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SCRAP RECOVERY OPERATIONS IN THE FUEL FABRICATION CAPACITY

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ABSTRACT

The GTRI program is currently engaged in development of nuclear fuel that enables conversion of high-performance research reactors to low enriched fuel (LEU). The fuel design is based on monolithic uranium-molybdenum alloy enclosed in Al-6061 cladding with a diffusion/bonding layer composed of zirconium or aluminum-silicon alloy material. The utilization of U alloy containing 10% (w/w) Mo as fuel requires a process for recovery and recycle of uranium from the scrap material in the fuel fabrication facility, and, potentially, uranium from irradiated fuel elements. To achieve this goal, a dissolution technique is required to prepare solutions suitable for processing. In this report, considerations in the fuel-element design relevant to scrap recovery and potential irradiated fuel processing such as limited solubility of Mo(VI) in uranyl nitrate solution and cases that require use of fluoride in the dissolution process are indicated. Also dissolution methods developed from literature reports are described and a uranium recovery process from U-10Mo alloy is designed.

1. Introduction

The U.S. high-performance research reactor (HPRR) fuel requires high fissile loadings needed to meet the HPRR performance requirements.[1] Existing LEU fuels can not meet the necessary fissile loadings for HPRR, so development of ultra-high density uranium LEU fuels has been underway in the program for several years. The monolithic ultra-high density U-10Mo alloy, identified for research reactor applications, contains 10% (w/w) Mo and has a U density of 15.3 gU/cm³. The baseline fuel-plate design consists of U-10Mo alloy foil in the 0.01 – 0.05 cm thickness range, clad in Al-6061 alloy in the 0.04 – 0.24 cm thickness range (sum of both sides) with Zr or Al-Si layers, most likely of 0.0025 cm maximum thickness, to control fuel/clad interaction.

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In this paper, the dissolution of U-10Mo fuel and its cladding is described. The product of the dissolution is a solution suitable for recovery of uranium by a tributyl-phosphate (TBP) based solvent extraction process. Special considerations and potential challenges in the dissolution of U-10Mo fuel, such as limited solubility of Mo in uranyl nitrate solution, formation of zirconium-uranium intermetallic compounds that might cause explosive reactions during the dissolution, and formation of gels during dissolution of aluminum-silicon alloy material are discussed. Flowsheets for recovery of uranium from dissolved scrap materials were designed using the AMUSE code (Argonne Model for Universal Solvent Extraction). Recovery of uranium from casting runners and uncoated U-Mo alloy is also discussed.

2. Literature Review

Dissolution of U-Mo Alloy

The most important consideration in designing a U-Mo fuel dissolution process is the limited solubility of molybdenum in uranyl nitrate solutions. It was first observed during U-Mo dissolution studies done at Hanford for processing of low-acid feed solutions.[2,3] When dissolution of 3-10% (w/w) U-Mo alloys in nitric acid was attempted, a flocculent, yellow precipitate (believed to be uranyl molybdate/polymolybdate) was obtained. It was noted that the solubility of Mo in uranyl nitrate solution is higher at higher nitric acid concentrations and at room temperature rather than at boiling temperatures. The precipitate from U-Mo alloy dissolution is easily centrifuged, and dissolved in 0.5 M Na_2CO_3 . Uranium can then be recovered as sodium diuranate solid from the molybdate solution by treatment with sodium hydroxide.

An extensive study on solubility of Mo in nitrate-based media in the presence of uranyl nitrate, ferric nitrate, and other metal salts was done by Faugeras.[4] This paper illustrates the complexity of U-Mo alloy dissolution. The study shows that Mo solubility in the presence of U peaks at ~1.5 M HNO_3 and that the solubility of Mo is dependent on uranium concentration and peaks at 120 g/L U (Figure 1). Thus, decreasing or increasing the uranium concentration from 120 g/L decreases the amount of Mo that can be dissolved in a solution containing uranyl nitrate and nitric acid. U-10Mo alloy can be dissolved up to 25 g/L U in ≥ 3 M HNO_3 , dissolution to higher concentrations will cause uranyl molybdate to precipitate out of the solution. Faugeras also studied the solubility of Mo in the presence of $\text{Al}(\text{NO}_3)_3$, without uranium. His data reveal that Mo solubility decreases above 0.5 M $\text{Al}(\text{NO}_3)_3$. Therefore, dissolution of Al cladding with the U-Mo alloy will require less concentrated solutions.

