

LA-UR- 97-4292

Approved for public release;
distribution is unlimited.

Title:

RECYCLE OF SCRAP PLUTONIUM-238 OXIDE
FUEL TO SUPPORT FUTURE RADIOISOTOPE
APPLICATIONS

CONF-980103 -

RECEIVED

FEB 02 1998

OSTI

Author(s):

L. D. Schulte
J. M. Espinoza
K. B. Ramsey
G. H. Rinehart
G. L. Silver
G. M. Purdy
G. D. Jarvinen

MASTER

Submitted to:

Space Technology & Applications International Forum
Albuquerque, NM
January 25-29, 1998

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

19980327 069

Los Alamos
NATIONAL LABORATORY

Los Alamos National Laboratory, an affirmative action/equal opportunity employer, is operated by the University of California for the U.S. Department of Energy under contract W-7405-ENG-36. By acceptance of this article, the publisher recognizes that the U.S. Government retains a nonexclusive, royalty-free license to publish or reproduce the published form of this contribution, or to allow others to do so, for U.S. Government purposes. Los Alamos National Laboratory requests that the publisher identify this article as work performed under the auspices of the U.S. Department of Energy. The Los Alamos National Laboratory strongly supports academic freedom and a researcher's right to publish; as an institution, however, the Laboratory does not endorse the viewpoint of a publication or guarantee its technical correctness.

DTIC QUALITY INSPECTED 3

Form 836 (10/96)

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

RECYCLE OF SCRAP PLUTONIUM-238 OXIDE FUEL TO SUPPORT FUTURE RADIOISOTOPE APPLICATIONS

Louis D. Schulte, Gary L. Silver,
 Geraldine M. Purdy and Gordon D. Jarvinen
 Advanced Technology Group NMT-6
 Nuclear Materials Technology Division
 Mail Stop E510
 Los Alamos National Laboratory
 Los Alamos, NM 87545
 (505) 665-1100

Kevin Ramsey, Jacob Espinoza,
 and Gary H. Rinehart
 Actinide Ceramics and Fabrication Group NMT-9
 Nuclear Materials Technology Division
 Mail Stop E502
 Los Alamos National Laboratory
 Los Alamos, NM 87545
 (505) 665-0162

Abstract

The Nuclear Materials Technology (NMT) Division of Los Alamos National Laboratory has initiated a development program to recover & purify plutonium-238 oxide from impure feed sources in a glove box environment. A glove box line has been designed and a chemistry flowsheet developed to perform this recovery task at large scale. The initial demonstration effort focused on purification of $^{238}\text{PuO}_2$ fuel by HNO_3/HF dissolution, followed by plutonium(III) oxalate precipitation and calcination to an oxide. Decontamination factors for most impurities of concern in the fuel were very good, producing $^{238}\text{PuO}_2$ fuel significantly better in purity than specified by General Purpose Heat Source (GPHS) fuel powder specifications. The results are encouraging for recycle of relatively impure plutonium-238 oxide and scrap residue items into fuel for useful applications. A sufficient quantity of purified $^{238}\text{PuO}_2$ fuel was recovered from the process to allow fabrication of a GPHS unit for testing. The high specific activity of plutonium-238 magnifies the consequences and concerns of radioactive waste generation. This work places an emphasis on development of waste minimization technologies to complement the aqueous processing operation. Results from experiments allowing more time for neutralized solutions of plutonium-238 to precipitate resulted in decontamination to about 1 millicurie/L. Combining ultrafiltration treatment with addition of a water-soluble polymer designed to coordinate Pu, allowed solutions to be decontaminated to about 1 microcurie/L. Efforts continue to develop a capability for efficient, safe, cost-effective, and environmentally acceptable methods to recover and purify $^{238}\text{PuO}_2$ fuel.

