# **PROCESS DESIGN & DEVELOPMENT FOR ELECTRON** ACCELERATOR-DRIVEN Mo-99 PRODUCTION FROM U<sub>3</sub>O<sub>8</sub> TARGETS

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in a dedicated target irradiation station.



The electron beam was converted to x-rays using water-cooled Ta. The beam energy was 40 MeV at various Kw-hr ranging from 1.6 to 30. Isotope yields were determined experimentally and compared with MCNP simulations.



Super conducting electron LINACS (linear accelerators) have the capability to induce neutron fluxes and photo-fission sufficient enough to produce medical isotopes from uranium targets. To facilitate this effort, Argonne has developed new chemical processes that are compatible with large  $U_3O_8$  targets and subsequent purification of <sup>99</sup>Mo from other fission and activation products.

The Argonne Model for Universal Solvent Extraction code simulates solvent extraction processes like those used to recover U-targets and purify <sup>99</sup>Mo. The simulations aid the researcher in optimizing flowrates, concentrations, and the number of stages to achieve a desired purity. The results are tested against contactor banks to be used in the actual processes.



to/from helium



# RADIOCHEMISTRY

The separation and purification of <sup>99</sup>Mo from target materials, fission products, and impurities is a daunting challenge. We developed a process that leverages solvent extraction and ion exchange to purify <sup>99</sup>Mo from the uranyl target solution. A new process: MoLLE (Molybdenum Liquid-Liquid Extraction) was specifically designed for this technology.



60

*Eluate (mL)* 

80

20

# ACKNOWLEDGMENT

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100







# REMOVAL OF <sup>99</sup>TC, <sup>95</sup>ZR, AND <sup>95</sup>NB FROM SOLUTIONS OBTAINED AFTER DISSOLUTION OF IRRADIATED MOLYBDENUM TARGETS

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## **OVERVIEW**

Removal of ground state <sup>99</sup>Tc that accumulates during <sup>99</sup>Mo production could allow the first elution of a <sup>99m</sup>Tc generator to be used instead of discarded. Side-reaction product impurities, <sup>95</sup>Zr and <sup>95</sup>Nb, should be removed simultaneously.

The existing process for <sup>95</sup>Zr and <sup>95</sup>Nb impurity removal utilizes an Fe(III) co-precipitation process in alkaline solutions of K<sub>2</sub><sup>99</sup>MoO<sub>4</sub>

Replacement of Fe(III) with Fe(II) under identical process conditions results in the removal of <sup>99</sup>Tc via redox reaction:

 $3Fe^{2+}_{(aq)} + Tc(VII)O_4^- + 7H_2O \leftrightarrow 3Fe(OH)_{3(s)} + Tc(IV)O_{2(s)} + 5H^+$ 

Reaction stoichiometry shows that the favorability of the redox

## **APPROACH**

The removal mechanism of Tc (reduction and incorporation) differs from that of Nb/Zr (adsorption), so we approached the problem first from a Tc-focused perspective and then explored Nb/Zr removals under the narrowed conditions.

Studies used radiotracers <sup>99</sup>Mo, <sup>99m</sup>Tc, <sup>99</sup>Tc, <sup>95</sup>Zr, and <sup>95</sup>Nb



reaction is strongly dependent Tc(VII) and Fe(II)<sub>ag</sub> activities and pН

## **LAB-SCALE RESULTS**

- Effects of several variables examined on Tc removal ( <sup>↑</sup> = very high
  - impact,  $\uparrow$  = moderate impact,  $\downarrow$  = little to no impact):
    - Mo concentration
    - Tc concentration
    - Fe(II) salt (Cl<sub>2</sub>, SO<sub>4</sub>,  $I_2$ )  $\uparrow$  Fe(III) concentration
    - Temperature









