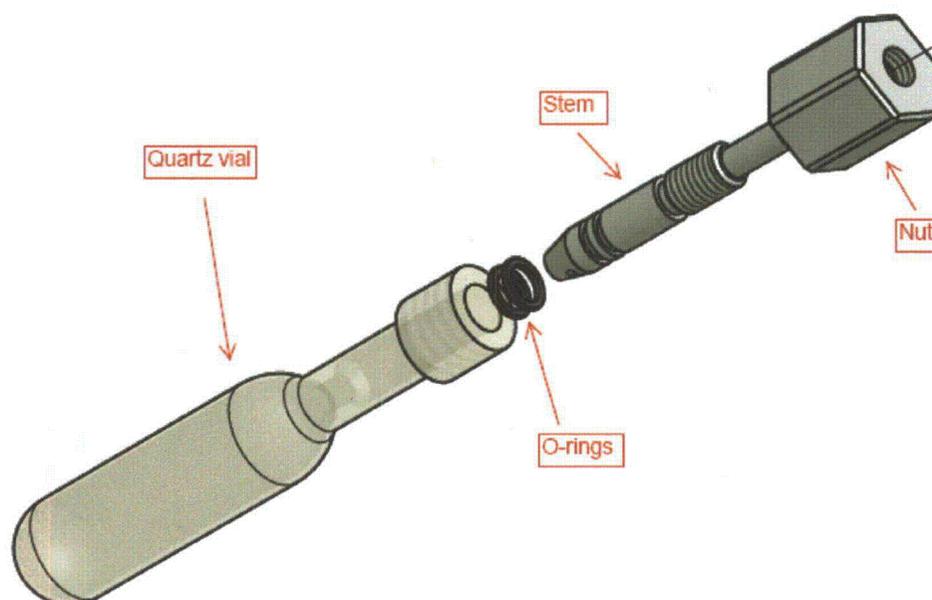


Figure 1. Sample vial design.

Choice of valve materials.

The commercial valve is manufactured using fluorinated polymers that provide good mechanical properties and great chemical resistivity but has poor radiation stability. Under radiation fields, fluorinated plastics lose mechanical integrity, so we had to replace materials used in the valve for more radiation resistant ones. In selection of the materials, we have used the guidelines from [1]. This book contains data for radiation stability of the polymers under x-ray irradiation. We have chosen medium density polyethylene for stem and for the nut manufacture and used Buna-N for the o-rings. Medium density polyethylene was chosen for the stem because of the mechanical properties; it is soft enough to make a reliable seal, but rigid enough to be machined. Buna-N o-rings were chosen because of the better radiation stability compared to

Viton. We have irradiated all materials at the Van de Graaff electron accelerator, and all materials have shown no degradation of the mechanical properties for radiation doses up to 200 MGy.

One can argue that inside of the reactor, the valve will be in the high intensity neutron field; however, there is discussion in the literature on applicability of the results obtained for one type of radiation for different forms of radiation on the stability of the polymers. Unfortunately, in the open literature, there are only a few studies of the mechanical stability of the polymers in neutron field. There are many studies that address changes in the physical properties of the polymers under different forms of radiation, but we are only interested in properties that affect mechanical stability. We were able to find one manuscript that compares the effects of different form of radiation on mechanical stability of the polymers [2]. According to this study, there is no significant degradation of the mechanical properties for polyethylene under proton, gamma, and neutron irradiation for the doses up to 20 Mrad. This literature review gives us confidence that the materials used in the sample vial with this valve will withstand irradiation in the reactor and ensure integrity of the valve. Each sample container will be used only once and then discarded.

References

1. Liesl K. Massey, The effects of sterilization methods on plastics and elastomers: the definitive user's guide and databook, William Andrew, 2005
 2. Guang-hou Wang, et. al. Proton beam modification of isotactic polypropylene. Nuclear Instruments and Methods in Physics research, B27, 1987
6. *Your application discusses some features of the Radiation Chemistry Laboratory. Some of these features appear to function as engineered safety features to limit the consequences if a separation experiment were to fail. It is not clear if the radiation area monitors discussed in your application are required by TSs. Please address. If not, please propose TS limiting conditions for operation and surveillance requirements. Please provide details on the ventilation system of the Radiation Chemistry Laboratory and its fume hoods. Discuss TS requirements to help ensure operation of equipment when required (e.g., ventilation fans, dampers and filters) and surveillances to help ensure equipment will be operable when needed. Your application states that the laboratory is maintained under negative pressure and contains alarms. Should this be a TS requirement when separation experiments are being performed? Please discuss. How would experiment failures be detected and releases controlled?*

A Limiting Condition of Operation and Surveillance Requirement have been added to the Technical Specifications and are provided in Attachment #4. The additional LCO ensures that the public and staff are not endangered by the discharge of radioactive materials in the event of an experiment malfunction and ensures that adequate radiation-monitoring equipment is available. The SR associated with the separation of byproduct material makes certain that all

radiation-monitoring equipment and ventilation systems are operating before initiating a separation experiment.

The Radiation Chemistry Laboratory consists of the main lab area in Building 42 Room 2165 of the AFRRRI complex. Within this main lab area, there are two fume hoods. Each hood is certified annually to have a flow across the sash of 125 cfm with the sash height at 18 inches. Additionally, there is a hot cell attached to the main lab area (labeled as Room 2166 in the FSAR). The hot cell is kept negative relative to the main lab area, which is in turn kept negative relative to the surrounding hallway. Within Room 2165 (3295 ft³), there are more than 20 air changes per hour. Air exhausting from the hot cell, main lab area, and fume hoods flows through stainless steel welded ductwork and passes through a high-efficiency particulate absorber (HEPA) filter and high-efficiency gas absorber (HEGA) filter. The HEPA filter captures 99.97% of particles 0.3 micron or larger. The HEGA filter has a mechanical efficiency of 99.9% and is used to remove gaseous contaminants from an airstream. The HEGA filter contains a nuclear grade carbon absorber which absorbs organic radioiodines, thus preventing their release from the stack. The differential pressure across the filters is measured in real time and there is a positive self-sealing damper downstream of the filters that can be used to prevent the flow of air to the stack. Additionally, there are two more dampers downstream surrounding the exhaust fan.

Releases of radioactive material associated with the accidental failure of a separation experiment would be controlled by the filter bank. Additionally, the radioiodine stack monitor will detect the presence of radioiodines and automatically cause the positive self-sealing dampers to close, preventing release to the public.

ATTACHMENT #1

Dose Rate Calculations in the Radiation
Chemistry Laboratory from Separation
Experiment Failure

ORIGEN model determination of nuclide content after irradiations at AFRRRI

In support of the Medical Isotope Production System (MIPS) project at Babcock and Wilcox (B&W) and Argonne National Laboratory (ANL), AFRRRI has irradiated three mock fuel pins with two sealed quartz vials in each that contained an LEU nitrate solution. In order to deal with the irradiated solutions safely, modeling of the fission and activation product content of the samples was performed using the ORIGEN depletion code. However, when the nuclide content was obtained through ORIGEN, an unexpectedly high burnup value was reported. The radionuclide quantities were corrected to the appropriate planned U burnup in the samples and those quantities of radionuclides were fed to MicroShield version 8 to model the dose rates as a function of time. In the example below (taken from an applicable ORIGEN output file), a $5.55\text{E-}07$ MWD burnup converts to 48 kW, which is twice the intended 24 kW exposure, and so the radionuclide quantities were divided by two for input to MicroShield. Though both the power in the sample ($1\text{E-}05$ MW, or 10 W, or 1 W/mL) and the flux ($6.99\text{E+}11$ n/cm²) are at the desired values, the burnup did not correlate

An independent validation of the dose rate using the statistical Way-Wigner method was performed by Dr. Russell Ball, with the predicted dose rates showing ~30% higher dose rates than the MicroShield calculations. This level of agreement led to a reasonable corroboration of the isotope amounts and the burnup correction applied to them.

Dose Calculations - Argonne National Laboratory LEU irradiation (MIPS)

23-Aug-10

A.J. Teachout

		cubic centimeters		cubic centimeters
Room 2165 (Rx Hot Laboratory)*			Room 3161 (Rx Deck)*	
Total cubic feet	3295	93304009.5	Total cubic feet	23,910
Supply air (CFM)	1150	32564373.5	Supply air (CFM)	2895
Exhaust air (CFM)	1551	43919429.06	Exhaust air (CFM)	3561
Air changes/hour#	28		Air changes/hour#	8.94
Iodine-131 Gamma Ray Dose Constant =		282.94 mR/hr/Ci @ 1 meter		
Ar-41 Gamma Ray Dose Constant=		695.97 mR/hr/Ci @ 1 meter		
Reference Breathing Rate =		1,200,000 ml/hr		

Concentration of isotopes in 2165 W/O air exchange			Repeated for 3161			Inhalation
		Intake@1 hr	W/O air exchange		Intake@1 hr	ALI in uCi##
Kr-83m	2.15E-07 uCi/cc	0.26 uCi	Kr-83m	2.97E-08 uCi/cc	0.04 uCi	Submersion
Kr-85m	1.83E-06 uCi/cc	2.20 uCi	Kr-85m	2.53E-07 uCi/cc	0.30 uCi	Submersion
Kr-87	9.15E-10 uCi/cc	0.001 uCi	Kr-87	1.26E-10 uCi/cc	0.000 uCi	Submersion
Kr-88	8.83E-07 uCi/cc	1.06 uCi	Kr-88	1.22E-07 uCi/cc	0.15 uCi	Submersion
Sr-89	1.14E-06 uCi/cc	1.36 uCi	Sr-89	1.57E-07 uCi/cc	0.19 uCi	800
Sr-90	6.78E-09 uCi/cc	0.01 uCi	Sr-90	9.35E-10 uCi/cc	0.001 uCi	20
Sr-91	2.95E-05 uCi/cc	35.37 uCi	Sr-91	4.06E-06 uCi/cc	4.87 uCi	6000
I-131	3.67E-06 uCi/cc	4.40 uCi	I-131	5.05E-07 uCi/cc	0.61 uCi	50
I-132	1.32E-05 uCi/cc	15.82 uCi	I-132	1.82E-06 uCi/cc	2.18 uCi	8000
I-132m	9.01E-12 uCi/cc	0.000 uCi	I-132m	1.24E-12 uCi/cc	0.00 uCi	8000
I-133	4.21E-05 uCi/cc	50.54 uCi	I-133	5.80E-06 uCi/cc	6.97 uCi	300
I-133m	1.03E-12 uCi/cc	0.000 uCi	I-133m	1.43E-13 uCi/cc	0.000 uCi	300
I-134	4.06E-11 uCi/cc	0.000 uCi	I-134	5.60E-12 uCi/cc	0.000 uCi	50000
I-135	2.05E-05 uCi/cc	24.56 uCi	I-135	2.82E-06 uCi/cc	3.39 uCi	2000
Xe-133	7.64E-06 uCi/cc	9.17 uCi	Xe-133	1.05E-06 uCi/cc	1.26 uCi	Submersion
Xe-133m	4.96E-07 uCi/cc	0.60 uCi	Xe-133m	6.84E-08 uCi/cc	0.08 uCi	Submersion
Xe-135	5.78E-05 uCi/cc	69.32 uCi	Xe-135	7.96E-06 uCi/cc	9.55 uCi	Submersion
Xe-135m	3.34E-06 uCi/cc	4.01 uCi	Xe-135m	4.61E-07 uCi/cc	0.55 uCi	Submersion
Xe-138	3.17E-34 uCi/cc	0.000 uCi	Xe-138	4.37E-35 uCi/cc	0.000 uCi	Submersion

Let us look at the concentrations and intakes if we assume double the quantity of radioactive material in the Argonne Experiment:

Concentration of isotopes in 2165 W/O air exchange			Repeated for 3161			Inhalation
		Intake@1 hr	W/O air exchange		Intake@1 hr	ALI in uCi##
Kr-83m	4.31E-07 uCi/cc	0.52 uCi	Kr-83m	5.94E-08 uCi/cc	0.07 uCi	Submersion
Kr-85m	3.67E-06 uCi/cc	4.40 uCi	Kr-85m	5.05E-07 uCi/cc	0.61 uCi	Submersion
Kr-87	1.83E-09 uCi/cc	0.002 uCi	Kr-87	2.52E-10 uCi/cc	0.000 uCi	Submersion
Kr-88	1.77E-06 uCi/cc	2.12 uCi	Kr-88	2.43E-07 uCi/cc	0.29 uCi	Submersion
Sr-89	2.27E-06 uCi/cc	2.73 uCi	Sr-89	3.13E-07 uCi/cc	0.38 uCi	800
Sr-90	1.36E-08 uCi/cc	0.02 uCi	Sr-90	1.87E-09 uCi/cc	0.002 uCi	20
Sr-91	5.89E-05 uCi/cc	70.74 uCi	Sr-91	8.12E-06 uCi/cc	9.75 uCi	6000
I-131	7.33E-06 uCi/cc	8.80 uCi	I-131	1.01E-06 uCi/cc	1.21 uCi	50
I-132	2.64E-05 uCi/cc	31.64 uCi	I-132	3.63E-06 uCi/cc	4.36 uCi	8000
I-132m	1.80E-11 uCi/cc	0.000 uCi	I-132m	2.48E-12 uCi/cc	0.000 uCi	8000
I-133	8.42E-05 uCi/cc	101.09 uCi	I-133	1.16E-05 uCi/cc	13.93 uCi	300
I-133m	2.07E-12 uCi/cc	0.000 uCi	I-133m	2.85E-13 uCi/cc	0.000 uCi	300
I-134	8.12E-11 uCi/cc	0.000 uCi	I-134	1.12E-11 uCi/cc	0.000 uCi	50000
I-135	4.09E-05 uCi/cc	49.13 uCi	I-135	5.64E-06 uCi/cc	6.77 uCi	2000
Xe-133	1.53E-05 uCi/cc	18.34 uCi	Xe-133	2.11E-06 uCi/cc	2.53 uCi	Submersion
Xe-133m	9.92E-07 uCi/cc	1.19 uCi	Xe-133m	1.37E-07 uCi/cc	0.16 uCi	Submersion
Xe-135	1.16E-04 uCi/cc	138.64 uCi	Xe-135	1.59E-05 uCi/cc	19.11 uCi	Submersion
Xe-135m	6.69E-06 uCi/cc	8.03 uCi	Xe-135m	9.22E-07 uCi/cc	1.11 uCi	Submersion
Xe-138	6.34E-34 uCi/cc	0.000 uCi	Xe-138	8.74E-35 uCi/cc	0.000 uCi	Submersion

Dilution by room air exchanges was not considered in these calculations. If we consider room air exchanges, the concentration of each isotope decreases by up to several orders of magnitude.

