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ORNL 936
Progress Report
Chemistry-Separation
Processes for Plutonium
and Uranium 8a

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CHEMICAL TECHNOLOGY DIVISION

F. L. Steahly, Division Director
F. R. Bruce, Laboratory Chief
J. O. Davis, Unit Operations Chief
D. G. Reid, Pilot Plant Chief
F. L. Culler, Process Design Chief

QUARTERLY PROGRESS REPORT

for Period Ending November 20, 1950

Edited by:

W. K. Eister

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ABSTRACT

The progress of the Chemical Technology studies for the quarter ending November 20, 1950 is reported on the following problems: TBP metal recovery process, Purex process, RaLa Process, Idaho Chemical Processing Plant, uranium extraction from ore, volatility process, radioactive waste, electrolytic processes, homogeneous reactor chemistry, Chalk River work, and equipment decontamination by electrostripping.

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INTRODUCTION

The major problems in the Chemical Technology Division were: the development of the Purex process for the recovery of uranium and plutonium from pile irradiated metal; the design of the Idaho Chemical Processing Plant; the development of the modified RaLa process for the separation of radioactive barium; and the construction of the TBP process building for the recovery of uranium from the ORNL metal waste. A study was initiated to evaluate alternate solvents for the extraction of uranium from ore.

The major problem in the Laboratory Section continued to be the Purex process, with the major studies concerned with the solvent extraction phase. In the RaLa development program, alternate procedures for barium extraction from the MTR fuel unit were evaluated, with work continuing on the modified ORNL process. The application of adsorption for the separation of plutonium from uranium was emphasized in the volatility process study. Studies of the preparation of a stable uranium slurry to be used as a fuel in the homogeneous type nuclear reactor were continued. Four new problems were initiated: (1) the development of the decontamination process for the uranyl sulfate solution now proposed for the Homogeneous Reactor Experiment; (2) the application of electrochemical methods to metal dissolution and to uranium, plutonium, and fission product separation processes; (3) the evaluation of alternate solvents for the extraction of uranium from ore; and (4) the determination of optimum chemical conditions for the dissolution of uranium-aluminum alloy slugs.

Work was begun by the Unit Operations Section on the dissolution of the metal to be processed in the Idaho Chemical Processing Plant, and will be continued in the attempt to eliminate the discharge of the radioactive fumes to the atmosphere. This work will also be applicable to the Purex program. Other work applicable at Idaho has included the development of the product evaporator and the evaluation of between-cycle pumps. Work is in progress to improve the operability of the product filter for the ORNL RaLa plant.

The construction program continued to be the major effort in the Pilot Plant Section. The modification of the 205 Pilot Plant Building was essentially complete, and modification of the equipment for the Purex process development was sufficiently advanced to start calibrating and testing the equipment.

The building for the ORNL Metal Recovery Plant has been completed, and the installation of the process equipment is in progress.

The design of the Idaho Chemical Processing Plant continued to be the major problem in the Design Section. Personnel of the architect-engineer, the Foster Wheeler Corporation, are working at the Laboratory on the design of the plant. Personnel of the construction contractor, the Bechtel Corporation, have also been located at the Laboratory to facilitate the purchase of the materials required in the construction of this plant. The design of the RaLa plant modification has continued while the responsibility for the erection of the ORNL metal recovery equipment has been shifted to the Pilot Plant Section.

SUMMARY

TBP Process for Waste Metal Recovery. Laboratory-scale investigations have shown that hydrofluoric acid is a satisfactory reagent for the partitioning of plutonium from uranium. The extent of corrosion that may be encountered is now being investigated.

Purex Process. 1. Satisfactory separation of the plutonium from the uranium in a countercurrent packed column was demonstrated.

2. The monobutyl and dibutyl phosphate in the solvent were found to reduce gross fission product decontamination in column IA by a factor of 7 with no significant effect on the plutonium or uranium losses.

3. A study of the distribution coefficient in column IA showed that the reduction of the acid concentration in the extraction section would reduce the zirconium contamination of the product. However, it was found also that the scrub was effective in removing the zirconium from the product stream. Ruthenium appeared to offer the most difficult decontamination problem.

RaLa Process. A preliminary laboratory evaluation of alternate methods for the extraction of barium from irradiated MTR fuel units indicated that the dissolution of the fuel unit in sulfuric acid gave the most satisfactory results from both the chemical and equipment standpoint.

Idaho Chemical Processing Plant Design. The design of the chemical plant for the Reactor Test Station progressed sufficiently so that construction was started and orders were placed for the stainless steel, the process equipment, the instruments, the piping, and the valves.

Volatility Process Study. A study was initated on the electrolytic fluorination of uranium metal.

Uranium Extraction from Ore. Laboratory study indicated that tributyl phosphate was the most outstanding alternate solvent to replace the diethyl ether now used in the extraction of uranium from ore.

1. TBP METAL RECOVERY PROCESS

Plutonium Stripping Study. In the Metal Recovery Program at ORNL it is proposed to recover the plutonium as well as the uranium by a modified Purex process. To accomplish this purpose a single-cycle three-column process is proposed with the uranium and plutonium decontaminated in the first column, the plutonium separated from the uranium in the second column, and the uranium stripped from the solvent in the third column. The successful use of ammonium fluosilicate for stripping the plutonium from the uranium in the second column was previously reported. In more recent work the effectiveness of hydrofluoric acid and sodium fluoride was compared with the ammonium fluosilicate as the plutonium stripping agent (see Table 1.1). On the basis of this study hydrofluoric acid was recommended as the stripping agent (see Fig. 1.1).

TABLE 1.1

Comparison of Uranium-Plutonium Partitioning Agents

BF: 26.5 g/l
0.25 *N* HNO_3
1.41×10^4 Pu α c/m/ml
15% TBP/Gulf BT

BS: 15% TBP/Gulf BT
Flow rates: BF:BS:BX = 6.25:1.04:1.125, 5 scrub, 6 strip stages in batch countercurrent equipment

PLUTONIUM PARTITION SOLUTION		URANIUM PRODUCT	PLUTONIUM PRODUCT
AGENT	HNO_3 (Molarity)	(Pu α c/m/mg U)	(mg U/mg Pu)
0.006 *M* $(NH_4)_2SiF_6$	0.75	1.65	2.18
0.005 *M* NaF	0.75	1.8	0.412 - 2.0
0.005 *M* HF	0.75	4.06	4.69
0.005 *M* HF	0	0.6	0.1

It was demonstrated in countercurrent batch tests that 1 volume of 0.005 *M* HF stripped the plutonium from 10 volumes of the organic stream. In six stripping stages, the plutonium concentration was reduced to less than 1 alpha disintegration per milligram of uranium in the organic stream. It was also found that the addition of nitric acid to the strip feed was not required

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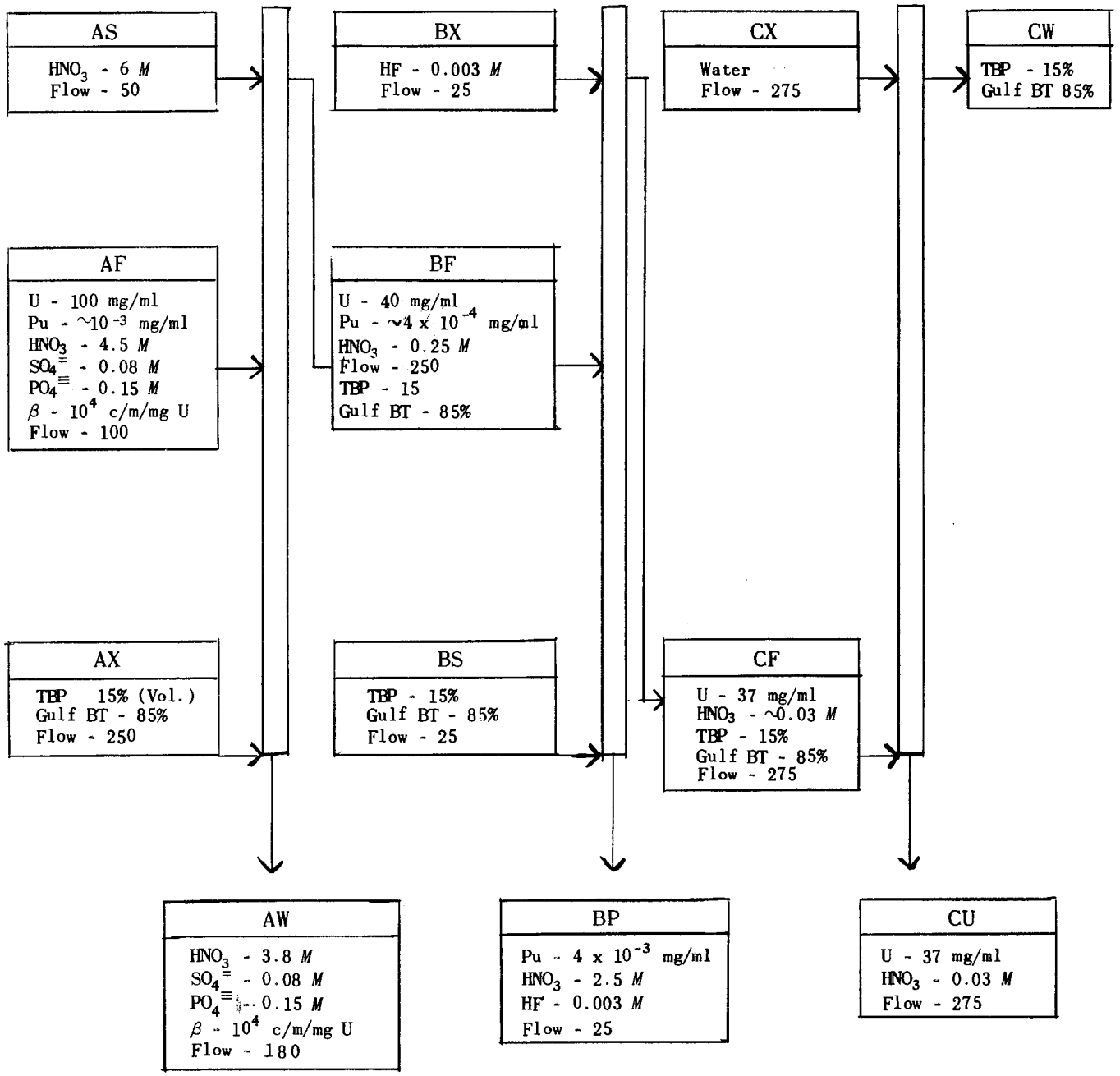
FIGURE 1.1

TBP Metal Recovery Process

ORNL Metal Recovery Plant Flow Sheet

November 20, 1950



because the organic stream from the extraction column contained about 0.25 *M* nitric acid which was almost quantitatively stripped into the aqueous phase along with the plutonium.

ORNL Metal Recovery Plant: Design and Construction. The construction of the building for the ORNL Metal Recovery Process equipment has been completed, and purchase orders have been placed for 75% of the equipment required for the building. The engineering design of the process is 85% complete. The major problem remaining is the design of the mechanism for the removal of the metal waste from the storage tank.