■ 0.05 M KOH ◆ 0.1 M KOH ▲ 0.15 M KOH





## **PILOT-SCALE RESULTS**

## The process was scaled up to 1-L, 200-g Mo batches of dissolved disk simulant solutions.

- Four trials examined the impacts of Mo and Tc concentrations, solution starting pH (7 or 13), and presence of  $H_2O_2$  on impurity removals
  - Mo concentration had no impact on impurity removals
  - Tc concentration had little impact on Tc removal and no impact on Nb/Zr removal
  - $-H_2O_2$  had a small (3.6%) impact on Tc removals and no impact on Zr/Nb removals
- All pilot-scale demonstrations resulted in excellent impurity removals, meeting required purity and radiopurity specifications



Photos of the large-scale apparatus showing (A) the full assembly, (B) the reaction vessel/evaporation tank fitted with heating mantle and drain valve on the bottom, and addition lines, vent valve, thermocouple, stirrer, and condenser + vacuum line on the top, and (C) the filter used for Fe-precipitate removal.

# CONCLUSIONS

Removal of <sup>99</sup>Tc, <sup>95</sup>Zr and <sup>95</sup>Nb from alkaline K<sub>2</sub>MoO<sub>4</sub> solutions resulting from dissolution of irradiated, enriched Mo disks can be accomplished using Fe(II) precipitation with very minimal changes to existing processes required.

- Fe(III) alone does not remove Tc but does remove Nb and Zr
- Replacing Fe(III)(NO<sub>3</sub>)<sub>3</sub> with Fe(II)Cl<sub>2</sub> or Fe(II)(SO<sub>4</sub>)<sub>2</sub> improves Tc-removal from ~1% to 99.5% under the same processing conditions, between pH 12.7-13.2 and ≥6 mM Fe(II) per 0.1 mM Tc
- Tc-removal is accomplished through the reduction and simultaneous immobilization Tc(VII) by reaction with Fe(II), producing an insoluble, Tc-incorporated Fe(II)-Fe(III) solid
- Tc, Nb, and Zr removals are all optimal at pH 13 (0.1 M KOH)
- Washing the Fe precipitate with 0.1 M KOH recovers 2.0-3.5% Mo while impurities remain immobilized

Average impurity removals from large-scale trials (n=3):

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<sup>99</sup>Tc (**97.6%** ± 1.7%), <sup>95</sup>Zr (**99.87%** ± 0.05%), <sup>95</sup>Nb (**99.40%** ± 0.16%)

# ACKNOWLEDGMENT

Argonne's Mo-99 Program is supported by the U.S. Department of Energy National Nuclear Security Administration's Material Management and Minimization Office.

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# DESIGN OF UNIVERSAL NEUTRON IRRADIATOR TARGET STATION AND EXECUTION OF EXPERIMENTS ON LONG DURATION IRRADIATION OF URANYL SULFATE SOLUTION

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## INTRODUCTION

- Argonne National Laboratory (Argonne) is providing technical support to SHINE Medical Technologies, Inc. (SHINE) to perform studies on uranium and iodine speciation in a neutronirradiated aqueous homogeneous uranyl sulfate target solution. SHINE plans to produce Mo-99 by fission of U-235 in a subcritical low enriched uranium solution irradiated with an accelerator-driven deuterium-tritium (DT) neutron source.
- In the past Argonne developed and commissioned scaled down (20L) AMORE experiment to demonstrate Mo-99 separation and purification. Those experiments, while successful, were limited in length of irradiation due to build up of fission product inventory.
- Universal Neutron Irradiator (UNI) experiment was commissioned to allow irradiation of the representative solution sample for up to 6 days to match SHINE irradiation cycle.
- The UNI irradiation setup was designed to irradiate 50 mL of a highly enriched uranium (HEU) target solution for up to 6 days at ~ 0.4 W/mL fission power using a Ta target assembly and beryllium reflector/moderator to produce photoneutrons by irradiation with an electron beam. The use of HEU is necessary to reach the desired fission power.
- Experiment was aimed at studying possible uranyl peroxide precipitation and iodine speciation in the solution under irradiation conditions.

