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CHEMISTRY-SEPARATION PROCESSES FOR PLUTONIUM AND URANIUM

AEC RESEARCH AND DEVELOPMENT REPORT

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CHEMICAL TECHNOLOGY DIVISION

PROGRESS REPORT FOR MONTH OF DECEMBER, 1949

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CHEMICAL TECHNOLOGY DIVISION

PROGRESS REPORT FOR MONTH OF DECEMBER, 1949

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1.0 Abstract

Chemical development is reported on the TBP Process, Purex Process, 25 Process, 23 Process, RaLa Process, Thorex Process, Fluoride Process studies, homogeneous reactor slurry study, liquid waste evaporation, and plutonium recovery processing.

[redacted]

2.0 Introduction

The development of the TBP Process for the recovery of uranium from the Hanford bismuth phosphate waste was continued as a major problem in the Laboratory and Semi-Works. A laboratory progress report on the TBP Process was issued and a meeting was held with Hanford representatives to coordinate the cooperative development program. Plutonium recovery continued to be studied in the laboratory, while pilot plant tests were made on solvent re-use and effect of scrub.

The Purex Process development in the laboratory included an evaluation of the use of Filtrol for improving zirconium and ruthenium decontamination, and column tests to determine the HETS values for the system.

The 25 Process development program was completed except for Uranium-235 purification and concentration phase of the program.

The first significant amounts of Uranium-233 were separated in the Pilot Plant development using the methyl isobutyl ketone flowsheet, 23 Process.

A high radiation level RaLa run using the Acetate-Citrate ion exchange barium purification procedure was successfully completed and plans were made for the first full level run.

The Thorex Process flowsheet has been revised to improve operability of the process in a packed column.

Introduction (continued)

The fluoride process study continued to evaluate reaction of plutonium fluoride with metallic surfaces of the equipment.

The addition of bentonite as a stabilizing agent for uranium oxide slurries was tested at high temperature and pressure to simulate conditions approaching those in a homogeneous nuclear reactor.

A method was developed for evaluating antifoam agents for the plant radiochemical liquid waste evaporator and scouting tests were made on numerous antifoam agents.

The plutonium recovery from the Redox IBP has proved to be a difficult job, as was expected, and short laboratory studies have been necessary to minimize the plutonium losses.

3.0 Summary

TBP Process

The elimination of the scrub decreased the fission product decontamination by a factor of approximately two, and did not effect the plutonium decontamination.

Summary (continued)

Purex Process

(1) The use of Filtrol adsorption for the removal of zirconium and ruthenium from the metal solution was limited by the fact that plutonium was also adsorbed.

(2) The extraction HETS for uranium was found to be approximately twice that for Redox.

23 Process

Approximately 50 grams of Uranium-233 has been separated from Hanford irradiated thorium metal.

RaLa Process

(1) A procedure for removal of the aluminum - silicon bonding alloy from uranium slugs by alternate sodium hydroxide - nitric acid treatments is being developed.

(2) Dowex 50 ion exchange resin was found to be more effective than IRC-50 for the barium-strontium separation.

Summary (continued)

Thorex Process

The column operability of the Thorex Process was improved by decreasing the feed concentration and the degree of solvent saturation without affecting the decontamination.

Radiochemical Waste Processing Evaporator Antifoam Agents

Foam reduction factors of three to four were obtained in laboratory tests with Alrowax-140, DC-200, Nonisol-300, Oleyl O-8, OPE-1 and Span 20 on representative samples of liquid radiochemical waste.

4.0 TBP Process

4.1 Laboratory Development

A progress report has been issued which covers the laboratory studies on the TBP Process through December, 1949 (ORNL-557, TBP Process for Uranium Recovery from Metal Waste - Laboratory Summary by T. C. Runion and C. V. Ellison). The subjects covered by this report are feed preparation, solvent studies, plutonium and fission product studies, waste evaporation, effect of phosphate and sulfate concentration, specific gravity and viscosity of process streams, and proposed process flowsheets.

4.2 Re-use of Solvent - Pilot Plant

It has now been demonstrated that the TBP Process solvent, a mixture of tributyl phosphate and Gulf BT, can be recycled 12 times with no significant effect on uranium loss or plutonium and fission products separation factors (see Table 4.1). The feed for these runs (HS-13 and 14) was Hanford 103 U supernatant and the solvent was recycled six times during each run.

