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UNITED STATES ATOMIC ENERGY COMMISSION
RICHLAND OPERATIONS OFFICE

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TO: H. E. Parker
DIRECTOR - DEPUTY DIRECTOR
CHEM. PROC. DIVISION - RLOO

DATE May 10, 1966

HANFORD 93855

Report No. 4

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CHEMICAL OPERATIONS BRANCH
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This document consists of
16 pages, No. 1 of
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SUBJECT: 200 AREA MONTHLY REPORT FOR April 1966

DC:AWL:FAA:MR

PROCESSING STATISTICS

PRODUCTION

PRODUCT					PERCENT EFFICIENCY		PERCENT WASTE LOSSES		ATM RELEASE
PUREX DOX URANIUM REDUCTION	URANIUM				OP	MECH	PU	U	CURIES / DAY
	NORMAL		ENRICHED						
	TONS	PERCENT FORECAST	TONS	PERCENT FORECAST					
	<u>278.6</u>	<u>107.2</u>	<u>80.3</u>	<u>94.5</u>					
	<u>355.1</u>	<u>98.6</u>	<u>281.9</u>	<u>141</u>					
PUREX REDOX	BRANDY OR NEPTUNIUM								
	KGS		PERCENT FORECAST						
	<u>885</u>	<u>--</u>	<u>--</u>	<u>--</u>					
	<u>--</u>	<u>--</u>	<u>--</u>	<u>--</u>					
	PLUTONIUM								
PLUTONIUM REDUCTION	KGS		PERCENT FORECAST						
	<u>225.4</u>	<u>90</u>	<u>42</u>	<u>95</u>					
	<u>69.2</u>	<u>--</u>	<u>--</u>	<u>--</u>					
RECOVERY	<u>69.2</u>	<u>--</u>	<u>--</u>	<u>--</u>	<u>--</u>	<u>--</u>	<u>--</u>	<u>--</u>	<u>--</u>

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INVENTORIES

PLUTONIUM			URANIUM	
SCRAP BACKLOG <u>861.7</u> KGS			NORMAL	ENRICHED
SCRAP COMPOSITION:			HANFORD CODE <u>C-65</u>	
PLUTONIUM PROCESSING	<u>40.3</u>		UNH <u>57.9</u> TONS	<u>30.8</u> TONS
FABRICATION	<u>12.2</u>			
RECOVERY	<u>0.1</u>		UO ₃ <u>11,723.4</u> TONS	
BNW LABORATORY	<u>13.0</u>			
OFF-SITE	<u>5.7</u>			
SUPPORTING LABORATORIES	<u>3.9</u>			
GROUP 2 TOR	<u>24.8</u>			
EXCLUDED FROM AUTOMATIC DOWNGRADING AND DECLASSIFICATION	<u>100 %</u>			

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PUREX

The plant was operated at 25.0 TU/D until April 11, 1966, on normal uranium metal. Cold feed was then introduced into the plant for cleanout preparatory to thorium processing. Pre-thorium plant flushing and equipment installation began April 13, 1966, after completion of cold feed processing and was still in progress at month end. The production of UNH was 107.2% of forecast.

The product quality for the uranium and plutonium was excellent during the month. Neptunium run #5-66 was purified and 932 grams were shipped. Another shipment, containing 408 grams made up from partial bottles from four separate runs, was also made during the month.

The three dissolvers and the other plant equipment are being flushed to remove uranium in preparation for thorium processing. "A" dissolver flushing has been in progress since March 30, and "B" and "C" dissolvers since April 10. The "C" dissolver downdraft condenser and knockout pot have been removed and replaced by a cleaned, hot spare. Flush solutions have been boiled in "A" dissolver a total of 595 hours. The uranium dissolution rate was five grams per hour at month end. The "B" dissolver was boiled for 543 hours, and the uranium dissolution rate was three grams per hour. The "C" dissolver was boiled 453 hours, and the uranium dissolution rate was 24 grams per hour. Comparison of the current results with those obtained during the test run is difficult because of changes in equipment and procedures. The "C" dissolver equipment is essentially the same as the test run. The uranium pick-up rate currently observed is about a factor of two lower than before. This is attributable to two changed procedures: (1) removing the dissolver tower for decontamination in separate facilities, and (2) changing flush solutions less so that they will be boiling a greater percent of the time. The design of the new annular dissolvers has simplified flushing operations. A uranium pick-up rate of 1.0 grams/hour or less is desired in the dissolvers before charging thorium. The dissolver rinse and head-end fission product tanks and equipment are being flushed with the spent dissolver solutions. The backcycle waste, HA column feed, feed makeup, and metal solution tanks are being flushed separately. The various tanks, columns, and concentrators in the solvent extraction system have been flushed with caustic-tartrate, oxalic-nitric acid, and dilute nitric acid solutions. With the exception of the H⁴ concentrator and the backcycle waste receiver tank, the system is just about at the desired uranium pick-up rate.