INTRODUCTION

Plutonium-238 is an excellent radioisotope for heat source applications in space because of availability, power density, useful lifetime, minimal shielding requirements, and oxide stability. General Purpose Heat Source (GPHS) units containing ~ 150 g $^{238}\text{PuO}_2$ are used in Radioisotope Thermoelectric Generator (RTG) units such as those used to provide electrical power for the Cassini mission to Saturn. Smaller amounts of $^{238}\text{PuO}_2$ fuel, typically ~ 2.7 g, are also used in Radioisotopic Heater Units (RHU) employed recently in Cassini and on the Mars Pathfinder Rover to keep critical components warm in space.

The United States is not presently producing new quantities of plutonium-238 isotope in reactors. One possible domestic feedstock of this nuclear fuel for heat source needs in the immediate future is the existing quantity of relatively impure plutonium-238 isotope residing in scrap and residues. Chemical separations resulting in recovery of high-purity plutonium-238 oxides from this feedsource could be used to produce a significant quantity of GPHS and RTG units. The PF-4 glove box facility operated by Nuclear Materials Technology (NMT) Division has been utilized for high-purity recovery of the weapons isotope, plutonium-239, from a wide variety of scrap materials. This report describes a portion of our efforts to demonstrate technologies for the efficient, safe, cost-effective, and environmentally acceptable recovery and purification of $^{238}\text{PuO}_2$ fuel.

DISSOLUTION AND PURIFICATION OF PLUTONIUM-238 DIOXIDE

The plutonium-238 oxide fuel used as feed for this study was fired at temperatures up to 1873 °K. Dissolution of plutonium oxide fired to this temperatures is slow and often incomplete in refluxing nitric acid (Cleveland 1979).

UNCLASSIFIED NOT UCNI

John Bopp NOV 06 1997
 FSS-18 date

Addition of a small amount of hydrofluoric acid increases the rate of dissolution, but this rate was too slow for our purposes. Better dissolution results were obtained for $^{238}\text{PuO}_2$ feed treated in a high-energy ball mill. Particle size analysis of the milled $^{238}\text{PuO}_2$ feed indicated that an average particle size of less than 1 micron could be obtained. The milled oxide was stirred in a mixture of nitric and hydrofluoric acids refluxing at ambient pressure. This provided good dissolution efficiency after 7-8 hours at reflux. Several dissolutions at 65 g scale have been accomplished. Reagent grade chemicals and distilled or deionized water were used to minimize introduction of impurities. Teflon or polypropylene apparatus was used for handling ^{238}Pu solutions to avoid leaching silicates from glassware. A flowsheet illustrating the process is shown in Figure 1.