From 10 CFR 20, Appendix B, Table 1, Column 2

*Room volumes and air exchange rates are from physical measurements taken in mid-August 2010.

**Radioisotopes and their quantities were provided by

Argonne National Laboratories/Babcock & Wilcox.

Activity Remaining After 24 Hours**

Isotope	Qty (Ci)	uCi	Half-life
Kr-83m	2.01E-05	20.1	1.8h
Kr-85m	1.71E-04	171	4.5h
Kr-87	8.54E-08	0.0854	1.3h
Kr-88	8.24E-05	82.4	2.8h
Sr-89	1.06E-04	106	50.6d
Sr-90	6.33E-07	0.633	29.1y
Sr-91	2.75E-03	2750	9.5h
I-131	3.42E-04	342	8.1d
I-132	1.23E-03	1230	2.3h
I-132m	8.41E-10	0.000841	83.6m
I-133	3.93E-03	3930	20.8h
I-133m	9.65E-11	9.65E-05	9s
I-134	3.79E-09	0.00379	52.6m
I-135	1.91E-03	1910	6.6h
Xe-133	7.13E-04	713	5.3d
Xe-133m	4.63E-05	46.3	2.2d
Xe-135	5.39E-03	5390	9.14h
Xe-135m	3.12E-04	312	15.3m
Xe-138	2.96E-32	2.96E-26	14.1m

**Radioisotopes and their quantities were provided by Argonne National Laboratories/Babcock & Wilcox.

Doubling the amount of material created:

Activity Remaining After 24 Hours**

Isotope	Qty (Ci)	uCi	Half-life
Kr-83m	4.02E-05	40.2	1.8h
Kr-85m	3.42E-04	342	4.5h
Kr-87	1.71E-07	0.1708	1.3h
Kr-88	1.65E-04	164.8	2.8h
Sr-89	2.12E-04	212	50.6d
Sr-90	1.27E-06	1.266	29.1y
Sr-91	5.50E-03	5500	9.5h
I-131	6.84E-04	684	8.1d
I-132	2.46E-03	2460	2.3h
I-132m	1.68E-09	0.001682	83.6m
I-133	7.86E-03	7860	20.8h
I-133m	1.93E-10	0.000193	9s
I-134	7.58E-09	0.00758	52.6m
I-135	3.82E-03	3820	6.6h
Xe-133	1.43E-03	1426	5.3d
Xe-133m	9.26E-05	92.6	2.2d
Xe-135	1.08E-02	10780	9.14h
Xe-135m	6.24E-04	624	15.3m
Xe-138	5.92E-32	5.92E-26	14.1m

**Radioisotopes and their quantities were provided by Argonne National Laboratories/Babcock & Wilcox.

ATTACHMENT #2

Dose Rate Calculations in the Reactor Room from Separation Experiment Failure

In order to obtain the quantity of fission product isotopes at reactor discharge after irradiating these samples, the quantities of isotopes that were used at 24 kW burnup to estimate dose were adjusted. This adjustment is based on the amount of burnup of 65 kW that was calculated by Dr. Russell Ball that would produce 1.3 Ci of iodine isotopes. Applying this factor and summing the iodine 131-135 isotopes does not yield 1.3 Ci, but this calculated burnup is used as a limiting variable. The quantities of fission product isotopes, in curies, are given below.

Dose Calculations - Argonne National Laboratory LEU irradiation (MIPS)

8-Dec-10

A.J. Teachout

		cubic			cubic
Room 2165 (Rx Hot Laboratory)*		centimeters	Room 3161 (Rx Deck)*		centimeters
Total cubic feet	3295	93304009.5	Total cubic feet	23,910	6.77E+08
Supply air (CFM)	1150	32564373.5	Supply air (CFM)	2895	81977271
Exhaust air (CFM)	1551	43919429.06	Exhaust air (CFM)	3561	1.01E+08
Air changes/hour#	28		Air changes/hour#	8.94	
Iodine-131 Gamma Ray Dose Constant =		282.94 mR/hr/Ci @ 1 meter			
Ar-41 Gamma Ray Dose Constant=		695.97 mR/hr/Ci @ 1 meter			
Reference Breathing Rate =		1,200,000 ml/hr			

Activity at Rx shutdown**				Inhalation	Concentration of isotopes in 2165 W/O air exchange			2165
Isotope	Qty (Ci)	uCi	Half-life	ALI in uCi##		Intake@1 hr		Fraction of ALI
Ga-73	1.48E-06	1.48	4.86 h	20000	1.59E-08 uCi/cc	0.02 uCi	Ga-73	9.52E-07
Ge-73m	1.45E-06	1.45	0.499 s	^	1.55E-08 uCi/cc	0.02 uCi	Ge-73m	0
Ga-74	2.21E-05	22.1	8.12m	^	2.37E-07 uCi/cc	0.3 uCi	Ga-74	0
Ge-75	4.01E-05	40.1	82.78 m	80000	4.30E-07 uCi/cc	0.5 uCi	Ge-75	6.45E-06
Ge-77	1.33E-05	13.3	11.3 h	6000	1.43E-07 uCi/cc	0.2 uCi	Ge-77	2.85E-05
As-77	1.10E-05	11	38.83 h	5000	1.18E-07 uCi/cc	0.1 uCi	As-77	2.83E-05
Ge-78	8.31E-04	831	88 m	20000	8.91E-06 uCi/cc	10.7 uCi	Ge-78	5.34E-04
As-78	2.33E-04	233	90.7 m	20000	2.50E-06 uCi/cc	3.0 uCi	As-78	1.50E-04
As-79	3.63E-03	3630	9.01 m	^	3.89E-05 uCi/cc	46.7 uCi	As-79	0
Se-79m	3.58E-03	3580	3.91m	800	3.84E-05 uCi/cc	46.0 uCi	Se-79m	5.76E-02
Se-81	1.48E-02	14800	18.45m	200000	1.59E-04 uCi/cc	190.3 uCi	Se-81	9.52E-04
Se-81m	6.26E-04	626	57.28m	70000	6.71E-06 uCi/cc	8.1 uCi	Se-81m	1.15E-04
Se-83	1.79E-02	17900	22.3m	100000	1.92E-04 uCi/cc	230.2 uCi	Se-83	2.30E-03
Br-83	1.14E-02	11400	2.4h	60000	1.22E-04 uCi/cc	146.6 uCi	Br-83	2.44E-03
Se-84	8.40E-02	84000	3.1m	^	9.00E-04 uCi/cc	1080.3 uCi	Se-84	0
Br-84	6.91E-02	69100	31.8m	60000	7.41E-04 uCi/cc	888.7 uCi	Br-84	1.48E-02
Br-84m	1.32E-03	1320	6.0m	^	1.41E-05 uCi/cc	17.0 uCi	Br-84m	0
Rb-88	5.74E-02	57400	17.78m	60000	6.15E-04 uCi/cc	738.2 uCi	Rb-88	1.23E-02
Kr-89	3.63E-01	363000	3.15m	Submersion	3.89E-03 uCi/cc	4668.6 uCi	Kr-89	0
Rb-89	3.68E-01	368000	15.15m	100000	3.94E-03 uCi/cc	4732.9 uCi	Rb-89	4.73E-02
Rb-90	3.52E-01	352000	153s	^	3.77E-03 uCi/cc	4527.1 uCi	Rb-90	0
Rb-90m	1.02E-01	102000	258s	^	1.09E-03 uCi/cc	1311.8 uCi	Rb-90m	0

Y-91	8.07E-06	8.07	58.51d	200	8.65E-08 uCi/cc	0.1 uCi	Y-91	5.19E-04
Y-91m	9.70E-03	9700	49.71m	200000	1.04E-04 uCi/cc	124.8 uCi	Y-91m	6.24E-04
Sr-92	1.48E-01	148000	2.71h	7000	1.59E-03 uCi/cc	1903.5 uCi	Sr-92	2.72E-01
Y-92	1.99E-02	19900	3.54h	8000	2.13E-04 uCi/cc	255.9 uCi	Y-92	3.20E-02
Sr-93	5.31E-01	531000	7.423m	^	5.69E-03 uCi/cc	6829.3 uCi	Sr-93	0
Y-93	4.12E-02	41200	10.10h	3000	4.42E-04 uCi/cc	529.9 uCi	Y-93	1.77E-01
Y-93m	1.88E-01	188000	0.82s	^	2.01E-03 uCi/cc	2417.9 uCi	Y-93m	0
Y-94	4.90E-01	490000	18.7m	80000	5.25E-03 uCi/cc	6302.0 uCi	Y-94	7.88E-02
Y-95	5.01E-01	501000	10.3m	200000	5.37E-03 uCi/cc	6443.5 uCi	Y-95	3.22E-02
Zr-95	2.50E-04	250	64.02d	100	2.68E-06 uCi/cc	3.2 uCi	Zr-95	3.22E-02
Nb-95	1.25E-07	0.125	34.97d	1000	1.34E-09 uCi/cc	0.002 uCi	Nb-95	1.61E-06
Nb-96	1.67E-06	1.67	23.35h	3000	1.79E-08 uCi/cc	0.02 uCi	Nb-96	7.16E-06
Zr-97	2.66E-02	26600	16.9h	2000	2.85E-04 uCi/cc	342.1 uCi	Zr-97	1.71E-01
Nb-97	8.42E-03	8420	1.227h	80000	9.02E-05 uCi/cc	108.3 uCi	Nb-97	1.35E-03
Nb-97m	2.50E-02	25000	58.8s	^	2.68E-04 uCi/cc	321.5 uCi	Nb-97m	0
Nb-98m	2.01E-03	2010	51.3m	^	2.15E-05 uCi/cc	25.9 uCi	Nb-98m	0
Mo-99	6.91E-03	6910	65.94h	3000	7.41E-05 uCi/cc	88.9 uCi	Mo-99	2.96E-02
Tc-99m	4.36E-04	436	6.01h	200000	4.67E-06 uCi/cc	5.6 uCi	Tc-99m	2.80E-05
Mo-101	4.12E-01	412000	14.6m	100000	4.42E-03 uCi/cc	5298.8 uCi	Mo-101	5.30E-02
Tc-101	3.76E-01	376000	14.2m	300000	4.03E-03 uCi/cc	4835.8 uCi	Tc-101	1.61E-02
Mo-102	3.47E-01	347000	11.3m	^	3.72E-03 uCi/cc	4462.8 uCi	Mo-102	0
Tc-102	3.52E-01	352000	5.28s	^	3.77E-03 uCi/cc	4527.1 uCi	Tc-102	0
Ru-103	2.32E-04	232	39.26d	600	2.49E-06 uCi/cc	3.0 uCi	Ru-103	4.97E-03
Rh-103m	8.21E-05	82.1	56.12m	^	8.80E-07 uCi/cc	1.1 uCi	Rh-103m	0
Tc-104	1.45E-01	145000	18.3m	^	1.55E-03 uCi/cc	1864.9 uCi	Tc-104	0
Tc-105	7.77E-02	77700	7.6m	^	8.33E-04 uCi/cc	999.3 uCi	Tc-105	0
Ru-105	1.26E-02	12600	4.44h	10000	1.35E-04 uCi/cc	162.1 uCi	Ru-105	1.62E-02
Rh-105	1.47E-04	147	35.36h	10000	1.58E-06 uCi/cc	1.9 uCi	Rh-105	1.89E-04
Rh-105m	3.55E-06	3.55	40s	^	3.80E-08 uCi/cc	0.05 uCi	Rh-105m	0
Ru-106	3.33E-06	3.33	373.59d	90	3.57E-08 uCi/cc	0.04 uCi	Ru-106	4.76E-04
Ru-107	1.25E-02	12500	3.75m	^	1.34E-04 uCi/cc	160.8 uCi	Ru-107	0
Rh-107	1.13E-02	11300	21.7m	200000	1.21E-04 uCi/cc	145.3 uCi	Rh-107	7.27E-04
Ru-108	4.88E-03	4880	4.55m	^	5.23E-05 uCi/cc	62.8 uCi	Ru-108	0
Rh-108	4.90E-03	4900	16.8s	^	5.25E-05 uCi/cc	63.0 uCi	Rh-108	0
Pd-109	1.60E-04	160	13.7h	6000	1.71E-06 uCi/cc	2.1 uCi	Pd-109	3.43E-04
Ag-109m	1.58E-04	158	39.6s	^	1.69E-06 uCi/cc	2.0 uCi	Ag-109m	0
Pd-111	1.28E-03	1280	23.4m	^	1.37E-05 uCi/cc	16.5 uCi	Pd-111	0

