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ORNL 708
Chemistry Separation Processes for Plutonium and Uranium

LABORATORY RECORDS
1954

CHEMICAL TECHNOLOGY DIVISION

PROGRESS REPORT FOR MONTH OF MARCH, 1950

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CHEMICAL TECHNOLOGY DIVISION

PROGRESS REPORT FOR MONTH OF MARCH, 1950

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Contents

		Page Number
1.0	Abstract	5
2.0	Introduction	6
3.0	Summary	7
4.0	TBP Process	8
4.1	Dissolution of Hanford Metal Waste Sludge	8
4.2	Plutonium Recovery from ORNL Metal Waste - Laboratory Study	9
4.3	Plutonium Recovery from ORNL Metal Waste - Pilot Plant Study	11
4.4	Uranium Product Analysis	11
4.5	ORNL Metal Waste Analysis	11
4.6	Nitric Acid Recovery Evaporator	11
4.7	Pumping Test on the ORNL Metal Waste Sludge	12
4.8	Sludge Concentrator Unit Development	14
4.9	Sludge Concentration by Centrifugation	14
5.0	Purex Process	15
5.1	Solvent Treatment	15
5.2	Evaluation of Varsol Type Diluents	15
5.3	Effect of Feed Pretreatment on High TBP Concentration Purex Runs	15

[redacted]

Table of Contents (continued)

		Page Number
	5.4 Effect of Increased TBP Concentration on the Process Volumetric Efficiency	20
	5.5 HETS Study	21
6.0	RaLa Process	23
	6.1 Coating Removal	23
	6.2 Barium Sulfate Precipitation Without Carrier	23
	6.3 One-Column Versene Process	23
	Iron Decontamination	23
	Color Removal from Product	24
	Sodium Acetate as a Feed Buffer	24
7.0	25 Process - Product Concentration	25
8.0	23 Process - Pilot Plant Development	25
9.0	Thorex Process - Use of Carbon Tetrachloride as the Solvent Diluent	26
10.0	Fluoride Process Study	26
11.0	Homogeneous Reactor Slurry Studies	27
	11.1 Stability of Bentonite	27
	11.2 Uranyl Phosphate Slurry	28
12.0	Solvent Extraction - Unit Operations Study	28
13.0	Special Equipment Tests	29
	13.1 Fulton Sylphon Packless Needle Valve	29
	13.2 Eco Gearless Pump	29
	13.3 Protective Coatings - G. E. Cocoon Test	30
14.0	Reports and Memos Issued Since January 1, 1950	31

1.0 Abstract

The progress of chemical technology studies is reported for: the TBP Process, the Purex Process, the RaLa Process, the 25 Process, the 23 Process, the Thorex Process, the Fluoride Process, slurry study for a homogeneous nuclear reactor, solvent extraction, and special equipment tests.

2.0 Introduction

The Laboratory Section continued the development of three solvent extraction processes employing tributyl phosphate as the solvent: the TBP Process for uranium recovery from the Hanford metal waste, the Purex Process for the recovery of uranium and plutonium from the Hanford reactor metal, and Thorex Process for the recovery of thorium from the 23 Process waste. The other major laboratory studies concerned an ion exchange process for barium purification, a fluorination study for uranium, plutonium and fission product separation, and a colloid study aimed at the preparation of a stable uranium slurry that would be useful in a homogeneous nuclear reactor.

The Unit Operations Section continued two pilot plant type studies: the TBP Process studies using Hanford and ORNL metal waste, and process improvement studies of the RaLa Process. Unit operations studies were in progress on continuous countercurrent solvent extraction, filtration, evaporation and evaluation of special equipment. A study of a falling-film type evaporator was started.

The Pilot Plant Section concluded the recovery of the plutonium from the Redox 1BP solution, and the equipment revisions for the CR job were in progress. The operation of the ORNL Radiochemical Waste Evaporator is being evaluated to establish a method for the improvement of its operation.

Introduction (continued)

The Design Section was concerned with problems on liquid and gaseous radioactive waste disposal, equipment design for the RaLa Process, and preliminary design studies for the MTR chemical plant.

3.0 Summary

TBP Process

A 99 percent plutonium recovery from ORNL metal waste was demonstrated in the Pilot Plant. For this recovery it was necessary to convert the plutonium from polymerized and VI state to the IV state.