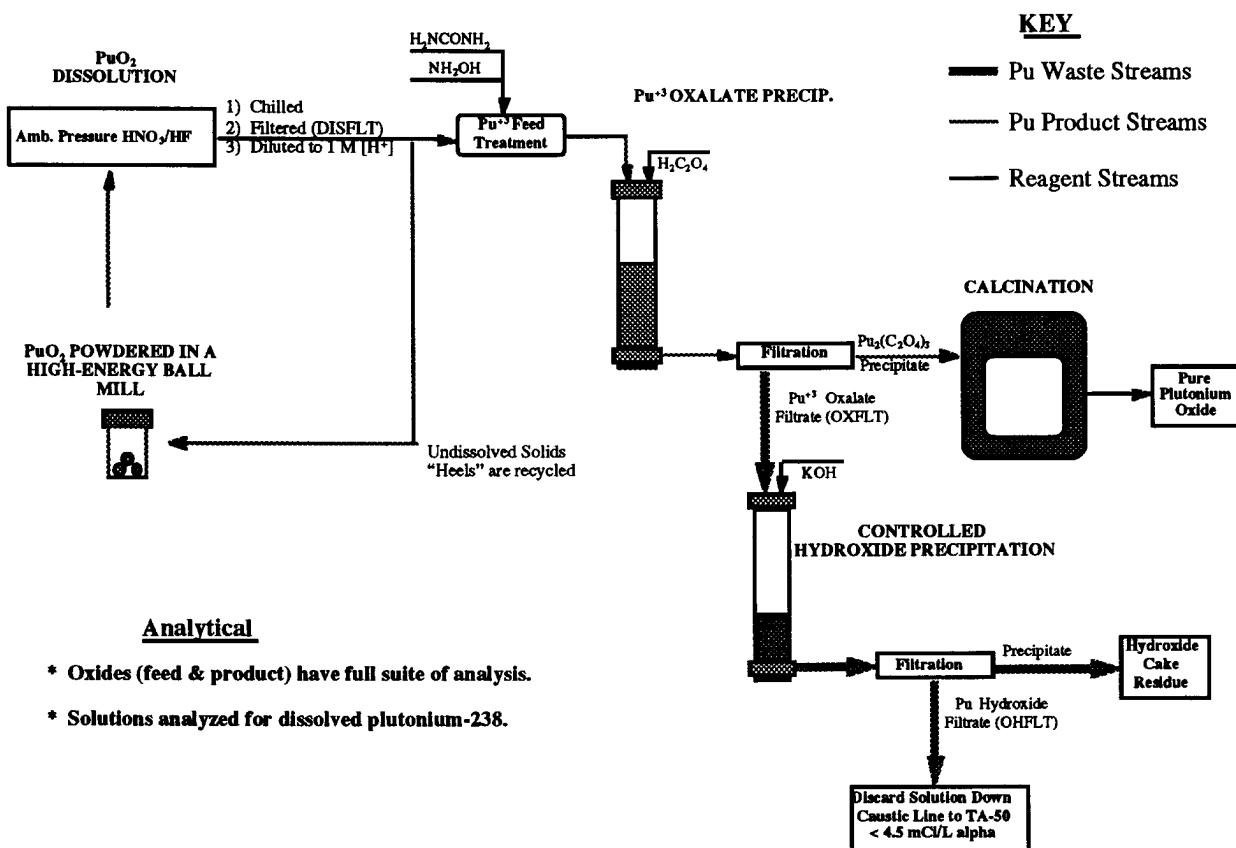


FIGURE 1. Current Plutonium-238 Aqueous Process Flow Diagram.

Plutonium (III) oxalate precipitation was used in this demonstration because of its simplicity, speed and adequacy of purification. In particular, plutonium (III) oxalate precipitation requires no temperature control, shows little detrimental effect from excess oxalic acid, and has a rapid reaction, precipitation, and filtration time. The large amount of radiolysis associated with plutonium-238 complicates control of the Pu oxidation state in nitric acid. In general, the radiolysis of concentrated nitric acid produces an oxidizing environment, and a larger proportion of Pu(VI) (Katz 1986). The presence of radiation-generated oxidizing agents in nitric acid increases the quantity of reducing agents required to effect complete reduction to Pu(III). Others have observed that ^{238}Pu is best reduced to Pu(III) by combinations of "holding reagents" and "reducing agents" (Greinetz 1980 and Burney 1984). Holding agents react with nitrite and other oxidizing species, slowing the rate of Pu(III) oxidation, while the reducing agent rapidly reduces Pu(VI) and Pu(IV) to Pu(III). Hydroxylamine nitrate was used as the primary reducing agent, and with the exception of one run, either sulfamic acid or urea as a holding agent. All steps were performed as rapidly as possible to minimize the effects of radiolysis.

For this study, recovered plutonium(III) oxalate was air-dried and then heated to dull red heat in a stainless steel, slip-top can on a hot plate for about 60 minutes to convert the plutonium to an oxide. Heating for longer times produced no significant weight loss.

RESULTS

Results were obtained on plutonium-238 oxide feed and purified products described in this study. These included measurement of actinide and non-actinide impurities in plutonium-238 oxide. Other results were obtained for the various process solutions based on radiochemical measurements of the plutonium-238 content.

