Mo-99 2013 TOPICAL MEETING ON
MOLYBDENUM-99 TECHNOLOGICAL DEVELOPMENT

April 1-5, 2013
Embassy Suites Downtown - Lakeshore
Chicago, Illinois

Development and Performance Testing of Two Frontend Processes for Mo-99 Production from LEU-Foil Targets

J. Jerden, A. Gelis, D. Stepinski, M. Brown, J. Bailey,
A. Hebden, L. Hafenrichter, and G. F. Vandegrift,
Chemical Sciences and Engineering Division
Argonne National Laboratory, 9700 South Cass Avenue, Argonne IL 60510 USA

ABSTRACT

Under the Global Threat Reduction Initiative Program, Argonne is developing two frontend options for Mo-99 production to allow the use of LEU-foil targets by current producers. Both options will produce a feed compatible with current Mo-99 purification operations. The two processes under development are: (1) ambient pressure dissolution of LEU-foil in nitric acid followed by extraction of Mo-99 using a titania column, and (2) electrochemical dissolution of LEU-foil in an aqueous bicarbonate solution to produce an alkaline Mo-99 feed. Equipment for these processes has been designed, fabricated, and tested using DU and fission product tracers. Procedures for the remote operation of these processes are being optimized based on tests performed using a hot-cell mock-up. Selected experimental results will be presented in this paper. Full-scale demonstration of these processes will be performed in summer 2013 in hot cells at Oak Ridge National Laboratory using foils irradiated in the High Flux Isotope Reactor.

1. Introduction

The Global Threat Reduction Initiative (GTRI) Conversion Program develops technology necessary to enable the conversion of civilian facilities using high enriched uranium (HEU) to low enriched uranium (LEU) fuels and targets. The conversion of conventional HEU dispersion targets to LEU for the production of Mo-99 production requires approximately five-times the uranium in a target to maintain the Mo-99 yield per target; therefore, processes involving high density metal LEU foil targets are a primary focus of this research and development effort.

Under GTRI, Argonne National Laboratory is involved in a number of research and development activities that support conversion of current Mo producers from HEU to LEU targets. This paper discusses our work on the development of two frontend options to current processes for Mo-99 production from LEU-foil targets: (1) ambient pressure dissolution of LEU-foil in nitric acid followed by extraction of Mo-99 using a titania column, and (2) electrochemical dissolution of LEU-foil in an aqueous bicarbonate solution to produce an alkaline Mo-99 feed.

Both the frontend-process options under development have two major goals. The first goal is to produce a Mo-product solution from the LEU-foil frontend that will be compatible with current purification operations and that will, with the same number of targets irradiated, provide the same or higher yield of Mo-99 at the end of processing. The second goal is to deliver a product solution that is of the same or higher Mo purity than the current solution and is of equal or better compatibility with current purification process.

For the nitric acid process, the irradiated LEU foil and associated 10-15 micrometer Ni fission-recoil barrier is removed from the annular target and dissolved in nitric acid at near ambient pressure. In this process, the uranium, nickel, and all fission and activation products are dissolved. The resultant solution (approximately 0.5 liters) will contain approximately 7 millimolar Mo, around 450 grams of uranium per liter of solution, and a 1 molar final nitric-acid concentration. After dissolution, the solution will be fed into a small column of titania sorbent, where Mo will be adsorbed on the column with minor amounts of other feed components. The column will be washed with nitric acid and then water; then Mo will be stripped into a sodium hydroxide solution.

For the electrochemical process, the LEU-foil target and an associated 40 micrometer aluminum fission recoil barrier are dissolved sequentially. The aluminum barrier is dissolved first in a basic solution followed by the low temperature, low pressure electrochemical dissolution of the LEU foil in a bicarbonate aqueous solution. The electrochemical dissolution involves anodic oxidation of the uranium metal within the bicarbonate aqueous matrix. After the target is completely dissolved the uranium, carbonate, and alkaline-insoluble fission and activation products are removed from solution by precipitation involving the addition of calcium oxide powder. Molybdenum remains in solution during the precipitation step and the resulting solution can be fed into the current purification processes.

These two frontend processes have been demonstrated in the laboratory and are currently being optimized for remote operation in a hot cell facility. This optimization is being performed at Argonne based on results from tests using our new hot-cell mockup apparatus, which is equipped with two fully operational hot-cell manipulators and a simulated hot cell window. This work in being done in preparation for a hot demonstration of both processes at Oak Ridge National Laboratory (ORNL) using fully irradiated foils. The demonstrations will involve irradiating uranium foil targets in the High Flux Isotope Reactor and performing the dissolutions and molybdenum separation steps (i.e. separation on titania column for acidic solution and selective precipitation for alkaline solution) in hot cells at ORNL.