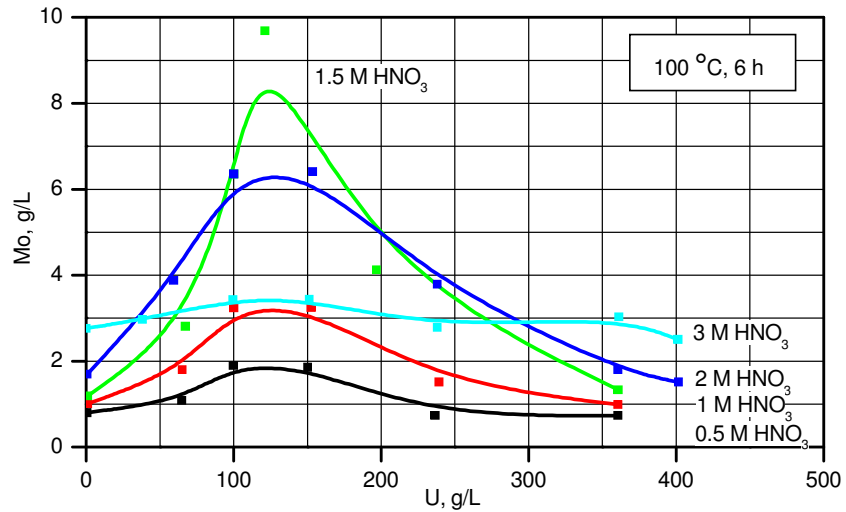


Figure 1. Solubility of molybdenic oxide in nitric acid as a function of uranium. Data obtained from reference 4.

Dissolution with the Addition of Ferric Nitrate

The extent of dissolution of U-Mo alloy increases significantly in the presence of $\text{Fe}(\text{NO}_3)_3$ (Figure 2).[2-6] The U-Mo alloys are dissolved to higher terminal uranium and lower terminal acids concentrations, without solid formation. U-10Mo can be dissolved in 2 M HNO_3 to terminal uranium concentration of 90 g/L without precipitate formation. The increased solubility of U-Mo alloy in presence of $\text{Fe}(\text{NO}_3)_3$ is attributed to formation of a negatively charged iron-molybdenum complex.[2,3] Corrosion of stainless steel by nitric acid is accelerated by the presence of ferric ions.[7] The maximum corrosion rate in boiling 4.5 M HNO_3 and 1 M $\text{Fe}(\text{NO}_3)_3$ is 0.025 mm per month (MPM) and decreases to 0.005 MPM as fuel dissolution progresses. The average corrosion rate is approximately 0.015 MPM. Manness suggests that dissolution of U-Mo alloy in solutions containing more than 0.7 – 0.8 M $\text{Fe}(\text{NO}_3)_3$ requires incremental addition of nitric acid due to corrosivity of stainless steel construction materials.[8]

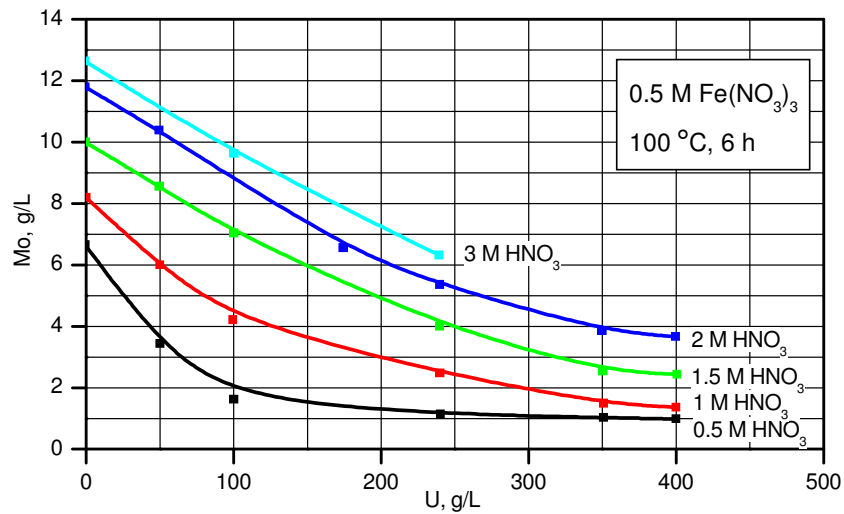


Figure 2. Solubility of molybdenic oxide in nitric acid as a function of uranium concentration in the presence of 0.5 $\text{Fe}(\text{NO}_3)_3$. Data obtained from reference 4.