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PUREX (continued)

The G5-2 (#1 organic system) pump was replaced.

The F-12 (acid waste feed) pump failed and was replaced.

The U5-2 acid pump was replaced with a reconditioned, hot spare after about four years' service. The failed unit was overhauled and will be used as a spare.

The #1 air sampling vacuum pump failed, was repaired, and returned to service.

The F8 (waste rework) agitator failed and was replaced with a new, slow speed (285 rpm) unit.

The F5 (acid rectifier) chloride purge jumper leaked at the nozzle so the jumper was removed and the nozzle blanked.

The IBXF DOV, 1BXF rotameter, 2AX, 2D column interface float, IC column interface, and 2E column interface jumpers (6) were replaced.

About eighty per cent of the new thorium jumpers were installed.

The HS column pulse generator failed and will be replaced.

The final plutonium concentrator and accumulator were replaced in the N-Cell hood.

Iodine emission was:

Month's total	8.75 curies
Daily Average	.29 curie
Maximum Daily	2.06 curies

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REDOX

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Production of enriched UNH was 94.5% of forecast. Dissolution and solvent extraction of the first production campaign of Zircaloy clad N-Reactor 94 fuel was completed during the month and dissolution and solvent extraction of N-Reactor 125 "spike" fuel was started April 19, 1966.

Processing Experience

Process performance data and operations experience are summarized below:

1. Over-all dissolution capacity continued to be about one-third lower than expected due to increased fuel handling and Zircaloy dissolution time. Zircaloy decladding, using 5.5 M NH_4F - 0.5 M HN_4NO_3 (ammonium fluoride-ammonium nitrate) and a 7 to 1 fluoride to zirconium mole ratio, was not complete after an 18-hour digest. Procedures were adjusted to a 12-hour decladding step leaving a Zircaloy heel, most of which is dissolved during the decladding of the subsequent charge. On occasion however, an additional Zircaloy dissolution has been required to remove the Zircaloy heel built up on the preceding 7 or 8 charges.
2. Sparging of Zircaloy waste during transfer from the dissolver was necessary to remove zirconium solids (fluozirconates and/or zirconium oxide). Zirconium content of first cut uranium dissolutions was reduced from 10 g/l to <1 g/l when the Zircaloy waste transfers include sparging. No abnormal waste losses were attributed to this procedure change.
3. Product losses to canister decladding, Zircaloy decladding and solvent extraction wastes were:

	<u>Waste Losses Percent</u>	
	<u>Pu</u>	<u>U</u>
Canister Dissolution	0.03	0.05
Zircaloy Decladding	0.45	0.73
Solvent Extraction Wastes	0.20	-.03

Losses to Zircaloy decladding were higher than those experienced with aluminum-clad 94 metal processing; however, solvent extraction losses are comparable to regular 94 metal processing.

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REDOX (continued)

Processing Experience (continued)

4. The plugging of the waste transfer line to the underground storage tank occurred on two occasions. This problem appears to have been solved by re-routing the wastes through an alternate line.

Dissolution of a production campaign of Zircaloy-clad N-Reactor 125 "spike" fuel was started on April 19, 1966. Dissolution of this fuel requires a three-step dissolution flowsheet similar to that used for N-Reactor 94 fuel:

1. Aluminum canister dissolution.
2. Zircaloy cladding dissolution.
3. Uranium core dissolution.

The quantity of uranium dissolved in each dissolver cut is calculated from volume measurements and uranium analyses. Uranium losses to canister and cladding dissolutions are similarly determined. These data and the charging records are used to calculate the heel size to determine the maximum quantity of new metal that can be charged without exceeding critical mass specifications. Normally the quantity of zirconium cladding on the new metal limits the charge size below that allowed for critical mass considerations.