Purex Process

(1) Washing the mixed solvent with sodium hydroxide solution increased the beta decontamination by a factor of four to ten.

(2) The uranium decontamination and yield was not affected by increasing the tributyl phosphate concentration from 15 to 30 percent.

(3) The uranium feed concentration was increased from 130 to 320 grams per liter by increasing the tributyl phosphate concentration from 15 to 30 percent.

RaLa Process

The silica content of uranium metal was found to be 0.005 to 0.01% which is high enough to cause rapid plugging of the filter medium proposed for process use. Pre-filtration will be required.

23 Process

The pilot plant program is nearing completion. Approximately 300 grams of Uranium-233 with 98.5% isotopic purity was separated.

4.0 TBP Process

4.1 Dissolution of Hanford Metal Waste Sludge

A sample of the Hanford metal waste from Tank 101 U was successfully dissolved by the addition of 70 percent nitric acid. The sludge sample as received from Hanford contained three distinct phases: supernate, soft sludge and hard sludge (see Table 4.1).

Approximately 650 milliliters of 70 percent nitric acid was required to dissolve 1 liter of sludge which gave the following solution composition: 2.5 molar nitric acid and 123 grams of uranium per liter.

The slow addition of nitric acid, for example, 7 milliliters of acid per liter of sludge per minute, prevented foaming that could have been caused by fast evolution of carbon dioxide. More rapid acid addition was possible after the carbon dioxide ceased to come off.

4.2 Plutonium Recovery from the ORNL Metal Waste - Laboratory Study

The plutonium in the ORNL metal waste was not completely extractable when the waste was dissolved in nitric acid to the 3.5 molar free acid concentration required for uranium recovery. This was attributed to the probable presence of plutonium in the VI state or as a polymer. By a treatment of the feed with a combination acid digestion, reduction and oxidation, 98 to 99 percent plutonium recovery from the ORNL metal waste was demonstrated in countercurrent batch experiments. The feed treatment used was as follows:

1. Plutonium depolymerization; digest from three to four hours in 4.5 to 5 molar nitric acid at 60° centigrade.
2. Plutonium valence adjustment; add 0.01 molar ferrous sulfamate and digest 15 minutes at room temperature to reduce the plutonium to the III state, then add 0.01 molar sodium nitrate and digest for one hour at 60° centigrade to oxidize the plutonium to the desired IV state.

It was found necessary to carry out the extraction at 4.5 to 5 molar nitric acid in order to recover more than 95 percent of the plutonium. The presence of some plutonium VI, which has a lower distribution coefficient than plutonium IV, required the increased acid concentration.

Table 4.1

Hanford Metal Waste Sludge Sample #1

Tank 101 U

Constituent	Sp G	U Concentration
Hard Sludge	2.87	359 mg U/g sludge
Soft Sludge	1.86 at 25°C	200 mg U/g sludge
Supernate	1.132 at 30°C	19.9 mg U/ml

[redacted]

4.3 Plutonium Recovery from ORNL Metal Waste - Pilot Plant Study

The operation of the TBP Process over the extraction and scrub column demonstrated plutonium and uranium losses of 1% and 0.001%, respectively. The feeds for both runs were given a ferrous sulfamate treatment as described in the preceding section. These feeds contained were approximately 100 g U/l and 15% TBP was used as the solvent. The complete results of these runs will be reported later.

4.4 Uranium Product Analysis

The product of the TBP pilot plant runs, using Hanford supernate as the feed material, has been analyzed to determine the concentration of other ions present with the uranium after the final evaporation concentration step (see Table 4.4).

4.5 ORNL Metal Waste Analysis

The contents of the metal waste tank at ORNL have been analyzed and these results, when checked with the SF accountability records, agree within approximately 10% with the analytical figures.

4.6 Nitric Acid Recovery Evaporator

An evaporator has been set up to evaporate the extraction wastes of the TBP Process pilot plant to demonstrate the concentration of the

Nitric Acid Recovery Evaporator (continued)

wastes and the recovery of the nitric acid from them. Chemical tests of the evaporator are now in progress. The "hot" runs are scheduled to be started in April.

4.7 Pumping Test on the ORNL Metal Waste Sludge

Another metal waste sludge pumping test was completed during this period using the centrifugal sump pump with a ten foot length of hose attached to the intake. The pump worked satisfactorily pumping out the material which analyzed 145 grams of uranium per liter after being dissolved to 2.8 normal nitric acid. This further indicates that the proposed "sweep" method for pumping the sludge from the waste tank will probably be satisfactory.