2.0 Ambient Pressure, Nitric-Acid-Dissolver System for LEU Foils

The ambient pressure, nitric-acid-dissolver system is designed for the dissolution of up to 250 grams of irradiated LEU-foil and associated fission recoil barrier metal (e.g., Ni) has been tested using non-irradiated uranium foils. The dissolver system components were tested at full-scale so that the design could be optimized in preparation for a full-scale demonstration. The key design criteria of the dissolver system that have been tested experimentally are summarized below:

- All water vapor, reaction products, and fission gases must be contained within the dissolver system at a maximum temperature of 125°C and 2 atmospheres of pressure (absolute) under both normal and off-normal (loss of cooling during reaction) conditions.
- The acid-feed system must be designed so that the thermally hot LEU foil (hot from decay heat) can be immersed in nitric acid without losing solution due to instantaneous boiling.
- All dissolver system components must designed for remote operation in a hot cell facility.
- Gas-trap components must be designed to trap/neutralize all nitrogen oxide and acid gases (NO, NO_2, HNO_2, HNO_3) as well as trap iodine gas for possible extraction of economically important iodine isotopes (noble fission gases will be passively contained).

The dissolver system design basis requirements imposed by the thermal and chemical properties of the LEU-foil dissolution process as well as the dissolver system design and plan for performance testing are discussed in previous reports.[1], A Schematic flow diagram showing the major components of the LEU-nitric acid front-end process is shown in Figure 1.

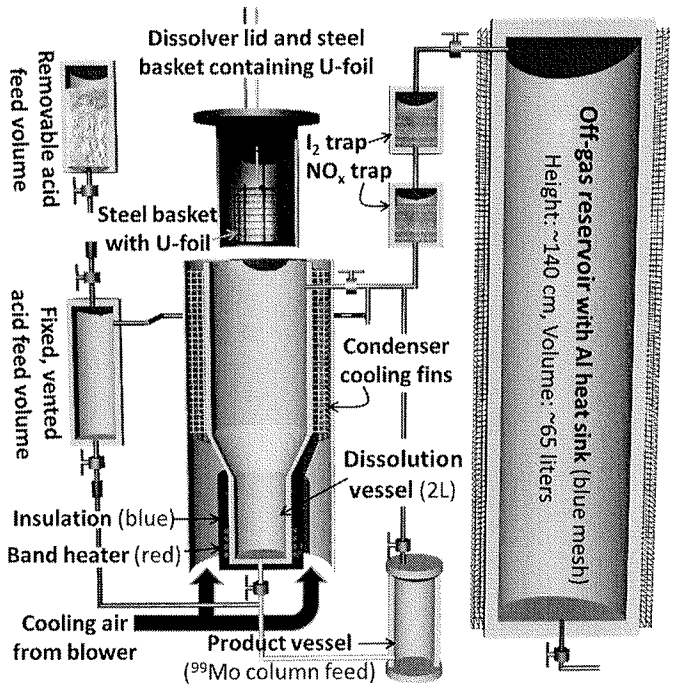


Figure 1. Schematic cross-section showing components of the LEU nitric acid dissolver system being designed at Argonne National Laboratory.

Dissolver Design Overview

The dissolver system consists of a 304 stainless steel vessel (2 liter volume) connected to an approximately 65 liter (30cm x 90cm) off-gas reservoir (Figure 1). The dissolver vessel is open to the off-gas reservoir during the dissolution process. The volume of the reservoir and the sizing of the aluminum heat sink (Figure 1) were chosen to provide passive containment of all water vapor and reaction products at a pressure less than 2 atmospheres (absolute), during both normal and off-normal (loss of cooling during reaction) conditions.

The dissolution process is started by first lowering the uranium foil (contained within a steel mesh basket) into the dissolver vessel and then sealing the vessel with a metal cap. Nitric acid is then added to the vessel using a two-chamber acid feed system that is designed to avoid pressurization of the acid bottle in the event that the dissolution reaction begins instantaneously when the acid is added. The dissolver vessel is cooled by forced air blown from the base of the unit. The temperature of the dissolver solution is monitored by a thermocouple. The dissolver vessel is insulated so that the top of the vessel is cooled continuously during the reaction. Heat loss from the top of the vessel is optimized by the presence of steel cooling fins attached to the condenser part of the dissolver system. This design causes the water and acid vapors to condense along the walls at the top of the vessel during the dissolution reaction (as acid is boiling).