The analytical procedure for determining Uranium has been changed from a colorimetric method to a fluorometric method, since the latter method is more accurate at low uranium concentration. This explains the apparent decrease in uranium loss in HS-14. Actually, it indicates that the uranium loss values reported previously are high by a factor of roughly two.

The variations in fission product decontamination were not significant, being within the limits of analytical accuracy. In any event, the final uranium product at equilibrium would have less than twice the natural amount of activity and the handling hazard would not be significantly higher.

It is planned to make final check runs with feed prepared from actual Hanford sludge.

4.3 Effect of Scrub - Pilot Plant

The complete elimination of the scrub section and solution feed from the TBP Process resulted in a decrease in fission product separation from uranium by a factor of approximately two (see Table 4.1). The addition of ferrous sulfamate to the metal feed immediately before entering the column was effective in yielding the usual satisfactory plutonium decontamination.

No difference in decontamination was observed between runs with ten and five foot scrub sections. With a five foot scrub section, the effect of substantially reducing the scrub-to-feed ratio will be studied in future runs.

Table 4.1

SEMI-WORKS TBP SOLVENT EXTRACTION RUNS

Equipment: 1 1/2" S/S extraction column packed with 1/4" x 3/8" split rings
extraction section 17.5'
scrub section 5.0'
2 1/2" S/S strip column packed with 1/4" x 3/8" split rings
strip section 18.0'

Materials: Hanford 103 U supernatant
Tributyl phosphate in hydrocarbon diluent (Gulf BT - high flash-point solvent naphtha)

Run #	AF			AS		AX		BX	Product BP			U Losses ***	
	U g/l	H^+ N	Flow rate cc/m	H^+ N	Flow rate cc/m	TBP %	Flow rate cc/m	Flow rate cc/m	β c/mg U	Pu α c/mg U	γ mv/mg U	Extraction %	Strip %
HS-13A	45.0	3.1	60	2.7	30	12.6	80	80	3.4	0.75	0.001	**	0.059
B	↓	↓	↓	2.7	↓	12.5	100	100	6.0	0.90	0.001	0.08	0.095
C	↓	↓	↓	3.1	↓	12.8	↓	↓	4.6	0.64	—	0.10	0.108
D	↓	↓	↓	3.0	↓	12.4	↓	↓	6.3	0.80	—	0.11	0.071
E	↓	↓	↓	3.0	↓	12.4	↓	↓	12.0	1.10	—	0.12	0.089
F	↓	↓	↓	3.15	↓	12.4	↓	↓	3.5	0.96	0.002	0.10	0.147
14A	48	3.0	60	2.6	30	13.2	108	120	6.5	0.25	0.004	0.047	0.057
B	↓	↓	↓	3.2	↓	13.9	↓	↓	13.5	0.37	0.006	0.050	0.068
C	↓	↓	↓	3.1	↓	13.8	↓	↓	17.0	0.55	0.008	0.028	0.077
D	↓	↓	↓		↓	13.8	↓	↓	15.0	0.45	0.007	0.025	0.076
E	↓	↓	↓		↓	13.7	↓	↓	14.4	0.47	0.007	0.031	0.048
F	↓	↓	↓		↓		↓	↓	13.0	0.43	0.004	0.047	0.056
Runs without scrub: Ext. section 22.5'; ferrous sulphamate fed into feed stream (5 cc/min.; 0.4 molar)													
15	58.9	2.9	60	—		13	90	100	28	0.45	0.012	<0.01	<0.01

* 0.05 M in ferrous sulphamate.
** High and erratic losses because of insufficient solvent to extract the uranium.
*** The uranium analytical procedure was changed from colorimetric to fluorometric after Run HS-13.

5.0 Purex Process

5.1 Filtrol Study for Zirconium and Ruthenium Separation

The filtration of the metal solution through Filtrol increased the gross fission product decontamination for uranium in the Purex Process from 8×10^3 to 10^5. This was due to the adsorption of the major part of the ruthenium and zirconium on the Filtrol, these being the limiting activities for the process (see Table 5.1).

However, to make this procedure useful, it will be necessary to develop a satisfactory method of eluting plutonium from Filtrol. The adsorption of plutonium on Filtrol is acid dependent; 2.9% was adsorbed at 3 N HNO_3 while 5 to 9.8% was adsorbed at 0.8 N HNO_3. Plutonium elution with 5 N HNO_3 has been demonstrated at Hanford.