Table 4.4

TBP Process Product Analysis

Feed: Hanford supernatant from Tank 103 U

Uranium	410 g/l (115 liters, 47 kg.U)
HNO_3	0.63 N
$PO_4^{\equiv}$	<0.005 g/g U
$SO_4^{=}$	<0.00003 g/g U
*β activity	25 counts/min/mg U (including UX_1-UX_2)
*γ activity	<0.005 mv/mg U (including UX_1-UX_2)
Pu α activity	1.4 counts/min/mg U
Ca	<0.00004 g/g U
Cu	"
Fe	"
Cr	"
Mg	"

* Age - 10 days from last separation

4.8 Sludge Concentrator Unit Development

A unique operation has been proposed for concentrating the sludge from the ORNL metal waste, dissolving it and then clarifying it in a single piece of equipment. Preliminary tests using a star sintered stainless steel filter medium of "G" porosity mounted in the bottom of a 2' x 6" Pyrex pipe has indicated satisfactory results. These tests were observed by a representative of the Niagara Filter Company, who has an applicable commercial unit. A small-scale model of the Niagara Unit was ordered.

A filter aid, Celite 545, was used in these filtration runs to yield higher filtration rates. Tests are to be made to determine the feasibility of re-using this filter aid.

4.9 Sludge Concentration by Centrifugation

Recent discussion with Sharples and DeLaval have indicated that the nozzle type centrifuge would probably not be satisfactory for the concentration of the sludge from the ORNL metal waste. In similar applications, two-stage centrifugation has been successful, using a Bird type continuous centrifuge preceding the nozzle type machine. No further work on this phase of the problem is anticipated.

5.0 Purex Process

5.1 Solvent Treatment

Washing the mixed TBP solvent with 0.1 molar sodium hydroxide was found to be effective in decreasing the beta distribution coefficient by a factor of 4 to 10 by removing butyl acid phosphates. Also the use of butanol-free TBP was demonstrated to be effective. The sulfonation of the Varsol was not effective in decreasing the beta distribution coefficients; however, preliminary results indicate that the chemical stability of the Varsol is markedly increased (see Table 5.1).

5.2 Evaluation of Varsol Type Diluents

A number of Varsol type petroleum fractions were evaluated for use as a TBP diluent. No definite correlation was established with respect to solvent constituents or physical characteristics on decontamination. However, the products of Esso and American Mineral Spirits Company were generally superior (see Table 5.2).

5.3 The Effect of Feed Pretreatment on High TBP Concentration Purex Runs

The combination of the acetone-nitrite feed treatment for ruthenium decontamination, along with the addition of ferrous sulphate as a reducing agent for plutonium and as a zirconium complexing agent, gave

The Effect of Feed Pretreatment on High TBP Concentration Purex Runs (continued)

the most satisfactory uranium decontamination results when a 25 percent TBP solvent was used. The acetone-nitrite treatment was also effective when followed by either digesting with an additional 1 percent of acetone or by zirconium dioxide scavenging. The use of ferrous sulfate alone was also effective; however, the use of ferrous sulfamate was not effective, probably because the sulfamate prevented the plutonium III oxidation to plutonium IV in the presence of 3 molar nitric acid (see Table 5.3).

Table 5.1

Effect of Solvent Treatment and the "Nitrite-Acetone" Feed Pretreatment on Beta Distribution Coefficients in the Purex Process

Conditions: Feed: A dissolver solution (0.143 M U, 0.3 N HNO_3, 0.04 M Hydroxylamine) was divided into two parts; (a) adjusted to feed containing 0.25 M U, 5.0 N HNO_3, 2×10^7 β/m/ml and (b) adjusted to 0.05 M $NaNO_2$, 1% acetone, digested at 90°C for two hours, and then adjusted 0.25 M U, 5.0 N HNO_3, 2×10^7 β/m/ml

Scrub: 5.0 N HNO_3

Organic: TBP/Gulf BT Varsol = 15/85, with varied pretreatment

Procedure: The feed was extracted with an equal volume of organic; the organic was then scrubbed with four successive equal volume passes of scrub solution.