Dissolver System Performance Tests

Two types of experiments were performed: (1) condensation performance tests were run using boiling water to confirm and quantify the performance of the condenser section of the dissolver system. An array of thermocouples was used in the initial testing to measure all relevant thermal gradients during the dissolution reaction process (inside and outside the dissolver vessel and condenser section). (2) Non-irradiated uranium dissolution tests were also performed to measure the loss of NO_x from the dissolver and to test the performance of the NaOH-based NO_x scrubber. The general procedures used for the experiments are summarized as follows:

An example of results from recent performance tests are shown in Figure 2. The linear flow velocity of the cooling fan was measured throughout each experiment. The cooling air flow remained constant at 26.1 meter per second where it entered the cooling fin section at the base of the dissolver. The cooling air flow measured at the top of the dissolver, where it exits through the cooling fins was consistently 19.8 meters per second. The loss of flow velocity is due to turbulence as the cooling air travels up through the dissolver duct work and is channeled into the cooling fins and perhaps by small leaks in the steel sheath that contains the fins.

For both tests, 1700 Watts of thermal power was supplied to the dissolver cup via band heaters. Key observations from these experiments are summarized below:

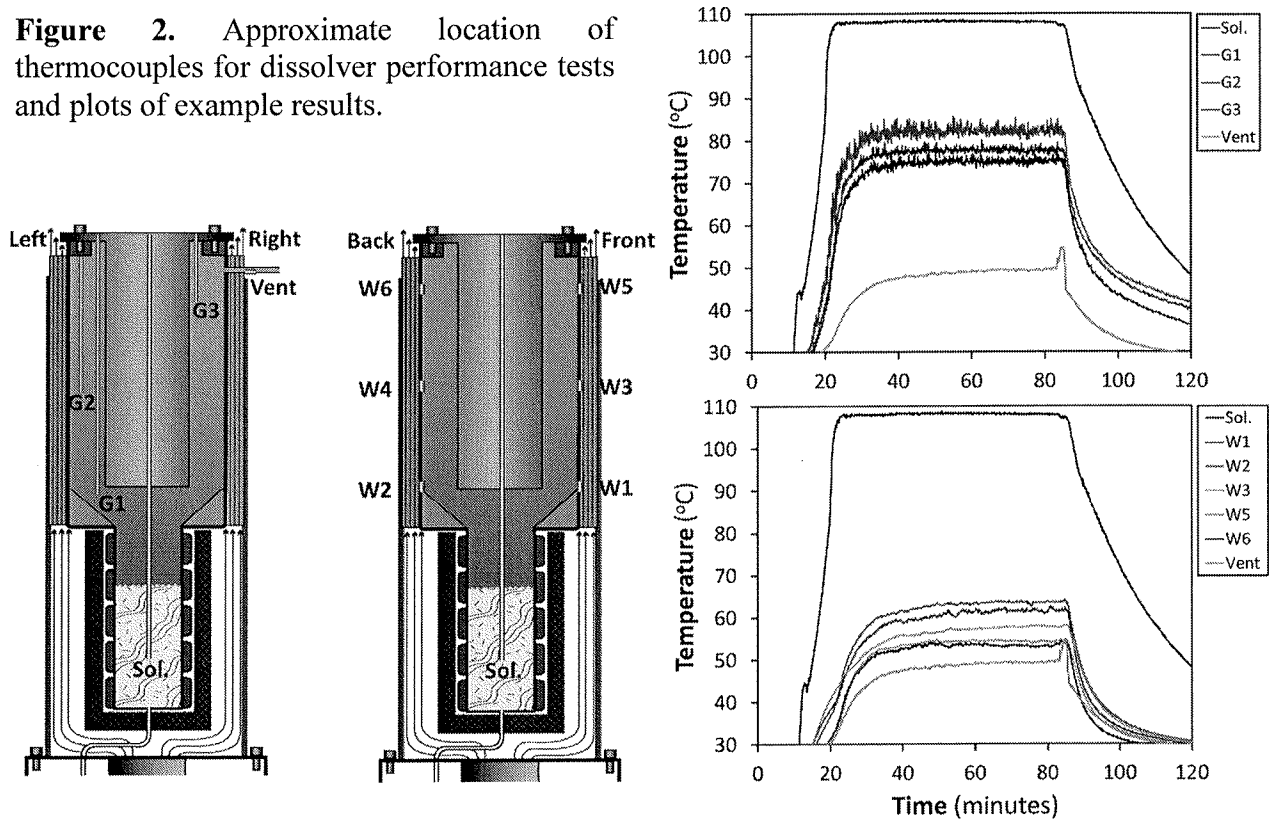
- No measurable water loss was detected from the DI condenser tests.
- Dissolution of 133 g of DU foil in 7 molar nitric acid (initial) was not complete after 1 hour under boiling conditions.
- Dissolution of 133 g of DU foil in 7 molar nitric acid (initial) was complete after 2 hours under boiling conditions.