On April 10, 1966, the pressure drop across the ventilation sand filter system exceeded normal operating levels, causing a 30-hour plant outage. Investigation revealed the problem resulted from ammonium nitrate being deposited on the prototype charcoal filter that had been installed in series with the sand filter some years ago. The charcoal bed was lifted from its normal position, effecting a bypass. Normal air flow was resumed without loss of filter effectiveness, and processing was resumed. Rerouting of the H-10 (coating waste tank) vent line via the C-2 dissolver, a condenser, two fiberglass filters, and to the ventilation stack has removed the major source of ammonia from the sand filter system.

One other short outage was sustained when the E-13 ozonator pressurized during an oxidation step and blew a small quantity (<1 qt.) of low level contamination into the operating gallery. Decontamination and cleanup were achieved without incident.

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REDOX (continued)

Maintenance Experience

There were three pump failures during the month: the E-5 (2A column feed) pump became erratic and was replaced with a repaired unit. The other two failures involved the pump on the D-14 (backcycle concentrator). The problem was with the electrical system. After brushing the contacts and checking the wiring system on the cold side, the problem appears to have been corrected. The leaking D-12 (waste concentrator) east heat exchanger was replaced with a helical type. Installation and sealing were greatly facilitated by the C-clamp receptacles which are now standard design for all heat exchangers.

Radiation Experience

Average Daily	0.013 Ci
Daily Maximum	0.05 Ci
Total for Month	0.382 Ci

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UO₃

Processing of normal and enriched oxide was 98.6 % and 141 % of forecast, respectively. There were 448 drums loaded with depleted normal uranium oxide during the month for a total of 15,327. This constitutes approximately 11,723 tons of depleted normal uranium in storage. There were 6 carloads of enriched oxide shipped to Fernald.

Processing was intermittent throughout the month, as several brief outages were experienced on individual calciners due to minor mechanical difficulties, loss of vacuum, and periods when UNH feed supplies were low. The month ended with all normal UNH feed being processed and a low inventory of N-Reactor UNH to be processed. "J", "K", "L", and "M" calciners will remain down until the end of July 1966, when Purex again starts processing normal uranium. "G" and "H" calciners will operate on enriched feed produced by Redox plant.

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PLUTONIUM FINISHING 234-5

The production of plutonium metal was 90% of forecast.

Processing Experience

Button production operations were scheduled in accordance with feed availability, except for the period April 11 through 17 when the line was idled while studies were made of an alleged discrepancy on Pu accountability between Purex and Plutonium Finishing operations. Maintenance problems were routine, consisting of difficulties with the fluorinator vacuum and with plugged filtrate lines. The button line operated 40% of the available time on a five-day week schedule. The average rate while operating was 1.44 kg per hour. Reduction yields were excellent, averaging 98.5% recovery on the plutonium.

Product Quality

Plutonium metal quality continued high with only 577 ppm impurities on the average. There were no buttons rejected for low density and no buttons rejected due to high impurities. The average metal density was 19.45 g/cc.

Nonweapons Plutonium

Beginning April 25, the weapons grade plutonium was cleaned from the button line and processing of nonweapons plutonium was initiated. Incomplete data on the nominal 12% Pu-240 campaign indicate the Pu-240 content to be approximately 11.5%.

Shipments of nonweapons plutonium included 333 grams of plutonium as metal and 867 grams of plutonium as oxide to Battelle-Ohio; 597 grams of plutonium as nitrate to Babcock and Wilcox; and 205 grams plutonium as nitrate to General Electric Company, Vallecitos. The latter two shipments and a previous shipment of plutonium nitrate to Battelle-Ohio, plus a previous nitrate shipment to General Electric at Vallecitos were troublesome in that the inner vessel leaked. In all four cases, the 2R vessel of the shipping container assured containment of the plutonium. Plastic bottle leakage apparently occurred at the cap. The problem and a solution are being investigated.

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PLUTONIUM RECLAMATION

Dissolution of slag and crucible cans reached a new record this month. The total plutonium recovered and returned to the production line was 69.5 kg. The scrap backlog at month end was 861.7 kg of plutonium.