Run No.	Organic Pretreatment			A Series				B Series			
				Beta D.C. (O/A)			β/m/ml	Beta D.C. (O/A)			β/m/ml
	TBP	Varsol	Mixture	Ext'n.	Sc.2	Sc.4	Org.4	Ext'n.	Sc.2	Sc.4	Org.4
1	Not "butanol-free"	-	-	0.01	1.30	5.50	21,000	0.01	0.72	0.75	4,000
2	Butanol-free	-	-	0.03	0.70	0.75	12,000	0.02	0.50	0.50	6,000
3	Not "butanol-free"	Sulfonated	-	0.02	1.70	3.80	21,000	0.01	1.14	2.65	8,000
4	Butanol-free	Sulfonated	-	0.05	0.56	0.73	12,000	0.02	0.58	0.60	4,700
5	Butanol-free	-	One 0.1M NaOH wash	0.02	0.60	0.45	1,550	0.02	0.54	0.43	900
[illegible]	Butanol-free	Sulfonated	One 0.1M NaOH wash	0.02	0.55	0.85	2,000	0.01	0.56	0.44	1,000

[redacted]

Table 5.2

Relative Effect of Various Varsols on Beta Decontamination in Purex Process

Conditions: Feed: 60 gm U/ml, 3 N HNO_3

Scrub: 3 N HNO_3

Organic: 15/85 = TBP/Varsol, 0.2 N HNO_3 (Varsols varied)

Procedure: Feed was extracted with equal volume of organic; organic was then scrubbed with four successive passes of scrub solution.

Varsol	Flash Point* °F	Kauri Butanol Valve* c.c.	β c/m/ml in Organic After: 2nd Scrub	4th Scrub
Esso 107	114	34.2	14,700	880
Esso 111	114	45.5	6,800	880
Amsco 1 and 9	119	30.0	6,600	890
Amsco 123	143	32.1	5,900	1000
Amsco 119-90	119	37.1	12,600	1300
Amsco 127	110	36.0	7,600	1600
Amsco 125	133	26.0	9,600	2500
Gulf BT	114	31.0	36,000	2800
BFMS (Shell)	119	39.0	12,000	3400
Amsco 122	103	32.4	33,000	5000
Amsco 119-92	114	31.9	51,600	5700
Gulf S.S.	104	38.0	57,000	5900
Dispersol (Shell)	155	-	126,000	7500

*Specified by the manufacturer

Table 5.3

Batch Countercurrent Data: Pu Loss and Beta Decontamination In A-Column with the Use of 25/75 Mixtures of TBP/Varsol

Conditions: Feed: 1.6 M U, 3.0 N HNO_3, $2x10^8$ β, $3.8x10^5$ Pu
Scrub: 4.0 N HNO_3
Organic: 25/75 = TBP/Varsol, equil. with 3.5 N HNO_3
Ratio: F/S/O = 1/1/5 Stages: 4 Ext'n. 6 Scrub Vol. Change: 2.5

Feed Pretreatment:
Run 1 - Reduced with 0.025 M ferrous sulfamate at 0.2 N acid deficient.
Run 2 - Reduced with 0.025 M ferrous sulfate
Run 3 - Nitrite-acetone effect, reduced with 0.025 M ferrous sulfate.
Run 4 - Nitrite-acetone effect, reduced by digestion with additional 1% acetone.
Run 5 - Nitrite-acetone effect, no reduction, scavenged with ZrO_2.

Run No.	1	2	3	4	5
β D.F. (Overall)	$1.2x10^5$	$1.0x10^5$	$2.3x10^5$	$2.8x10^5$	$3.8x10^5$
β D.F. (Ext'n)	$1.5x10^3$	$7.1x10^2$	$1.0x10^4$	$2.5x10^3$	$5.7x10^3$
% U Loss (3 Stages)	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1
% U Saturation (Scrub)	80-90	80-90	80-90	80-90	80-90
% Pu Loss (3 Stages)	125	0.23	0.2	0.05*	52.0
Pu D.C. (Ext'n) O/A	0.08***	9.3*	12.0*	-	1.9 **
Pu D.C. (Scrub) O/A	0.54**	1.6*	1.3*	-	0.47**

* Indicates presence of Pu^{+4}.
** Indicates presence of Pu^{+6}.
*** Indicates presence of Pu^{+3}.