The effects of temperature and particle size on the zirconium and ruthenium adsorption on Filtrol were found to be not important (see Table 5.2).

Table 5.1

Comparison of Decontamination Factors in "A" Column Obtained Before and After Filtrol Filtration of Purex Feeds

Conditions:

Feed: 130 mg U/ml, 3 N HNO_3
Scrub: 3 N HNO_3
Organic: 85/15 mixture sulfonated varsol/butanol-free TBP, 0.3 N HNO_3
Flow Ratios: F/S/O = 3/2/10
Volumn Changes: 2 to 2.5
Stages: 4 ext'n, 6 scrub

Run No.	Treatment	Decontamination Factors					
		Extraction			Overall		
		Tot. β	Ru β	Zr β	Tot. β	Ru β	Zr β
D-24	Control	$8x10^2$	$1.7x10^2$	$1.2x10^3$	$8x10^3$	$6.0x10^2$	$3.3x10^3$
D-38	(a)	-	$5.0x10^2$	$5.0x10^2$	-	$2.4x10^3$	$1.2x10^4$
D-44	(b)	$2.2x10^4$	$1.63x10^3$	$1.5x10^4$	$1.1x10^5$	$7.2x10^3$	$3.3x10^4$

(a) Filtrol filtration at 3 N HNO_3

(b) Filtrol filtration at 0.8 N HNO_3

[redacted]

Table 5.2

Effect of Temperature and Particle Size on the Carrying of Zirconium on "Super" Filtrol

Conditions: Feed: 130 mg U/ml, 3 N HNO_3, $2x10^7$ Tot. β, $2x10^6$ Zr β
Filtrol: 50 gm/l "Super" Filtrol, qualitatively graded into particle sizes designated as coarse, medium, fine.
Contact: Ten minute agitation period.

Particle Size	% Zr Carried	
	at 25°C	at 90°C
Coarse	51.5%	55.5%
Medium	57.0%	52.0%
Fine	49.5%	55.5%

5.2 Extraction HETS

The height of an equivalent theoretical stage, the HETS, was determined for the extraction section under Purex Process conditions to be 1.9 and 2.6 feet, respectively, at 15 and 50 percent of flooding (see Table 5.3). The column was 1-1/2 inch diameter glass column with a 12 foot packed height using 1/4 x 1/4 inch split stainless steel rings.

The HETS was determined by comparing stage-wise data obtained from countercurrent batch runs with the column data. During the column runs, aqueous samples were taken at intermediate points along the column which indicated that the first stage below the aqueous feed point was about half the height of the subsequent two stages (see Table 5.4 and 5).

Table 5.3

HETS for Tributyl Phosphate Extraction of Uranium in Column 1A

Column: 1 1/2 inch glass pipe, packed with 1/4 x 1/4 split rings

Solvent: 15% TBP, 85% Gulf BT, 0.15 M HNO_3

Feed: 79 g U/l, 3 M HNO_3 corresponds to three volumes of 1AF and two volumes of 1AS

Percent of flooding	Length of first stage (feet)	Length of second stage (feet)	Length of third stage (feet)	Average length over three stages (feet)
15	1.0	2.8	2.8	1.9
50	1.5	3.3	3.0	2.6

[redacted]

Table 5.4

Uranium Concentration in the Aqueous Phase of Column Runs

Column: 1 1/2 inch glass pipe 12 feet long packed with 1/4 x 1/4 inch split rings and equipped with aqueous phase samplers at 2 1/4, 4 3/4, 7 1/4 and 9 3/4 feet below the feed plate.

Aqueous Feed: 3 M HNO_3, 79 g U/l corresponding to 3 volumes of 1AF and 2 volumes of 1AS

Solvent: 15% TBP, 85% "Gulf BT", 0.15 M HNO_3

Flow ratio: Feed: solvent = 1:2

Percent of flooding*	Uranium concentration in the aqueous phase (g/l)			
	2 1/4 feet below feed plate	4 3/4 feet below feed plate	7 1/4 feet below feed plate	9 3/4 feet below feed plate
15	2.83	0.084	0.0032	
50	5.08	0.32	0.046	0.0038

* Flooding in this system is 750 gals/hr./ft.^{2}

[redacted]