Ag-111	4.33E-06	4.33 7.45d	900	4.64E-08 uCi/cc	0.1 uCi	Ag-111	6.19E-05
Ag-111m	1.26E-03	1260 64.8s	^	1.35E-05 uCi/cc	16.2 uCi	Ag-111m	0
Pd-112	4.58E-05	45.8 21.03h	^	4.91E-07 uCi/cc	0.6 uCi	Pd-112	0
Ag-112	6.20E-06	6.2 3.13h	9000	6.64E-08 uCi/cc	0.1 uCi	Ag-112	8.86E-06
Ag-113	1.70E-04	170 5.37h	^	1.82E-06 uCi/cc	2.2 uCi	Ag-113	0
Ag-115	6.09E-04	609 20m	90000	6.53E-06 uCi/cc	7.8 uCi	Ag-115	8.70E-05
Cd-115	1.27E-05	12.7 53.46h	1000	1.36E-07 uCi/cc	0.2 uCi	Cd-115	1.63E-04
In-115m	1.08E-06	1.08 4.486h	50000	1.16E-08 uCi/cc	0.01 uCi	In-115m	2.78E-07
Cd-117	2.87E-04	287 2.49h	20000	3.08E-06 uCi/cc	3.7 uCi	Cd-117	1.85E-04
Cd-117m	4.36E-05	43.6 3.36h	20000	4.67E-07 uCi/cc	0.6 uCi	Cd-117m	2.80E-05
In-117	3.95E-05	39.5 43.8m	200000	4.23E-07 uCi/cc	0.5 uCi	In-117	2.54E-06
In-117m	5.58E-05	55.8 116.5m	40000	5.98E-07 uCi/cc	0.7 uCi	In-117m	1.79E-05
Cd-118	6.39E-04	639 50.3m	^	6.85E-06 uCi/cc	8.2 uCi	Cd-118	0
In-118	6.42E-04	642 5.0s	^	6.88E-06 uCi/cc	8.3 uCi	In-118	0
In-119	4.39E-04	439 2.4m	^	4.71E-06 uCi/cc	5.6 uCi	In-119	0
In-119m	5.90E-04	590 18m	100000	6.32E-06 uCi/cc	7.6 uCi	In-119m	7.59E-05
Sn-121	3.25E-05	32.5 27.06h	20000	3.48E-07 uCi/cc	0.4 uCi	Sn-121	2.09E-05
Sn-123m	8.56E-04	856 40.08m	100000	9.17E-06 uCi/cc	11.0 uCi	Sn-123m	1.10E-04
Sn-125	3.01E-06	3.01 9.64d	900	3.23E-08 uCi/cc	0.04 uCi	Sn-125	4.30E-05
Sn-125m	1.97E-03	1970 9.52m	^	2.11E-05 uCi/cc	25.3 uCi	Sn-125m	0
Sb-126	1.99E-07	0.199 12.4d	500	2.13E-09 uCi/cc	0.003 uCi	Sb-126	5.12E-06
Sb-126m	1.28E-04	128 19m	200000	1.37E-06 uCi/cc	1.6 uCi	Sb-126m	8.23E-06
Sn-127	2.64E-03	2640 2.1h	2000	2.83E-05 uCi/cc	34.0 uCi	Sn-127	1.70E-02
Sn-127m	4.58E-03	4580 4.13m	^	4.91E-05 uCi/cc	58.9 uCi	Sn-127m	0
Sb-127	6.20E-05	62 3.85d	2000	6.64E-07 uCi/cc	0.8 uCi	Sb-127	3.99E-04
Te-127	2.17E-06	2.17 9.35h	20000	2.33E-08 uCi/cc	0.03 uCi	Te-127	1.40E-06
Sn-128	1.60E-02	16000 59.1m	40000	1.71E-04 uCi/cc	205.8 uCi	Sn-128	5.14E-03
Sb-128	1.07E-04	107 9.01h	400000	1.15E-06 uCi/cc	1.4 uCi	Sb-128	3.44E-06
Sb-128m	1.43E-02	14300 10.4m	^	1.53E-04 uCi/cc	183.9 uCi	Sb-128m	0
Sn-129m	1.57E-02	15700 6.9m	^	1.68E-04 uCi/cc	201.9 uCi	Sn-129m	0
Sb-129	7.69E-03	7690 17.7m	9000	8.24E-05 uCi/cc	98.9 uCi	Sb-129	1.10E-02
Te-129	1.93E-03	1930 69.6m	70000	2.07E-05 uCi/cc	24.8 uCi	Te-129	3.55E-04
Te-129m	7.75E-07	0.775 33.6d	600	8.31E-09 uCi/cc	0.01 uCi	Te-129m	1.66E-05
Sn-130	4.79E-02	47900 3.72m	^	5.13E-04 uCi/cc	616.1 uCi	Sn-130	0
Sb-130	4.66E-02	46600 39.5m	80000	4.99E-04 uCi/cc	599.3 uCi	Sb-130	7.49E-03
Sb-130m	7.61E-02	76100 6.3m	^	8.16E-04 uCi/cc	978.7 uCi	Sb-130m	0
I-130	1.15E-06	1.15 12.36h	700	1.23E-08 uCi/cc	0.01 uCi	I-130	2.11E-05

Sb-131	1.84E-01	184000	23m	20000	1.97E-03	uCi/cc	2366.5	uCi	Sb-131	1.18E-01
Te-131	1.33E-01	133000	25m	5000	1.43E-03	uCi/cc	1710.5	uCi	Te-131	3.42E-01
Te-131m	8.18E-04	818	30h	400	8.77E-06	uCi/cc	10.5	uCi	Te-131m	2.63E-02
I-131	3.09E-04	309	8.04d	50	3.31E-06	uCi/cc	4.0	uCi	I-131	7.95E-02
Sb-132	1.03E-01	103000	78.2h	^	1.10E-03	uCi/cc	1324.7	uCi	Sb-132	0
Te-132	3.82E-03	3820	78.2h	200	4.09E-05	uCi/cc	49.1	uCi	Te-132	2.46E-01
I-132	9.56E-04	956	2.3h	8000	1.02E-05	uCi/cc	12.3	uCi	I-132	1.54E-03
I-132m	3.49E-04	349	83.6m	8000	3.74E-06	uCi/cc	4.5	uCi	I-132m	5.61E-04
Te-133	2.70E-01	270000	12.5m	20000	2.89E-03	uCi/cc	3472.5	uCi	Te-133	1.74E-01
Te-133m	1.69E-01	169000	55.4m	5000	1.81E-03	uCi/cc	2173.5	uCi	Te-133m	4.35E-01
I-133	1.27E-02	12700	20.8h	300	1.36E-04	uCi/cc	163.3	uCi	I-133	5.44E-01
I-133m	2.38E-02	23800	9s	^	2.55E-04	uCi/cc	306.1	uCi	I-133m	0
Xe-133	3.60E-05	36	5.243d	Submersion	3.86E-07	uCi/cc	0.5	uCi	Xe-133	0
Xe-133m	5.12E-06	5.12	2.19d	Submersion	5.49E-08	uCi/cc	0.1	uCi	Xe-133m	0
Te-134	4.06E-01	406000	41.8m	20000	4.35E-03	uCi/cc	5221.6	uCi	Te-134	2.61E-01
I-134	2.25E-01	225000	52.6m	50000	2.41E-03	uCi/cc	2893.8	uCi	I-134	5.79E-02
I-134m	2.87E-02	28700	3.69m	^	3.08E-04	uCi/cc	369.1	uCi	I-134m	0
I-135	6.50E-02	65000	6.57h	2000	6.97E-04	uCi/cc	836.0	uCi	I-135	4.18E-01
Xe-135	4.63E-03	4630	9.14h	Submersion	4.96E-05	uCi/cc	59.5	uCi	Xe-135	0
Xe-135m	2.12E-02	21200	15.29m	Submersion	2.27E-04	uCi/cc	272.7	uCi	Xe-135m	0
Cs-135m	1.26E-05	12.6	53m	200000	1.35E-07	uCi/cc	0.2	uCi	Cs-135m	8.10E-07
Cs-136	1.27E-06	1.27	13.16d	700	1.36E-08	uCi/cc	0.02	uCi	Cs-136	2.33E-05
Ba-136m	4.93E-05	49.3	0.3084s	^	5.28E-07	uCi/cc	0.6	uCi	Ba-136m	0
Xe-137	4.88E-01	488000	3.818m	Submersion	5.23E-03	uCi/cc	6276.3	uCi	Xe-137	0
Cs-137	1.61E-06	1.61	30.1y	200	1.73E-08	uCi/cc	0.02	uCi	Cs-137	1.04E-04
Ba-137m	1.19E-05	11.9	0.3s	^	1.28E-07	uCi/cc	0.2	uCi	Ba-137m	0
Xe-138	4.88E-01	488000	14.08m	Submersion	5.23E-03	uCi/cc	6276.3	uCi	Xe-138	0
Cs-138	3.74E-01	374000	32.2m	60000	4.01E-03	uCi/cc	4810.1	uCi	Cs-138	8.02E-02
Cs-139	5.04E-01	504000	9.27m	^	5.40E-03	uCi/cc	6482.0	uCi	Cs-139	0
Ba-139	2.12E-01	212000	83.06m	30000	2.27E-03	uCi/cc	2726.6	uCi	Ba-139	9.09E-02
Ba-140	1.45E-03	1450	12.752d	1000	1.55E-05	uCi/cc	18.6	uCi	Ba-140	1.86E-02
La-140	2.56E-05	25.6	1.678d	1000	2.74E-07	uCi/cc	0.3	uCi	La-140	3.29E-04
Ba-141	4.41E-01	441000	18.27m	70000	4.73E-03	uCi/cc	5671.8	uCi	Ba-141	8.10E-02
La-141	6.83E-02	68300	3.92h	9000	7.32E-04	uCi/cc	878.4	uCi	La-141	9.76E-02
Ce-141	3.30E-05	33	32.501d	700	3.54E-07	uCi/cc	0.4	uCi	Ce-141	6.06E-04
Ba-142	4.60E-01	460000	10.6m	100000	4.93E-03	uCi/cc	5916.1	uCi	Ba-142	5.92E-02
La-142	1.82E-01	182000	91.1m	30000	1.95E-03	uCi/cc	2340.7	uCi	La-142	7.80E-02

La-143	4.63E-01	463000	14.2m	100000	4.96E-03 uCi/cc	5954.7 uCi	La-143	5.95E-02
Ce-143	9.83E-03	9830	33.1h	2000	1.05E-04 uCi/cc	126.4 uCi	Ce-143	6.32E-02
Pr-143	1.15E-05	11.5	13.57d	800	1.23E-07 uCi/cc	0.1 uCi	Pr-143	1.85E-04
Ce-144	5.80E-05	58	284.893d	30	6.22E-07 uCi/cc	0.7 uCi	Ce-144	2.49E-02
Pr-144	4.14E-05	41.4	17.28m	100000	4.44E-07 uCi/cc	0.5 uCi	Pr-144	5.32E-06
Pr-144m	1.71E-06	1.71	7.2m	^	1.83E-08 uCi/cc	0.02 uCi	Pr-144m	0
Ce-145	3.25E-01	325000	3.01m	^	3.48E-03 uCi/cc	4179.9 uCi	Ce-145	0
Pr-145	4.41E-02	44100	5.984h	9000	4.73E-04 uCi/cc	567.2 uCi	Pr-145	6.30E-02
Ce-146	2.36E-01	236000	13.52m	^	2.53E-03 uCi/cc	3035.2 uCi	Ce-146	0
Pr-146	1.91E-01	191000	24.15m	^	2.05E-03 uCi/cc	2456.5 uCi	Pr-146	0
Pr-147	1.82E-01	182000	13.6m	200000	1.95E-03 uCi/cc	2340.7 uCi	Pr-147	1.17E-02
Nd-147	4.82E-04	482	10.98d	900	5.17E-06 uCi/cc	6.2 uCi	Nd-147	6.89E-03
Nd-149	3.74E-02	37400	1.72h	30000	4.01E-04 uCi/cc	481.0 uCi	Nd-149	1.60E-02
Pm-149	3.39E-04	339	53.08h	2000	3.63E-06 uCi/cc	4.4 uCi	Pm-149	2.18E-03
Pm-150	6.99E-07	0.699	2.68h	20000	7.49E-09 uCi/cc	0.01 uCi	Pm-150	4.49E-07
Nd-151	3.82E-02	38200	12.44m	200000	4.09E-04 uCi/cc	491.3 uCi	Nd-151	2.46E-03
Pm-151	8.29E-04	829	28.4h	4000	8.88E-06 uCi/cc	10.7 uCi	Pm-151	2.67E-03
Nd-152	2.07E-02	20700	11.4m	^	2.22E-04 uCi/cc	266.2 uCi	Nd-152	0
Pm-152	2.07E-02	20700	4.1m	^	2.22E-04 uCi/cc	266.2 uCi	Pm-152	0
Pm-152m	1.47E-04	147	7.52m	^	1.58E-06 uCi/cc	1.9 uCi	Pm-152m	0
Pm-153	1.28E-02	12800	5.4m	^	1.37E-04 uCi/cc	164.6 uCi	Pm-153	0
Sm-153	2.25E-04	225	46.27h	3000	2.41E-06 uCi/cc	2.9 uCi	Sm-153	9.65E-04
Sm-155	2.33E-03	2330	22.3m	200000	2.50E-05 uCi/cc	30.0 uCi	Sm-155	1.50E-04
Sm-156	1.10E-04	110	9.4h	9000	1.18E-06 uCi/cc	1.4 uCi	Sm-156	1.57E-04
Eu-156	1.44E-07	0.144	15.19d	500	1.54E-09 uCi/cc	0.002 uCi	Eu-156	3.70E-06
Sm-157	4.88E-04	488	8.07m	^	5.23E-06 uCi/cc	6.3 uCi	Sm-157	0
Eu-157	2.47E-05	24.7	15.18h	5000	2.65E-07 uCi/cc	0.3 uCi	Eu-157	6.35E-05
Sm-158	2.56E-04	256	5.51m	^	2.74E-06 uCi/cc	3.3 uCi	Sm-158	0
Eu-158	1.77E-04	177	45.9m	60000	1.90E-06 uCi/cc	2.3 uCi	Eu-158	3.79E-05
Eu-159	7.66E-05	76.6	18.1m	^	8.21E-07 uCi/cc	1.0 uCi	Eu-159	0
Gd-159	2.58E-06	2.58	18.56h	8000	2.77E-08 uCi/cc	0.03 uCi	Gd-159	4.15E-06

4.56

^Not Listed

Dilution by room air exchanges was not considered in these calculations. If we consider room air exchanges, the concentration of each isotope decreases by up to several orders of magnitude.