2. PUREX PROCESS

Study of the IB Column Operation. The satisfactory operation of the first-cycle plutonium partitioning column, column IB, was demonstrated using a stainless steel column, 1-1/2 in. in diameter, with a 25-ft plutonium stripping section and a 16-ft scrub section. The column was packed with 1/4- by 3/8-in. stainless steel rings. The plutonium loss was 0.1% with only 0.0005% of the uranium accompanying the plutonium. For this study, because of equipment limitations, it was necessary to prepare the feed for the IB column in the laboratory extraction column and then transfer the feed to the larger pilot plant column. During the first run made in the large IB column, the plutonium loss was satisfactorily low, 0.04%; however, 1 to 2% of the uranium was lost to the plutonium product stream. Analysis of the data indicated that the nitric acid build-up in the scrub section stream was quite slow, which suggested this as the reason for the large uranium loss. In the second run the column was brought to equilibrium with respect to nitric acid prior to the start-up. Column IB was then operated under flowsheet conditions, with the results indicated.

A run was made in the small laboratory columns using 0.01 *M* oxalic acid instead of 0.035 *M* ferrous sulfamate as the plutonium partitioning agent, but there was no significant difference in the plutonium and uranium results. This partitioning column had a 13-ft strip section and a 2-ft scrub section.

Effect of Monobutyl and Dibutyl Phosphate on the Uranium, Plutonium, and Fission Product Distribution. It was found that when 0.1% by volume of monobutyl phosphate or dibutyl phosphate was added to the 30% tributyl phosphate—70% Gulf BT solvent, the fission product decontamination factor in the first-cycle extraction column was decreased by a factor of 6; however, there was no significant effect on the uranium or plutonium losses in the extraction or partitioning column.

Fission Product Distribution in the First-cycle Scrub. The distribution coefficients of the individual fission products were determined under conditions to be found in the scrub section of column IA; analysis of these data

indicated that ruthenium is the principal fission product following uranium and plutonium (see Table 2.1). The separation of zirconium and the total rare earths under the scrub conditions should be much better than for ruthenium. Therefore, for the separation of ruthenium, the important factors will be the degree of uranium saturation and the concentration of nitric acid at the feed plate.

TABLE 2.1

Distribution Coefficients of Fission Products in Scrub Section of Column IA

Conditions IA Feed - 310 g of Uranium per liter; 2 *M* HNO_3; 3.34×10^6 β c/min/ml
IA Scrub - 3 *M* HNO_3
IA Extractant - 30% TBP—70% Gulf BT
Flow Ratio F:S:X = 3:2:10
Stages - 5 extraction, 9 scrub
Countercurrent Batch Extraction, 5 volume changes
Equilibration Time - 3 min

STAGES	GROSS β	TOTAL RARE EARTHS	Ru	Zr
DISTRIBUTION COEFFICIENT (O/A)				
S-8	0.19			0.11
S-6	0.17	0.16		0.09
S-4	0.12	0.10	0.16	0.15
S-2	0.10	0.02	0.15	0.03
E-1	0.026	0.002	0.008	0.06
E-3	0.059	0.03	0.08	0.33
E-5	0.063	0.03	0.14	0.06
DECONTAMINATION FACTOR (Based on U)				
S-9	1.12×10^3	4.37×10^4	170	780
S-8	620	4.37×10^4	131	760
S-6	237	2.2×10^4	125	430
S-4	152	1.2×10^4		134
S-2	51	4.3×10^3	97	63

Reflux of Nitric Acid and Fission Products in the Extraction Section of Column IA. The tributyl phosphate—Gulf BT solvent entering the bottom of column IA extracts the nitric acid from the aqueous phase. However, higher up in the column near the feed plate, where the solvent is more saturated with uranium, the nitric acid is re-extracted into the aqueous phase, thus causing a reflux of nitric acid in the extraction section. This causes a build-up of the nitric acid concentration in the extraction section to about 4 *M* when there is 2 *M* nitric acid in the feed and 3 *M* nitric acid in the scrub. The effect of this high nitric acid concentration in the lower stages of the extraction section on plutonium and the individual fission products was evaluated, and it was thought that the zirconium, whose distribution coefficient was approximately the same as that of the nitric acid, would reflux similarly to the nitric acid. The evaluation of zirconium data from batch countercurrent runs indicates that this was the case (see Table 2.2).

TABLE 2.2

Distribution Coefficients of Fission Products in Extraction Section of Column IA

Conditions: IA Feed - 310 g of Uranium per liter; 3.5, 2, and 1.5 *M* HNO_3; 4 x 10^6 β c/min/ml
IA Scrub - 4.5, 3, and 2 *M* HNO_3
IA Extractant - 30% TBP—70% Gulf BT
Flow Ratio F:S:X = 3:2:10
Stages - 5 extraction, 9 scrub
Countercurrent Batch Extraction - 5 volume changes
Equilibration Time - 3 min

FISSION PRODUCT	HNO_3 CONCENTRATION* (*M*)		DISTRIBUTION COEFFICIENT (O/A)		
	AF	AS	STAGE 1*	STAGE 3*	STAGE 5*
Gross β	3.5	4.5	0.11	0.19	0.03
	2.0	3.0	0.026	0.059	0.063
	1.5	2.0	0.0032	0.052	0.023
Total Rare Earths	3.5	4.5	0.0013	0.024	0.026
	2.0	3.0	0.002	0.03	0.03
	1.5	2.0	0.0009	0.024	0.024
Ruthenium	3.5	4.5	0.004	0.16	0.071
	2.0	3.0	0.008	0.08	0.14
	1.5	2.0	0.0097	0.13	0.28
Zirconium	3.5	4.5	0.15	0.94	0.024
	2.0	3.0	0.06	0.33	0.06
	1.5	2.0	0.0035	0.089	0.036

* Extraction stage.

The refluxing of zirconium can be significantly reduced by reducing the nitric acid concentration in the feed and scrub to 1.5 and 2.0 *M*, respectively. This will yield a feed-plate nitric acid concentration of 3 *M*, which has been found to give adequate separation of fission products from uranium and plutonium.

Uranium-Plutonium Partitioning Agents. The separation of plutonium from uranium in column IB has been satisfactorily demonstrated using ferrous sulfamate as the partitioning agent. Here the ferrous sulfamate reduces the plutonium and allows it to be stripped from the uranium. A second general method for the separation of plutonium from the uranium was investigated using complexing agents which would make the plutonium more aqueous-soluble. These were oxalic acid, hydrofluoric acid, fluosilicic acid, and phosphoric acid. All these reagents, with the exception of the hydrofluoric acid, caused a precipitate in the aqueous phase after the solution had stood for several hours. Batch countercurrent runs with hydrofluoric acid used as the partitioning agent successfully stripped the plutonium. The zirconium activity also followed the plutonium, leaving only the ruthenium in the uranium stream. The plutonium stream was then evaporated to give a volume 1/5 of the original. Prior to the second plutonium extraction cycle, the solution was made 0.05 *M* in aluminum ion, which had a greater affinity for the fluoride ion than did the plutonium. In this cycle the zirconium was extracted along with the plutonium; however, a ruthenium decontamination factor of 350 was obtained. In future work the concentration of the aluminum will be decreased in the hope of obtaining separation of plutonium from the zirconium.

Plutonium HETS in the IA Column Extraction Section. The height of an equivalent theoretical stage (HETS) for plutonium in the first-cycle extraction section at 20% of flooding was found to be 2.7 ft. This HETS was determined in a 1-in. glass column packed with ¼- by ¼-in. rings with a bed depth of 10 ft. Under these conditions the plutonium loss was 0.09%.

Second Plutonium Cycle Study. A gross beta decontamination factor of 300 was obtained in the second plutonium cycle with a feed approaching the Hanford beta activity level to give an overall plutonium decontamination through two cycles of approximately 10^6. This result was obtained in a batch countercurrent experiment using six scrub stages. The flowsheet was presented in the last quarterly report, ORNL-846, Fig. 3.

The addition of 0.01 *M* sodium fluoride to complex the zirconium reduced the decontamination from 300 to approximately 50 in a batch countercurrent extraction study with five extraction and six scrub stages. To permit the use of fluoride ion it was necessary to oxidize the plutonium to the hexavalent state.

Flooding Rate Study. The flooding rates have now been determined for the first-cycle extraction and scrub to be 610 gal/hr/sq ft and 210 gal/hr/sq ft. Flooding rates for the second-cycle plutonium extraction and strip have been found to be 850 gal/hr/sq ft and 520 gal of total throughput (organic + aqueous) per hour per square foot, respectively. These flooding rates were determined in a 1-in.-diameter glass column packed with ¼- by ¼-in. stainless steel rings, under the process conditions described on the flowsheet found in the last quarterly report (ORNL-846).

Equilibrium Data for a Tributyl Phosphate—Carbon Tetrachloride Purex Flowsheet. The use of carbon tetrachloride instead of a saturated hydrocarbon such as Gulf BT as the diluent for the tributyl phosphate in the Purex process has been suggested by the Chemistry Division and by work carried out at other sites, which indicated improved separation and decreased fire hazard. However, there were three disadvantages of carbon tetrachloride to be evaluated: (1) its toxicity, (2) its radiation decomposition products which could be corrosive to the standard materials of construction, (3) the carbon tetrachloride—tributyl phosphate mixed solvent was heavier than the aqueous phase and its operability in packed or pulsed columns was unknown.

The equilibrium data for the extraction of uranium, plutonium, nitric acid, zirconium, ruthenium, and total rate earths under feed plate conditions were determined as functions of (1) the percent uranium saturation of the solvent, and (2) the percent TBP in the tributyl phosphate—carbon tetrachloride mixed solvent. The uranium saturation varied from 35 to 99% while the solvent composition was varied from 20 to 40% TBP by volume. These data indicate a factor of 2 advantage in fission product separation from plutonium in the carbon tetrachloride system (see Table 2.3).

The specific gravity differential between the carbon tetrachloride—tributyl phosphate solvent and the aqueous phase as the function of the tributyl phosphate content and uranium content was evaluated. The resulting information indicated that for an operable countercurrent solvent extraction system the maximum amount of tributyl phosphate in the carbon tetrachloride diluent would

be 25 to 30%, which is comparable to the 30% tributyl phosphate now used in the hydrocarbon diluent system.