Table 5.5

Uranium Concentration in Aqueous Phase Versus Stage Number

Equipment: Five stage countercurrent batch extractor to simulate 1A column

Feed: 3 M HNO_3, 79 g/U/l

Solvent: 15% TBP, 85% "Gulf BT", 0.15 M HNO_3

Flow ratio: Feed: solvent = 1:2

Run	Uranium Concentration in the Aqueous Phase (g/l)				
	1st Stage	2nd Stage	3rd Stage	4th Stage	5th Stage
1	11.2	0.35	0.03	0.0003	0.0001
2	11.2	-	0.02	-	0.0004

[redacted]

23 Process - Pilot Plant (continued)

The thorium metal used in this work was irradiated approximately 100 days in the Hanford reactor and has been cooled about 20 months. The radiation level of the product, Uranium-233, is very low and will be purified and concentrated in unshielded laboratory equipment designed for this purpose.

8.0 RaLa Process

8.1 Metal Solution

A tentative procedure has been developed to remove the aluminum silicon bonding from the Hanford uranium slugs. Following the solution of the aluminum jackets in sodium hydroxide-sodium nitrate solution, the slugs were boiled six hours in 6 M NaOH solution, and then boiled for three hours in 1 M HNO_3 solution containing 0.005 M mercuric nitrate.

This procedure does not dissolve the bonding material but undercuts it, allowing it to be washed from the dissolver. Less than one percent of the uranium is dissolved.

The addition of hydrogen peroxide to the sodium hydroxide digestion seemed to be effective in reducing the time required.

8.2 Acetate-Citrate Process - Effect of pH and Sodium Citrate Concentration on the Barium Strontium Separation

Dowex 50 cation exchange resin was four times as effective as IRC-50 for the barium-strontium separation and the optimum chemical condition for the elution solution was 0.1 molar sodium citrate at pH 9. The barium-strontium separation factor for Dowex 50 ranged from 7 to 10, while for IRC-50, it was only 2 to 3. Increasing the sodium citrate concentration has a small effect on the barium-strontium separation factor but there was no significant effect for pH.

The major effect of increasing the sodium citrate concentration was to decrease the distribution coefficient (resin to solution) which decreased the separation of strontium from barium and increased the barium loss.

8.3 Versene Process - Reproducibility Laboratory Runs

Laboratory studies have now established process conditions for a two-column Versene procedure which, if successful in larger scale studies, should eliminate the metathesis step, decrease the time cycle by 10 hours, and reduce the amount of activity initially adsorbed on the ion exchange column. The barium loss under optimum conditions is approximately two percent.

Versene Process - Reproducibility Laboratory Runs (continued)

The general procedure is as follows:

(1) The lead-barium sulfate precipitate is dissolved in 0.1 molar Versene pH 6.3 and adsorbed on a Dowex 50 bed. The barium and 50 percent of the strontium is adsorbed, all the rest of the ions pass through the column as anion complexes (i.e., lead, rare earths, iron, etc.).

(2) The barium is eluted with 0.23 molar Versene pH 11.

(3) The barium solution is adjusted to pH 4 and adsorbed on a second Dowex 50 bed.

(4) Strontium is eluted with 0.1 molar sodium citrate pH 9.

(5) Sodium is eluted with 0.5 molar hydrochloric acid.

(6) The barium product is eluted with six molar nitric acid.

9.0 Thorex Process

9.1 Effect of Feed Concentration

The concentration of the thorium and aluminum extraction aqueous feed was reduced from one molar and 0.26 molar, respectively, to 0.75 molar and 0.2 molar to increase the packed column flooding rates.

Effect of Feed Concentration (continued)

A countercurrent batch run (six extraction and six scrub stages) with 15 month cooled feed gave a beta decontamination factor of 600 and at 0.004% thorium loss which was comparable to that obtained in the previous flowsheet.

The flooding rates in a 1-1/2 inch diameter column packed with 1/4 x 1/4 inch split rings was 250 and 520 gallons per hour per square foot of packed column cross section, respectively, for extraction and stripping. In the stripping column it was possible to maintain steady operation at a rate as high as 520 gallons, but at 335 gallons packing flooding was noted.