## **MONTE CARLO SIMULATIONS**



- No precipitation was observed in the solution.
- Iodine speciation studies are reported in other poster by Anna Service and coauthors.



## **SOLUTION CAPSULE**



## **REFLECTOR DESIGN**



## **IRRADIATIONS**



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Comparison of the predicted and measured fission power in solution

*Predicted from Monte Carlo simulations for as build system: 0.425 W/cc at 20 kW beam power Measured during experiment: 0.329 W/cc at 20 kW beam power* 

## ACKNOWLEDGMENT

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# **DESIGN OF THE TEST BENCH FOR URANIUM FOILS** DISSOLUTION

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## **OVERVIEW**

Argonne National Laboratory, with support from The United States Department of Energy (DOE) National Nuclear Security Agency's (NNSA's) Office of Material Management & Minimization (M3) is developing technologies for a number of domestic entities in their pursuit of production of <sup>99</sup>Mo/<sup>99m</sup>Tc medical isotope without the use of highly enriched uranium (HEU). One of the potential domestic <sup>99</sup>Mo producers, Eden Radioisotopes (Eden), is developing a Triga-sized, all-target reactor system. This reactor system uses up to 60 of the Argonne-designed annular Low Enriched Uranium (LEU)-foil target. The annular target contains approximately 137 g  $(\pm 12 \text{ g})$  of LEU as a 250-µm  $(\pm 25 \text{ µm})$  thick uranium foil covered on all sides by a thin Ni-foil fission barrier, and two aluminum tubes that form the outer and inner walls of the cladding cylinder. Eden expects to process between three and five targets per process batch, limiting the <sup>99</sup>Mo batch inventory to less than 5000 Ci, within a few hours of their discharge from the reactor, meeting U.S <sup>99</sup>Mo demand with 10 process batches per week. The cladding is mechanically removed before the uranium foil and Ni fission-recoil barrier are dissolved in nitric acid for molybdenum recovery.

The aim of the dissolution demonstrations is to achieve efficient dissolution at subatmospheric pressures. To control gas pressure build-up dissolver is equipped with condenser coils to reflux water and to capture NOx gasses. It also incorporates pump for NOx gas recirculation and oxygen addition system for NOx conversion, which will contribute to decreased pressure, lower amount of acid used for dissolution, and decrease the dissolution time.

## **FUMELESS DISSOLUTION**

The goal for this project was to design an acidic dissolver that can handle up to 750 g of LEU metal foils and total mass of 16 g of Ni that is present as a fission fragment barrier to produce the feed solution with final desired concentration of 450 g-U/L in 1M HNO<sub>3</sub>. The volumes and concentrations of acid as well as the amount of nitrogen oxide gas (NOx: NO, NO<sub>2</sub>, N<sub>2</sub>O<sub>4</sub>) that is produced are determined by the following general reactions:

 $U + 4HNO_3 \rightarrow UO_2(NO_3)_2 + 2H_2O + 2NO_3O_3$ 

 $Ni + 8/3HNO_3 \rightarrow Ni(NO_3)_2 + 4/3H_2O + 2/3NO$ 

In the presence of oxygen, the NO(g) produced in these dissolution reactions can be rapidly converted to  $NO_2(g)$ :

 $NO + 0.5O_2 \rightarrow NO_2$ 

In the presence of water vapor, NO, and oxygen gas, NO<sub>2</sub> can be readily converted to nitrous and nitric acid vapors  $[HNO_2(g)]$  and  $HNO_3(g)$ , which will dissolve in condensed water and flow back down into the dissolver:

$$3NO_2 + H_2O \rightarrow 2HNO_3 + NO_3$$

 $NO + NO_2 + H_2O \rightarrow 2HNO_2$ 

If the NO is fully transferred back to acid, fumeless dissolution can be achieved.