Non-Actinide Element Decontamination

The first row in Table 1 lists the element limits in $\mu\text{g/g}$ that are specified in the GPHS standard (Los Alamos 1996). Three different sources of plutonium-238 oxide fuel, as designated by Feed #1, Feed #2, and Feed #3, were used in the experiments. Table 1 shows analytical results for element impurities ($\mu\text{g/g}$) in the feed materials and plutonium oxide products produced by oxalate precipitation. The numbers in boldface type in Table 1 represent a value from an average of duplicate analysis for the impurities. Italicized numbers indicate analytical results that are at or below the indicated detection limits.

Good decontamination was achieved for silicon, chromium, iron and nickel, with mixed results for aluminum. The column entitled Sum non-Actinides is the summation of all elemental impurities listed above the detection limit. Entries marked “*” in the Sum non-Actinides column indicate an indeterminate value, typically because one element exceeded the range of the measurement method. Product # 1A, for example, was an experiment in which hydroxylamine nitrate was added as a reducing agent, but no holding reagent was included. A poor precipitation occurred leaving a highly colored solution. NaOH was added to force more plutonium to precipitate, resulting in a high Na content in Product # 1A in this preliminary experiment.

TABLE 1. Non-Actinide Impurities ($\mu\text{g/g}$) in Plutonium-238 Oxide Fuel.

	Al	B	Be	Ca	Cd	Cr	Cu	Fe	Mg	Mn	Mo	Na	Ni	Pb	P	Si	Sn	Zn	Sum non-Actinides
GHPS Rev A	500	5	5	300	50	500	200	800	100	50	250	250	500	100	25	750	50	50	2550
Feed # 1	145	5	<i>1</i>	9	<i>10</i>	390	<i>1</i>	830	18	13	20	50	120	20	710	560	5	5	2203
Pdt. # 1A	200	5	<i>1</i>	3.5	<i>10</i>	285	<i>1</i>	335	10	9	20	<i>*</i>	5	10	20	5	5	5	<i>*</i>
Pdt. # 1B	34	5	<i>1</i>	65	<i>10</i>	19	<i>1</i>	67	10	2	20	50	9	10	20	53	5	5	377
Feed # 2	155	5	<i>1</i>	95	<i>10</i>	820	13	1500	10	18	20	50	455	20	20	850	5	23	4051
Pdt. # 2A	185	5	<i>3</i>	40	<i>10</i>	15	20	39	<i>10</i>	1.5	20	50	8	10	20	7	5	5	436
Pdt. # 2B	155	5	<i>3</i>	40	<i>10</i>	19	18	78	<i>10</i>	3	20	50	14	10	20	5	5	5	453
Pdt. # 2C	55	5	<i>2</i>	450	<i>10</i>	23	100	62	<i>10</i>	6	20	50	18	10	20	23	5	5	656
Pdt. # 2D	62	5	<i>2</i>	<i>>750</i>	<i>10</i>	30	<i>3</i>	67	<i>10</i>	1	20	50	10	<i>10</i>	20	11	5	<i>*</i>	
Pdt. # 2E	225	5	<i>2</i>	<i>>750</i>	<i>10</i>	56	<i>3</i>	145	<i>10</i>	1	20	50	25	<i>10</i>	20	91	5	<i>*</i>	
Feed #3	155	5	<i>1</i>	7	<i>10</i>	540	100	555	<i>10</i>	10	20	63	99	<i>10</i>		515	5	15	2121
Pdt. # 3A	176	5	<i>1</i>	350	<i>10</i>	194	75	52	<i>10</i>	10	20	50	17	<i>10</i>		32	5	5	1024
Pdt. # 3B	110	5	<i>1</i>	500	<i>10</i>	25	50	54	<i>10</i>	8	20	50	19	13		23	5	15	919
Pdt. # 3C	5	5	<i>1</i>	300	<i>10</i>	17	50	32	<i>10</i>	1	20	50	14	15		12	5	5	553
Pdt. # 3D	144	5	<i>1</i>	300	<i>10</i>	220	50	255	<i>10</i>	6	20	50	97	25		13	5	5	1217