Dissolution Rates

The dissolution rates for U-Mo alloys in nitric acid are higher than those for ingot uranium metal.[2,6] The dissolution rate at boiling temperature for U-10Mo alloy follows a third-power dependence on nitric acid concentration and is $\sim 100 \text{ mg/min-cm}^2$ in 6 M HNO_3 . It drops to about 2 mg/min-cm^2 in boiling 2 M HNO_3 . Additionally, iron catalyzes the dissolution of U-Mo alloy at low acidities. At comparable pH (0.2) and nitrate-ion concentrations, the dissolution rate in 1.0 M $\text{Fe}(\text{NO}_3)_3$ is about 1000 times greater than in $\text{NaNO}_3 - \text{HNO}_3$ solution.

Dissolution Processes of U-Mo Alloy

U-Mo alloy containing 10% (w/w) Mo was dissolved at SRNL[7,9] and ORNL.[10] At SRNL, U-10Mo alloy in carbon steel cans was dissolved at $\sim 100^\circ\text{C}$, to 15-20 g/L U and 3-5 M HNO_3 . Carbon steel cans contributed 1.0-2.3 g/L iron to the solution. The range of nitric-acid concentration gave a maximum margin for avoiding precipitate formation while providing a solvent-extraction feed without adjusting nitric-acid concentration. U-10Mo alloy fuel slugs enclosed in aluminum cans, from Hallam Nuclear Power Facility, were dissolved in the H-Canyon dissolver at 100°C . [7] The aluminum cans were dissolved in the nitric acid solution, along with the fuel, in the presence of mercury catalyst. The dissolvent was initially $\sim 4.5 \text{ M HNO}_3$ and 1 M $\text{Fe}(\text{NO}_3)_3$. The composition of the final solution was not described.

The dissolution flowsheet designed at ORNL for a TBP-based solvent-extraction process for U-10Mo alloy involved dissolution in boiling 11 M HNO_3 after the aluminum can was removed in boiling NaOH-NaNO_3 solution (with less than 0.1% uranium losses to the decladding solution).[10] The 1.5-cm. diameter core was dissolved in 5 hours, producing a solution containing 260 g/L U and 6 M HNO_3 . Most of the molybdenum precipitated as MoO_3 during dissolution. However, the precipitate was difficult to separate and contained 5-10% of uranium. Recovery of uranium was improved when the solids were separated by vacuum filtration and

washed with 1 M HNO₃. Uranium can be recovered from partially washed precipitates by treatment with 19 M NaOH and then 11 M HNO₃ at 90 °C. Sodium hydroxide metathesizes the precipitate to soluble molybdate and insoluble Na₂U₂O₇, which is then dissolved in 11 M HNO₃.

Additionally, CEA/VALRHO in France studied dissolution of U-Mo along with aluminum clad/can in nitric acid.[11] CEA/VALRHO dissolved irradiated and non-irradiated U-Mo fuels containing 7% Mo along with the Al cladding; however, cladding dissolution details were omitted. To obtain a solution free of precipitate, non-irradiated materials were dissolved in boiling 5 M HNO₃. The target aluminum concentration was 15 g/L and the combined U-Mo concentration was also 15 g/L. When irradiated fuel was dissolved under the same conditions, the solution became slightly cloudy after overnight cooling. Upon filtration, 0.6 weight% of the spent fuel was collected. The insoluble species were composed of more than 90% (w/w) molybdenum and aluminum.

Dissolution flowsheets for 3-4% (w/w) Mo alloys for a TBP-based process were designed at Hanford and ORNL and SRNL.[2,3,5,6,12] However, conditions for dissolution of 3-4% alloys are significantly different than those of 10% alloys.

Dissolution of Al-6061 Cladding

Aluminum cladding can be rapidly removed from U-Mo alloy by caustic solution.[12] In 10% NaOH and 20% NaNO₃ and 100°C, the penetration rate for aluminum is 0.4-0.6 cm/hr.[13] Dissolution of unirradiated U-Mo alloy containing 3% (w/w) Mo in this solution is 0.15 mg/cm²hr.[2,3] During the dissolution of Al-canned U-Mo fuel containing 3% (w/w) Mo at ORNL, the aluminum can was dissolved in boiling NaOH-NaNO₃ solution followed by dissolution of the alloy in boiling 6 M HNO₃. [5] Less than 0.1% of the uranium was lost to the decladding solution.