## **P&I DIAGRAM OF THE DISSOLUTION SYSTEM**



## **DESIGN PARAMETERS**

- Operating pressure: 14 300 psi
- Dissolution temperature: 100 180 C
- Uranium mass: 375 750 g
- Uranium enrichment: 19.8 %
- Uranium foil thickness: 250 ±25 µm
- Number of targets: 3 5
- Uranium mass per target: 137±12 g
- Fission fragment barrier: Ni
- Ni mass: 4 25 g
- Initial nitric acid concentration: <68 %</p>
- Final desired HNO<sub>3</sub> concentration: 1 M
- Final desired U concentration: 450 g-U/L
- Nitric acid volume: 0.5 2.5 L
- Total dissolution time: ~2.5 h
- NOx recirculating or trapping
- Iodine trapping
- Cooling coil above the solution for vapor condensing
- Electrical heating for dissolution initiation
- Automated control for pressure/flow between volumes
- Connection to MS/RGA for gas composition analysis

## **DISSOLVER OPERATIONS**



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## **DISSOLVER PROTOTYPE**



## ACKNOWLEDGMENT

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# **ARGONNE SUPPORT FOR ACCELERATOR-BASED PRODUCTION OF MO-99**

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## **OVERVIEW**

- With support from NNSA Material Management and Minimization office (M3) Argonne pursue development of technologies to establish domestic supply of Mo-99 medical isotope.
- NorthStar Medical Radioisotopes, LLC, is in the process of commissioning a medical isotope production facility, which will utilize high-power electron accelerators for production of molybdenum-99 (Mo-99), the parent of technetium-99m (Tc-99m), through photonuclear reactions in molybdenum-100 (Mo-100).



- In this approach, a target comprising multiple thin disks of enriched molybdenum metal is bombarded with a 40-MeV electron beam. Electrons that impinge on the molybdenum target produce bremsstrahlung x-rays that cause the nuclear reaction. Because enriched Mo-100 is expensive, there is a desire to utilize as much beam power as possible to achieve maximum production yield and minimize the size of the target. This requirement leads to very high beam power density (heat deposition in the target), which creates challenging requirements for cooling of the target.
- In electron accelerator facility built by NorthStar, a stack of target (sintered molybdenum) disks is

## **MOTE CARLO SIMULATIONS OF THE TARGET**

Energy deposition in the target and target housing due to electron beam





Energy deposition in the target disks for ideal and realistic beam profiles



Mo-99 production in full scale target



irradiated from two sides by a 40-MeV electron beam, with a total power of 250 kW (125 kW from each side). The target disks are cooled by pressurized helium gas flowing through channels between the disks. Inconel windows in the target housing form a pressure boundary between the helium within the housing and the evacuated beam tube

## **MOTE CARLO SIMULATIONS FOR THE FACILITY**

Model of accelerator-based Mo-99 production facility



Radiation filed outside the shielding due to air duct



Radiation fields inside and outside accelerator vault during accelerator operations





Residual radiation fields inside the target room at different cool down times after 1 year of operations



## **CFD ANALYSIS OF THE TARGET**

Range of operating parameters for target system

Force on first disk due to Venturi effect













Max: 2.605

Min: 0.00030618

2.3156 2.0262 1.7368 1.4474 1.158 0.86855 0.57914 0.28972 0.00030613







## STRUCTURAL ANALYSIS FOR CYCLIC LOAD



The ratcheting analysis using a linear elastic-perfectly plastic material model shows that within 30 cycles, the window enters the plastic shakedown regime and the maximum deformation towards the targets approaches a stabilized value of 0.38 mm.