Only in the case of calcium has there been noted an apparent increase in the amount of a measured impurity in the PuO_2 fuel. It is anticipated that calcium would coprecipitate with plutonium under the conditions of the plutonium(III) oxalate precipitation process (Cleveland 1979). High-purity water, acids and other reagents were used in this demonstration specifically to avoid the possibility of introducing contaminants such as Ca. In spite of these precautions, an increase in Ca was observed in some runs while not in others. Separate analysis of reagents used was instituted in a search for a possible source of calcium contamination, with results to date for all reagents and water below detection limits. Pu can act as an interferent with Ca measurements in the DC arc analysis technique

used on the PuO₂ samples. Work is continuing to determine if the calcium data are significant or a measurement artifact.

Actinide Element Decontamination

The first row in Table 2 lists the actinide impurities in $\mu\text{g/g}$ as specified in the GPHS standard. The numbers in boldface type represent values ($\mu\text{g/g}$) observed for the actinide impurities. Those numbers presented in italics indicate results that are at or below the indicated detection limits. ²³⁴U was by far the largest actinide impurity observed in the feed material because it is the daughter product of ²³⁸Pu by alpha decay. Good decontamination was achieved for U. Np, Am and Th showed little or no decontamination from Pu under the conditions of plutonium (III) oxalate precipitation. As these actinides are well below the GPHS standard in this feed, they were not a major concern. Analysis for actinide impurities of several more ²³⁸PuO₂ product samples, including those originating from Feed # 3, is pending.

TABLE 2. Actinide Impurities ($\mu\text{g/g}$) in Plutonium-238 Fuel.

Sample	Am-241	Np-237	Pu-236	U-234 ^a	Th	Sum of Actinides
GHPS Rev A	5000	5000	2	5000	5000	10,000
Feed # 1	350	978	<i>1</i>	14000	1600	16929
Pdt. # 1A	331	936	<i>1</i>	1900	1600	4768
Pdt. # 1B	340	237	<i>1</i>	60	1700	2338
Feed # 2	135	637	<i>1</i>	7000	530	8303
Pdt. # 2A	57	891	<i>1</i>	60	560	1569
Pdt. # 2B	91	603	<i>1</i>	70	530	1295
Pdt. # 2C	132	517	<i>1</i>	130	350	1130
Pdt. # 2D	136	577	<i>1</i>	280	410	1404
Pdt. # 2E	108	315	<i>1</i>	0	290	714

a. Values listed in the table for U-234 and Am-241 are decay corrected for the time interval between precipitation and analysis. The rapid ingrowth of U-234 adds to the uncertainty in these values.

Process Effluents Solution Data

TABLE 3. Data for Plutonium-238 Oxalate Filtrate Solutions.

Sample	Volume (L)	[H+] (M)	²³⁸ Pu (g/L)	alpha (mCi/L)	Total Pu (g/L)	²³⁸ PuO ₂ Recovered (g)	Soln. Pu % Recovered
OXFLT-1A	1.22	0.57	3.26E-02	558.01	4.09E-02	9.03	99.38
OXFLT-1B	1.05	1.04	2.87E-01	4912.58	3.60E-01	7.79	94.78
OXFLT-2A	1.90	1.33	2.51E+00	42963.67	3.05E+00	16.99	72.08
OXFLT-2B	1.74	1.39	6.28E-01	10749.48	7.64E-01	9.40	86.18
OXFLT-2C	1.90	1.14	1.72E-01	2944.12	2.09E-01	28.87	98.46
OXFLT-2D	1.90	1.14	1.73E-01	2961.24	2.10E-01	27.68	98.39
OXFLT-2E	0.86	1.19	2.25E-01	3851.33	2.74E-01	4.97	94.90
OXFLT-3A	2.80	0.98	1.64E-01	2810.00	2.00E-01	14.35	95.76
OXFLT-3B	2.35	1.10	5.74E-02	982.00	7.00E-02	13.36	98.62
OXFLT-3C	3.24	0.92	4.30E-02	736.00	5.24E-02	27.63	99.31
OXFLT-3D	3.00	1.05	4.76E-02	814.00	5.80E-02	29.04	99.33
OXFLT-3E	2.90	0.88	1.07E-01	1840.00	1.31E-01	28.78	98.52
OXFLT-3F	2.90	1.02	9.06E-02	1550.00	1.10E-01	31.81	98.87