Mercury-catalyzed nitric acid dissolution of aluminum-jacketed uranium fuel has been successfully demonstrated; [7,12,14] however, mercury is a U.S. Environmental Protection Agency (EPA) listed Resource Conservation and Recovery Act (RCRA) hazardous waste. To obtain complete dissolution, mercuric nitrate is added in amount equal to 5% of aluminum to be dissolved. The advantages of the simultaneous dissolution of Al clad and fuel meat is that the production rates are increased through continuous dissolution; however, the amount of aluminum limits the volume to which the nitric acid waste can be reduced.[14]

At concentrations up to 0.2 M, HF enhances aluminum dissolution, with more additional aluminum dissolved than is stoichiometrically associated with HF alone. That effect diminishes at HF concentrations 0.5 M and higher as the dissolved aluminum ions complexed most of the fluoride ions.[15] The dissolution rate of aluminum increases linearly with fluoride concentration.[16] However, as HF is also gradually consumed during the course of Al dissolution, its effect diminishes. Laboratory test results show that complete dissolution is achieved with Al/F ratio of 1.3 or less.

Dissolution of a Bonding Layer

The fuel/clad interaction is controlled by a Zr or Si-Al bonding layer, which most likely will be 0.0025 cm thick. The presence of zirconium introduces a risk of explosive reactions, which are a result of formation of zirconium-uranium intermetallic compounds and may be initiated by a

mechanical shock. These reactions were observed when zirconium-uranium alloys containing low percentage of zirconium are dissolved in nitric acid.[17-19] The presence of fluoride eliminates the hazard of explosive reactions. Uranium-zirconium alloys are completely dissolved in nitric acid when 4 moles of fluoride ion are added per mole of zirconium. The source of fluoride ion can be hydrofluoric acid or ammonium fluoro-silicate.

Sodium hydroxide dissolves the aluminum-silicon bonding alloy.[12] The siliceous material derived from the bonding alloy forms gelatinous solids during caustic dissolution, which might have to be separated by centrifugation. When fuel elements containing aluminum, silicon, and uranium are dissolved in nitric acid, the silicon remains as an insoluble residue, however aluminum and silicon are soluble in HF and HF/HNO₃ solutions.[20]

Gas Formation

The off-gas from dissolution of U-Mo alloys in nitric acid is mainly NO_x and only traces of H₂ are evolved (0 – 0.2 volume percent).[2] The main components of the off-gas (> 1%) are NO ~ 70 - 72%; NO₂ ~ 21 - 25% and N₂ ~ 3 - 5%. The hydrogen generation for dissolution of Hallam Fuel (U-Mo alloy containing 10% (w/w) Mo) in an Al can is less than 2% and is attributed to the dissolution of aluminum in presence of Hg catalyst.[9] Hydrogen-gas generation for dissolution of U-Mo alloy with aluminum in nitric acid did not approach the safety limits of 4% in the dissolver off-gas.

3. Proposed Conditions for Fuel-Scrap Dissolution and Uranium Recovery

Batch Dissolution of U-10Mo with Cladding

U-10Mo alloy can be dissolved to 20 g/L U in 3 M HNO₃ (Table 1). Those conditions afford highest uranium/molybdenum concentration, while providing a margin for avoiding the precipitate formation. At that concentration, up to 2.8 g/L Mo can be dissolved at 100 °C when final nitric acid concentration is between 2.2 and 5 M. To ensure complete dissolution, the initial ratio of HNO₃/U should be ~6; therefore, initial HNO₃ concentration should be 3.5 M. The maximum ratio of aluminum to uranium, which was calculated using number of plates required annually and plate geometries for the MURR, NIST and ATR HPRR, is about 0.6 g-Al/g-U. Therefore, at 20 g-U/L, the aluminum concentration would be ~0.45 M and, at those concentrations, will not have a significant effect on Mo solubility. When cladding is dissolved with the U-Mo alloy, the initial nitric acid concentration will have to be increased to 5.5 M to accommodate cladding dissolution.

When aluminum cladding is dissolved in one-batch with the fuel meat in nitric acid media, addition of a catalyst is necessary to penetrate the protective oxide film that forms on the Al surface. Fluoride is preferred to mercury because its presence is also required for the dissolution of Zr or Al-Si bonding layers. A 1.3 molar ratio of Al/F is used for calculating the F⁻ concentration required for timely dissolution of cladding. In presence of sufficient HF, Si-Al bonding layer will be soluble and removal of solids will not be necessary. However, the amount of fluoride necessary to dissolve fuel with Zr barriers will need to be confirmed experimentally. The uncomplexed fluoride concentration in the dissolved U-10Mo fuel target solution is expected to drop to 10⁻³ M and, at this concentration, will have a corrosive effect on stainless steel components.