TABLE 2.3

Comparison of Plutonium and Fission Product Distribution Coefficients in 30% TBP—Gulf BT and 30% TBP—CCl_4

Conditions: Batch Extractions Simulating Column IA Feed Plate Conditions
Aqueous - Variable uranium concentration; 3 *M* nitric acid; 1×10^6 Pu α counts/min/ml; 1×10^6 gross β counts/min/ml in Gulf BT system, 1×10^7 in CCl_4 system
Organic - 30% TBP—Gulf BT and 30% TBP—CCl_4
Organic/aqueous Ratio - 2/1
Equilibration Time - 10 min

URANIUM SATURATION OF SOLVENT (%)	DISTRIBUTION COEFFICIENT (O/A)				
	PLUTONIUM	GROSS FISSION PRODUCTS	ZIRCONIUM	RUTHENIUM	TOTAL RARE EARTHS
30% TBP—GULF BT SYSTEM					
37.0	3.7	0.011	0.026	0.039	0.0073
54.9	1.3	0.0050	0.025	0.016	0.0032
61.7	1.6	0.0035	0.020	0.0096	0.0021
72.0	1.1	0.0018	0.012	0.0037	0.0011
77.2	1.0	0.0015	0.013	0.0027	0.00071
82.4	0.79	0.0011	0.0090	0.0016	0.00040
86.8	0.57	0.00067	0.0073	0.00069	0.00016
30% TBP—CCl_4 SYSTEM					
35.4	2.8	0.0061	0.035	0.011	0.0044
52.8	2.1	0.0027	0.015	0.0034	0.0014
59.4	1.8	0.0021	0.018	0.0023	0.0010
71.7	1.2	0.00092	0.010	0.00075	0.00033
75.2	0.78	0.00075	0.0074	0.00046	0.00028
81.6	0.48	0.00047	0.0053	0.00015	0.00002
85.0	0.31	0.00026	0.0044	0.00007	0.00001

Alternate Flowsheet Study. The evaluation of the Purex process at Hanford has resulted in the suggestion that certain process modifications might be

made to reduce significantly the number of solvent extraction cycles required for the separation of uranium, plutonium, and fission products. In the flowsheet presented here the uranium and plutonium are separated in the first cycle from the bulk of the fission products and, subsequently, separated from each other. Then the plutonium is passed through a second cycle for its final separation from the fission products, and the uranium stream goes through a second cycle for final separation from fission products and plutonium. This is similar to the flow of the uranium, plutonium, and fission products in the Redox process. This is now considered as our No. 1 Purex flowsheet.

The No. 2 Purex flowsheet, which has been suggested by the Hanford workers, involves only two cycles of solvent extraction. In the first cycle the uranium and plutonium would be separated from the gross fission products. In the second cycle further separation of the plutonium and uranium from the remaining fission products and their subsequent separation from each other would be accomplished (see Figs. 2.1 and 2.2). Laboratory evaluation of this process in countercurrent batch extraction equipment has indicated satisfactory results, and alterations in the pilot plant equipment have been made to allow the operation of the pilot plant with either the No. 1 or No. 2 flowsheet. This process has three major advantages: (1) the elimination of one cycle of solvent extraction, (2) the elimination of the hydrocarbon scrub column for the removal of the tributyl phosphate prior to the volume reduction of the first-cycle plutonium product stream required for the second plutonium cycle, and (3) the use of the same solvent mixture, 30% tributyl phosphate, throughout the process. The major disadvantage of this process may be the degree of separation of uranium and plutonium.

Status of the Pilot Plant. The pilot plant phase of the Purex process development will employ the equipment in Building 205 that was previously used for the 25 and the Redox process pilot plant studies, as well as the equipment in 706HB previously used for the 23 process pilot plant studies. In the course of the conversion of Building 205 equipment, the cells and equipment have been decontaminated and the floors and walls of the cells have been clad with stainless steel to permit greater ease of decontamination in the future.

The solvent extraction equipment in the Building 205 cells will include a pulse column and a packed column in parallel to permit pilot plant evaluation of their operability. In addition, there will be sufficient flexibility to demonstrate the alternate Purex processes that are now being considered. The

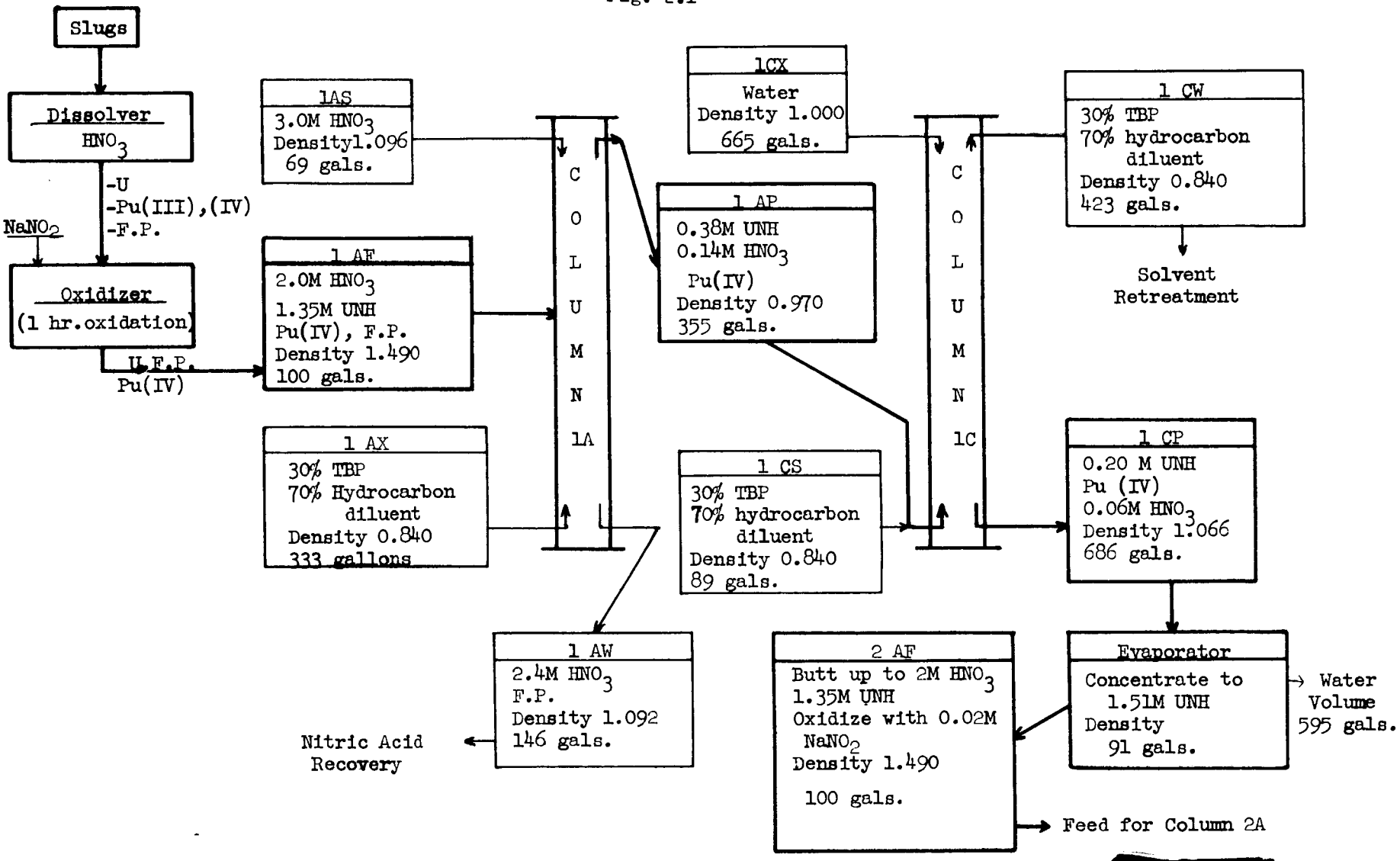
[redacted]

PUREX PROCESS - FIRST CYCLE

Flowsheet No. 2

November 1950
Drawing No. 10041

Fig. 2.1



[redacted]

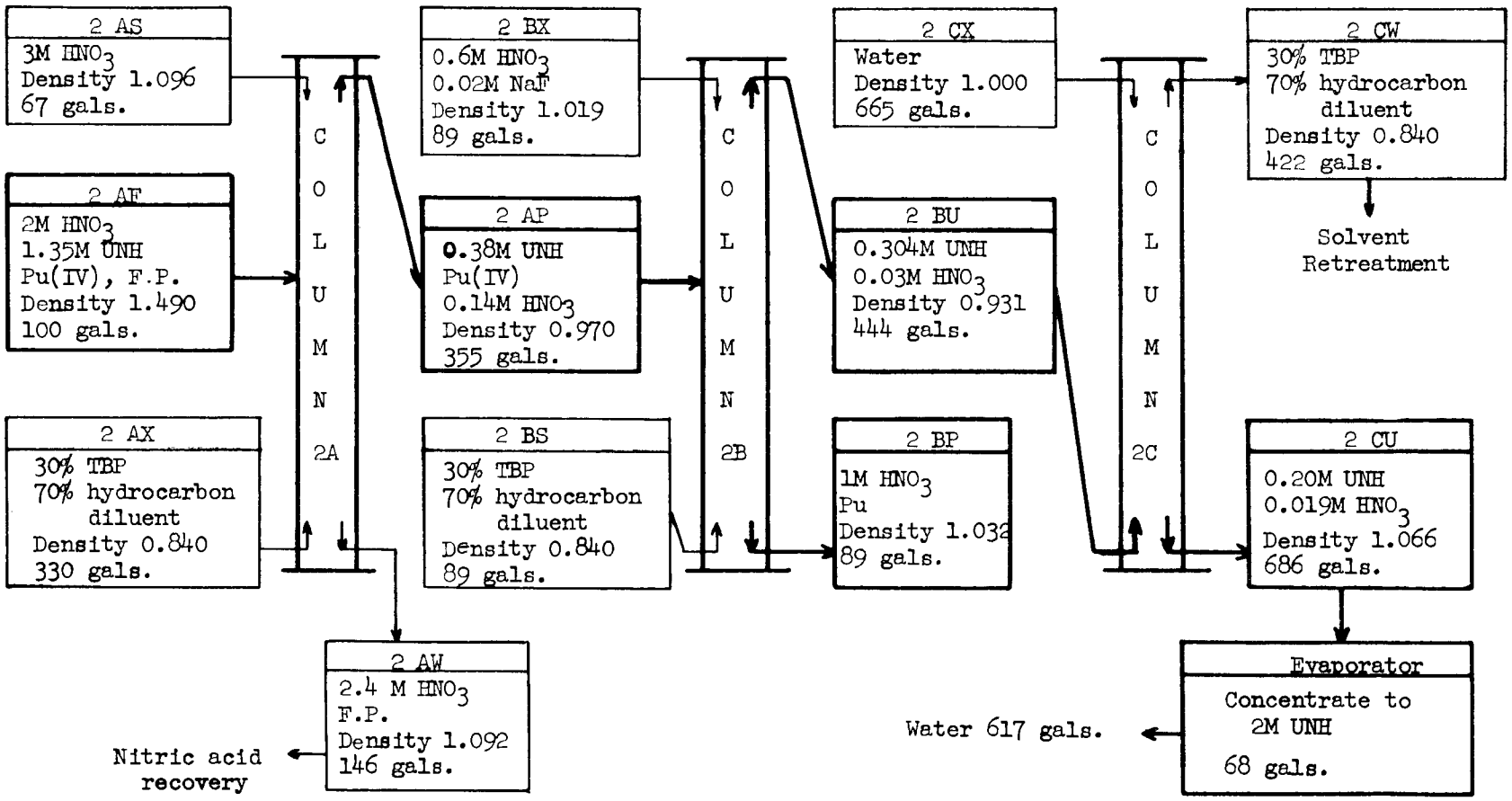
[redacted]

PUREX PROCESS - SECOND CYCLE

Flowsheet No. 2

November, 1950
Drawing No. 10042

Fig. 2.2



[redacted]

alterations of the solvent extraction equipment are essentially complete, and equipment testing and calibration are in progress. Work is still in progress on the acid recovery equipment and the first-cycle feed makeup equipment, which is scheduled for completion by Jan. 1, 1951. New sampling equipment has also been installed for the first-cycle process which will greatly reduce the radiation hazard which existed in the past (see ORNL-885 for pilot plant conversion details).