Window deformation toward the target disk



# ACKNOWLEDGMENT

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## SUPPORTING DEVELOPMENT OF DOMESTIC **SUPPLY OF MO-99**

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### **OVERVIEW**

- With support from NNSA Material Management and Minimization office (M3) Argonne pursue development of technologies to establish domestic supply of Mo-99 medical isotope.
- Argonne supports four domestic partners, namely:
- NorthStar - NorthStar Medical Radioisotopes
- SHINE Medical Technologies
- - Niowave
- Eden

### Argonne's expertise:

- Irradiations services, beam transport, shielding and target design, **MCNPX**
- Radiochemistry separation, purification and recycle processes
- Development of analytical methods for process quality control

## **IRRADIATION CAPABILITIES**

Low-Energy Accelerator Facility (LEAF) houses 53MeV electron linac, and 3MeV Van de Graaff (VDG) generator to study activation and radiation damage to materials



- Irradiated liquid and solid targets at electron linac for Mo-99 production need to be processed and purified to meet required purity specs
- Irradiated targets can be handled in bldg. 211 hot cell
- Irradiations at VDG allow to study radiation damage on targets, materials, electronic components, and equipment without activation. This can be utilized in predicting corrosion rates of stainless steel or other materials under radiation field, durability of various materials and equipment in radiation field for their use in hot cells





### **RADIOCHEMISTRY R&D**

The separation and purification of <sup>99</sup>Mo from target materials, fission products, and impurities is a daunting challenge. Argonne's chemists and chemical engineers leverage their expertise to design radiochemical processes and experimentally validate that produce high-purity <sup>99</sup>Mo from a variety of targets.







Final purification step of Mo-99 from MOLLE raffinate

### ENGINEERING DESIGN AND MODELING

- · Having ability to simulate multiple options before designing, and building experimental setup is extremely important
- Argonne engineers help with design and testing of experimental setup
- Monte Carlo simulations are used to predict activation products and dose rates of irradiated targets and components





rgy deposition, showing: 1) Ta converter and beam window ner holder, 4) beam dump, 5) beam pipe and the assembly U<sub>3</sub>O<sub>8</sub> pellet, 3) target's conta interconnector, 6) flange

### FROM LAB-SCALE TO FULL-SCALE PROCESSES

- Developed processes are first tested at small scale, and then scaled-up to allow full-scale testing and ability to perform work in hot cell
- Many challenges arise during scale-up process, and modifications need to be made to obtain desired product quality or allow for hot cell remote manipulation of the designed equipment









## ACKNOWLEDGMENT

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## DEVELOPMENT OF LOW SPECIFIC ACTIVITY MO-99/TC-99M GENERATOR

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## **OVERVIEW**

With support from NNSA Material Management and Minimization office (M3) Argonne is pursuing development of Low Specific Activity Mo-99/Tc-99m generator.

There are several alternatives to fission-based production of <sup>99</sup>Mo/<sup>99m</sup>Tc. While neutron capture via <sup>98</sup>Mo(n, γ)<sup>99</sup>Mo reaction still requires a nuclear reactor as a neutron source, alternative production methods using <sup>100</sup>Mo(γ, n)<sup>99</sup>Mo or direct production of <sup>99m</sup>Tc via <sup>100</sup>Mo(p, 2n)<sup>99m</sup>Tc can be achieved using commercially available accelerators. A generator system that can handle high chemical concentration of Mo is needed to separate the trace amount of <sup>99m</sup>Tc from low-specific-activity <sup>99</sup>Mo.

#### Dealing with high Mo content different way:

 Instead of designing high-capacity column that can capture Mo, bulk Mo is removed by selective precipitation, while Tc remains in liquid phase. This concept allows for g-Mo content and potentially 10-20Ci generator strength.