From 10 CFR 20, Appendix B, Table 1, Column 2

*Room volumes and air exchange rates are from physical measurements taken in mid-August 2010.

**Radioisotopes and their quantities were provided by Argonne National Laboratories/Babcock & Wilcox.

	Repeated for 3161	W/O air exchange Intake@1 hr	3161 Fraction of ALI
Ga-73	2.19E-09 uCi/cc	0.003 uCi	1.31E-07
Ge-73m	2.14E-09 uCi/cc	0.003 uCi	0
Ga-74	3.26E-08 uCi/cc	0.04 uCi	0
Ge-75	5.92E-08 uCi/cc	0.1 uCi	8.88E-07
Ge-77	1.96E-08 uCi/cc	0.02 uCi	3.93E-06
As-77	1.62E-08 uCi/cc	0.02 uCi	3.90E-06
Ge-78	1.23E-06 uCi/cc	1.5 uCi	7.36E-05
As-78	3.44E-07 uCi/cc	0.4 uCi	2.06E-05
As-79	5.36E-06 uCi/cc	6.4 uCi	0
Se-79m	5.29E-06 uCi/cc	6.3 uCi	7.93E-03
Se-81	2.19E-05 uCi/cc	26.2 uCi	1.31E-04
Se-81m	9.25E-07 uCi/cc	1.1 uCi	1.59E-05
Se-83	2.64E-05 uCi/cc	31.7 uCi	3.17E-04
Br-83	1.68E-05 uCi/cc	20.2 uCi	3.37E-04
Se-84	1.24E-04 uCi/cc	148.9 uCi	0
Br-84	1.02E-04 uCi/cc	122.5 uCi	2.04E-03
Br-84m	1.95E-06 uCi/cc	2.3 uCi	0
Rb-88	8.48E-05 uCi/cc	101.7 uCi	1.70E-03
Kr-89	5.36E-04 uCi/cc	643.4 uCi	0
Rb-89	5.44E-04 uCi/cc	652.2 uCi	6.52E-03
Rb-90	5.20E-04 uCi/cc	623.9 uCi	0
Rb-90m	1.51E-04 uCi/cc	180.8 uCi	0

Y-91	1.19E-08 uCi/cc	0.01 uCi	7.15E-05
Y-91m	1.43E-05 uCi/cc	17.2 uCi	8.60E-05
Sr-92	2.19E-04 uCi/cc	262.3 uCi	3.75E-02
Y-92	2.94E-05 uCi/cc	35.3 uCi	4.41E-03
Sr-93	7.84E-04 uCi/cc	941.1 uCi	0
Y-93	6.09E-05 uCi/cc	73.0 uCi	2.43E-02
Y-93m	2.78E-04 uCi/cc	333.2 uCi	0
Y-94	7.24E-04 uCi/cc	868.5 uCi	1.09E-02
Y-95	7.40E-04 uCi/cc	888.0 uCi	4.44E-03
Zr-95	3.69E-07 uCi/cc	0.4 uCi	4.43E-03
Nb-95	1.85E-10 uCi/cc	0.0002 uCi	2.22E-07
Nb-96	2.47E-09 uCi/cc	0.003 uCi	9.87E-07
Zr-97	3.93E-05 uCi/cc	47.1 uCi	2.36E-02
Nb-97	1.24E-05 uCi/cc	14.9 uCi	1.87E-04
Nb-97m	3.69E-05 uCi/cc	44.3 uCi	0
Nb-98m	2.97E-06 uCi/cc	3.6 uCi	0
Mo-99	1.02E-05 uCi/cc	12.2 uCi	4.08E-03
Tc-99m	6.44E-07 uCi/cc	0.8 uCi	3.86E-06
Mo-101	6.09E-04 uCi/cc	730.2 uCi	7.30E-03
Tc-101	5.55E-04 uCi/cc	666.4 uCi	2.22E-03
Mo-102	5.13E-04 uCi/cc	615.0 uCi	0
Tc-102	5.20E-04 uCi/cc	623.9 uCi	0
Ru-103	3.43E-07 uCi/cc	0.4 uCi	6.85E-04
Rh-103m	1.21E-07 uCi/cc	0.1 uCi	0
Tc-104	2.14E-04 uCi/cc	257.0 uCi	0
Tc-105	1.15E-04 uCi/cc	137.7 uCi	0
Ru-105	1.86E-05 uCi/cc	22.3 uCi	2.23E-03
Rh-105	2.17E-07 uCi/cc	0.3 uCi	2.61E-05
Rh-105m	5.24E-09 uCi/cc	0.01 uCi	0
Ru-106	4.92E-09 uCi/cc	0.01 uCi	6.56E-05
Ru-107	1.85E-05 uCi/cc	22.2 uCi	0
Rh-107	1.67E-05 uCi/cc	20.0 uCi	1.00E-04
Ru-108	7.21E-06 uCi/cc	8.6 uCi	0
Rh-108	7.24E-06 uCi/cc	8.7 uCi	0
Pd-109	2.36E-07 uCi/cc	0.3 uCi	4.73E-05
Ag-109m	2.33E-07 uCi/cc	0.3 uCi	0
Pd-111	1.89E-06 uCi/cc	2.3 uCi	0

Ag-111	6.40E-09 uCi/cc	0.01 uCi	8.53E-06
Ag-111m	1.86E-06 uCi/cc	2.2 uCi	0
Pd-112	6.76E-08 uCi/cc	0.1 uCi	0
Ag-112	9.16E-09 uCi/cc	0.01 uCi	1.22E-06
Ag-113	2.51E-07 uCi/cc	0.3 uCi	0
Ag-115	8.99E-07 uCi/cc	1.1 uCi	1.20E-05
Cd-115	1.88E-08 uCi/cc	0.02 uCi	2.25E-05
In-115m	1.60E-09 uCi/cc	0.002 uCi	3.83E-08
Cd-117	4.24E-07 uCi/cc	0.5 uCi	2.54E-05
Cd-117m	6.44E-08 uCi/cc	0.1 uCi	3.86E-06
In-117	5.83E-08 uCi/cc	0.1 uCi	3.50E-07
In-117m	8.24E-08 uCi/cc	0.1 uCi	2.47E-06
Cd-118	9.44E-07 uCi/cc	1.1 uCi	0
In-118	9.48E-07 uCi/cc	1.1 uCi	0
In-119	6.48E-07 uCi/cc	0.8 uCi	0
In-119m	8.71E-07 uCi/cc	1.0 uCi	1.05E-05
Sn-121	4.80E-08 uCi/cc	0.1 uCi	2.88E-06
Sn-123m	1.26E-06 uCi/cc	1.5 uCi	1.52E-05
Sn-125	4.45E-09 uCi/cc	0.01 uCi	5.93E-06
Sn-125m	2.91E-06 uCi/cc	3.5 uCi	0
Sb-126	2.94E-10 uCi/cc	0.0004 uCi	7.05E-07
Sb-126m	1.89E-07 uCi/cc	0.2 uCi	1.13E-06
Sn-127	3.90E-06 uCi/cc	4.7 uCi	2.34E-03
Sn-127m	6.76E-06 uCi/cc	8.1 uCi	0
Sb-127	9.16E-08 uCi/cc	0.1 uCi	5.49E-05
Te-127	3.21E-09 uCi/cc	0.004 uCi	1.92E-07
Sn-128	2.36E-05 uCi/cc	28.4 uCi	7.09E-04
Sb-128	1.58E-07 uCi/cc	0.2 uCi	4.74E-07
Sb-128m	2.11E-05 uCi/cc	25.3 uCi	0
Sn-129m	2.32E-05 uCi/cc	27.8 uCi	0
Sb-129	1.14E-05 uCi/cc	13.6 uCi	1.51E-03
Te-129	2.85E-06 uCi/cc	3.4 uCi	4.89E-05
Te-129m	1.14E-09 uCi/cc	0.001 uCi	2.29E-06
Sn-130	7.07E-05 uCi/cc	84.9 uCi	0
Sb-130	6.88E-05 uCi/cc	82.6 uCi	1.03E-03
Sb-130m	1.12E-04 uCi/cc	134.9 uCi	0
I-130	1.70E-09 uCi/cc	0.002 uCi	2.91E-06

Sb-131	2.72E-04 uCi/cc	326.1 uCi	1.63E-02
Te-131	1.96E-04 uCi/cc	235.7 uCi	4.71E-02
Te-131m	1.21E-06 uCi/cc	1.4 uCi	3.62E-03
I-131	4.56E-07 uCi/cc	0.5 uCi	1.10E-02
Sb-132	1.52E-04 uCi/cc	182.6 uCi	0
Te-132	5.64E-06 uCi/cc	6.8 uCi	3.39E-02
I-132	1.41E-06 uCi/cc	1.7 uCi	2.12E-04
I-132m	5.15E-07 uCi/cc	0.6 uCi	7.73E-05
Te-133	3.99E-04 uCi/cc	478.5 uCi	2.39E-02
Te-133m	2.50E-04 uCi/cc	299.5 uCi	5.99E-02
I-133	1.88E-05 uCi/cc	22.5 uCi	7.50E-02
I-133m	3.52E-05 uCi/cc	42.2 uCi	0
Xe-133	5.32E-08 uCi/cc	0.1 uCi	0
Xe-133m	7.56E-09 uCi/cc	0.01 uCi	0
Te-134	6.00E-04 uCi/cc	719.6 uCi	3.60E-02
I-134	3.32E-04 uCi/cc	398.8 uCi	7.98E-03
I-134m	4.24E-05 uCi/cc	50.9 uCi	0
I-135	9.60E-05 uCi/cc	115.2 uCi	5.76E-02
Xe-135	6.84E-06 uCi/cc	8.2 uCi	0
Xe-135m	3.13E-05 uCi/cc	37.6 uCi	0
Cs-135m	1.86E-08 uCi/cc	0.02 uCi	1.12E-07
Cs-136	1.88E-09 uCi/cc	0.002 uCi	3.22E-06
Ba-136m	7.28E-08 uCi/cc	0.1 uCi	0
Xe-137	7.21E-04 uCi/cc	864.9 uCi	0
Cs-137	2.38E-09 uCi/cc	0.003 uCi	1.43E-05
Ba-137m	1.76E-08 uCi/cc	0.02 uCi	0
Xe-138	7.21E-04 uCi/cc	864.9 uCi	0
Cs-138	5.52E-04 uCi/cc	662.9 uCi	1.10E-02
Cs-139	7.44E-04 uCi/cc	893.3 uCi	0
Ba-139	3.13E-04 uCi/cc	375.7 uCi	1.25E-02
Ba-140	2.14E-06 uCi/cc	2.6 uCi	2.57E-03
La-140	3.78E-08 uCi/cc	0.05 uCi	4.54E-05
Ba-141	6.51E-04 uCi/cc	781.6 uCi	1.12E-02
La-141	1.01E-04 uCi/cc	121.1 uCi	1.35E-02
Ce-141	4.87E-08 uCi/cc	0.1 uCi	8.36E-05
Ba-142	6.79E-04 uCi/cc	815.3 uCi	8.15E-03
La-142	2.69E-04 uCi/cc	322.6 uCi	1.08E-02