The conversion of the 23 process pilot plant equipment was completed and was used during this period for the demonstration of the operability of the first-cycle uranium-plutonium partitioning-cycle column. Cold runs in the Building 205 pilot plant are scheduled to begin in December, with hot runs to be started in February, 1951. A five-month experimental program to develop and demonstrate the major features of the process will then be conducted. Starting about July, 1951, the pilot plant equipment will then be modified as necessary to obtain all the data required for the detailed design and operation of equipment for a Purex production plant.

3. RaLa PROCESS

Comparison of Feed Preparation Procedures from MTR Fuel. The initial step of the RaLa Process is the dissolution of the MTR fuel unit and the recovery of the barium from the resulting solution. Three procedures for this phase of the RaLa Process were evaluated: (1) the sulfuric acid procedure, (2) the nitric acid procedure, and (3) the caustic procedure. The results of the laboratory investigations have indicated that the sulfuric acid procedure is the least complicated from both the equipment and chemical viewpoint and that it yields a barium product that is the most readily processed through the final ion-exchange purification step.

In the sulfuric acid procedure the MTR fuel unit was dissolved in 3 *M* sulfuric acid with 2% mercury, based on the weight of the fuel unit, as a catalyst. In addition, the solution was maintained at about 0.3% hydrogen peroxide, which served to redissolve the mercury that was deposited during the amalgamation-dissolution cycle (see Fig. 3.1). Complete solution of the fuel unit was obtained in about 2 hr with a final aluminum concentration of about 1 *M*. After the solution was cooled to room temperature, the barium, present as a barium sulfate crystal suspension, was recovered by simple filtration In laboratory tests the barium sulfate has been separated from the metal solution by filtration through a ¾-in. bed of analytical grade Celite filter aid or a ½-in. bed of cellulose BW-40 Solka-flock filter aid on an H porosity sintered stainless steel filter. The filter cake was then washed with 1 *M* sulfuric acid followed by water, and the barium precipitate was dissolved in dilute Versene. This solution contained approximately 96% of the barium, 0.1% of the aluminum, and 0.05% of the uranium. It was possible to substitute 1.6 *M* nitric acid for the hydrogen peroxide in the dissolution step. However, it was found that less complete precipitation of the barium sulfate resulted with the nitrate present in the solution.

In the nitric acid procedure, the assembly was dissolved in nitric acid, using 2% mercury as a catalyst and 4 moles of acid per mole of aluminum. This was followed by three crystallization cycles (see Fig. 3.2). In the first cycle, the bulk of the aluminum nitrate was crystallized from a 70% nitric acid solution at -15°C. The supernatant from the first crystallization containing the barium was evaporated, and then two more crystallizations were made from

Dwg. # 9991

Fig. 3.1

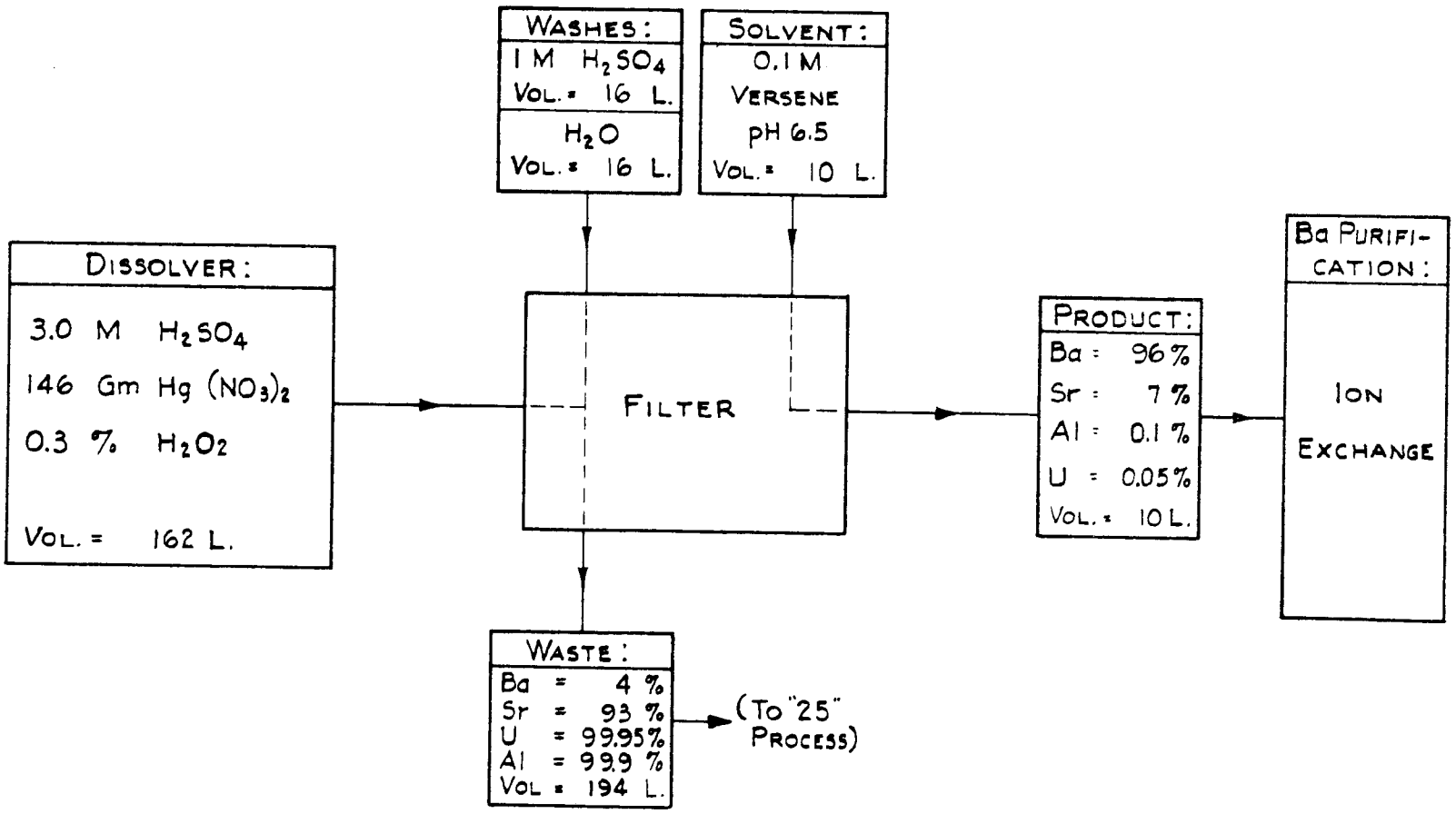
TENTATIVE FLOWSHEET: MTR-RALA SULFURIC ACID PROCESS

FULL SCALE = 1 MTR ASSEMBLY *

162 MOLES Al

0.63 MOLES U

7.28×10^{-3} MOLES Ba



* DOES NOT INCLUDE ENDS WHICH WILL BE REMOVED BY SHEARING

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Dwg. # 9992

Fig. 3.2

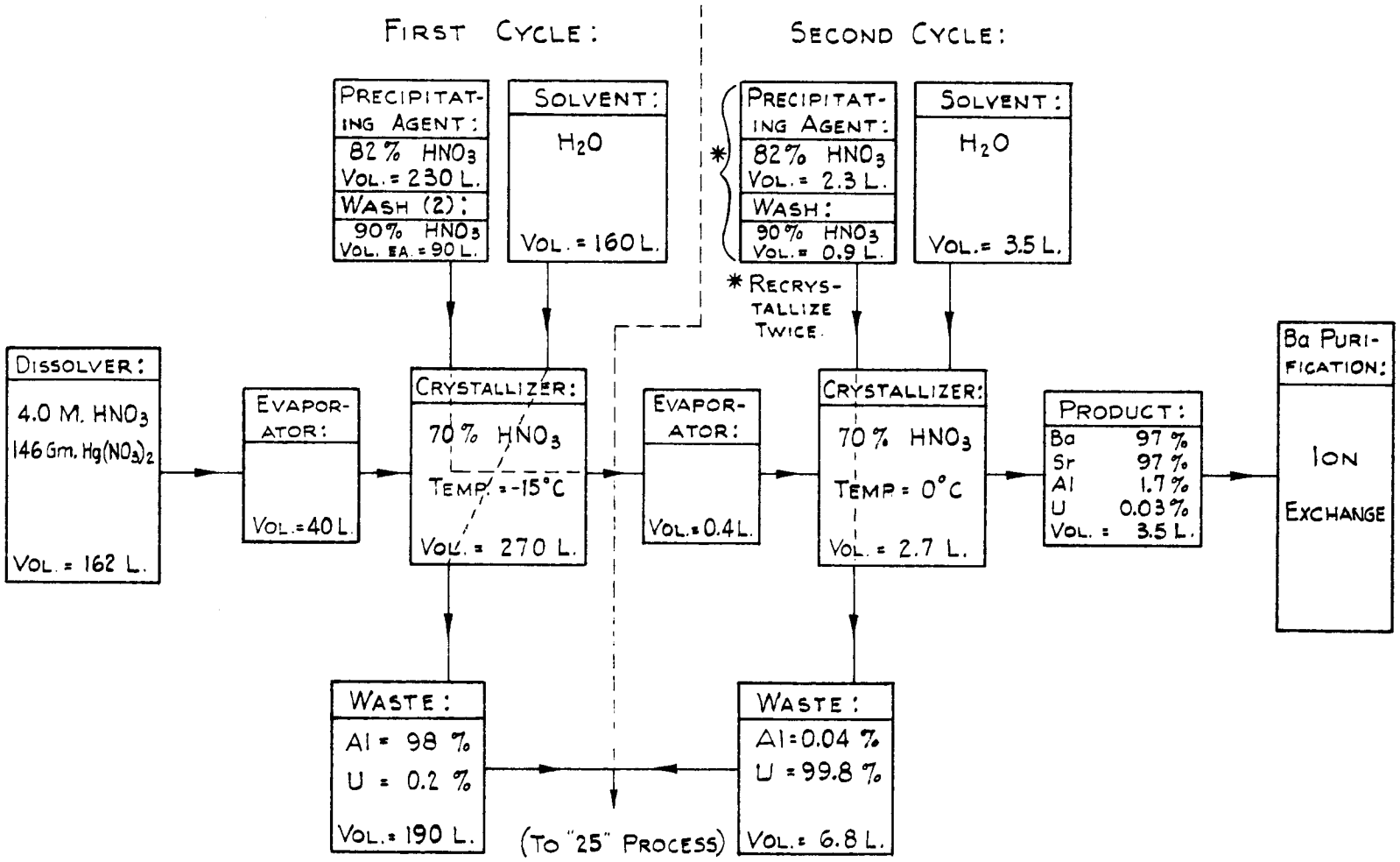
TENTATIVE FLOWSHEET : MTR-RALA NITRIC ACID PROCESS

FULL SCALE = 1 MTR ASSEMBLY *

162 MOLES Al.

0.63 MOLES U

7.28×10^{-3} MOLES Ba.