## MO REMOVAL AND TC RECOVERY

Solubility of Mo in EtOH decreases as the concentration of EtOH increases

- Tc doesn't co-precipitate with Mo
- Tc yield can be increased by adding EtOH wash
- Precipitation yield of Mo can be further increased by addition of inorganic salt, leading to removal of >99.99% of Mo





### **CONCLUSIONS AND FUTURE WORK**

- Precipitation of bulk Mo with ethanol is a very simple, quick and efficient way to remove 99.99% of g-Mo quantities from Mo/Tc mixture
- γ-alumina serves as a guard column to capture any remaining Mo
- Activated charcoal shows strong affinity for Tc from ethanol, but de-sorption of Tc remains difficult. If Tc is acceptable in EtOH, charcoal could be fully eliminated
- Although regeneration of Mo (re-dissolution of Mo ppt) has been demonstrated during multiple cycles, formation of fine Mo particles eventually leads to some Mo break-throughs (as solids) inside the filter
- Filter material and properties, as well as selecting right charcoal material are main areas that need further optimization

Crystallization of 650 mg  $Na_2MOQ_4$  in fritted reservoir with showing precipitate (black box) and clear supernatant solution (red box). Filtration occurs by evacuated vial with flow rate ~1-2 mL/min through alumina column.

## PRECIPITATE-BASED LSA GENERATOR CONCEPT

Sodium molybdate Na<sub>2</sub>MoO<sub>4</sub> (or other molybdate salt) is not soluble in ethanol (EtOH); pertechnetate is highly soluble in EtOH.

- Starting Mo/Tc solution in water is mixed with EtOH. This leads to precipitation of Mo, while Tc remains in liquid phase
- Mo & Tc are separated by filtration
- Mo in precipitate is recovered by dissolution with water and ready for next milking

## Application: non-fission-based Mo-99 production or direct cyclotron production of Tc-99m (patent pending)



## LSA GENERATOR CYCLES

Typical Tc recoveries after Mo precipitation and  $\gamma$ -alumina are 90.9% ± 1.7% after seven cycles

- Regeneration experiments concentrated on consistently removing 99+% Mo by precipitation, which is not possible by re-dissolving Mo in water only
- Various sparger frits were tested for precipitation/re-dissolution of Mo

Cycle #	Mo in supernatant (µg/mL)	Mo in supernatant (%)	Mo in precipitate (%)	Cycle #	<sup>99m</sup> Tc recovered
1	144.5	0.151%	99.85%	1	91.0%
2	35.1	0.037%	99.96%	2	89.0%
3	28.7	0.030%	99.97%	3	94.4%
4	12.8	0.013%	99.99%	4	89.5%
5	7.73	0.008%	>99.99%	5	91.8%

PE/PTFE/PVDF v SS v Al<sub>2</sub>O<sub>3</sub> frits

frit

Precipitation occurs rapidly and mixing can be accomplished by bubbling air through dispersion tube

89.4%

 Tested activated charcoal shows that both Mo and Tc strongly sorb from EtOH/water mixtures

- Saline itself was not effective in stripping Tc from tested charcoal sorbents
- Tc can be effectively removed from the <u>charcoal</u> by dilute NaOH
- Selective Mo precipitation with EtOH is a great alternative to recover Tc from cyclotron production and recover enriched Mo

## ACKNOWLEDGMENT

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# IMPACT OF IRRADIATION AND DECAY TIME ON CHEMICAL SPECIATION IN URANYL SULFATE SOLUTION

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# INTRODUCTION

- SHINE Medical Technologies, Inc. (SHINE) plans to produce <sup>99</sup>Mo by fission of <sup>235</sup>U in a neutron-irradiated subcritical low enriched uranium (LEU) solution
- Aqueous homogeneous subcritical assemblies for <sup>99</sup>Mo benefit from inherently safe operation, efficient fuel utilization, and simplified target fabrication and processing
- Argonne National Laboratory (ANL) is investigating 2 problems:
  - Radiolytically-generated H<sub>2</sub>O<sub>2</sub> reacts with uranium to form insoluble uranyl peroxide
  - Nonvolatile iodine species in solution react with water radiolysis products to form iodine gas