When 0.5 M $\text{Fe}(\text{NO}_3)_3$ is added to the dissolvent, U-10Mo can be dissolved to 50 g/L U without precipitate formation (Table 1). We anticipate, this is the maximum concentration of U that can be obtained when cladding is dissolved with fuel meat because under those conditions aluminum concentration reaches 1.1 M. At 50 g/L U, up to 11 g/L of Mo can be dissolved at 100 °C and 3 M HNO_3 . Under those conditions, nitric acid concentration can deviate between 1-4 M without precipitate formation. However, the solubility of Mo in the presence of aluminum and ferric nitrate has not been investigated and will have to be confirmed experimentally.

U-10Mo Dissolution without Cladding

If cladding and Al-Si alloy bonding layer is removed with a caustic solution, then U-Mo alloy can be recycled directly into the melting process. Alternately, U-Mo alloy can be dissolved up to 75 g/L U, without precipitate formation, in the presence of 0.5 M $\text{Fe}(\text{NO}_3)_3$ (Table 1). Under those conditions, up to 9 g/L Mo can be dissolved in 2 M HNO_3 at 100 °C, and the acid concentration can vary between 1.5 and 3 M HNO_3 . As discussed above, in the absence of $\text{Fe}(\text{NO}_3)_3$, U-Mo can be dissolved up to 20 gU/L. When the cladding from fuel plates containing Zr bonding layer is removed with a caustic solution, the Zr layer will remain with the fuel meat. In this scenario, fluoride will also have to be added to the dissolvent, in the molar ratio of 4 F/Zr to prevent formation of explosive uranium-zirconium intermetallic compounds.

Feed Compositions

The potential feed compositions to the recovery process are summaries in Table 1. Lines 1 & 2 show feed compositions when cladding is dissolved with the fuel meat and Zr bonding layer. Lines 3 and 4 show feed compositions when cladding is dissolved with the fuel meat and Si-Al bonding layer. Lines 5 and 6 feed show feed composition when cladding and Si-Al bonding layer are removed prior to fuel meat dissolution. Lines 7 and 8 show feed composition when cladding is removed but the Zr bonding layer is dissolved with the fuel meat. Table 2 presents concentrations of cladding components, which are not already included in Table 1, present in the feed where the cladding is dissolved with the fuel meat.

Table 1. Potential feed compositions for dissolved scrap fuel plates

Feed #	U (g/L)	Mo (g/L)	Max. Mo g/L 100 °C	Al-6061 Clad (g/L)	Final HNO ₃ (M)	Min. Final HNO ₃	Max. Final HNO ₃	F ⁻ (M)	Al (M)	Fe (M)	Zr (M)	Si (M)
1	20	2.2	2.8	12.4	3	2.2	5	0.34	4.5E-01	6.2E-04	1.7E-02	3.4E-03
2	50	5.6	10	31.0	2	1	4	0.86	1.1E+00	5.0E-01	4.2E-02	8.4E-03
3	20	2.2	2.8	12.4	3	2.2	5	0.35	4.6E-01	6.2E-04	-	1.3E-02
4	50	5.6	10	31.0	2	1	4	0.88	1.2E+00	5.0E-01	-	3.3E-02
5	20	2.2	2.8	-	3	2.2	5	-	-	-	-	-
6	75	8.3	9	-	2	1.5	3.5	-	-	5.0E-01	-	-
7	20	2.2	2.8	-	3	2.2	5	0.07	-	-	1.7E-02	-
8	75	8.3	9	-	2	1.5	3.5	0.25	-	5.0E-01	6.3E-02	-

Table 2. Concentrations of cladding components, which are not already included in Table 1, in the feed solution where the cladding is dissolved with the fuel meat

Al 6061Clad (g/L)	Mg (M)	Cu (M)	Ti (M)	Cr (M)	Zn (M)	Mn (M)	Be (M)	V (M)
12.4	4.7E-03	4.3E-04	2.6E-04	1.7E-04	1.1E-04	9.0E-05	4.1E-05	2.4E-05
31.0	1.2E-02	1.1E-03	6.5E-04	4.2E-04	2.8E-04	2.3E-04	1.0E-04	6.1E-05

The same chemical approach could be used for dissolution of irradiated fuel plates; however, irradiated fuel dissolution would require a shielded facility. One possible facility is Savannah River Plant, which is designed for batch dissolutions, where cladding is dissolved with the fuel meat using a mercury catalyst. Therefore, dissolution of plates with Si-Al bonding layer would most conveniently be done in the presence of mercury and dissolution of irradiated plates with Zr bonding layer can be done in the presence of just fluoride or both mercury and fluoride catalysts.