Table 3 lists data for plutonium-238 oxalate filtrate solutions (OXFLT) in this work. An acid-base titration was performed on feed solutions prior to oxalate precipitation to measure and allow adjustment of free acidity in the solution to approximately 1 Molar. Dissolved ^{238}Pu was adjusted to the trivalent state and precipitated with oxalic acid, followed by a filtration step for recovery. Plutonium-238 in the filtrate was measured radiochemically by gas proportional alpha counting, and converted to a total plutonium value, based on measured plutonium isotopic values for the feed material. The "Soln. Pu % Recovered" data is based on the amount of Pu recovered as oxide in the process and the amount of Pu remaining in the oxalate filtrate solution following filtration. Recovery percentages were generally 95% or greater for all runs with the exceptions of products OXFLT-2A and OXFLT-2B that were recovered from higher molarity acid.

Solutions from the oxalate filtration process were neutralized with KOH, allowed to stand at least overnight and then filtered through Whatman 42 paper. Table 4 lists data for plutonium-238 hydroxide filtrate solutions (OHFLT) in runs where a sample aliquot was immediately removed and diluted in 0.1 M HNO₃. The alpha activity observed in hydroxide filtrate solutions after a single filtration ranged from approximately 3–170 millicurie/L. Solutions below the alpha activity limit (4.5 millicurie/L) of the PF-4 caustic drain line are allowed to be transferred to a centralized collection point in the wastewater treatment facility for further decontamination.

The "Soln. Pu % Recovered" data is based on the amount of Pu initially present in the oxalate filtrate and the amount of Pu remaining in the filtrate following the hydroxide precipitation process. The "Overall Soln. Pu % Loss" data shows the total loss of ^{238}Pu in process effluent streams from the point of dissolution through the sequential precipitation and filtration steps.

TABLE 4. Data for Plutonium-238 Hydroxide Filtrate Solutions Sampled Immediately After Filtration.

Sample	Volume (L)	Pu-238 (g/L)	alpha (mCi/L)	Total Pu (g/L)	Soln. Pu % Recovered	Overall Soln. Pu % Loss
OHFLT-1A	1.59	3.13E-04	5.36	3.93E-04	98.75	0.0078
OHFLT-1B	1.44	4.32E-04	7.39	5.42E-04	99.79	0.0108
OHFLT-2A	2.30	1.62E-04	2.77	1.97E-04	99.99	0.0022
OHFLT-2B	2.10	8.85E-04	15.15	1.08E-03	99.83	0.0235
OHFLT-2C	2.20	8.68E-04	14.86	1.06E-03	99.42	0.0090
OHFLT-2D	2.20	4.53E-04	7.75	5.51E-04	99.70	0.0049
OHFLT-2E	1.00	7.30E-04	12.50	8.88E-04	99.62	0.0192
OHFLT-3A	4.00	1.02E-02	173.80	1.24E-02	91.16	0.3749
OHFLT-3B	4.00	1.83E-03	31.35	2.23E-03	94.57	0.0748
OHFLT-3C	4.00	7.00E-03	119.90	8.54E-03	79.89	0.1393
OHFLT-3D	4.00	6.09E-03	104.28	7.43E-03	82.92	0.1153

Several of the hydroxide filtrate solutions were allowed to stand for a few weeks and then supernatant solution sampled. Data for these solutions, designated with a W at the end of the Sample name, are presented in Table 5. Alpha activity for solutions sampled in this manner were considerably decreased than the same solutions (OHFLT-3A through OHFLT-3D in Table 4) sampled immediately after filtration. A longer period of time appears to allow additional plutonium to precipitate or settle from the solution.