• No measurable solution loss was detected during DU dissolution tests; however, NOx gas was observed escaping intermittently during the dissolution process. The amount of NOx lost from the dissolver and its effect of the dissolution rate and efficiency is being further investigated in on-going experiments.

Summary and Future/On-going Work for Nitric Acid Dissolver

• A full-scale prototype of the ambient pressure, nitric acid LEU-foil dissolver system (capable of dissolving 250 grams of irradiated LEU for Mo-99 extraction) was designed and tested.
• Tests results show that the cooling system of the dissolver (reflux condenser) is sufficient to remove all heat produced by the exothermic dissolution reaction and continuous decay heat of the irradiated foil.
• The dissolution of 133 grams of depleted uranium foil was completed successfully in approximately 2 hours. The dissolution time can be minimized by optimizing the starting acid concentration to account for the effects of the loss of NO_x and acid gases from the dissolver during dissolution. These optimization dissolution experiments, using test set-up described in this report, are on-going.
• The performance tests of the off-gas traps for NO_x (NaOH trap) and iodine (copper metal or silver loaded zeolite) will be completed and these gas traps will be added to the dissolver system design.
• The complete dissolver system (including gas traps and off-gas reservoir) will be tested in a manipulator mock-up facility to ensure that the dissolver system can be easily operated in a production scale in a hot cell facility.

Figure 2. Approximate location of thermocouples for dissolver performance tests and plots of example results.



3.0 Design of Column Separation Processes for Recovery of Molybdenum from Dissolved High Density LEU Target

The product solution produced by dissolution of the LEU foil in the ambient pressure, nitric acid dissolver will then be run through a column to recovery Mo from other target components; the Mo will be stripped from the column with a sodium hydroxide solution that will be compatible with the current purification processes used for dispersion targets. The volume of dissolved-foil solution is assumed to be 500 mL, and the pressure drop, ΔP, constraint of 0.8 atm is based on use of vacuum/gravity fluid transfer system. Therefore pressure loss is limited < 1 atm; hence a value of 0.8 was chosen.

A full-scale design for the separation column was chosen based on target total Mo-recovery-process time of ~1 h. The design employs a 2.5 x 8 cm column loaded at 5 cm/min. In this process, 500 mL of solution is loaded in 20.4 min. Subsequently, the column is washed with 10 BV (393 mL) of 1M HNO_3 for 8 min at 10 cm/min. This step is followed by a water wash, using 196 mL of water loaded for 4 min at 10 cm/min. Finally, Mo is stripped from the column using 16 BV (629 mL) of 0.1 M NaOH at 5 cm/min for 25.6 min. This design was tested experimentally using 500 mL of 450 g-U/L uranyl nitrate solution, 1 M H^+, 7 mM stable Mo, and tracer Mo-99. In this run 96 ± 5% of Mo was loaded on the column and 90 ± 5% is recovered. The total process time is 58 min.

In this paper, we discuss the testing of the recovery column design with irradiated target solution to monitor partitioning of Mo and tracer level fission products. In addition, due to discontinuation of market availability of S80 sorbent, S40 and S110 were evaluated for recovery of Mo from LEU target solution. Description of new column designs, and results of verification experiments are also presented in the results section.

S80 Column Design Verification with Irradiated Target Solution

A 4-g depleted uranium (DU) foil encapsulated in aluminum casing was irradiated at the Argonne Linear Particle Accelerator (linac). Irradiation was done using a tungsten photoconverter to generate high-energy x-rays that generated U-238 photofissions in the target. The target was irradiated for 5.5 hours at 18 MeV and 200 microA. The calculated ^{99}Mo yield, after 14 h cooling time was 0.5 mCi. After a 15-h cooling time, the aluminum casing was removed, and 4 g of 125 μm thick DU foils were dissolved in 12 mL of 10 M HNO_3. The dissolution took ~3 h and was maintained at ~100 °C using a water bath. The dissolution flask was equipped with a dry ice/acetone filled condenser to minimize loss of water and nitric acid. The resulting 10 mL of dissolved target solution was added to 500 mL of UNH solution containing 450 g-U/L uranyl nitrate, 1 M H^+, and 7 mM Mo.