The flowsheet conditions for these studies was as follows:

Extraction aqueous	0.75 M $Th(NO_3)_4$
	0.2 M $Al(NO_3)_3$
	0.05 M NaF
	100 volumes
Extractant	50% TBP
	50% Varsol
	1.0 M HNO_3
	250 volumes
Scrub	6.0 M HNO_3
	25 volumes

Effect of Feed Concentration (continued)

Organic Diluent added to solvent before stripping to obtain favorable density	100% Varsol
	250 volumes
Strip	Water
	500 volumes

10.0 Fluoride Process Study - Surface Area Effect

The study of the fluorination of irradiated uranium has been continued to develop the procedure for the separation of the uranium, plutonium and fission products. It has been found that the plutonium fluoride reacts with the metallic equipment surfaces but may be removed by flushing with fluorine. The plutonium hold up in the fluorinator was reduced to 8.4 percent by decreasing the surface area (see Table 10.1).

When the product gases pass through a 70°C trap, 70.8 percent of the plutonium fluoride was removed with probably less than 1 percent uranium fluoride. More than 90 percent of the uranium fluoride with 5.8 percent of the plutonium fluoride was caught in the cold trap where the plutonium fluoride was separated by distilling the uranium fluoride over into a second-cold trap.

Fluoride Process Study - Surface Area Effect (continued)

The fission product decontamination continued to be approximately 10 across fluorination and approximately 100 across distillation. Decreasing the size of the reactor has decreased the decontamination, further indicating that the fission products are probably physically entrained in the gas stream.

The future program will investigate a more efficient plutonium trap and the effect of filtration on the removal of the fission product activities from the gas stream.

Table 10.1

Uranium Fluorination and Distillation Distribution of Uranium

Plutionium and Fission Products

Procedure:

1. Uranium metal, irradiated in the ORNL pile 335 days and cooled for two years, was reacted with elemental fluorine.

2. Volatile products were passed through a copper trap at 70-80°C (This trap was omitted in Runs 18 and 19). Remaining volatile products were condensed in a cold trap at -80°C

3. Cold trap removed from system and fluorides redistilled into a second cold trap.

Run No.	Fluorination %Hold Up		Plutonium Trap % Hold Up		Fluorination Decontamination		Distillation					
							Still Residue				Decontamination	
	U	Pu	U	Pu	β D.F.	γ D.F.	% U	% Pu	% U	% Pu	β	γ
18	0.03	46.0			25	15	1.00	48.2	89.8	2.7	1300	100
19	0.08	40.8			20	15	1.6	41.5	92.8	0.41	400	300
21	1.69	26.6	0.24	18.6	10	10	-	-	88.0	49.9	12	13
29	0.08	8.4		70.8	7	5	0.8	5.8	90.0	0.014	1000	370

11.0 Homogeneous Reactor Slurry Study - Effect of Slurry Stabilizing Agent

The addition of bentonite has been effective in holding uranium oxide slurries in suspension after being heated for one hour at 250°C, at a pressure of 575 pounds per square inch gauge. In the absence of bentonite, previous tests resulted in the complete caking out of the uranium.

The neutron cross section of the bentonite in a 1 to 3% slurry has been estimated to be less than that of the equivalent amount of sulfur to yield a 40 gram per liter solution of uranium as uranyl sulfate. Irradiation measurements will be made to check this.

It has been demonstrated that the smaller bentonite particles are the more effective for increasing the viscosity of water, and size separations have been made using centrifugal techniques. A uranium oxide slurry (40 grams of uranium per liter) stabilized with 1.4 percent of the fine bentonite was more stable than a similar slurry containing 3.0 percent of the normal bentonite.

Further studies on the effect of bentonite are proposed.

12.0 Radiochemical Waste Processing - Evaluation of Antifoam Agents for Plant Waste Evaporator

Laboratory evaluation of numerous antifoam agents showed that the following gave foam reduction factors of three to four Alrowax-140, DC-200, Nonisol-300, Oleyl O-8, OPE-1, and Span 20 (see Table 12.1). Foaming is the major operating problem related to the pot evaporator used for concentrating the radiochemical wastes at ORNL. It is operated on a semi-continuous basis and the amount of concentration that can be obtained is that degree at which a foam-over occurs.

The antifoam agents were evaluated using two solutions from the representative evaporator runs, one which foamed in the plant at the start of evaporation (#190) and one which did not foam (#196) (see Table 12.2).