TABLE 5. Data for Plutonium-238 Hydroxide Filtrate Solution Supernatant Sampled Weeks Later.

Sample	Volume (L)	Pu-238 (g/L)	alpha (mCi/L)	Actual Pu (g/L)	Soln. Pu % Recovered
OHFLT-3AW	4.00	5.67E-05	0.97	6.92E-05	99.95
OHFLT-3BW	4.00	5.50E-05	0.94	6.70E-05	99.84
OHFLT-3CW	4.00	6.19E-06	0.11	7.55E-06	99.98
OHFLT-3DW	4.00	1.32E-04	2.26	1.61E-04	99.63
OHFLT-3EW	4.00	8.35E-04	14.30	1.02E-03	98.93
OHFLT-3FW	4.00	4.06E-05	0.70	4.95E-05	99.94

The bulk of the effluent solutions generated were combined, treated with additional caustic and iron sulfate, again filtered, and transferred to the wastewater treatment facility when decontamination below the PF-4 caustic drain line alpha limit (4.5 millicurie/L) was achieved.

Ultrafiltration Solution Data

Passing small samples of each of the aged OHFLT supernatent solutions through an 0.2 micron filter resulted in no additional decontamination. Samples of supernatent OHFLT-3BW adjusted to different pH values were also passed through an ultrafilter with a 10,000 molecular weight cut-off (~0.008 microns). This also resulted in no significant decontamination. Additional samples of this supernatent solution were treated using Polymer Filtration. This technique employs a water-soluble chelating polymer to bind selected metal ions and then uses an ultrafiltration step to separate the polymer-metal ion complex from the bulk of the solution (Jarvinen 1997). A concentrated solution of a water-soluble polymer containing phosphonic acid chelating groups was added to the supernatent solution to give a 0.2 to 0.4 weight % solution of the polymer. The pH of the solution was adjusted between 4 and 13 and the solution filtered through a 10,000 MW cut-off filter. As shown in Table 6, up to 98% of the remaining alpha activity could be removed at pH 6.5. These preliminary results indicate that two stages of polymer filtration could allow the solutions to be decontaminated to less than 0.5 microcurie/L which is the upper limit for discharging aqueous solutions to the industrial waste line at Los Alamos. The concentrated Pu-238 bound to the polymer could be recovered by acidifying the solution to release the metal and adding a relatively small volume of wash solution to remove the metal by allowing it to pass through the ultrafilter. The polymer solution could then be reused. Another option involves sending the loaded polymer solution directly to a disposal process such as evaporation followed by cementation.

TABLE 6. Removal of Alpha Activity from Hydroxide Filtrate Supernatent OHFLT-3BW by Polymer Filtration

Sample	wt % polymer	pH	Final alpha activity (mCi/L)	% alpha removal
3BW1	0.20	4.0	1.33E-02	44
3BW2	0.40	4.0	9.50E-03	59
3BW3	0.20	6.5	6.87E-04	97
3BW4	0.40	6.5	5.55E-04	98
3BW5	0.20	8.5	1.06E-03	95
3BW6	0.40	8.5	8.09E-04	96
3BW&	0.20	13.0	4.88E-03	79
3BW8	0.40	13.0	2.17E-03	91