[redacted]

5.4 Effect of Increased TBP Concentration on the Process Volumetric Efficiency

It was desirable to increase the process volumetric efficiency of the Purex Process in order to reduce the size of the extraction columns that would be required for this process. To do this, the concentration of the TBP in the mixed solvent was increased to increase the solvent capacity for uranium. It was found that the specific gravity difference limited the TBP concentration in the Varsol type solvent to 30 percent. Equilibrium data obtained under these conditions indicated that a feed containing 320 grams of uranium per liter could be used with the resulting distribution coefficients as follows: uranium, 5.9; plutonium, 1.02; gross beta, 2.3×10^{-3}. The uranium distribution coefficient for stripping into water was 1.2; however, a 30 percent increase in the distribution coefficient was obtained by increasing the stripping temperature from 25° to 60°C.

Further work will be carried out using carbon tetrachloride as a diluent rather than Varsol. This will give a solvent which at all times will be heavier than the aqueous phase; therefore, increasing uranium concentration will aid rather than hinder the hydraulic operability of the column (see Table 5.4).

5.5 HETS Study

Average HETS values were determined for the Purex Process using 15% TBP as the solvent in a one-inch glass column packed with 1/4" x 1/4" split rings. The average uranium HETS were found for the 1A, 1B, and 1C columns to be 3, 2, and 2 feet, respectively. The average plutonium HETS for the extraction section of the 1A column was found to be 2 feet.

[redacted]

Table 5.4

Plutonium (IV) and Gross Beta Distribution Coefficients as Functions of Percent Uranium Saturation of Solvent

Conditions:

Aqueous Feeds: 3 M HNO_3, β = ca. 10^6 c/m/ml, Pu (IV) α = 10^6 c/m/ml, uranium as indicated.

Solvents: TBP as indicated diluted with Varsol. Equilibration times were 10 minutes using two volumes of solvent to one volume of feed at room temperature.

Percent TBP in Solvent	Uranium Concentration Feed g/l	Distribution Coefficients			Percent U Saturation of Solvent
		Gross Beta	Plutonium	Uranium	
20	218	1.54×10^{-3}	0.79	2.5	71
	238	1.21×10^{-3}	0.63	2.0	77
	258	7.9×10^{-4}	0.53	1.7	79
	278	8.2×10^{-4}	0.42	1.3	81
25	238	2.48×10^{-3}	1.08	6.1	65
	258	1.70×10^{-3}	0.91	5.2	70
	278	1.77×10^{-3}	0.72	4.2	76
	297	1.24×10^{-3}	0.59	3.6	82
30	40	-	2.50	22.4	10
	119	-	1.73	16.7	29
	159	-	1.37	15.7	41
	198	-	1.20	11.8	49
	278	3.98×10^{-3}	1.29	7.7	64
	318	2.32×10^{-3}	1.02	5.9	72
	357	1.75×10^{-3}	0.83	4.0	76
	397	1.18×10^{-3}	0.62	2.9	84

6.0 RaLa Process

6.1 Coating Removal

The development of a chemical procedure for removing the siliceous coating from Hanford slugs was concluded. It was found that silica is present in the slug as well as in the bonding material to an extent of 0.005 to 0.01%; therefore, the removal of the siliceous coating from the slug itself could not be completely effective. Therefore, a pre-filter is required.

6.2 Barium Sulfate Precipitation Without Carrier

Work previously reported indicated that the barium sulfate could be separated from the metal solution in the absence of the lead sulfate carrier by centrifugation. These tests have been made with a 12" Bird Centrifuge which develops a pull of 2200 G with a two-minute hold-up of the solution in the bowl. During the report period, equipment alterations were completed to install a metering pump which would allow 4 minutes hold-up in the centrifuge bowl.

6.3 One Column Versene Process

Iron Decontamination

The iron decontamination of the one-column Versene Process was barely adequate. Therefore, Fe(3) Specific (N-di-ethanol-glycine), an iron complexing agent, was added to the feed. The use of this complex

One Column Versene Process (continued)

allowed the iron to go through the column without being adsorbed. Preliminary results using 0.10 molar complexing agent have shown a 15 percent barium loss, the use of a 0.01 molar solution is to be studied.