The irradiated target solution was loaded on a 2.5 x 8 cm column containing 51 g of S80 sorbent at 24.5 mL/min for 20.4 min, at 60 °C. The loading was followed by 1 M HNO_3 and H_2O washes. Mo was stripped from the column with 0.1 M NaOH at 5 cm/min at 25 °C. The strip was divided into 4 fractions containing 4, 6, 6, and 14 BV of strip solution.

The counting results indicate that 98.3% of Mo was loaded on the column. Subsequently, 85.0% of Mo was in strip fraction 2, and 10.0% of Mo was found in strip fraction 3. Strip fraction 4 contained 0.2% of Mo. Therefore, stripping with 16 BV of 0.1 NaOH is sufficient to recover Mo from the column. About 1.5 % Mo remained in the sorbent after stripping with 30 BV of 0.1 M NaOH. Nitric acid wash, water washes, and strip fraction 1, each contained <0.1% Mo. Partitioning of actinides and fission products between process streams was determined using gamma counting.

It was found that majority of radionuclides, including uranium, iodine, alkaline (Ba, Sr), transition metals (Cd, Rh and Ru) and lanthanides (Ce, Nd) partition into the eluent and nitric acid wash streams. The fission products found in the Mo product stream are Te (~ 18% in the strip fractions 2 and 3), I (~8% from fission + growing-in from Te), as well as < 3% of Ru, Rh, Sb and Zr. Te (~50%), Sb (> 90%) and Zr (~80%, most likely as a precipitate) remained on the sorbent. Since 14% of Te was in strip fraction 4, which is not projected for the actual process conditions, ~ 60% of Te is expected to remain on the sorbent.

Activity of Ag-112, I-131, I-132, La-140 and Nb-97 in the product stream was determined at t = 26 h. Their activities were not corrected to t = 0 since they are both fission products and are growing in from their parent radionuclides; Pd-112, Te-131m, Te-132, Ba-140 and Zr-97, respectively. Their activities are presented as μCi/ mCi Mo-99 at t = 26 h.

During the actual production of Mo-99, ~99% of I is expected to be removed during the dissolution step. Therefore, iodine activity in the product stream can be expected to be 100 times lower.

Sorbent Selection

Two sorbents, S40 and S110 were considered for new column designs for recovery of Mo from dissolved LEU target solution. VERSE simulation program was used to determine MTZ (Mass Transfer Zone) for S40 and S110 at 3 and 5 cm/min, when Mo concentration in the bulk reached 0.1% of the feed concentration. Both sorbent have the same titania composition as S80 and 60 Å pore size. For both sorbent isotherms, sorbent and mass transfer parameters were assumed to be the same as those for S80 sorbent. Based on the MTZ length for S40 and S110, column volumes were calculated for both sorbents. The results indicate that for recovery of Mo from dissolved LEU target solutions, columns based on S110 would be ~ 2.5 times larger than those based on S40 sorbent. Hence S40 was chosen for recovery of Mo from LEU targets.

Column designs for Mo recovery from 450 g-U/L $UO_2(NO_3)_2$ and 1 M HNO_3, 7mM Mo

Column designs were calculated for loading 500 mL LEU target solution in 20 min to limit the total column process time to 1 h. Therefore, for loading the LEU target solution at 25 mL/min inner column diameters were chosen to be 5, 3.5 and 2.5 cm. These diameters reflect dimensions of commonly manufactured columns. VERSE simulations were carried out to estimate mass transfer zone (MTZ) at resulting linear velocities for 20 min loadings (Table 1). For the 2.5 x 9 cm column design, ΔP exceed the 0.8 atm pressure drop limit. The 5 x 2.5 and 3.5 x 5 cm are viable column designs.

Table 1. Column designs for recovery of Mo from 500 mL of LEU target solution in 20 min.

column ID (cm)	velocity (cm/min)	$MTZ_{0.1\%}$ (cm)	Column length (cm)	Column volume (mL)	Sorbent weight (g)	ΔP (atm)
5	1.27	1.19	2.50	49	64	0.09
3.5	2.60	2.41	5.00	48	63	0.35
2.5	5.09	4.70	9.00	87	113	1.24

The 3.5 x 5 cm column design was tested experimentally at 60 °C using 450 g-U/L solution with 7 mM Mo, 1 M H^+ and tracer levels of Mo-99. Loading velocity and column length were kept constant at 2.60 cm/min and 5 cm, respectively, while the column ID and solution volume was reduced to 1 cm and 40.8 mL, respectively. Mo was stripped from the column with 0.1 M NaOH at 3 cm/min at 25 °C. 99.4 ± 3% of Mo was loaded on the column and 99.9 ± 3% was recovered in strip fraction #2 and 3. Mo-99 activity in the nitric acid and water wash fractions and strip fraction 1 and 4 was below detectable levels.