* DOES NOT INCLUDE ENDS WHICH WILL BE REMOVED BY SHEARING

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Dwg. # 9993

Fig. 3.3

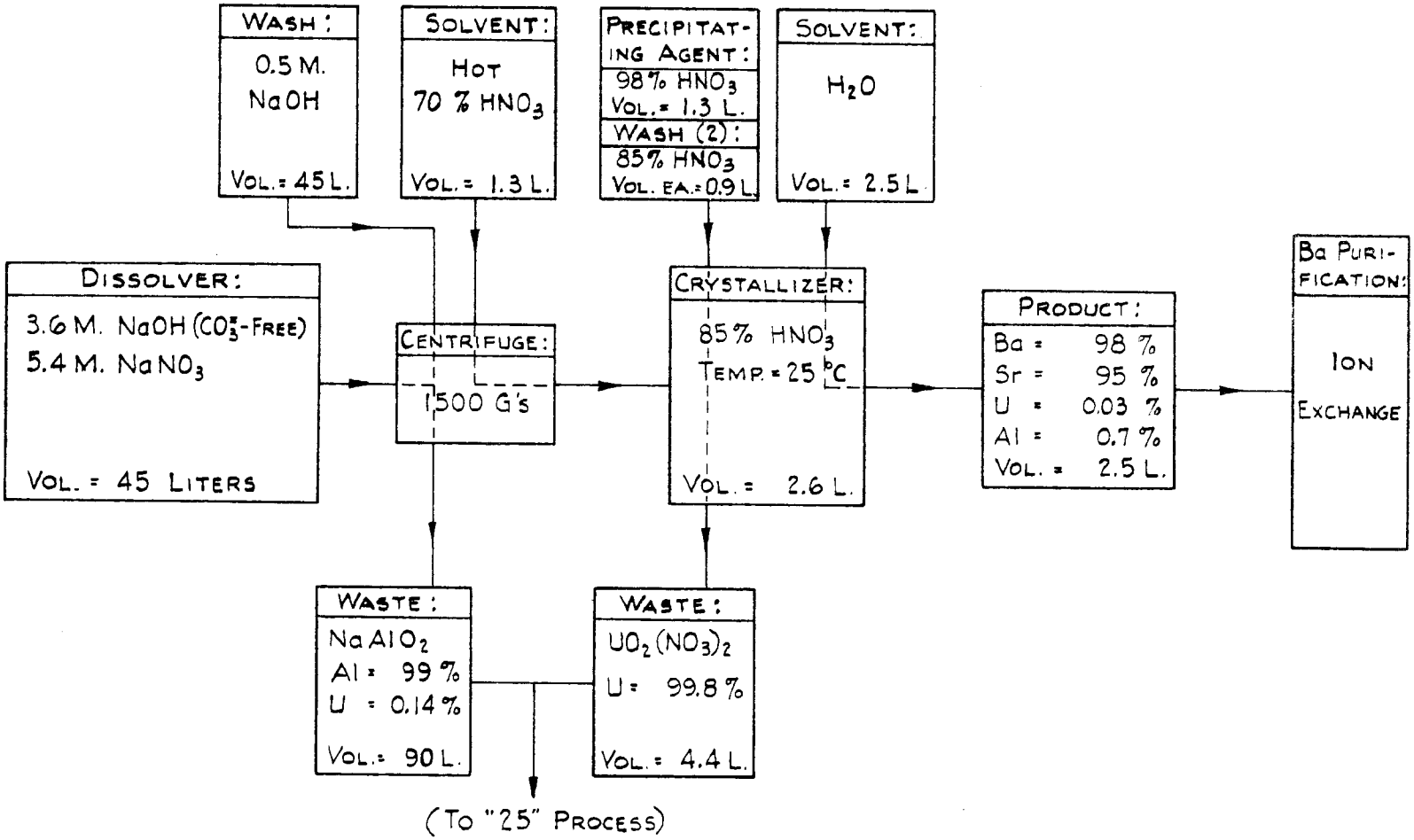
TENTATIVE FLOWSHEET : MTR-RALA CAUSTIC PROCESS

FULL SCALE = 1 MTR ASSEMBLY *

162 MOLES Al

0.63 MOLES U

7.28×10^{-3} MOLES Ba.



* DOES NOT INCLUDE ENDS WHICH WILL BE REMOVED BY SHEARING

[redacted]

a 70% nitric acid solution at 0°C. Here the barium nitrate was crystallized, separating it from the soluble uranium. The crystalline barium nitrate was then dissolved in water with the resulting barium product solution containing 97% of the barium, 1% of the aluminum, and 0.05% of the uranium. It was found that by increasing the nitric acid concentration from 70 to 85% at 25°C in the initial crystallization step, the barium loss in the initial aluminum nitrate crystallization step was significantly increased.

In the caustic procedure the assembly was dissolved in a sodium hydroxide—sodium nitrate solution to yield a final solution containing 3.6 *M* aluminum, 3.6 *M* sodium hydroxide, and 5.4 *M* sodium nitrate. Work at Hanford had indicated that these would be the optimum solution conditions. In this procedure the uranium, barium, and fission products were not dissolved but could be separated from the aluminate solution by centrifugation, washing with 0.5 *M* sodium hydroxide, and dissolving in nitric acid. The barium and residual aluminum were then separated from the uranium by crystallization, in the same manner as described previously for the nitrate procedure (see Fig. 3.3). The final product solution contained 98% of the barium, 0.7% of the aluminum, and 0.05% of the uranium. Return of both the caustic wastes and the crystallization supernatant to the 25 process would be required for recovery of the uranium.

The work on the sulfuric acid procedure is continuing with studies to determine the optimum conditions for crystal growth in order to obtain an improved barium yield in the filtration step.

ORNL RaLa Plant Modification. The modification of the ORNL RaLa plant has been completed, with the exception of the equipment required for the ion-exchange purification cycle. Full-scale runs using cold metal feed indicated a lower sulfate precipitate filtration rate than was expected. Work has been initiated in the Laboratory and Unit Operations Sections to develop a solution to this problem.

4. IDAHO CHEMICAL DESIGN

Status of Design and Construction. A facility is under construction at the Reactor Test Station near Arco, Idaho, for the purpose of recovering U^{235} from the spent fuel units of the various reactors now proposed for construction at that site. The chemical processing area includes four major buildings: the SF Storage Building, the Chemical Processing Building, the Waste Processing Building, and the Utilities Building. The completion of this facility has been scheduled for Oct. 1, 1951 (see Fig. 4.1).

The SF Storage Building, designed for the storage of source and fissionable material prior to processing, is essentially a covered water basin for the storage of spent fuel. The concrete for this building has been poured, and the steel work will be started soon. It is expected that the construction of this building will be completed by March, 1951.

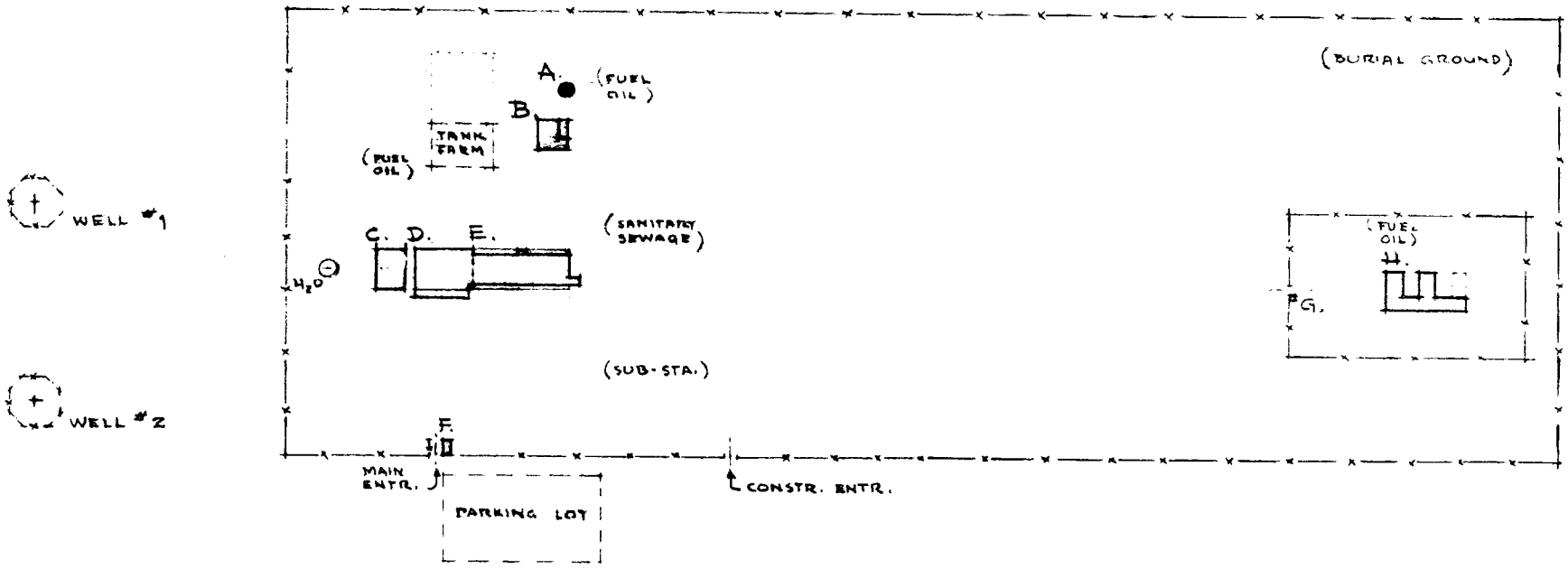
The design of the Chemical Processing Building is in progress, and 70% of the concrete design has been completed, along with 40% of the steel work (see Fig. 4.2). Construction has been started, with the concrete poured for the drain ditch as well as for the deep cells required for the waste collection area in the building. Design sketches of all the vessels and heat exchangers for process equipment have been completed, and bids have been received on representative sketches that were previously submitted to fabricators. On the basis of these bids, tank fabricators will be chosen during the next quarter and the required information supplied to them in order to allow them to make the detailed design drawings on these pieces of equipment. This significantly reduces the amount of detailed design required on the individual pieces of equipment. The equipment layouts for the feed makeup area and the dissolver cells have been completed, and work is in progress on the other processing areas in the plant. The mill orders have been placed for the 750 tons of stainless steel that will be required for the vessels and piping, and the order has also been placed for the structural steel required in the building.