Color Removal from Product

Preliminary studies were made to determine an adequate method for removal of the color that might result in the final product from ion exchange processes. Adequate decolorization was obtained using 30% peroxide solution or aqua regia. However, these processes were not desirable, as the peroxide causes the precipitation of insoluble barium oxide, while aqua regia presented a corrosion problem. Other techniques are to be considered.

Sodium Acetate as a Feed Buffer

To eliminate the necessity of determining the pH of the "hot" feed for the Versene process, sodium acetate was considered as a feed buffer agent. Test runs using Versene feed 0.5 molar in sodium acetate showed that the column size would have to be doubled in order to achieve adequate barium yield. Therefore, this does not appear to be a practical solution to the problem.

7.0 25 Process - Product Concentration

Approximately 2 kilograms of Uranium^{235} from pilot plant testing of the 25 Process have been purified and concentrated in seven runs with a loss of approximately 0.0012%. The process being used to do this work is a batch solvent extraction process, and it is planned in the future to develop such a process for purification of both fissionable uranium and plutonium.

8.0 23 Process - Pilot Plant Development

The 23 pilot plant program is essentially complete. One additional run using an alternate solvent, diisopropyl ether, will be made, and about three cleanup runs will be required to recover 23 held up in the system and to decontaminate the equipment. It is estimated that 300 grams of 98.5% isotopic pure Uranium^{233} will have been recovered by the end of the program.

The results of twelve previous runs with methyl isobutyl ketone solvent demonstrated that the one-cycle solvent extraction process yielded a fission product decontamination factor of 2×10^4, with a 23 loss of less than 0.2%.

The most significant results of Runs 13T-18T were as follows:

(1) Gross decontamination was decreased to 1×10^3 when the metal feed acidity was increased from 0.2 N acid deficient to 0.05 acid deficient; decontamination was not significantly increased above 2×10^4 when the metal feed acidity was decreased from 0.2 N acid deficient to 0.34 N acid deficient.

23 Process - Pilot Plant Development (continued)

(2) Decontamination was not affected when untreated solvent, methyl isobutyl ketone, was made 0.5% in methyl isobutyl carbinol.

(3) Except for the inextractable 23 loss (0.04%), no extraction loss could be detected when only 9.5 ft of extraction section was used at 50% of flooding. A 23 loss of 0.06% (excluding inextractable 23) was observed using the same extraction length at 15% of flooding.

9.0 Thorex Process - Use of Carbon Tetrachloride as the Solvent Diluent

A study was started on the use of carbon tetrachloride as the diluent for tributyl phosphate in the Thorex Process in order to increase the hydraulic operability of the countercurrent extraction column. A 30 percent TBP - 58 percent carbon tetrachloride - 12 percent nitric acid solvent was found to be optimum. Countercurrent batch extraction with this solvent showed about 0.01% thorium extraction loss. The stripping loss was of the same order of magnitude. Decontamination data is not available at present.

10.0 Fluoride Process Study

The fluorination experiment, the reaction of elemental fluorine with irradiated uranium metal, continued to show at least 10^3 overall uranium decontamination with an approximate 0.2 percent uranium loss. The specific

Fluoride Process Study (continued)

activity of the product uranium was less than 30% that of the background of natural uranium. The separation of the plutonium has continued to be the major problem. During this period, greater hold-up of the plutonium in the reactor and in the filter than was previously experienced was unexpected and will be investigated.

11.0 Homogeneous Reactor Slurry Studies

11.1 Stability of Bentonite

Preceding work has shown that Bentonite stabilized uranium oxide slurries are stable when heated for one hour at 250°C. However, this slurry was not stable after heating for about 72 hours. This instability of the slurry appeared to be associated with the presence of iron liberated from the Bentonite during the course of heating, and a test has been completed which shows that bentonite slurried in water and heated 72 hours at 250°C remains stable as verified by X-ray analysis.

Iron was eluted from Bentonite with 6 molar hydrochloric acid. Several samples of Bentonite have been treated in this way, but as yet none of these samples have been used for slurry. It is also planned to utilize complex phosphates such as Calgon and sodium pyrophosphate to complex liberated iron.

These studies are to be continued.

11.2 Uranyl Phosphate Slurry

Uranyl phosphate is a relatively insoluble uranium salt which may be used in a homogeneous reactor slurry. Uranyl phosphate was precipitated from an acetate solution, and made up into a slurry containing 40 grams of uranium, 22.6 grams of bentonite, and 1000 milliliters of water. This resulting slurry has a very high viscosity and was stable at room temperature. After being heated for the total of 100 hours at 250°C in a sealed quartz tube, there was no visual evidence of the separation of the solids and the viscosity of the slurry had further increased. This program is to be continued.