Based on the results, Mo recovery process was designed using a 3.5 x 5 cm column with S40 sorbent. In this process, 500 mL of solution is loaded at 60 °C in 20 min at 25 mL/min. The loading is followed by a 1 M HNO_3 and H_2O washes at 48.1 mL/min at 25 °C. Mo is stripped at 25 °C with 770 mL of 0.1 M NaOH at a flow rate of 28.9 mL/min for 26.7 min. The total process time is 58 min. The calculated pressure drop for the loading, washing and strip are 0.35, 0.43 and 0.26 atm respectively.

Summary of results for Mo recovery column work

Partitioning of actinides, and fission products between different process streams, using a representative titania-based column was determined. The majority of activity from radionuclides, including iodine, actinides, lanthanides, alkaline and transition metal ions reports in the eluent and wash fractions (waste). Fission products found the Mo product stream are transition metals; Te (~30%) and <3% of Ru, Rh, Sb and Zr. About 8% of fission iodine was found in the Mo product stream. However, the activity of iodine in the product stream is expected to be ~ 100 times lower when iodine is removed during the target dissolution step.

A new column recovery process was developed based on S40 sorbent. In this process, a 3.5 x 5 cm column, with 63 g of sorbent is utilized. The loading time is 20 min and the column recovery process is designed to be complete in 1 h. This process design was verified with solution containing 450 g-U/L, 1 M H^+, 7 mM Mo and tracer levels of Mo-99. The experimental results indicate that 99.4 ± 3% of Mo was loaded on the column and 99.9 ± 3% was recovered in strip fractions #2 and 3.

4.0 Uranium Anodic Dissolution under Slightly Alkaline Conditions: Full Scale Demonstration with DU foil

A low-temperature, low-pressure procedure employing anodic oxidation of the uranium metal into basic bicarbonate media is under development, with consequent precipitation of uranium and carbonate from the solution by the addition of calcium oxide, followed by the sorption of Mo-99 on an anion exchanger. The process schematic is shown in Figure 3. The goal of this study was to verify the feasibility of this method and to identify key optimizing factors for the development of an industrial-scale procedure. The dissolver design is shown in Figure 4.

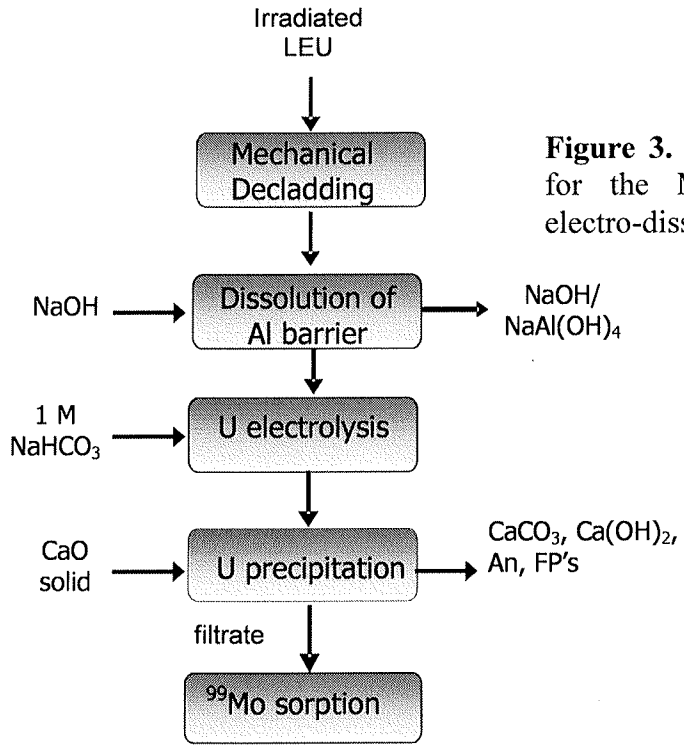


Figure 3. Block-diagram of a novel process for the Mo-99 production via LEU-foil electro-dissolution.