La-143	6.84E-04 uCi/cc	820.6 uCi	8.21E-03
Ce-143	1.45E-05 uCi/cc	17.4 uCi	8.71E-03
Pr-143	1.70E-08 uCi/cc	0.02 uCi	2.55E-05
Ce-144	8.57E-08 uCi/cc	0.1 uCi	3.43E-03
Pr-144	6.11E-08 uCi/cc	0.1 uCi	7.34E-07
Pr-144m	2.53E-09 uCi/cc	0.003 uCi	0
Ce-145	4.80E-04 uCi/cc	576.0 uCi	0
Pr-145	6.51E-05 uCi/cc	78.2 uCi	8.68E-03
Ce-146	3.49E-04 uCi/cc	418.3 uCi	0
Pr-146	2.82E-04 uCi/cc	338.5 uCi	0
Pr-147	2.69E-04 uCi/cc	322.6 uCi	1.61E-03
Nd-147	7.12E-07 uCi/cc	0.9 uCi	9.49E-04
Nd-149	5.52E-05 uCi/cc	66.3 uCi	2.21E-03
Pm-149	5.01E-07 uCi/cc	0.6 uCi	3.00E-04
Pm-150	1.03E-09 uCi/cc	0.001 uCi	6.19E-08
Nd-151	5.64E-05 uCi/cc	67.7 uCi	3.39E-04
Pm-151	1.22E-06 uCi/cc	1.5 uCi	3.67E-04
Nd-152	3.06E-05 uCi/cc	36.7 uCi	0
Pm-152	3.06E-05 uCi/cc	36.7 uCi	0
Pm-152m	2.17E-07 uCi/cc	0.3 uCi	0
Pm-153	1.89E-05 uCi/cc	22.7 uCi	0
Sm-153	3.32E-07 uCi/cc	0.4 uCi	1.33E-04
Sm-155	3.44E-06 uCi/cc	4.1 uCi	2.06E-05
Sm-156	1.62E-07 uCi/cc	0.2 uCi	2.17E-05
Eu-156	2.13E-10 uCi/cc	0.0003 uCi	5.10E-07
Sm-157	7.21E-07 uCi/cc	0.9 uCi	0
Eu-157	3.65E-08 uCi/cc	0.04 uCi	8.76E-06
Sm-158	3.78E-07 uCi/cc	0.5 uCi	0
Eu-158	2.61E-07 uCi/cc	0.3 uCi	5.23E-06
Eu-159	1.13E-07 uCi/cc	0.1 uCi	0
Gd-159	3.81E-09 uCi/cc	0.005 uCi	5.72E-07

0.63

ATTACHMENT #3

Experimental Protocol and Safety Features of MIPS Experiment

Scope of the Work

Solutions of LEU (19.8% U-235) and/or DU (0.22% U-235) as uranyl nitrate will be prepared for use in irradiation tests at AFRRRI using a TRIGA reactor. These solutions will be irradiated at AFRRRI in core to allow various chemical and physical measurements in support of proof of principle tests for MIPS (Medical Isotope Production System) for domestic production and purification of Mo-99 from a LEU uranyl nitrate solution for medical use. The irradiated solutions will be used to examine solution behavior (both physical and chemical) during and after irradiation.

Small volumes (1 – 5 mL) of LEU and/or DU solutions will be irradiated. The solutions will be kept in quartz vials with plastic valves for gas analysis (Appendix I), and two vials will be irradiated at a time. The two quartz vials will be placed in an aluminum sleeve inside an aluminum mock fuel pin (Appendix II), which will serve as the secondary container. The secondary container is sealed at the top end by a viton O-ring on the top plug. The aluminum mock fuel pin will be placed in a specially fabricated lead-shielded cask for transport to the radiation chemistry laboratory. Dose rate calculations have been performed for the solutions that will be the most and least radioactive (Table I and II). The samples will not be handled until 24 hours after irradiation for the solutions predicted to be the most radioactive and at least 6 hours after irradiation for the solutions predicted to be the least radioactive.

Table I. Projected dose rates for aluminum mock fuel pin and vials containing irradiated uranyl nitrate solution under conditions that will produce the most radioactive samples (24 kW).

Hours After Irradiation	Mock Fuel Pin Exposure Rate (mR/hr at 1 foot)	Vials Exposure Rate (mR/hr at 30 cm)	Mock Fuel Pin Exposure Rate (mR/hr at 3 cm)	Vials Exposure Rate (mR/hr at 3 cm)
0	415900	19450	59889600	2800800
1	4100	5908	590400	850752
6	1156	666	166464	95890
12	304	308	43790	44395
24	63	138	9125	19901
36	31	85	4419	12177
48	17	59	2411	8516
60	9	45	1331	6467
72	5	36	741	5193
96	2	26	236	3725
120	1	20	83	2922

Table II. Projected dose rates for vials containing irradiated uranyl nitrate solutions under the conditions that will produce the least radioactive samples (3 kW_s).

Hours After Irradiation	Vials Exposure Rate (mR/hr at 30 cm)	Vials Exposure Rate (mR/hr at 3 cm)
0	5915	592900
1	1599	160000
6	151	15120
12	58	5820
24	25	2490
36	15	1501
48	10	1038
60	8	781
72	6	623
96	4	443
120	3	346

The first irradiation that will take place will be the one that is predicted to produce the least radioactive samples, which is a 10 minute irradiation using 50 kW of power. The dose rates for the aluminum fuel pin and quartz vials will be measured to assess the reliability of the predicted dose rate calculations before any experiments are performed using the irradiated solutions. If the actual dose rates are close to the calculated values, experiments will be initiated. Longer and higher power irradiations predicted to produce more radioactive samples will follow.

Experimental Matrix

Table III is a compilation of the test matrix planned for phase Ia. irradiations in the AFRRI reactor core. This matrix was designed to measure effects of system variables on (1) the radiolytic-gas generation rate and composition (2) chemistry of the fuel solution after irradiation (peroxide concentration, iodine speciation, solution pH, and sorption behavior of Mo and other fuel components on the Mo recovery column). System variables to be studied are irradiation time, reactor flux, uranium concentration, LEU vs. DU concentration, pH, and presence or absence of stainless steel (material of construction).

All of the tasks to be performed will allow analysis of the composition and rate of generation of the radiolytic gas production during irradiation. These results can be directly related to the operation of a full-sized AHR and will aid in design of the gas-handling loop that will circulate the hydrogen, oxygen, oxides of nitrogen, nitrogen, and fission gases that are formed during operation. In this loop, a chilled reflux condenser will return easily condensable vapors as well as oxides of nitrogen that will react with water to the reactor. Also, a catalytic recombiner will react the hydrogen with oxygen to add water back to the reactor vessel. Understanding the gas composition and generation rates will be critical to maintaining the AHR in a pH range that will avoid uranium precipitation.

In most tests, a set of batch equilibrium sorption partitioning experiments will be performed to determine the effect of that parameter on the Mo extraction column. Iodine speciation analyses will also be performed on some of the irradiated solutions to determine the ratios of I^- , I_2 , and IO_3^- in the solution versus process variables. The pH and peroxide concentration will also be measured for most samples.

The mock fuel pin will hold two sample vessels at a time. Therefore, two samples vials will be run under the same external flux and irradiation time. Eleven separate irradiations are planned. The two samples are designated as xa and xb, x is a number between 1 and 11.

Samples 1a, 2a, 3a, 4a, 5a, and 6a all contain 145g-LEU/L and are run at the same external flux. The variable being varied is irradiation time. Note that sample volumes in 5a and 6a are reduced to keep radioiodine production below reactor limits.

Sample 1b will be irradiated with 1a and will show the effects of higher LEU concentration in the fuel solution. The same effect will be studied in samples 7a and 7b at a lower external flux. Sample 2b will be irradiated with 2a and will use the same uranium concentration, but 55 g/L of LEU will be replaced by DU, therefore looking at the effects of a lower fission rate for the same uranium and nitrate concentrations. Sample 3b replaces 100g/L of LEU with DU, showing the same effect. Comparison of samples 8a and 8b will show the same effect at a higher external flux.

The effects of varying the initial solution pH are being examined in samples 4a, 4b, 10a, and 10b. The effects of varying external reactor flux are being measured by a comparison of samples 1a, 7a, 8a, and 9a. The effects of the presence of a stainless-steel screen (material of construction) will be made comparing results of 6a to 6b, 10a to 10a, and 10b to 11b.

Table III. Eleven irradiations will be performed at AFRRRI for phase Ia. experiments. Two samples will be irradiated at the same time.

Sample	Sample Volume (mL)	LEU Concentration (g-U/L)	DU Concentration (g-U/L)	Irradiation Time (min)	External Neutron Flux ($\times 10^{12}$)	Power in Sample (W/mL)	pH	SS
1a	5	145	0	10	0.7	~1	1	No
1b	5	200	0	10	0.7	~1	1	No
2a	5	145	0	20	0.7	~1	1	No
2b	5	90	55	20	0.7	~1	1	No
3a	5	145	0	40	0.7	~1	1	No
3b	5	45	100	40	0.7	~1	1	No
4a	5	145	0	80	0.7	~1	1	No
4b	5	145	0	80	0.7	~1	0.3	No
5a	2	145	0	160	0.7	~1	1	No
5b	2	145	0	160	0.7	~1	1	Yes
6a	1	145	0	320	0.7	~1	1	No
6b	1	145	0	320	0.7	~1	1	Yes
7a	5	145	0	*	0.1	TBC	1	No
7b	5	300	0	**	0.1	TBC	1	No
8a	5	145	0	*	3	TBC	1	No
8b	5	45	100	**	3	TBC	1	No
9a	1	145	0	*	10	TBC	1	No
9b	1	145	0	**	10	TBC	2	No
10a	5	145	0	80	0.7	0.7	1.5	No
10b	5	145	0	80	0.7	0.7	2	No
11a	2	145	0	160	0.7	TBC	1.5	Yes
11b	2	145	0	160	0.7	TBC	2	Yes

*Time to achieve the same dose as 1a.

**Times set by 7a, 8a, and 9a.

Phase Ia. Experiments

- 1) GC-MS analysis of gases generated after irradiation
- 2) pH determination
- 3) H₂O₂ determination
- 4) K_d measurements – 6 for each 5 mL sample
- 5) Iodine speciation

Phase Ib. Experiments

- 1) Column Run
- 2) Purification of Mo-99 product

The experiments that will be done are separated into phase Ia. and Ib. Irradiated solutions will be used for both phase Ia. and phase Ib. experiments. There are several features of the phase Ia. and Ib. experiments that will decrease the likelihood for release of radioactive materials in the AFRRRI radiation chemistry laboratory. Phase Ia. experiments include an analysis of the gases produced from irradiation, a pH measurement, determination of peroxide concentration, batch sorption studies, iodine speciation experiments, and examination of stainless steel screen material after irradiation. Phase Ib. experiments consist of a column run and purification of the Mo-99 product. A detailed procedure for the phase Ia. and phase Ib. experiments that will be performed along with built-in engineering constraints to minimize the risk for release of radioactive materials is discussed below.

Experimental design features to minimize risk for release of radioactive materials

It is important to note that most experimental work will be performed in closed vessels except when transfers are being made between containers, a pH measurement is made, and fractions are collected during the column run (bottles will be capped immediately after each fraction is collected). Tungsten shielding containers, trays, acidic absorbent paper, and sealed ziplock bags will serve as secondary means of containment for liquid spills. Fission gases will be collected in an empty gas cylinder (method of collection is described below).

GC-MS analysis of gases generated after irradiation

After the samples are removed from the aluminum mock fuel pin, they will be immediately placed in specially fabricated tungsten shielding containers. The samples will be gently clamped on a ring stand for gas analysis. The headspace gas will be inserted into the GC-MS system using a gas manifold (Appendix III for a diagram of the setup). A vacuum pump and He gas will be used to introduce the sample to the system. The tungsten shielding container has a small opening on the lid to allow connection to the GC-MS for gas analysis. The shielding container with vial will be kept in a tray at all times to serve as a secondary container. A large tray that encompasses the entire floor of the hood is also present. The hood will be lined with absorbent paper that readily absorbs acidic solutions. The shielding container, trays, and absorbent paper should serve as adequate protection to mitigate contamination from a liquid spill. The gases that will be introduced into the GC-MS instrument will be vented into two vacuum pumps, which will be connected to a 10 gallon holding tank that can handle a pressure up to 175 psi. The holding tank will be connected to a pressure switch (rated to 13.2 – 16.2 psi), which will be connected to a relay to turn on a compressor pump to fill the empty cylinder with the exhaust gases from the GC-MS. The compressor pump runs on a 230V/30A current with a discharge pressure of 5000 psi. All gases will be pumped into gas cylinders (certified to 5000 or 5500 psi), which will be stored in the AFRRI hot cell room until Kr and Xe isotopes have decayed enough for release. Any gaseous iodine that is produced will stick to the molecular sieve columns in the GC-MS instrument.^{1,2} By pumping the gases directly into an empty cylinder, gases will not be released in the hood and up the stack.

pH measurement

Immediately after the gaseous samples have been introduced into the GCMS, the top of the shielding container will be removed, the plastic valve will be disconnected from the quartz vial, and the solution will be transferred to a scintillation vial in a tungsten shielded container that was specially fabricated to hold 4 scintillation vials. A long plastic transfer pipette (chosen to maximize distance to minimize dose to the worker) will be used to transfer the solution from the shielded quartz vial to the shielded scintillation vial. The shielded container will serve as a secondary container for the scintillation vial now containing the irradiated solution, but it will also be kept in a tray. A pH measurement will be taken by placing the micro pH electrode directly into the shielded scintillation vial containing the irradiated solution. The electrode will be wiped after the measurement has been recorded and it will be kept covered. The pH meter and electrode will be placed in one of the hoods for all measurements.