12.0 Solvent Extraction - Unit Operations Studies

The solvent extraction data obtained from the TBP Process pilot plant runs was studied analytically and indicated the following facts:

(1) that the rate of diffusion of the uranium controlled the extraction section behavior, not the rate of reaction or solvation;

(2) the HTU values are a function of the operating variables, and do not remain constant for a given system. The basic data was not well suited for this type of study because of the great length of the column, the insufficient accuracy of the analyses, the uncertain equilibrium data, and the uncertainties connected with the rate of flows.

Solvent Extraction - Unit Operations Studies (continued)

(3) It is proposed to continue this study using shorter columns for the experimental runs. Future correlations shall be based on the HTU values rather than those of the HETS because the HTU is derived from the fundamental process of diffusion behavior, and is, therefore, more readily susceptible to theoretical analysis of correlation methods.

13.0 Special Equipment Tests

13.1 Fulton-Sylphon Packless Needle Valve (Cat. #314, 1/8" STP, 3347 SS)

A Fulton-Sylphon packless needle valve was tested against water at 50 lbs/sq. in. gauge for 500 cycles and no leakage occured. The maximum and minimum continuous flow at this pressure was 1735 ml/min and 63 ml/min, respectively. Further tests of this valve are planned; however, this valve is recommended for use in aqueous systems.

13.2 Eco Gearless Pump, Model PP-1M 316, 3/8" SPT

The Eco pump, having a gearless dual eccentric system, failed under tests for pumping a simulated sludge solution containing filter aid. Normally the pump pistons are supplied of nylon, but for this application special Teflon pistons were made. The pump was satisfactory when used with water pumping 3 gallons per minute against a 15 pounds per square inch head. However, when an aqueous slurry containing 5 grams per liter of Celite 545 filter aid was pumped under the same conditions, leakage owing to bearing

Eco Gearless Pump, Model PP-1M 316, 3/8" SPT (continued)

failure occurred after approximately 10 minutes of operation.

13.3 Protective Coatings - G. E. Cocoon Test

The effectiveness of strippable coatings to prevent contamination bleed-through was demonstrated recently in the Unit Operations Building, 706-HB.

A section of the cell wall was accidentally contaminated after the wall had been coated with Cocoon for a period of approximately 6 months. During this period the wall had been contaminated several times from leaks. The Cocoon was stripped from the wall, and quite complete decontamination was effected as evidenced by smear tests. This confirms previous data as determined from laboratory tests, and justifies using Cocoon in areas liable to contamination. More recent microscopic examination of various types of paints applied to floors indicates that in all instances, strippable coatings are less porous than non-strippable chemically resistant paints.

14.0 Reports and Memos Issued by Chemical Technology Division During the Period 1/4/50 - 4/10/50

TBP Process

Report No.	Author	Date Issued	Title
50-1-28	HEGoeller	1/5/50	Trip to Hanford to Sample Hanford Metal Waste Sludge.
50-1-41	ECStewart	1/11/50	Cost Survey for Proposed TBP Metal Recovery Plant.
ORNL-557	TCRunion, CVEllison	1/18/50	TBP Process for Uranium Recovery from Metal Waste - Laboratory Summary.
50-1-87	ACJealous	1/19/50	Recovery of Waste Uranium from ORNL Tank Farm.
ORNL-543	RPMilford	1/27/50	TBP Process Pilot Plant Design Report.
50-1-167	ACJealous	1/31/50	Semi-Works TBP Pilot Plant Program.
50-3-77	ACJealous	3/16/50	UNH from TBP Process.
50-3-142	ECStewart	3/27/50	Revision to Cost Estimate for ORNL Metal Recovery Plant.
50-4-6	ACJealous	4/5/50	Results of the First "Total Waste" Run in the ORNL TBP Pilot Plant.

Purex Process

50-1-39	DGReid	1/11/50	Capital Addition Expenditures for Purex Development Cost Center 3435( )-39.
50-2-112	FRBruce	2/9/50	Conference on Plutonium Chemistry in the Purex Process at ORNL, January 18, 1950.