Uranium Recovery Process Flowsheet

Once the U-Mo scrap is dissolved, uranium purification will be performed using a PUREX-type solvent extraction process, which uses 30% (v/v) tributyl phosphate (TBP) in n-dodecane. Our flowsheet design assumes that the process will be performed in a 28-stage Argonne-design annular centrifugal contactor. The size of the contactor will be determined by how often the scrap recovery operation is conveniently run. The number of stages in the extraction section (8), scrub section (8), and strip (12) will remain constant for all potential feeds. The composition and flow rates of aqueous feeds to the scrub and strip sections will also remain constant, as will the organic-phase flow rate and composition. The flow rate of the extraction feed is set by its uranium concentration. Table 3 provides the relative flow rates for each of the feed compositions described in Table 1.

Table 3. Relative flow rates for the uranium recovery flowsheet described in Figure 3 for each of the feed compositions described in Table 1

	Flow Rate Relative to that of Solvent		
	Extraction	Scrub	Strip
1	3	0.5	1.5
2	1.2	0.5	1.5
3	3	0.5	1.5
4	1.2	0.5	1.5
5	3	0.5	1.5
6	0.8	0.5	1.5
7	3	0.5	1.5
8	0.8	0.5	1.5

The flowsheet depicted in Figure 3 is a countercurrent solvent-extraction flowsheet. In this flowsheet, the organic phase travels from left to right, entering at stage 1 and exiting the contactor at stage 28. The aqueous feeds enter at stage 8 for the extraction feed, stage 16 for the scrub feed, and stage 28 for the strip feed. The aqueous scrub solution exiting stage 9 combines with the extraction feed in stage 8, and the combined raffinate exits the contactor at stage 1. The U-loaded strip solution (product) exits the contactor at stage 17. Solvent-wash steps are not included in this flowsheets, but two additional stages could be added in which the solvent will be contacted with 0.025 M sodium carbonate to strip dibutyl phosphoric acid (the hydrolysis product of TBP) from the organic phase. Because of the projected low degradation to the organic phase during scrap operation, carbonate washing could also be done periodically in a batch stirred tank.

The flowsheets were designed to recovery >99.999% of the uranium, with a decontamination factor from other fuel and cladding components of $>1 \times 10^8$. The flowsheets were conservatively designed to be robust, thus allowing significant variation in dissolver-solution composition and feed flow rates. The only fuel component in unirradiated fuel, other than uranium, that is significantly extracted by TBP is zirconium; however, the addition of fluoride to the dissolver solution ties up zirconium in the form of inextractable fluoride complexes, making purification from zirconium an easy task.

Examples of AMUSE-calculated stage profiles of feed components are presented below. It is important to remember that these results are calculations and are used to predict behavior. The flowsheet, as designed by AMUSE, needs to be validated experimentally. Even though AMUSE calculations are based on chemically-sound equilibria and use of thermodynamic activities and rigorous mass-balance equations, they are based on solution chemistry and can not predict real-world effects such as colloid formation. As can be seen from the stage profiles, AMUSE predicts uranium recovery far greater than 99.999% and decontamination factors far greater than 1×10^8 .