Peroxide concentration measurement

A semi-automatic pipette will be used to transfer an aliquot (0.1 – 1 mL) of the irradiated solution to a solution in a shielded scintillation vial that will have a final volume of 10 mL. The

full procedure is described in Appendix IV. A 10 – 100 fold dilution will be performed on the aliquot, and ~2 mL of that solution will be transferred to a plastic cuvette with cap using a long plastic transfer pipette. The cuvette will be placed in a shielded container. All transfers will be made in a tray in a hood. Before the shielded container containing the cuvette is removed from the hood, health physics will smear/survey the containers to be sure that nothing is contaminated. The container will then be placed in a sealed ziplock bag. The cuvette will quickly be placed in the spectrophotometer (in a tray on the benchtop), and the absorbance at 350 nm will be measured. The cuvette will be removed from the spectrophotometer, placed in the shielding container in a clean ziplock bag, and brought back to the hood.

K_d measurements

The remaining irradiated solution will be used to measure distribution ratios for the uptake of Mo-99 and other fission products by S80 (pure TiO₂), T52 (75% TiO₂ & 25% SnO₂), and T5M (95% TiO₂ & 5% ZrO₂). Two different sets of batch studies will be performed for each 5 mL irradiated sample. About 10 mg of each sorbent will be contacted with 0.8 – 1 mL of irradiated solution, and about 100 – 500 mg of each sorbent will be contacted with 0.2 – 0.5 mL of irradiated solution. The process for the batch study tests is described below. The sorbents will be pre-weighed in culture tubes, and a semi-automatic pipette will be used to transfer the irradiated solution from the shielded scintillation vials to the culture tubes with sorbent. The culture tubes will be in a rack in a tray surrounded by lead bricks for shielding. All work will be done in a hood. After the irradiated fuel solution has been added to the culture tubes, the tubes will be capped and the area around the cap will be wrapped with parafilm. Next, the culture tubes with irradiated solution will be surveyed by health physics, placed in a shielded container, and put in a clean sealed ziplock bag for transport to a thermostated shaker bath. The thermostated shaker bath will be kept on a table that can hold up to 1500 pounds. The shaker bath will be shielded using lead bricks. The samples will shake for about 24 hours and then they will be removed from the bath, placed in a shielded container in a clean ziplock bag, and placed in a rack in a tray surrounded by lead bricks in one of the hoods. Shell vials will be placed in the same rack. Long plastic transfer pipettes will be used to transfer the solution into a syringe fitted with a 0.22 µm filter in a syringe shield. After the solution has been added to the syringe, the plunger will be pushed back into the syringe to filter the solution into the shielded shell vial. Small aliquots (10 – 100 µL) will be collected for counting, and the remaining solutions will be kept in the shielded shell vials with caps behind lead shielding in the hood.

Iodine speciation

The procedure for examining the speciation of radioactive iodine in the gaseous and solution phases is described below. The new hazards involved are working with small amounts of chloroform, handling solutions that contain a significant amount of fission products, and reducing volatile I₂ to non-volatile I⁻. Personal air monitoring systems equipped with filter cartridges designed specifically for iodine capture will be placed in front of both hoods in the laboratory. Proper gloves such as poly vinyl alcohol gloves will be worn when handling chloroform, and all work will be done in a hood. Samples will not exceed the 2% DAC limits for working in the hoods. Specialized tungsten shielding and lead bricks will be used to shield samples at all times.

The first step in the analysis of iodine species involves reducing volatile I_2 to non-volatile I^- in a closed system in a hood. After reduction, the solution will be kept in a shielded container in a tray in a hood. Transfers in and out of the hood will be surveyed by health physics, kept in a shielded container, and transferred in a sealed ziplock bag.

1. Using a manifold equipped with a metering valve, the I_2 -containing gas mixture will be passed through a bubbler containing 10 mL of 0.1 M Na_2SO_3 in 0.1 M NaOH (to reduce I_2 to non-volatile I^-). Six 100 μ L samples will be collected and counted.

2. Using a shaker bath preset to a high mixing frequency, 0.5 mL of the irradiated uranyl nitrate solution will be contacted with 0.5 mL of chloroform for 5 minutes at room temperature. This will be repeated three times. A 10 μ L aliquot of each organic phase will be added to the counting vials that contain 100 μ L of 0.1 M Na_2SO_3 in 0.1 M NaOH (to reduce I_2 to non-volatile I^-). The samples will be counted.

3. 0.5 mL of the irradiated solution that has been contacted with the chloroform will be used and contacted 3 times with 0.1 M I_2 in chloroform for 5 minutes at room temperature. This will be repeated for a total of 3 contacts. A 10 μ L aliquot of each organic phase will be added to the counting containers that contain 100 μ L of 0.1 M Na_2SO_3 in 0.1 M NaOH (to reduce I_2 to non-volatile I^-). The samples will be counted.

4. The activity remaining in the aqueous phase is expected for iodate ion, and a 10 μ L aliquot will be used for counting.

Analysis of stainless steel screen

For irradiation experiments that contain a small piece of stainless steel, all other experiments (gas analysis, pH measurement, peroxide concentration determination, and batch studies) will still be performed. When the solution is transferred from the shielded quartz vial to the shielded scintillation vial using a long transfer pipette, the piece of stainless steel will be left in the quartz vial to examine its integrity at a later date. The quartz vial will be capped. A small aliquot of the solution will be submitted for ICP-MS analysis to determine if the stainless steel material dissolved or degraded.

Column run

Approximately 2 L of depleted uranium as uranyl nitrate solution will be passed through a 1.5 cm ID x 4 cm L S80 column in 2 hours. It will be spiked with 10 mL of irradiated solution. The quartz vials will be placed in shielded containers after removal from the fuel pin, and the plastic valving top will be removed after gas analyses described above have been performed. Each 5 mL sample will be poured directly into 2 L of depleted uranium solution. The column will be kept at 60°C using heat tape connected to a thermocouple with the over temperature set at 70°C (if the heat tape reaches a temperature above 70°C, it will turn off power to the heat tape), and it will be shielded using tungsten ribbon (can withstand temperatures up to 450°C). The feed solution will be stirred on a stirring hot plate and shielded with lead bricks. The feed solution will be loaded onto the S80 column in the up-flow direction at a flow rate of 16 mL/min. The column effluent will be collected in 13 fractions (150 mL each) using a fraction collector that will be shielded using lead bricks. The column will subsequently be washed with 10 CV

(column volumes) of 0.1 M HNO₃, 10 CV of H₂O, 30 CV of 1 M NH₄OH, and 10 CV of H₂O, and 1 fraction (80 mL) of the nitric acid wash and water washes will be collected. Three fractions of the ammonium hydroxide strip will be collected (80 mL each). Elution of Mo-99 will be done with 1 M NH₄OH in the down-flow direction, and the nitric acid and water washes will be done in the up-flow direction. The entire setup will be kept in a tray in a hood with absorbent paper that quickly absorbs acidic solutions. The sample collector also has its own tray.

Purification of Mo-99 product

The 240 mL of ammonium hydroxide strip from the column run will be evaporated using a rotary evaporator. The rotary evaporator will be in a tray in a hood. The vacuum pump will be exhausted into the hood. Three bottles each containing 80 mL of 1 M NH₄OH with the Mo-99 product from the column run will be poured into a 1 L round bottom flask. Once the Mo-99 is in a solid form, it will be dissolved in 1 M nitric acid, and the remaining steps in the Cintichem process (that will take 2.5 – 3 hours) will be performed. A flow chart of the procedure is shown in Appendix V. Again, all work will be done in a hood in a tray with lead bricks and tungsten ribbon serving as the main sources of shielding.

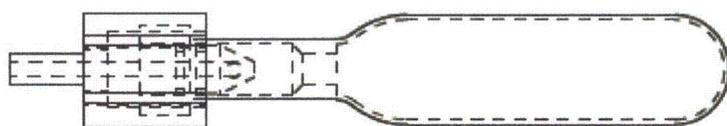
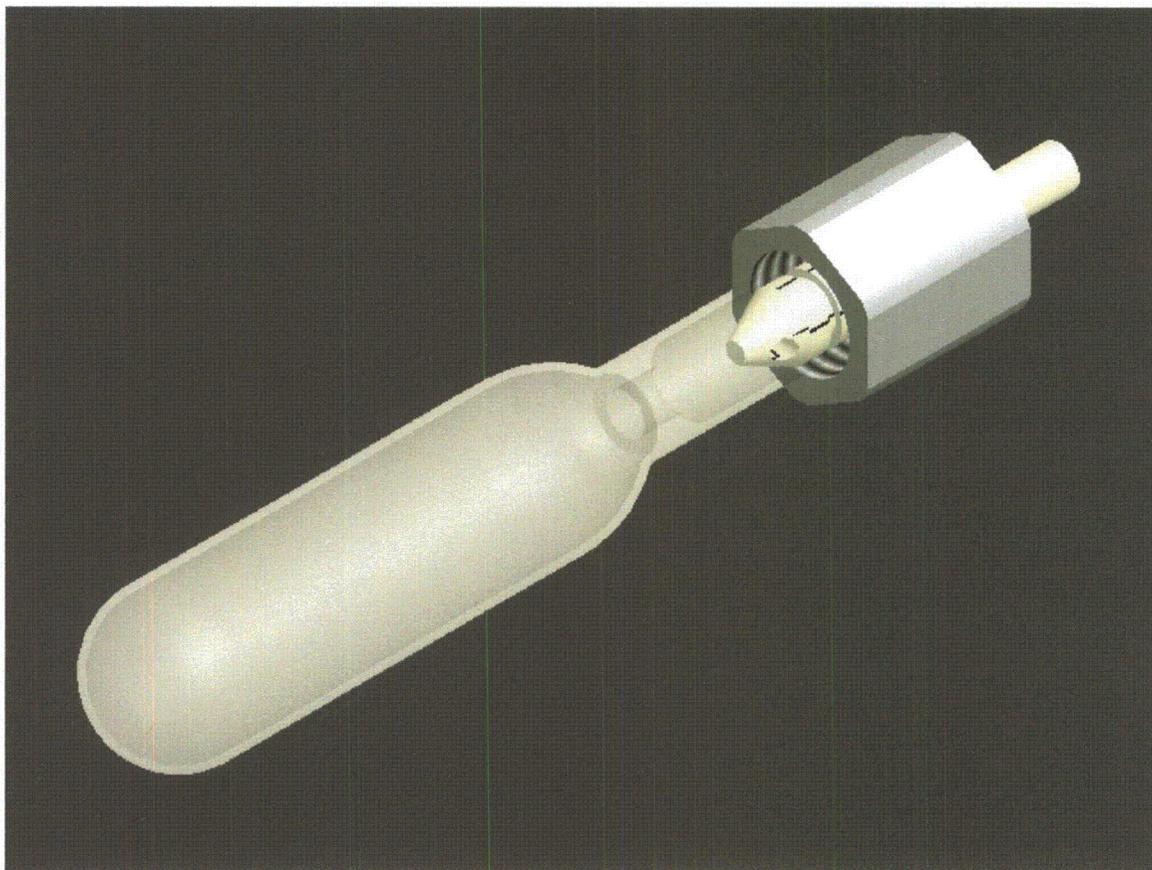
Radiation exposure to workers

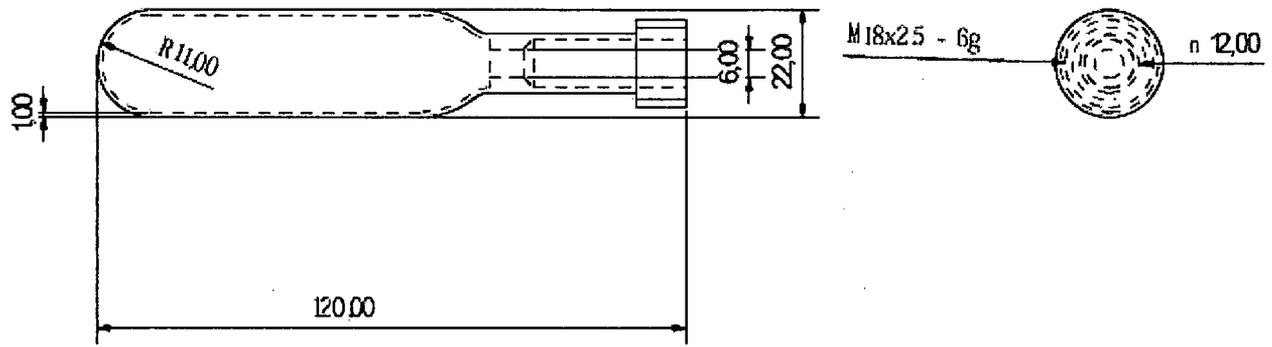
A dry run of the experiments has been performed at AFRRI to help determine maximum worker dose rates. Time and distance measurements were determined for all experiments. Phase Ia. experiments can all be performed at a distance of 30 cm, but hands will be in direct contact with shielded solutions. The amount of time required for all phase Ia. experiments is 31 minutes and 37.5 minutes for all phase Ib. experiments. The maximum unshielded dose rate for the samples will be 140 mR/hr. Multiple workers will be performing all experiments and samples will be shielded at all times to minimize the dose. Phase Ia. and Ib. experiments will not be performed by the same individual and a total of 6-8 workers will be completing the experiments. It is important to note that most samples that will be handled will be much less than 140 mR/hr, as this is predicted to be the hottest sample. The maximum amount of dose for phase Ia. and Ib. experiments is ~160 mR split between 6-8 workers. Each individual would obtain a dose of 20-27 mR for the hottest samples without shielding.