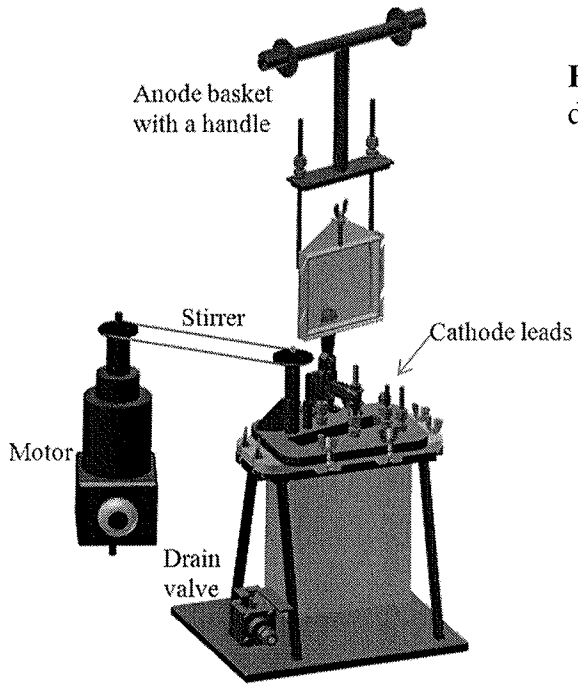


Figure 4. Sketch of the electrochemical dissolver with a motor and the anode basket.

Electrolysis Experiments

The oxidation of uranium metal is assumed to proceed quickly through U(III), producing U(IV) oxide. Then, according to literature data, the oxidation of UO_2 proceeds through the formation of UO_{2+x} species on the surface until $UO_3 \cdot 2H_2O$ is formed. Hexavalent uranium has rather high solubility in the presence of carbonate/bicarbonate ions. The dissolution occurs when these negative ions reach the positively charged surface of the uranium anode.

Our preliminary tests showed that 20- 22 g DU foils are completely dissolved in 1M $NaHCO_3$ / 90 °C within 1.5 hours. The foils were connected to anode lead via a clip, hanging freely in the solution, while the solution was intensively stirred. However, hanging the irradiated foils during the dissolution may not practical, as there is chance of dropping the foil in the dissolution vessel. Therefore, an anode basket or a cage should be utilized.

The following dissolution tests were conducted using a high current DC power supply by Magna Power with a voltage range of 0- 32 V and a current range of 0-62 Amps. The foil was placed in the anode basket between two SS coarse screens, which were tightened together; see figure 5.

The vertical bolt pushes the outer frame with the welded screen toward the inner frame with the attached screen, thus providing the necessary contact between the anode lead and the foil. We have shown that using a basket it takes about 2.5 hours to dissolve nearly 99% of the full size foil due to the partial reduction of the free foil surface area, caused by the mesh interference.

A dissolver required for this process has been designed and fabricated. The material used for the prototype is the 304 SS (Figure 5). Nitrogen gas will be used to sweep the dissolver. The fission gases will be collected; hydrogen gas, which is generated during the electrolysis, will need to be recombined.

The first step is the dissolution of Al barrier in 1 M NaOH as $NaAl(OH)_4$. Then the solution is drained through a ball valve on the side of the dissolver. After a wash step, a sodium bicarbonate solution is added in the dissolver through one of the ports on the lid, and the uranium foil electrolysis is conducted at elevated temperature (85 - 95°C) and intense stirring. Heating is provided by a heat gun (not shown).

The approximate volume of the solution, required to cover a 24-g uranium foil (10x8 cm), is about 1.2 L. However, the solution/U-metal ratio will be significantly smaller if a batch of several foils is dissolved in a vessel, modified for the full-scale process. The solubility of uranium in bicarbonate medium is 80 g-U/L (It is soluble as neutral and anionic complexes of uranyl carbonate.), so for a 225 g batch of U foils at least 3 L of the solution is required. A concept of the dissolver with the anode for 225 g batch dissolution has been conceived; however; its implementation is pending the experimental results of the dissolver shown above.

The fabricated prototype dissolver includes a stand, a heat gun, and a stirring motor. The stand serves also as a shroud for the hot/cool air direction in order to control the temperature. For the full size dissolver, the anode basket(s) will be fabricated from nickel or nickel alloys, resistant to the oxidation in alkaline medium. 304 SS is acceptable for several applications, so it was chosen

to minimize the cost of the prototype. The cathode and the dissolver body of the production dissolver can be made from the stainless steel, as there should be no significant corrosion during the processing.

A photo of the dissolver in the shroud and the anode basket is presented in Figure 5.