The Waste Processing Building will house the evaporator for the low-level liquid waste and the blowers and decontamination equipment for the radioactive off-gas system. The evaporation of the high-level liquid waste will be carried out in the Chemical Processing Building, with the condensate going to the low-level evaporator along with other low-level wastes for re-evaporation. The

Dwg. 10301

KEY:

A. EXHAUST STACK.
B. WASTE PROCESS BUILDING.
C. UTILITY BUILDING: SERVICES.
D. ANALYTICAL LABORATORIES & OFFICES.
E. PROCESS CELLS & FACILITIES.
F. MAIN GUARD HQTRS & ENTRANCE CONTROL.
G. ACCESS TO STORAGE AREA.
H. S.F. STORAGE BUILDING.



NORTH

SCALE: 1" = 400'-0"

PLOT PLAN FOR CHEMICAL PROCESSING AREA AT IDAHO SITE

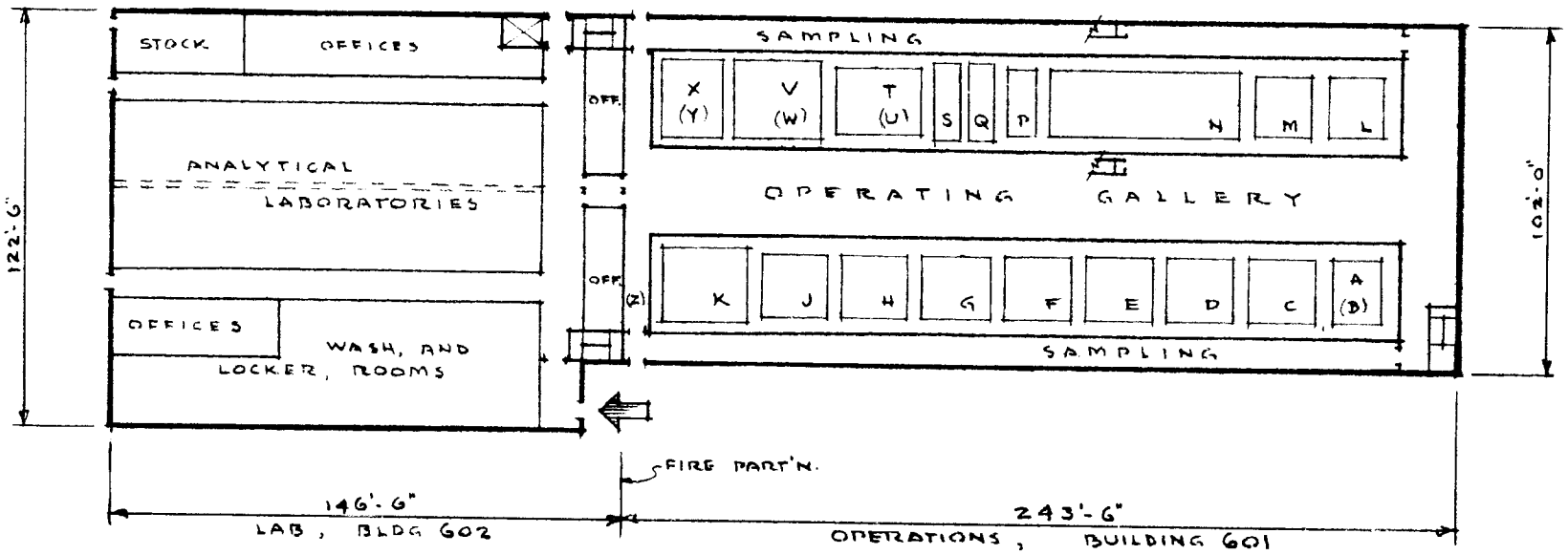
A.D M 12.30.50

Fig. 4.1

Dwg. 10302

CELL DESIGNATIONS

A. E.B.R. MAKE-UP
B. E.B.R. STORAGE
C. N.P.-M.T.R. MAKE-UP
D. " " " "
E. " " " "
F. SPARE
G. SPARE
H. SPARE
J. HOT SALVAGE
K. SOLV. RECOVERY
L. RALA
M. RALA
N. N.P.-M.T.R. STORAGE
P. 1ST CYCLE EXTRAC.
Q. 2ND " "
R. UPPER PART OF "S"
S. 3RD CYCLE EXTRAC.
T. SOLVENT PUMP RM.
U. 1ST CYCLE AQ. RAFF TREAT.
V. COLD SALVAGE
W. 1ST CYCLE SOLV. RAFF. TREAT.
X. DECONTAMINATION
Y. 2ND, 3RD CYCLE RAFF TREAT.
Z. 3RD CYCLE PROD STORAGE.



NORTH

SCALE: 1" = 50'-0"

FIRST FLOOR PLAN OF CHEM. PROC. BLDG.

A.D.M., 12-30-50

Fig. 4.2

tank farm associated with these facilities will consist of two collection tanks, each with a 20,000-gal capacity; three collection tanks, each with a 40,000-gal capacity; and two permanent storage tanks, each with a 300,000-gal capacity. All tanks will be stainless steel and installed in concrete pits.

The design specifications for the equipment to be installed in the Utilities Building have been transmitted to the Foster Wheeler company, and the design of this building is underway.

Dissolution of Uranium-Aluminum Alloy. Cast uranium-aluminum alloy slugs from the P-10 program are to be processed in the Idaho Chemical Plant. For accountability and criticality reasons, it is desirable to dissolve completely each batch of slugs in a single operation. On the basis of the work completed to date, the preferred procedure for the dissolution of the cast slugs requires about 12 hr for complete dissolution and results in a solution 1 *M* in aluminum nitrate and 1 *M* in nitric acid. This procedure requires 4.35 moles of nitric acid per mole of alloy and 0.004 mole of mercury per mole of alloy. The total acid is added in three equal parts, each portion being heated to boiling, digested for 4 hr, and withdrawn before the addition of the subsequent portion. This work is being continued in an effort to reduce the time required and the acidity of the final product solution.

Preliminary dissolution studies made with slugs produced by an extrusion process indicated that 93% of the slug was dissolved in 4 hr in the presence of 4.0 moles of nitric acid per mole of alloy and 0.002 mole of mercury per mole of alloy. However, under similar conditions, slugs produced by a casting procedure were only 50% dissolved at the end of the same period. Many factors have been found to be effective in increasing the dissolution rate of the cast slugs, including (1) increasing the moles of nitric acid per mole of alloy, (2) increasing the concentration of the mercury catalyst, (3) heat-treating or cold-working the cast slugs prior to dissolution, and (4) the addition of such ions as potassium, calcium, or sodium, which decreased the tendency of the cast alloys to become passive during dissolution.

Comparison of Filter Aids for the Clarification of the Uranium-Aluminum Solution. The satisfactory removal of insoluble aluminum silicate alloy from uranium-aluminum metal solution was satisfactorily demonstrated by a vaccum filtration technique. The filter medium was a 700 by 60 stainless steel Dutch twill weave filter cloth, precoated with diatomaceous earth or cellulose

filter aid, which gave an average filtration rate of 5 ml/sq cm/min. Maximum filtrate throughput was 66 ml/sq cm for the diatomaceous earth and 89 ml/sq cm for the cellulose. Work on this problem is being continued to determine if a bare filter cloth or a coarse sintered stainless steel plate may not be sufficiently effective for this application.

The diatomaceous earth filter aid that was used in this study was Johns Manville Celite-545. A satisfactory precoat of this material was 0.44 g of Celite per square centimeter. The cellulose filter aid was Alpha Floc grade AW-40, made by the Browns Company, and 0.22 g of this Alpha Floc per square centimeter was required for a satisfactory precoat. An advantage of the alpha Floc filter aid was the fact that it could be dissolved in 15 ml of 65% sulfuric acid per gram of cellulose fiber in 10 min at room temperature. The resulting solution, when filtered through a medium fritted glass filter, left no solids on the filter.

An important consideration in the use of filters in radiochemical processes is their ability to be decontaminated. It was demonstrated that, by electrostripping, a gross beta decontamination factor of 30 was obtained. This procedure involved the application of 0.2 amp-min per square centimeter through the filtered disk as the cathode in a 1 *N* hydrochloric acid solution. After rinsing with water, the disk was made the anode in an 85% phosphoric acid solution and 2.2 amp per square centimeter of surface area was applied. The amount of surface removed from the metal was found to 10^{-5} in. of metal.

Continuous First-cycle Evaporator. An evaporator was required in the Idaho Chemical Processing Plant, following the first solvent extraction, to reduce the volume of the uranium product stream by a factor of 30 prior to pumping it to the second cycle. Additional requirements for this evaporator were as follows: (1) the feed rate was to be 300 ml/min, (2) the hexone dissolved in the product stream was to be flashed off to eliminate its decomposition during evaporation, and (3) because of criticality conditions, no diameter of the evaporator could be larger than 5 in. A falling-film evaporator was designed which incorporated these features, and satisfactory operation under atmospheric pressure was demonstrated. The evaporator consists of a preheater for the feed, a 1½-in.-diameter stainless steel pipe, 10 ft long, for the falling-film section of the evaporator, and a 6-in.-diameter reboiler pot on the bottom of the falling-film tube (5-in.-diameter pipe was not readily available for fabrication of an experimental evaporator). Optimum operation

was obtained with 75% of the evaporation taking place in the falling-film section and the remaining 25% in the reboiler. In the final product, no trace of hexone or its decomposition products could be found. The entrainment for this evaporator varied from 0.01 to 0.05%. A baffle chamber in the off-gas line served as the entrainment unit.

Control of the evaporator is automatic. The product is pumped from the reboiler at a constant rate to the second-cycle solvent extraction column. The height of liquid in the reboiler is maintained at a constant rate by varying the amount of steam pressure on the steam jacket and thereby regulating the amount of evaporation.

Further work on this unit is planned to evaluate its operation under vacuum and to improve the de-entrainment operation.

Pump Evaluation for Second-cycle Feed. A 250-hr check run was made on a Lapp Pulsafeeder diaphragm pump at a rate of 10 ± 0.3 ml/min. The operation of this pump was the most satisfactory of those tested, and a more thorough test of the pump is in progress.

Two other pumps were considered for this application: (1) the Milton Roy piston pump, and (2) a "pneumatic-piston" pump of local design. The Milton Roy pump proved to be unsatisfactory because of leakage of the packing gland. To correct this situation the gland was doubled in length and water at a pressure of 1 to 2 lb/sq in. in excess of the discharge pump pressure was injected at the mid-point. However, under these conditions as much as 0.3% of the uranium leaked from the gland.