References

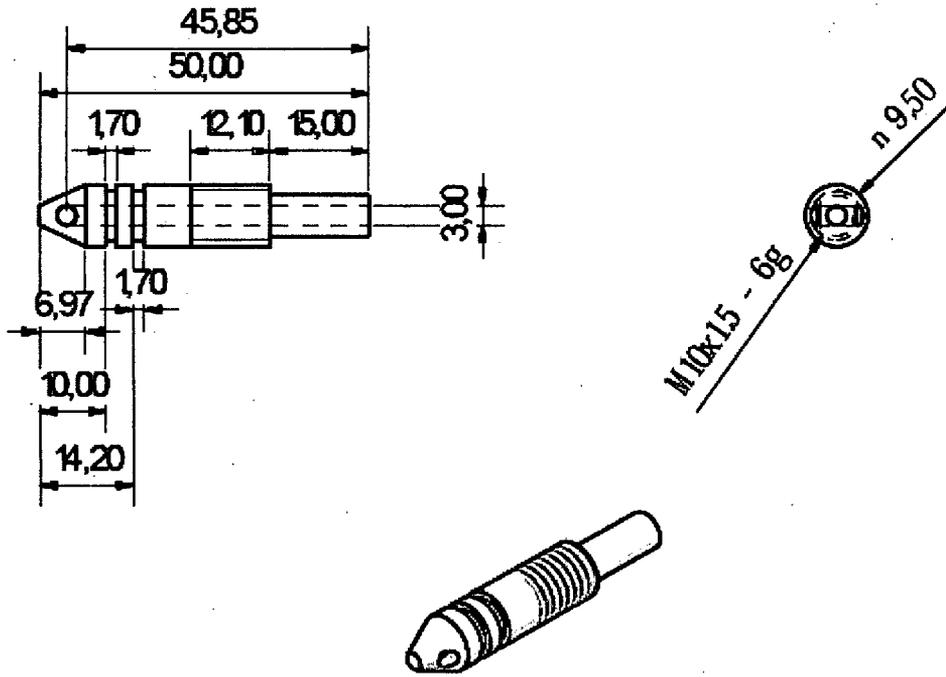
1. Ackley, R.D. and Z. Combs. "Applicability of Inorganic Sorbents for Trapping Radioiodine from LMFBR Fuel Reprocessing Off-Gas." **ORNL/TM-4227**, Oak Ridge National Laboratory, (1973).
2. Atkinson, L. P. and Natoli, G. "Gas Chromatographic Determination of Inorganic Sulfide and Organic Thiol Compounds Using the Catalyzed Azide-Iodine Reaction." *Analytical Chemistry*, **46**, (1974), p. 1316-1319.

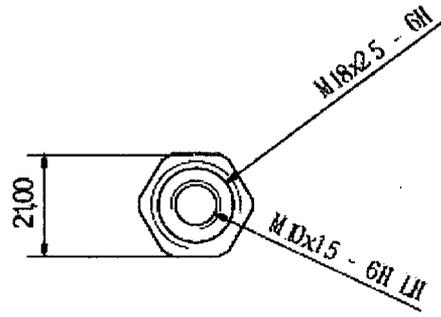
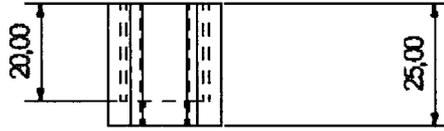
Appendix I. – Quartz primary containment vial drawings.



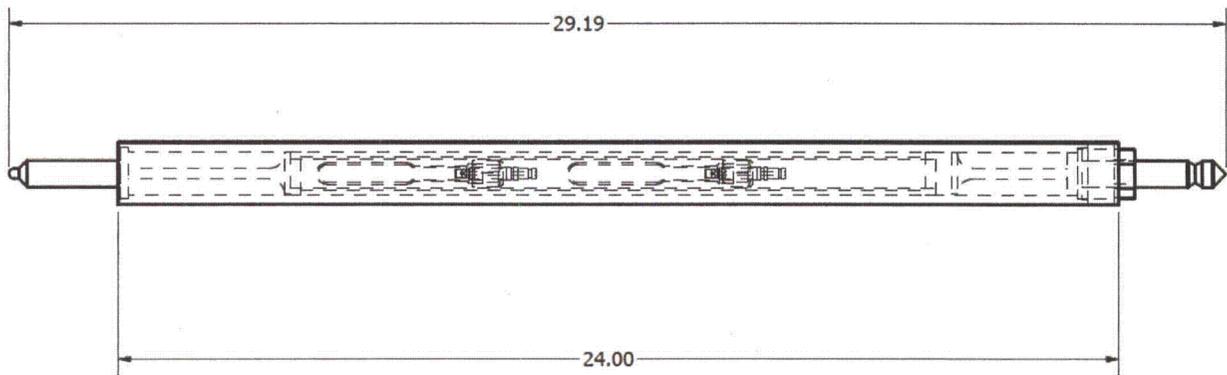
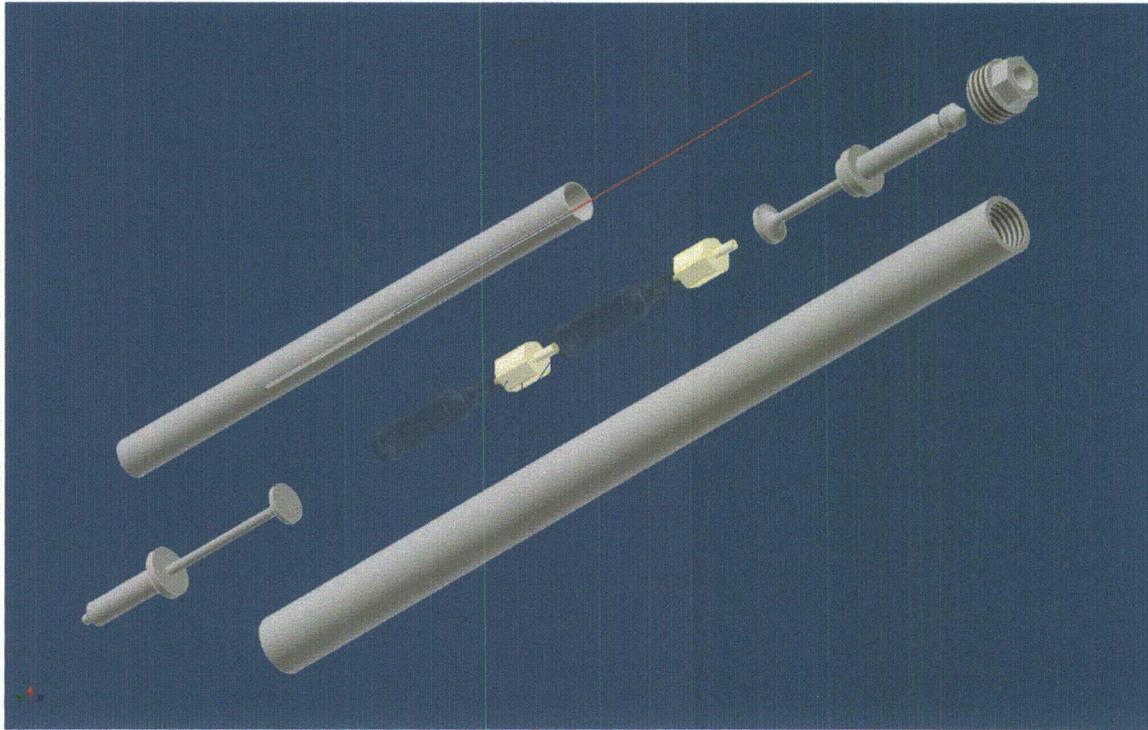


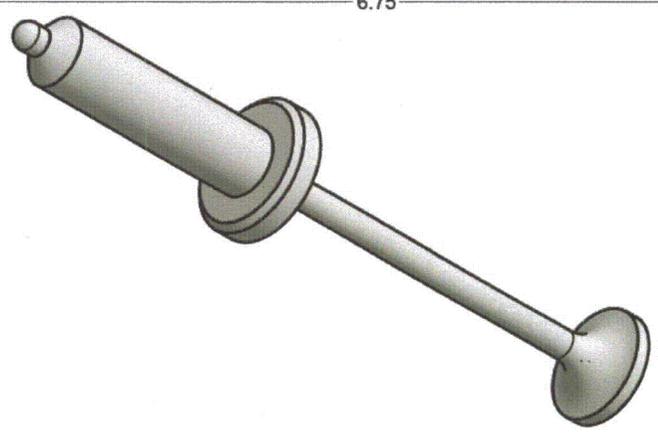
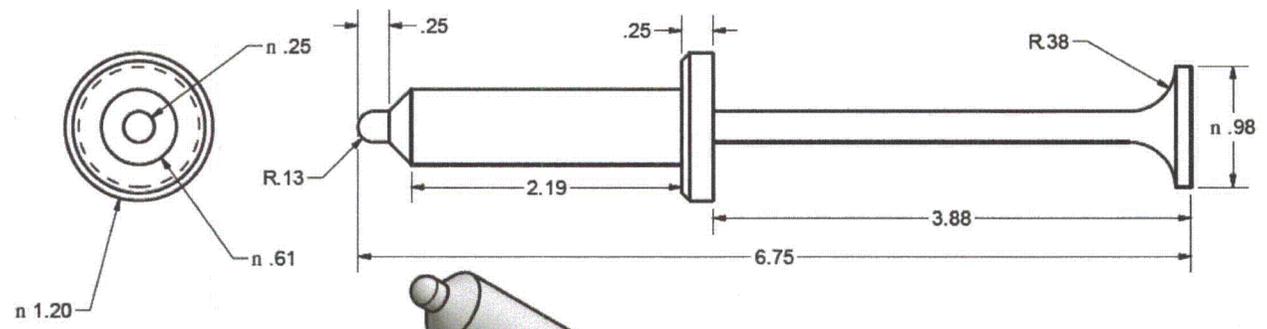
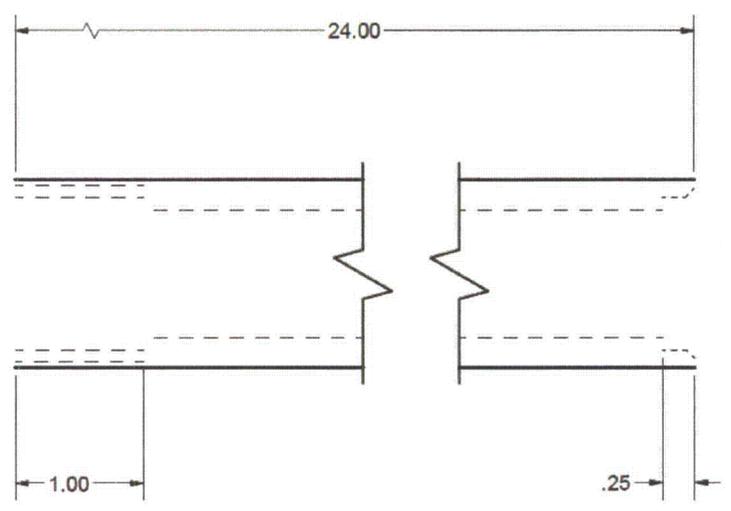
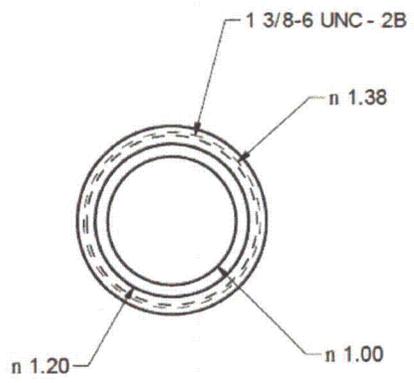
Material: Quartz
 Mass: 18 g