# **ANL IRRADIATION SETUP**

- At SHINE, fission in the LEU target solution will be sustained by a deuterium-tritium (D-T) accelerator driven neutron source
- ANL's Universal Neutron Irradiator (UNI) an electron accelerator and Ta bremsstrahlung converter to produce photoneutrons
- UNI goal was to irradiate a 50 mL target solution 5 times to reach at least 144 hours of cumulative irradiation time at ~0.3 W/mL



## **URANIUM PRECIPITATION**

- A pH 1, 200 g U/L highly enriched uranyl sulfate (HEU) target solution was irradiated in UNI for steadily increasing lengths of time
  - Before irradiation, 200 ppm Fe<sup>2+</sup> (final concentration) was added to catalyze the destruction of  $H_2O_2$
- After each irradiation, the target solution was photographed in situ with a fiberscope, and a 3 mL sample was collected
- No precipitation was observed in the target capsule or in collected samples for all irradiations



Representative fiberscope images of the HEU solution target in the UNI capsule: (a) photo of solution before irradiation (b) photo after 3<sup>rd</sup> irradiation (c) photo after 5<sup>th</sup>, final irradiation

- Total beam energy deposited on the target exceeded the 2160 kWh target
  - Total of 2365 kWh deposited
  - A maximum fission power density of 0.26 W/mL was reached in the final UNI irradiation



## **IODINE SPECIATION**

- Iodine speciation in solution was determined using series of liquid-liquid extractions to isolate I<sub>2</sub> and I<sup>-</sup> from the initial aqueous phase
  - Volatile iodine (I<sub>2</sub>) partitions to a chloroform phase
  - Iodide (I<sup>-</sup>) partitions to a 0.1 M I<sub>2</sub> in chloroform phase
  - Iodine remaining in the aqueous phase is iodate  $(IO_3^-)$



- The distribution of iodine in solution between I<sub>2</sub>, I<sup>-</sup>, and IO<sub>3</sub><sup>-</sup> species remained constant following an initial transition period
  - The iodide (I<sup>-</sup>) fraction is dominant
  - $I_2$  is the next most abundant species
  - lodate is lowest abundance  $(IO_3^-)$  species
- Very small amounts of I<sub>2</sub> likely being continuously lost to gas phase
  - Iodine was not detected in gas samples until the  $5^{\text{th}},$  final irradiation





#### ◆I<sub>2</sub> (#4) % ▲ IO<sub>3</sub><sup>-</sup> (#4) 20% ■I<sup>-</sup> (#4) ♦ I<sub>2</sub> (#5) ▲ IO<sub>3</sub><sup>-</sup> (#5) 100 200 300 400 500 600 700 I<sup>−</sup> (#5) Time Since EOB (hr)

# **GAS ANALYSIS SYSTEM**

Irradiation Time | Fission Power

- UNI has a gas analysis system to monitor hydrogen and contain fission gases
  - No hydrogen detected during irradiations



## **GAS SAMPLE CONTENTS**

- Sample of the UNI capsule headspace was collected during each irradiation
- Gas cylinder contents:
  422
- <sup>85m</sup>Kr, <sup>133m</sup>Xe, <sup>133</sup>Xe, <sup>135</sup>Xe
- Gas cylinder activation products:
   <sup>24</sup>Na, <sup>51</sup>Cr, <sup>57</sup>Ni, <sup>58</sup>Co, <sup>59</sup>Fe, <sup>60</sup>Co, <sup>187</sup>W
- No iodine detected until final irradiation



# CONCLUSIONS

- UNI with an HEU target was close to the ~0.3 W/mL fission power density expected by SHINE (using D-T and LEU)
- No uranyl peroxide (or other) precipitation was observed after 162 hours irradiation
- Iodine in solution is predominantly (>70%) nonvolatile iodide after an initial transitional period
- Iodine in gas sample not observed until final irradiation (after >100 hours irradiation)

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