The pneumatic-piston pump consisted of two pressure cylinders connected to Milton Roy pump heads. In operation, the pressure cylinders were alternately filled and emptied by the operation of solenoid valves and electric probes, with the flows from the pump controlled by a Hammel-Dahl control valve. Operation of this system was not satisfactory because leakage through the check valves led to unbalance between the two cylinders. Also, the Hammel-Dahl valve was easily plugged.

5. EXTRACTION OF URANIUM FROM ORES: EVALUATION OF ALTERNATE SOLVENTS

The present procedure for uranium extraction from ores involves a diethyl ether solvent extraction process (Mallinckrodt). Recently the AEC has suggested that the solvents now commonly used for the purification of the uranium from the spent reactor fuels be evaluated for use in the recovery of uranium ore. Using a representative feed supplied by Mallinckrodt, the laboratory evaluation of these solvents was carried out in batch countercurrent equipment. The results of this study indicated that only tribuyl phosphate showed any promise of replacing diethyl ether with a minimum of short-range development. Other solvents included in the study were methyl isobutyl ketone, dibutyl cellosolve, pentaether, and diisopropyl ether. The advantages of tributyl phosphate over diethyl ether as a solvent in this procedure include: (1) increased safety of operation since the flashpoint of tributyl phosphate—hydrocarbon diluent mixtures is greater than 130°F as compared to -40°F in the case of ether; and (2) improved uranium extraction in the presence of phosphate and sulfate ions. On the basis of chemical and equipment costs, the two processes are comparable. The results of the work are reported in detail in ORNL-874.

The remaining major problem is the evaluation of tributyl phosphate with slurry type feeds in the jet-mixer type solvent extraction columns now in use at Mallinckrodt. This program is now in progress at Mallinckrodt and the preliminary results have been satisfactory.

6. VOLATILITY PROCESS STUDY

Plutonium Separation from Uranium Hexafluoride by Alundum Adsorption. The separation of plutonium fluoride from uranium hexafluoride following reaction of uranium metal with fluorine gas to produce gaseous uranium hexafluoride was found in earlier work to be accomplished by the adsorption of the plutonium on alundum. In subsequent studies the most effective variable has been found to be the pressure under which the adsorption takes place. By increasing the pressure from 14.7 to 20 lb/sq in. absolute, the adsorption of plutonium and beta activity was increased by a factor of 10 to 100 (see Table 6.1). Other variables investigated included the geometry of the bed, the

TABLE 6.1

Effect of Pressure on Alundum Adsorption of Plutonium and Fission Products

Alundum: Norton RR (90-mesh crystals)
Temperature: 100°C
Feed: Uranium metal (irradiated 2 years, cooled 4 years)

PRESSURE (lb/sq in. absolute)	BED SIZE	ALUNDUM WEIGHT (g)	DECONTAMINATION FACTOR	
			Pu	β
14.7	3/4 in. diameter by 21 in. long	253	34	13
20 - 25	3/8 in. diameter by 21 in. long	45	3,100	1,060
		50.8	28,000	2,800

weight of the alundum adsorber, and the type of alundum adsorber, but no significant difference in results was noted (see Table 6.2).

Electrofluorination of Uranium Metal. An exploratory investigation of the electrofluorination of uranium metal in molten fluoride systems was initiated. A two-step process was demonstrated in which the uranium was electrolytically dissolved in KF·2HF and then electrolytically fluorinated to evolve the UF_6. The work on this process is to be continued.

TABLE 6.2

Comparison of Alundum Type on Plutonium and Fission Product Decontamination

Feed: 49- to 77-g samples of uranium (irradiated 12 months, cooled 27 months)

Adsorption Conditions: 100°C and atmospheric pressure

ADSORBENT	BED SIZE	ALUNDUM WEIGHT (g)	DECONTAMINATION FACTOR	
			Pu	β
Norton RA-84 chips (crushed crucibles)	1 in. diameter by 7 in. long	80	39	1.6
		90	16	1.2
		105	58	1.3
Norton 1221-1222 1/4-in. spheres	1 in. diameter by 9 in. long	80	37	1.6
Norton RR 90-mesh crystals	1 in. diameter by 9 in. long	158.4	7	22
	3/4 in. diameter by 21 in. long	253	34	13

7. RADIOACTIVE WASTE TREATMENT

Radiochemical Waste Evaporator. During August, September, and October 615,000 gal of mixed waste was concentrated by the Operations Division in the radiochemical waste evaporator with an average beta decontamination factor of 2×10^3 and an average condensate activity of 2×10^{-4} microcurie per cubic centimeter. While processing waste containing 10^7 beta c/min/ml, an average decontamination factor of 10^5 was obtained. This is the highest average decontamination that has been obtained to date.

To further improve the decontamination obtained in the evaporator, it is planned to install a three-stage Peabody impingement baffle plate scrubber on the vapor line from the evaporator. It was previously proposed to use a five-plate bubble cap vapor scrubber for this purpose; however, the Peabody scrubber has the advantages of lower initial costs, smaller size, and anticipated greater vapor scrubbing efficiency. The instruments previously reported on order for foam level indication and steam and water consumption determination are scheduled for delivery during the coming quarter.

Evaluation of the Cottrell Electrostatic Precipitator. The Cottrell electrostatic precipitator has now been placed in operation on the off-gas system from the iodine production unit. This precipitator is a stainless steel unit designed to process 2000 cu ft of air per minute with an applied voltage of 75 kv. The collecting tubes of this unit are continuously flushed with water in order to prevent the build-up of activity in the unit. Preliminary tests with the unit have indicated that without the wash system in operation, only 50 to 60 kv may be applied without arcing. With the unit flushed with 30 gal of water per minute, which is the minimum flow rate for complete flushing, the applied voltage dropped to 40 kv. The contaminated air going to the unit contained 10^{-7} to 10^{-5} microcurie of activity per cubic centimeter, of which better than 90% was I^{131}. To evaluate the efficiency of the precipitator, sample streams were pulled off before and after the precipitator, and the sampled gas was filtered through CWS-6 filter paper. Based on the assumption that this filter paper is 100% efficient, the efficiency of the Cottrell unit has varied from 30% without water flowing, to 94% with 60 gal of water per minute flushing the unit. The apparent low efficiency of this unit does not seen reasonable. Tests on the unit are to be continued to evaluate the accuracy of the sampling procedure.

8. ELECTROLYTIC DISSOLUTION STUDY

Electrolytic Dissolution of Uranium Metal in Sodium Bicarbonate Solution. The feasibility of an electrochemical process for separation of uranium, plutonium, and fission products was reported in K-587 and K-601, based on work carried out at K-25. This process involved the electrolytic dissolution of uranium in a sodium bicarbonate solution to form a black uranium oxide which carried the plutonium out of solution, resulting in concentration of the plutonium with respect to uranium and its separation from the fission products. In connection with this work, the effect of hydrogen peroxide and anode rotation on the dissolving efficiency and the dissolving rate was studied. The results of this study indicate that (1) the addition of 0.4 *M* hydrogen peroxide increased the amount of uranium dissolved from 1.37 to 1.53 g/amp/hr, and (2) the rotation of the anode had no effect in the presence of the peroxide solution.

Electrolytic Dissolution of Uranium-Aluminum Alloy in Sodium Hydroxide. The dissolution of uranium-aluminum alloy in sodium hydroxide has been considered in the past because the insoluble uranium could be separated from the aluminate solution, which would reduce the volume to be processed by solvent extraction. However, the major objection to this procedure has been that hydrogen was evolved, which constituted an explosion hazard, and that the uranium was sufficiently soluble in the sodium hydroxide solution to cause a significant loss. Dissolution of the uranium-aluminum alloy was demonstrated in an electrolytic cell with a porous diaphragm between the cathode, in which the hydrogen was evolved, and the anode cell, in which the aluminum alloy served as the anode. This permitted the hydrogen gas to be evolved free of radioactive gases. The radioactive gases, evolved from the uranium-aluminum alloy, could then be treated in the presence of a minimum of other gases.

A second electrolytic dissolution procedure was demonstrated in which the aluminum served as the anode, a nickel oxide plate such as used in an Edison cell as the cathode, and sodium hydroxide as the electrolytic solution. In this system no hydrogen was evolved. In addition it was not necessary to supply current to the cell since this system acts as a battery; however, current was applied to decrease the dissolving time. Current densities of up to 12 amp/sq cm were obtained with both cast and extruded rods.

In the second phase of the problem it was found that the caustic solution, after a sample of uranium-aluminum alloy was anodically dissolved in 20% hydroxide, contained 30 ppm of uranium. Electrodeposition, using approximately 20 amp-hr per liter of solution and a smooth platinum-wire electrode, reduced the uranium concentration to 3 ppm with the remainder of the uranium deposited on the cathode. Earlier work had shown that by the use of coarse electrodes instead of smooth electrodes lower concentrations of uranium in solution could be obtained.

9. HOMOGENEOUS REACTOR CHEMISTRY

Preparation of Stable Slurries. A uranium slurry was proposed as the optimum fuel for a homogeneous reactor. In order to be satisfactory, this slurry must be stable under the conditions of 250°C and 100 lb/sq in. pressure and a neutron flux of about 10^{15}. This slurry is also required to remain in suspension under both flow and still conditions. Earlier work demonstrated that uranium oxide slurries were not stable. Therefore it has been necessary to investigate the use of stabilizing agents, and, of these, bentonite has proved most successful. The bentonite was most effective when its particle sizes were below 25 millimicrons and when all the exchangeable cations such as iron, magnesium, and calcium had been removed.

Slurries composed of 40 g of uranium as uranium trioxide and 12.5 g per liter of the specially prepared bentonite have been found to be stable under the proposed thermal and pressure conditions for a period of 330 hr. However, at the end of 526 hr, the uranium had settled out in a fairly hard cake.

There was evidence that ferric hydroxide was present, probably from the corrosion of the stainless steel bomb in which the tests were carried out. Earlier work showed that iron flocculated the bentonite; however, the addition of phosphate would complex the iron, holding it in solution. Another approach to be examined is conversion of the hydrogen form of the bentonite to another suitable bentonite, such as beryllium bentonite, that might be less sensitive to the presence of foreign cations.

Colloidal silic acid was investigated as an alternate slurry stabilizer in the place of bentonite. It appeared from qualitative observation on the degree of settling that over 4% silicic oxide as colloidal silicic acid would be necessary to achieve a satisfactory suspension of uranium trioxide. The colloidal silicic acid was prepared by passing sodium silicate through a cation exchange resin in the hydrogen form, and the resulting clear solution was allowed to age until it became colloidal. Slurries were then prepared containing 40 g of uranium as uranium trioxide, and the concentration of silicic acid was varied from 20 to 40 g per liter. The slurry was tested in a bomb at 250°C for 22 hr and allowed to settle for 24 hr.

The brown uranium dioxide was known to be unreactive with water at 250°C. Slurries of uranium dioxide and bentonite were therefore prepared and tested at 250°C for 138 hr in a sealed quartz tube. After standing 24 hr at room temperature, analysis for uranium distribution showed that all the uranium had settled out. This instability indicated the presence of some impurities.