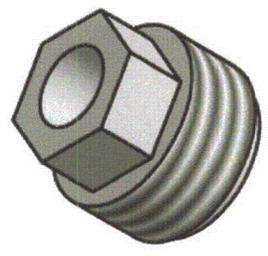
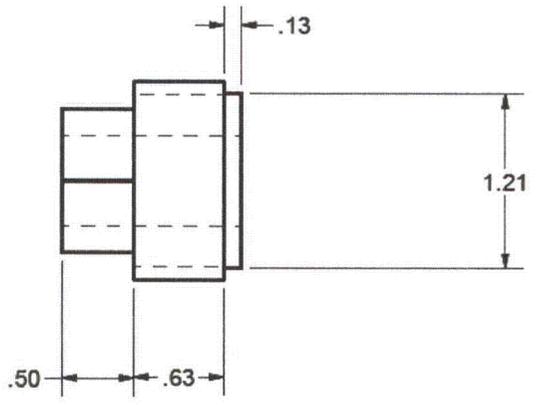
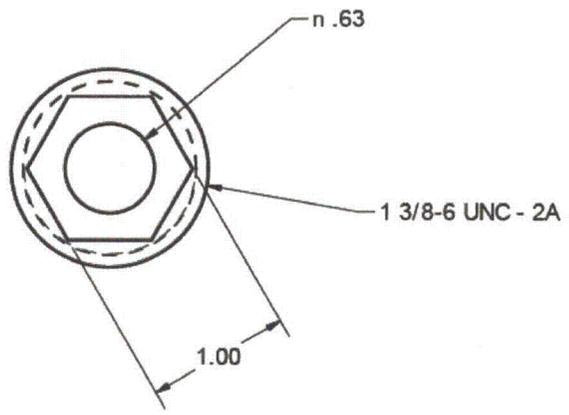
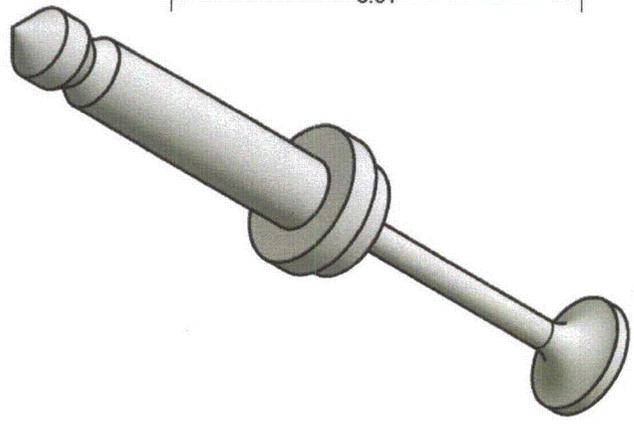
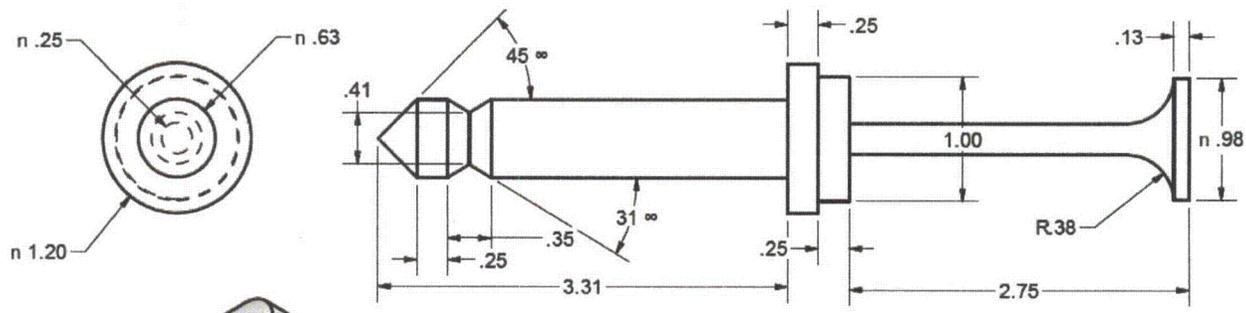




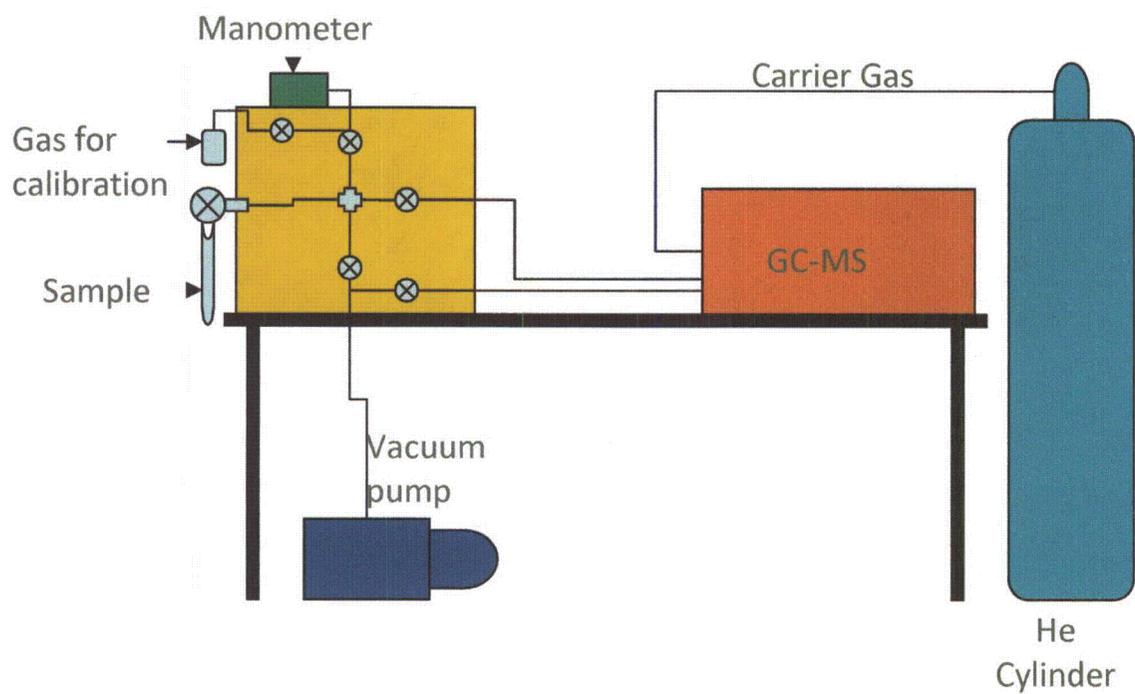
Appendix II. – Aluminum mock fuel pin for secondary containment.







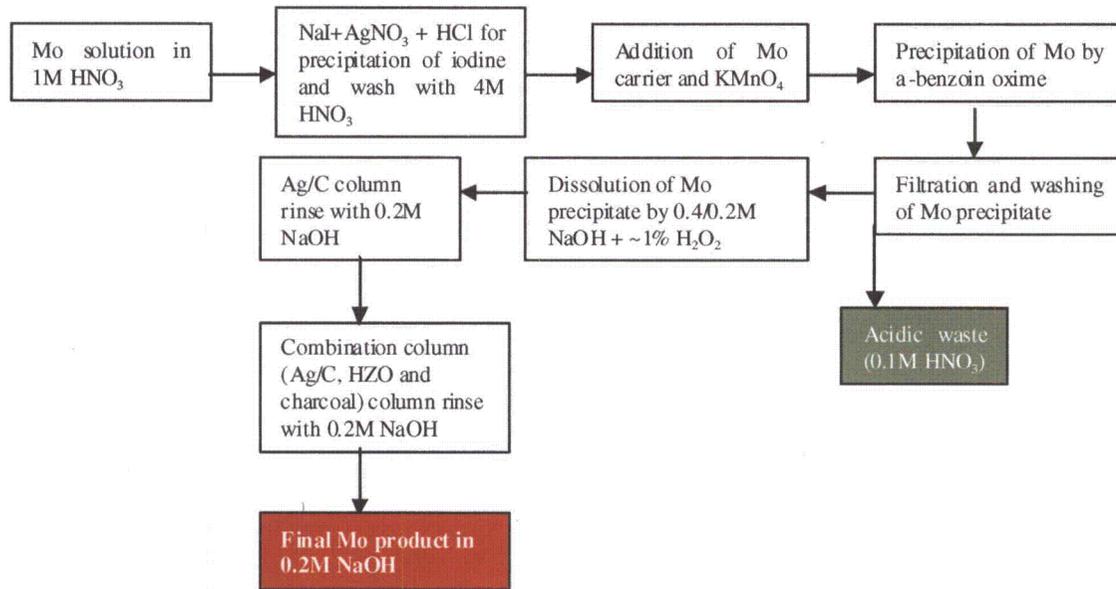
Appendix III. - Diagram for sample introduction into GC-MS.



Appendix IV. - H₂O₂ Determination in 145 g-U/L as Uranyl Nitrate

1. Prepare 100 mL of a solution containing 0.05 M HNO₃ and 0.02 M sulphanilamide. This will be Solution A.
2. Prepare 100 mL of a solution containing 1.3 g KI, 0.2 g NaOH, and 0.02 g (NH₄)₂MoO₄. This will be Solution B.
3. Prepare 100 mL of a solution containing 2 g of acid potassium phthalate. This will be Solution C.
4. Take an aliquot of irradiated fuel solution (0.1 – 1 mL) and add it to Solution A for a total volume of 5 mL.
5. Prepare 5 mL of a 1:1 mixture of Solution B:Solution C.
6. Combine solutions from steps 4 and 5 for a total volume of 10 mL.
7. Take 0.1 - 1 mL and measure the absorbance at 350 nm for the production of I₃⁻.
8. Prepare a blank with DU (non-irradiated) for each sample that contains a different volume of irradiated uranyl nitrate solution.

Appendix V. – Flow chart for Mo-99 purification process



ATTACHMENT #4

Limiting Condition of Operation 3.7 and
Surveillance Requirement 4.6

3.7 BYPRODUCT MATERIAL SEPARATION EXPERIMENTS

Applicability

This specification applies to experiments involving the separation of byproduct materials at the TRIGA reactor facility.

Objective

The objective is to ensure that health and safety of the public and staff are not endangered by the discharge of radioactive materials in the event of an experiment malfunction and ensure that adequate radiation-monitoring equipment is available.

Specification

The following limitations shall apply to the separation of byproduct materials:

- a. All separations must take place in the Radiation Chemistry Laboratory of the AFRRRI facility.
- b. Area Radiation-Monitoring System. The area radiation-monitoring (ARM) system within the Radiation Chemistry Laboratory shall have two detectors located in the main area and, when in use, one detector located in the hot cell.
- c. Air Particulate Monitor. The air particulate monitor (APM) will sample the air in the Radiation Chemistry Laboratory.
- d. Radioiodine Stack Monitor. The radioiodine stack monitor will sample and measure radioiodines in the building exhaust system. Alarm of this unit will cause closure of the positive self-sealing dampers, preventing release of radioiodines to the public.
- e. Air exhausting from the Radiation Chemistry Laboratory, including from the fume hoods, will pass through high-efficiency particulate (HEPA) and high-efficiency gas absorber (HEGA) filter banks prior to discharge from the TRIGA reactor facility.
- f. The Radiation Chemistry Laboratory will be maintained under negative pressure while separations are being performed.

Bases

- a. This specification is intended to ensure that all separations of byproduct material are performed in a facility equipped with adequate monitoring systems and engineered controls to provide safety to the public and staff.
- b. The radiation monitors provide information to personnel of any existing or impending danger from radiation, to give sufficient time to evacuate the facility and take necessary steps to prevent the spread of radioactivity to the surroundings.
- c. The air particulate monitor provides information to personnel of any existing or impending danger from radiation as a result of malfunction of an experiment and gives time to evacuate the facility if necessary.

- d. The automatic closure of the ventilation system prevents the release of radioiodines to the atmosphere outside the facility boundary.
- e. Passage of exhaust air from the Radiation Chemistry Laboratory through HEPA and HEGA filter banks ensures that radioactive materials released in the event of an experiment malfunction are not released to the atmosphere outside the facility boundary.
- f. This specification is intended to ensure that in the event of an experiment malfunction in the Radiation Chemistry Laboratory the staff within the AFRRRI facility is not exposed to the spread of radioactivity.

4.6 SEPARATION OF BYPRODUCT MATERIAL IN THE RADIATION CHEMISTRY LABORATORY

Applicability

These specifications apply to the surveillance requirements for radiation-monitoring and ventilation systems related to the Radiation Chemistry Laboratory.

Objective

The objective is to assure that the radiation-monitoring equipment and ventilation system are operating and to verify appropriate alarm settings when the separation of byproduct material is to be performed.

Specification

- a. The area radiation-monitoring system and the air particulate monitor shall be verified operable by a channel check daily when separations of byproduct material are being performed. They shall be calibrated annually, not to exceed 15 months, and channel tested quarterly, not to exceed 4 months, when separation experiments are being performed.
- b. The radioiodine stack monitor shall be verified operable by a channel check daily when separations of byproduct material are being performed. It shall be calibrated annually, not to exceed 15 months, and channel tested quarterly, not to exceed 4 months, when separation experiments are being performed.
- c. The operating mechanism of the positive sealing dampers in the Radiation Chemistry Laboratory ventilation system shall be verified operable and visually inspected monthly, not to exceed 6 weeks, when separation experiments are being performed.
- d. The differential pressure across the HEPA filter bank shall be checked weekly when separation experiments are being performed. When the pressure is twice the initial pressure at time of installation, the filters will be replaced. The HEPA filters will be tested biennially and replaced when necessary, when separation experiments are being performed.
- e. The negative pressure of the Radiation Chemistry Laboratory will be verified daily when separations of byproduct material are being performed.

Bases

- a. Experience has shown that daily verification of area radiation-monitoring and air monitoring system set points in conjunction with a quarterly channel test is adequate to correct for any variation in the system due to a change of operating characteristics over a long time span. Annual calibration ensures that the units are within the specification demanded by the extent of use.
- b. Experience has shown that daily verification of radioiodine stack monitoring system set points in conjunction with a quarterly channel test is adequate to correct for any variation in the system due to a change of operating characteristics over a long time span. Annual

calibration ensures that the units are within the specification demanded by the extent of use.

- c. Experience has demonstrated that the tests of the ventilation system on a monthly basis are sufficient to assure proper operation of the system and control of the release of radioactive material.
- d. The manufacturer's specification for the HEPA filter banks indicates that an increase in differential pressure across the filter less than a factor of two from the time of installation indicates a properly functioning filter bank. Any changes greater than twice the initial pressure requires that the unit be replaced. The average lifetime of a HEPA filter is approximately three years. By testing the unit for its gas absorbing properties biennially, the units will be assured to function as designed while separation experiments are ongoing.
- e. Daily verification of the negative pressure difference of the Radiation Chemistry Laboratory relative to the surrounding areas will ensure that the release of any radioactive material will be confined to the controlled area and exhausted through the appropriate filters.