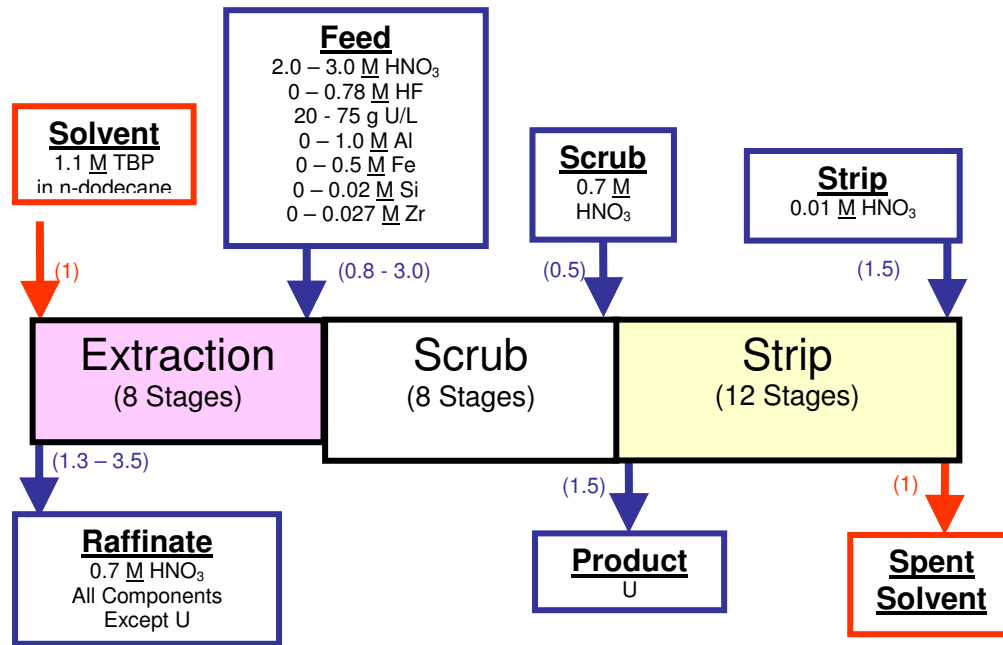


Figure 3. Flowsheet for uranium recovery from dissolved U-Mo fabrication scrap. (The extraction-feed flow rate depends on the concentration of uranium in the feed; the higher is the uranium concentration, the lower must be the flow rate. The ranges of feed-component concentrations vary with the scrap feed to the dissolver and whether iron is added to the feed.)

Figure 4 is an example of the AMUSE-projected stage profile for uranium. Uranium concentration in the aqueous phase decreases as the aqueous solution travels from stage 8 to stage 1, and the organic-phase uranium concentration increases as it travels from stage 1 to stage 8. The concentration of uranium in the extraction feed is 8.4×10^{-2} M, while its concentration in the raffinate is 4×10^{-9} M, a decrease of seven orders of magnitude. The organic phase, which entered the contactor with essentially no uranium, exits the feed stage with a uranium concentration of 3×10^{-1} M. In the scrub section, conditions are chosen so that uranium remains in the organic phase while impurities are scrubbed from the solvent into the aqueous phase, eventually to exit the contactor at stage 1 in the raffinate. The concentration of uranium in the organic phase is essentially constant throughout the scrub section (stages 9-16). In the strip section, conditions are set so the uranium partitions to the aqueous phase. The uranium concentration in the aqueous solution leaving the contactor at stage 17 (the uranium product) is 1.7×10^{-1} M (twice the concentration in the feed, as dictated by the strip/feed flow-rate ratio of 3/1.5). As the solvent moves from stage 17 to stage 28, more and more uranium is stripped from it, finally exiting the contactor with a uranium concentration of 2×10^{-21} M.* The stripped solvent would be recycled (with or without solvent washing) to stage 1 to perform its next cycle through the contactor.

* Remember, the concentration is calculational and the actual concentration would be below detectability.

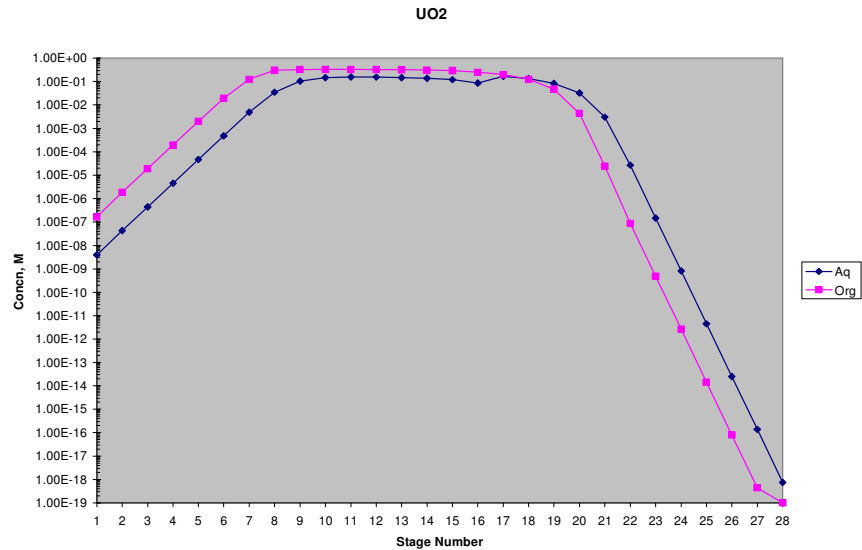


Figure 4. Stage profile of UO_2^{2+} concentration in the organic and aqueous solutions in each contactor stage for flowsheet number 1.