Reports and Memos Issued by Chemical Technology Division During the Period 1/4/50 - 4/10/50 (continued)

RaLa Process

Report No.	Author	Date Issued	Title
50-1-65	WKEister	1/16/50	RaLa Meeting with Hanford Personnel at ORNL.
50-1-66	WKEister	1/16/50	RaLa Development Run Meeting.
50-2-3	FLCuller	2/1/50	Conference on RaLa Progress.
Dwg. TD-1550	WEUnger	1/23/50	RaLa Modification Project Process Summary.
50-2-93	REBlanco	2/16/50	Processes Used at Los Alamos for the Separation of Lathanum and Barium; Specifications for Oak Ridge RaLa Product.
50-3-85	WEUnger	3/16/50	Letter Re: Preliminary Process Summary TD-1550.
ORNL-623	REBlanco, IRHiggins, WEUnger	3/22/50	Preliminary Demonstration of RaLa Ion Exchange Processes at the Kilocurie Level.

25 Process (Arco Chemical Plant)

50-1-136	FLSteahly, FLCuller	1/27/50	Process for Recovery and Decontamination of U-235 Reactor Fuel from Fission Products.
50-2-67	FLCuller	2/14/50	Letter Re: Comments of Processing Advisory Committee Meeting, January 11, 1950.

Reports and Memos Issued by Chemical Technology Division During the Period 1/4/50 - 4/10/50 (continued)

25 Process (Arco Chemical Plant) (continued)

Report No.	Author	Date Issued	Title
50-3-30	WKEister	3/9/50	Proposed Construction for MTR Chemical Processing Area at Arco.
ORNL-611	HEGoeller	3/15/50	The Cooling of Irradiated 25 MTR Fuel Assemblies and its Effect on 25 Inventory in the MTR Projcet.
ORNL-612	HEGoeller	3/15/50	The Production of Transuranic Isotopes in the MTR.
ORNL-343	FRBruce	3/21/50	The Laboratory Development of a Continuous Solvent Extraction Process for the Recovery of U-235.

23 Process

ORNL-564	DEFerguson, RJKlotzbach	2/15/50	Orientation Manual for 23 Pilot Plant.
50-3-52	RJKlotzbach	3/3/50	Initial Equipment Modifications and Additions for the 23 Pilot Plant.
ORNL-651	DEFerguson, HKJackson, ELNicholson	3/27/50	Pilot Plants Section Status Report on the 23 Pilot Plant Program.

Reports and Memos Issued by Chemical Technology Division During the Period 1/4/50 - 4/10/50 (continued)

CR Job

Report No.	Author	Date Issued	Title
50-2-111	DGReid	2/20/50	CR Separations Program for the ORNL-205 Pilot Plant.

Waste Processing

50-3-61	EMShank	3/9/50	Proposed Resin Column Test Installation on the ORNL Radiochemical Waste Evaporator Condensate.
50-3-179	FLCuller	2/30/50	Review of Resin Column Installation.

Homogeneous Reactor Chemical Process

ORNL-648	WKEister	3/27/50	Recovery and Decontamination of U-235 and Heavy Water from a Homogeneous Pile Slurry.

Redox Process

ORNL-598	WBLanham	4/7/50	Laboratory Development of an Acid Deficient Redox Process for the Recovery of Uranium and Plutonium.

Reports and Memos Issued by Chemical Technology Division During the Period 1/4/50 - 4/10/50 (continued)

General Progress Reports

Report No.	Author	Date Issued	Title
ORNL-530	WKEister	1/13/50	Progress Report for Quarter Ending 11/30/49.
50-1-127	DGReid	1/20/50	Pilot Plants Section Report for December, 1949.
ORNL-580	WKEister	2/9/50	Progress Report for Month Ending December 31, 1949.
ORNL-592	DGReid	2/13/50	Pilot Plants Section Report for December, 1949.
50-2-66	DGReid	2/20/50	Chemical Technology Division - Pilot Plants Section Report for January, 1950.
ORNL-640	WKEister	3/22/50	Progress Report for December, 1949.
50-3-76	DGReid	3/23/50	Pilot Plants Section Report for February, 1950.

Miscellaneous

50-1-31	WKEister	1/16/50	Re: Drawings of Three One-Slug Dissolvers.
50-2-109	WKEister	2/22/50	Headings for the TBP and the Thorex Process.