Table 12.1

Effect of Antifoam Agents on Tank Farm Evaporator Feeds

Temperature: 110°C
Volume of Feed: 5 ml
Foam Tube Diameter: 1" I.D.
Gas Flowrate: 630 cc per min.
Antifoam Agents Concentration: 100 p.p.m.

Code No.	Antifoam Agent	Foam Volume (ml)		
		Diluent	#190	#196
-	-	-	45-55	40
1	Nalco 70	Benzene	25	17-22
2	Nalco 71	Benzene	35-40	25-40
3	Yelkin TTS	Benzene*	20-25	10-15
4	Atco Hydrofoam	CCl	35-40	40-50
5	Defoamer SB	Benzene	35-40	25-30
6	Oleyl 0-8	Benzene	16	14
7	Antifoam F-1	Benzene	35-40	23
8	Nacconal NRSF	Benzene *	>100	>100
9	DC Antifoam A	H_2O	30	20
10	DC 200	Benzene	17	25
11	OPE-1	Benzene	21	17
12	Carnation Oil	Gulf BT	30	30-35
13	Petronate	Gulf BT*	30	30-40
14	Alrowax 140	Gulf BT*	13	14
15	Nomisol 300	Gulf BT*	25	11
16	Amine 0	Gulf BT	25	25-30
17	Span 20	H_2O	18	15
18	Span 85	Gulf BT	30	25-30
19	Drew 87-L	H_2O	40-45	20-25
20	Antifoam 8	H_2O*	35-40	15-20
21	Antifoam A	H_2O*	35-45	25-30
22	Alkaterge	C BT	30-35	30-40
23	Dibutyl Phthalate	Gulf BT	35-40	30-40
24	TBP	Gulf BT	30-35	25-35
25	Ultrawet	H_2O	45-60	40-60
26	Aerosol OT*	H_2O	>100	>100
27	Defoamer 15	H_2O	>100	>100
28	Defoamer 16*	Gulf BT	15	35-50
-	Benzene 16*	-	40	25-30
-	Gulf BT	-	No effect	35
-	CCl_4	-	No effect	40

* These were % by weight, others by volume.

[redacted]

Table 12.2

Ionic Analysis of Tank Farm Evaporator Feeds

Ions Analyzed	Concentration (mg/ml) Feed #190	Feed #196
$CO_3^{=}$	2.4	11.1
$PO_4^{\equiv}$	0.37	2.6
$SO_4^{=}$	0.5	6.0
Cl^-	0.08	0.18
NO_3^-	12.4	21.7
R.N.*	0.0027	0.0058
Na^+	5.3	20.3
Total Solids	13.7	52.1

R.N.* refers to the reducing normality as titrated with ceric ion.

#190 foamed badly in the plant evaporator

[redacted]

13.0 Plutonium Recovery - Pilot Plant Assistance

The plutonium from the IBP solution from the Redox development program has been recovered from Tank F-2 with fair success and the processing of Tank E-2 is in progress. During the progress of the recovery runs, unusually high plutonium losses have occurred which required numerous flowsheet changes (see Table 13.1).

The process changes were as follows:

(1) The sodium dichromate was eliminated from the scrub in the IA to prevent carry over into IB. No effect on plutonium loss in IA was noted.

(2) The IB hexone scrub was made 0.05 M HNO_3 to prevent iron precipitation.

(3) The plutonium loss in the IBU stream was reduced by decreasing the aluminum nitrate concentration in IBX.

(4) The IBP solution when processed through the second plutonium cycle operated satisfactorily at the lower salt and acid condition.

(5) Decreasing the nitric acid concentration in the ID scrub from 0.3 to 0.05 molar decreased the acid carry over into the strip column with seriously effecting the plutonium loss.

W. K. Eister

W. K. Eister

WKE/rcp

[redacted]

Table 13.1

Revised Conditions of Column Solutions Used in the Plutonium Recovery Process

Stream	$Al(NO_3)_3$ M	HNO_3 N	$Na_2Cr_2O_7$	Fe^{++} Sulf. M
1AF	1.0	0.3	0.1	
1AS (F-2)	1.3	0.3	0.0	
1AS (E-2)	1.3	0.05	0.0	
1AX		0.05		
1BX	0.9			0.05
IBS		0.05		
1CS		0.1		
1DF	0.9	0.05	0.1	
1DS	1.3	0.05	0.0	
1DX		0.05		
1ES		0.05		