CONCLUSIONS

NMT Division has the expertise and facilities to make large scale recovery and purification of plutonium-238 for oxide fuel a practical option. Additional supplies of plutonium-238 oxide fuel from many existing sources, including scrap and materials that fail the GPHS standard, can be recovered in a glove box environment. For much of the scrap $^{238}\text{PuO}_2$ available as feed, including that with significant ^{234}U ingrowth, the plutonium (III) oxalate precipitation procedure is adequate to meet the GPHS fuel standard. Good dissolution efficiencies were observed for high-fired $^{238}\text{PuO}_2$ material that was first treated in a high energy ball mill. Overall losses of the ^{238}Pu isotope to aqueous effluent streams were small. More than enough purified $^{238}\text{PuO}_2$ fuel has been recovered to begin the process for fabrication of a 150g GPHS. The flowsheet shown in Figure 1 will be adapted to allow anion exchange recovery of ^{238}Pu from leaner feed residues, and to incorporate additional waste minimization technologies for treatment of process effluent.

Acknowledgments

Acknowledgments to Larry Avens, Elizabeth Folty, Tim George, Keith Fife, Mark Dinehart, Steve Yarbro & Steve Schreiber for helpful conversations. Thanks to Charles V. Puglisi, Carlos D. Dozhier, Christina M. Lynch, Paul F. Moniz, Robert W. Mathews, Richard T. Romero, and Mary Severinghaus of LANL Group NMT-9 for the help with "hot" jobs and sample removal. Thanks to Mary Ann Reimus for particle size analysis measurements. Thanks to Jim Jones for help with hardware and glove box development. Thanks to Johnny N. Quintana, Margaret

T. Trujillo and Nelson D. Stalnaker of LANL group CST-8 for solution radiochemical analyses. Thanks to numerous others in LANL groups CST-8 and CST-9 for elemental and radiochemical analysis of the plutonium-238 oxide products.

References

Burney, G. A. and P. K. Smith (1984) "Controlled PuO_2 Particle Size from Pu(III) Oxalate Precipitation," Dupont/Savannah River Laboratory Report DP-1689.

Cleveland, J. M. (1979) "The Chemistry of Plutonium," Published by The American Nuclear Society, LaGrange Park, IL 60525. 573-575.

Greinetz, R. M. and D. H. Neal (1980) "Plutonium(III) Oxalate Precipitation and Calcination Process for Plutonium Nitrate to Oxide Conversion," Rockwell International/Rocky Flats Plant Report RFP-2603.

Jarvinen, G. D. (1997) "Water-Soluble Chelating Polymers for Removal of Actinides from Wastewater," in Proceedings of the Efficient Separations and Processing Crosscutting Program 1997 Technical Exchange Meeting, January 28-30, Gaithersburg, Maryland, Pacific Northwest National Laboratory report PNNL-SA-28461 (Jan. 1997), pp. 2.25-2.30.

Katz, J. J., G. T. Seaborg and S. R. Morss (1986) "The Chemistry of the Actinide Elements," Second Edition. Published by Chapman and Hall Ltd., 1:825

Los Alamos National Laboratory Drawing Number 26Y-318180 Rev A (1996) "GPHS 238-Plutonium Dioxide Feed Powder Specification."

Mullins, L. J. (1972) "Preparation and Development of Medical-Grade Plutonium-238 Fuels, July 1, 1967 - June 30, 1971," Los Alamos National Laboratory Report LA-4940.

Rogers, D. R. and B. R. Kokenge (1973) "Hydroxide and Oxalate $^{238}\text{PuO}_2$ Radioisotopic Fuel Processes at Mound Laboratory," Mound Laboratory Report MLM-MU-7364-0100.

Shupe, M. W., L. J. Mullins, A. N. Morgan, A. L. Gonzales, A. E. Ogard, A. M. Valentine and J. A. Leary, (1970) "Operating Experience on an Economical Plutonium-238 Processing Facility," In *Radiation Safety in Hot Facilities*, IAEA-SM-125/45.

M98002663



Report Number (14) LA-UR-97-4292
CONF-980103--

Publ. Date (11) 199711
Sponsor Code (18) DOE/DP, XP
JC Category (19) UC-713, DOE/ER

DOE