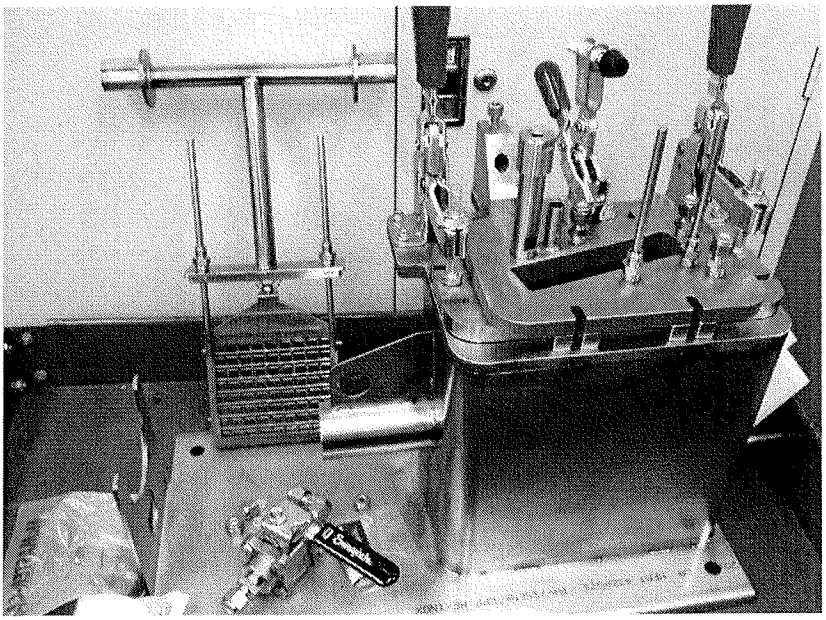


Figure 5. Fabricated anode basket and the dissolver, inserted in the shroud.

A 23.7 g piece of DU foil, 0.125 mm thick (10x7.5 cm) was wrapped in 0.1 mm Al foil (W=4.5 g) and was placed in the anode basket (Figure 5).

To dissolve the Al recoil barrier, 1.2 L of 1.1 M NaOH was added to the dissolver. The solution was intensely stirred and heated up to 60 °C using a heat gun. The temperature was maintained for 5 min, then heating was stopped, and the resultant solution was drained by gravity. The basket was removed from the dissolver to ensure the completeness of Al dissolution. No Al foil remained in the basket. A small amount of black solids, presumably hydrous UO_2, was found in the solution. The UO_2 was likely removed from the DU foil surface due to the agitation. No significant UO_2 formation is expected in alkaline medium under the experimental conditions.

For the tests with the irradiated U foil, the $NaOH/NaAl(OH)_4$ solution will be analyzed for ^{99}Mo content. It could be combined with the filtrate (the feed to the purification process) or processed separately to prevent any loss of the product from the Al dissolution step.

To dissolve the uranium foil, 1.2 L of 1M $NaHCO_3$ was added to the dissolver after 100 mL H_2O rinse. The solution was heated up to 85 °C, and the electrolysis was started. Nitrogen gas was continuously used for sweeping the dissolver head space. The temperature was maintained at 90±5 °C. For the first hour, the current was maintained at 29A, while the voltage was in the 6.0-6.3 V range. After 1 hour, the process was stopped, the foil removed from the basket, washed with water, dried and weighted on the scale. Exactly 10 g was dissolved. This dissolution rate was considered to be unsatisfactory, so for the remaining time the experiment was conducted at ≈ 8.5 V and 40-45 A. After 1.5 h from the restart (2.5 h total) the electrolysis was terminated.

After inspection, only a small piece of 0.78 g of DU left. Therefore about 97% of the foil was dissolved in 2.5 hours, although the rate can be further improved by applying higher voltage all the time.

After the dissolution, the uranium containing solution was transferred to a stirring tank under vacuum, then while stirring solid CaO was added to it through a funnel. About 86 g (1.5 mol) of CaO was used to ensure a complete precipitation of uranyl carbonates and $CaCO_3$. After 10 min of stirring, the suspension was filtered through 1 micron Whatman filter with 2000 cm^2 surface area under vacuum.

We are planning to conduct a series of tests using depleted uranium foil as well as the low-burn-up irradiated foil. We will study molybdenum recovery and the behavior of fission products including iodine to find the ways to separate it from the other fission products. The study will also focus on the waste characterization including the formation of fission-product gasses and the means of their storage, utilization, and/or disposal. We will also test the prototype system operations in a hot cell mockup facility at Argonne.

Once these laboratory studies are completed, our plans are to test this process in a hot cell at Oak Ridge National Laboratory using a fully irradiated foil.