Figure 5 shows the typical behavior of an impurity in this process. In this case, it is the stage profile for Zr in flowsheet number 8. Notice the great difference in this profile (for a non-extracting species) compared to that of uranyl ion (an extracting species). The aqueous-phase concentration of zirconium is essentially flat in the extraction section. This is because the distribution ratio for zirconium in this section is low. The distribution ratio, which is the ratio of organic-phase concentration (blue diamonds) to aqueous-phase concentration (magenta squares), is ~ 0.001 in the extraction section. (In the case of uranium, the distribution-ratio range in the extraction section is 8-42.) Also note the large difference between zirconium and uranium behavior in the scrub section. Where the uranium profile was essentially flat, both the organic- and aqueous-phase concentrations decrease as the solvent moves from stage 9 to stage 16. The organic-phase concentration of zirconium entering the strip section at stage 17, where the uranium product exits the contactor, is below detectability. The behaviors of all other impurities in the eight flowsheets are equivalent to that of zirconium.

4. Recovery of Uncoated U-10Mo Alloy

Cast runners and uncoated U-Mo scrap material can be recycled to the melting step in the alloying process. The U-10Mo alloy has a melting point of around 1200°C , which is much lower than the melting point of Mo (2623°C).^[1] Re-melting scrap material is a common practice but should be kept to 50% virgin materials and 50% re-meltable scrap in order to keep carbon to acceptable levels.^[21]

5. Conclusions

Means to dissolve and recover uranium from U-10Mo scrap were developed using literature data and the AMUSE code. Validation of these results by experimental means is essential before being incorporated in a fuel fabrication facility.

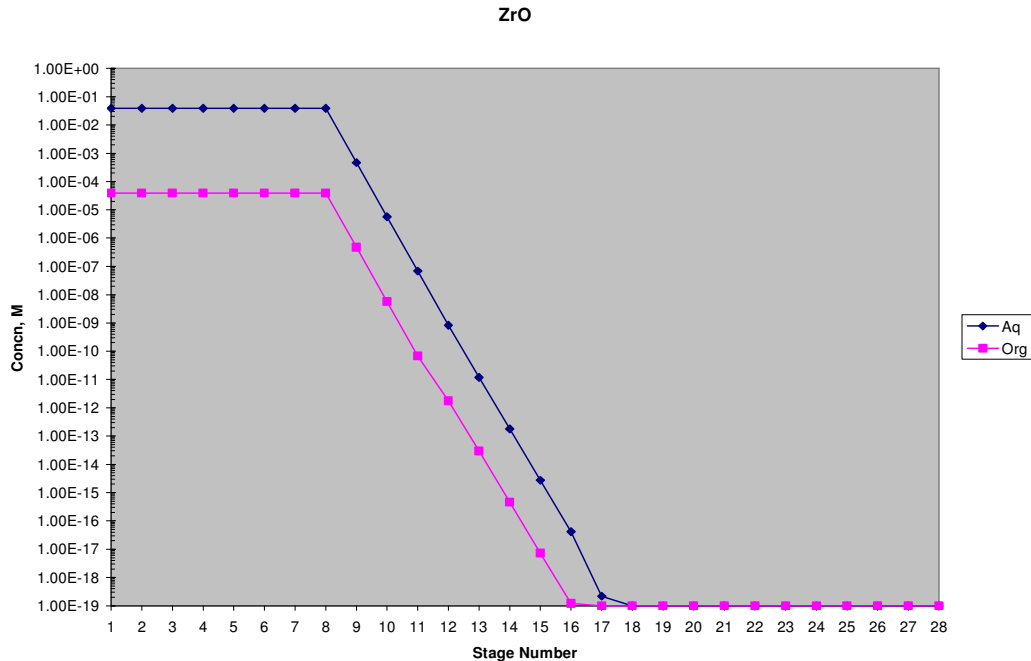


Figure 5. Stage profile of Zr(IV) concentrations in the organic and aqueous solutions in each contactor stage for flowsheet number 8.

6. Acknowledgements

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