Fuel Processing. The ideal process for spent fuel from a nuclear reactor would include the following characteristics: (1) no change in the fuel, (2) no fuel holdup, (3) a continuous process, (4) no radiation effect, (5) maximum separation, and (6) low waste volume. With a homogeneous reactor, using uranyl sulfate solution as fuel, it may be possible to approach an ideal process. In the design of the homogeneous reactor provision has been made for the separation of the radioactive gases krypton and xenon. For the processing cycle the following techniques are being evaluated: (1) filtration, (2) adsorption, (3) electrolysis, and (4) ion exchange. A preliminary study has been made to evaluate each of these procedures on the basis of its removal efficiency for each specific radioisotope. Filtration of the pile solution should be effective for the removal of the barium and zirconium, and, by the use of a filter aid such as Super Filtrol, adsorption can be combined with the filtration to remove the rubidium, zirconium, niobium, and cesium. It is thought that in electrolysis molybdenum would be plated out on the cathode while iodine, tellurium, and ruthenium would be plated out at the anode. The final step in the process, ion exchange, would be required to remove the rare earths and the other fission products not removed in the earlier steps. While ion exchange is probably the most efficient process of those considered for the homogeneous fuel processing cycle, the ion-exchange resin is not stable under high radiation. Radiation stability tests on Dowex 50 are in progress using Co^{60} as the radiation source. Preliminary results indicate that the major effect was the loss in capacity, and that the uranium on the resin was easily eluted. Therefore a high uranium loss on the irradiated resins would not be expected. This study will be continued.

10. CHALK RIVER: FINAL CONCENTRATION AND PURIFICATION

Two of the six batches of Chalk River plutonium have been processed through the final concentration and purification cycle and have been shipped to the customer. The plutonium loss for the first batch was 4.2%; however, only 0.5% was lost when the second batch was processed. The final product showed the following impurities, per milligram of plutonium: 0.08 mg of uranium, 0.02 mg of iron, 1.85 mv of gamma, and 2.7×10^4 beta counts per minute at 10% geometry. The presence of gross amounts of organic decomposition products and corrosion products in the feeds to the final cycle has caused considerable difficulty.

11. PROCESS VESSEL DECONTAMINATION BY ELECTROSTRIPPING

During the conversion of the pilot plant equipment from the Redox process to the Purex process, it was necessary to decontaminate the equipment used for metal dissolution and feed preparation. Chemical methods were used to reduce the level of the radiation to approximately 2 r inside the vessels. At this point, chemical treatment was not further effective. However, it was found that a decontamination factor of 10 to 40 was obtained when the process vessels were electrostripped. The electrostripping operation was carried out with the stainless steel tank or vessel as the anode and with the vessel filled with 2% sulfuric acid. A copper pipe served as the cathode, and a standard direct-current welding machine served as the source of power. The output of this machine was 200 amp at 35 volts. Three tanks were decontaminated by this procedure with volumes from 55 to 360 gal, and the time requirement for this operation was 20 to 40 min.

It should be recognized that the conditions used in this preliminary application of electrostripping are not known to be the optimum conditions. However, the success obtained does suggest that large volumes of contaminated waste resulting from process vessel decontamination may be drastically reduced in the future by the use of a modification of this electrostripping procedure.

12. PUBLICATIONS

REPORT NO.	TITLE	AUTHOR	DATE
	TBP METAL RECOVERY		
ORNL-743	Application of Purex Process to ORNL Metal Waste Recovery	C. V. Ellison T. C. Runion	8/24/50
CF 50-9-112	Countercurrent Batch Decontamination Runs on TBP Process Using Shell Deodorized Spray Base as a Diluent	T. C. Runion	9/25/50
ORNL-603	Recovery and Decontamination of U from Hanford Wastes: ORNL PP Demonstration	A. C. Jealous	11/13/50
	PUREX		
CF 50-9-61	Letter, F. S. Chambers from F. L. Steahly 9/18/50	F. L. Steahly	9/18/50
	IDAHO CHEMICAL PLANT DESIGN		
ORNL-725	Preliminary Criticality Study, Idaho Chemical Processing Plant	R. J. Klotzbach	8/24/50
CF 50-8-74	Notes on AEC Conference in Washington on August 9, 1950	F. L. Culler	8/21/50
CF 50-8-96	Criticality Control in SF Storage Building	R. J. Klotzbach	8/11/50
CF 50-9-8	Minutes of Meetings on Storage Building and Chemical Process Plant at ORNL, August 16-18, 1950	H. A. Ohlgren	9/5/50
CF 50-8-132	Stack and Stack Location for the Idaho Chemical Processing Plant	H. A. Ohlgren	8/15/50
CF 50-9-64	Trip to Knolls Atomic Power Laboratory 9/6/50	D. G. Reid	9/15/50
CF 50-9-63	Trip to Washington, D.C., 9/13/50	D. G. Reid	9/15/50
CF 50-9-78	Minutes of Meeting on Third Cycle Extraction-25 Process	F. L. Culler	9/14/50
CF 50-9-95	IDO SF Storage Building - Capacity	D. G. Reid	9/19/50
CF 50-10-44	Metallurgical Treatment of Alloy Slugs Versus Dissolving Rate	A. T. Gresky E. O. Nurmi	10/9/50
CF 50-10-69	IDO Chemical Plant - Estimated Hanford P-10 Slug Inventory and IDO Plant Capacity Schedule	D. G. Reid	10/16/50
CF 50-9-168	Minutes of Meeting Held at Oak Ridge 9/1/50	H. A. Ohlgren F. L. Culler	9/27/50
CF 50-10-88	Corrosion Studies of ELC Stainless Steel	R. P. Milford	10/18/50
CF 50-11-36	IDO Chemical Processing Plant - Capacity	D. G. Reid	11/10/50
CF 50-11-177	Comments on B.P. Shepherd's Report of Visit to Oak Ridge and to New York	F. L. Culler	11/30/50

PUBLICATIONS (Cont'd)

REPORT NO.	TITLE	AUTHOR	DATE
	MONTHLY AND QUARTERLY REPORTS		
ORNL-850	Pilot Plants Section Report for July - Aug. 1950	D. G. Reid	10/10/50
ORNL-846	Quarterly Progress Report for Period Ending August 31, 1950	W. K. Eister	10/13/50
ORNL-885	Pilot Plants Section Report for Sept. - Oct. 1950	D. G. Reid	11/28/50
	MISCELLANEOUS		
CF 50-9-52	Material Balance for Processing of Special Chalk River and Hanford Materials	D. G. Reid per N. J. Rigstad	9/15/50
ORNL-801	TBP as an Alternate Solvent for Extraction of 25 from Spent MTR Fuel Assemblies	T. C. Runion	9/26/50
CF 50-9-170	Cost Estimation Procedure to be used in Preliminary Cost Analysis	D. E. Ferguson	9/30/50
CF 50-11-82	Comments on Ion Exchange Resin Bed Proposal	F. L. Steahly	11/17/50
CF 50-11-121	A Method for the Preliminary Analysis of Percolation Data	S. H. Jury	11/20/50

13. PERSONNEL

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C. W. Schersten, Assistant to Director	Virginia Wells, Secretary J. W. Clark, Technician C. J. Shipman, Technician Aola Johnson, Janitress
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R. E. Blanco, Problem Leader - RaLa	
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J. A. McLaren, Problem Leader - Electrochemistry	
J. H. Goode H. L. Peeler	
D. E. Ferguson - Ion Exchange	R. C. Lovelace
A. T. Gresky - "25" Process	
I. R. Higgins - HR Processing	W. E. Shockley
L. E. Morse - Fuel Slurries	
T. C. Runion - Blanket Slurries	J. L. Bamberg V. L. Fowler
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F. L. Culler,(2) Problem Leader - Idaho Chemical Plant	
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R. P. Milford
E. L. Nicholson
A. M. Rom
D. H. Stewart
W. G. Stockdale
A. J. Sulzback[3]
L. W. Zahnstecher[3]

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J. M. O'Keefe[3]
F. L. Peishel[3]
Salvatore Piraino[3]
William Richman[3]
E. P. Stolarski[3]
C. M. Thompkins[3]
W. J. Veltung[3]
W. T. Watson[3]

W. E. Unger, Problem Leader - RaLa

B. F. Bottenfield[6]
J. B. Ruch
R. H. Vaughan
E. Wischhusen[6]

R. V. Foltz[6]

F. C. McCullough - Cost Analysis

BECHTEL CORPORATION REPRESENTATIVES

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Virginia Malone, Secretary
Winzle Shockley, Secretary
} ORNL

Robert Digges, Assistant Project Manager

J. A. Richardson, Project Engineer

R. K. Phillips, Manager of Purchases
John Kuhns, Engineer
L. T. McCarthy, Expediter

H. E. Weber, Project Coordinator

J. O. Davis, Section Chief - Unit Operations

Ruth Pennington, Secretary

A. C. Jealous[7] - Solvent Extraction

Technicians

C. V. Ellison

G. B. Dinsmore

T. A. Arehart[8] - Evaporation, Filtration

F. Mills

T. D. Napier
R. O. Payne
J. C. Rose

C. D. Watson[9] - Equipment Development

J. C. Suddath
G. A. West

G. R. Guinn
F. L. Rogers

E. O. Nurmi[10] - Metal Dissolution

M. R. Bennett
D. L. Foster

G. Jones
D. B. Masters
H. F. Soard

J. O. Blomeke[11] - Electrolysis

(6) On Loan from Engineering Department.
(7) Responsible for Metal Recovery.
(8) Responsible for Purex and Liquid Waste Treatment.
(9) Responsible for RaLa and Gaseous Waste Treatment.
(10) Responsible for "25" Process.
(11) Responsible for Volatility Process.

[redacted]

D. G. Reid,[12] Section Chief - Pilot Plants

Agnes Hair, Secretary
R. B. Waters, Draftsman

H. K. Jackson,[12] Associate Chief

H. K. Jackson,[12] Problem Leader - Purex

F. E. Harrington - Bldg. 205 Conversion

R. B. Lindauer

J. W. Landry - Engineering

J. W. Ullman - Engineering

D. O. Darby - Chemistry

W. H. Lewis - Bldg. 706-HB Operations

G. S. Parker

G. S. Sadowski - Bldg. 205 Operations

C. D. Hylton
R. H. Winget

"A" Operators

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H. L. Sexton
W. E. Ledbetter
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Operators

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R. E. Purkey
R. C. Shipwash
D. E. Spangler
B. J. Strader
D. H. Summers
H. C. Thompson
C. H. Tipton
J. T. Wiggins

E. C. Stewart, Problem Leader - Metal Recovery

J. P. Jackson,[13] Draftsman

E. M. Shank, Problem Leader - Waste Treatment

D. G. Reid,[12] Problem Leader - Idaho Chemical Plant

Phyllis Groover, Secretary

N. J. Rigstad

(12) Dual capacity,

(13) On loan from Engineering Department,

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