Plutonium Reclamation Facility

The Plutonium Reclamation Facility continued on a seven-day week schedule. There were 1,716 cans of slag and crucible material containing 52 kg of plutonium dissolved and recovered. The O6 dissolver operated all month except for minor down time due to plugging of the condenser off-gas line. On the O8 dissolver, the upper ball valve actuator failed and the unit was removed from service, repaired, and readied for reinstallation. The O4 dissolver was installed, and it operated satisfactorily for the last half of the month. A process test was initiated interspersing magnesia fabrication line crucible fragments among the cans of slag and crucible materials.

The increased operating schedule was evident in the failure of 27 pumps which were removed from service: 24 were repaired and returned to service, whereas 3 were beyond repair and were buried. The breakdown is tabulated below:

<u>No.</u>	<u>Make</u>	<u>Type of Failure</u>
18	Deanline	Mechanical seals failed.
2	Deanline	Larger impellers were installed for testing.
3	Chempump	Holes in case of each, buried.
1	Chempump	Pump bearings replaced.
2	Deanline	Motor bearings replaced.
1	Deanline	Bad electrical connector.

There were 25 new mechanical seals received which are being installed on the pumps as failures require. These seals were discussed in the report for January.

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PLUTONIUM RECLAMATION (continued)

Waste Treatment

Treatment of 122,585 liters of aqueous waste yielded 77 grams of Pu. An additional 24 grams of Pu were recovered from two batches of solution from the waste sump. There were 22.6 grams of americium-241 recovered in the waste treatment facility.

Incinerator Building

This facility was not operated. Fabrication is 95% complete on replacement parts for the incinerator facility. On April 28, the fabricator began trial assembly of the new incinerator furnace components. There are now 585 boxes of dry waste awaiting processing.

Recovery of Plutonium from Fabrication Oil

Revisions to hood HC-30 were completed and a process test initiated to recover plutonium associated with the stored Fab oil (lard oil plus carbon tetrachloride). The objective is to recover the plutonium from the Fab oil without fouling the organic solvent of the plutonium reclamation facility and causing organic losses. The first test with nitric acid and 164.2 grams of Pu in Fab oil resulted in the following: 81.0 grams recovered, 17.5 grams (10.8%) into cribbed waste; 32 grams remaining in the HC-30 tank and 33.7 grams unaccounted for, or in the 19 unrinsed cans in which the Fab oil had been stored. Development work continues.

Recuplex Cleanup

Continued alternate acid and organic cleanouts of old Recuplex vessels have reduced the plutonium in two of the hoods to the point that further recovery is uneconomical. The R&B hood cleanout has yielded 9,928 grams Pu. Gamma and neutron counting instrumentation indicate that the tanks now contain only 350 grams, whereas approximately 500 grams are on the outside of tanks, on lucite panels, supports, services, etc. The SE hood has yielded 18,717 grams Pu and the counting instrumentation indicates 1400 grams remain scattered throughout the hood. Solutions are now being removed from the Slag and Crucible Hood as the cleanup operation continues.

Miscellaneous Plutonium Scrap

Hood 227 in the 234-5 Building was prepared for test burning of the polystyrene-Pu cubes requiring recovery after use in the Critical Mass Laboratory.

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PLUTONIUM RECLAMATION (continued)

Miscellaneous Plutonium Scrap (continued)

Packaging for shipment of the Pu-U scrap scheduled for recovery at the NFS Plant in Erwin, Tennessee, has been discussed at length with Isochem. Richland has not been notified by NFS of a desired shipping date.

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FISSION PRODUCTS

Purex head end fission product operated during the first part of the month to recover 630 kci of Pm-147 and 140 kci of Sr-90. Processing was discontinued on April 12 to enable 1WW loadouts to Pacific Northwest Laboratory.

At the Strontium Semiworks, promethium processing was continued for the removal of nonradioactive inerts. Flowsheet and plant equipment performance continued to be excellent. More than one megacurie of promethium-147 was successfully processed during the report period by the D2EHFA (di-2-ethylhexyl phosphoric acid) solvent extraction flowsheet. Lead and sodium decontamination factors were 3,000 and 1,000, respectively. An estimate of the plutonium content based upon recent laboratory analysis is 4×10^{-7} molar, or 0.06 g Pu/mci promethium. The total amount of Pm-147 (2.5 year aged) processed during the past two months is 3,476 kci, all of which will be shipped to PNL.

Cask Loadings and Shipments

Two 300-gallon batches of centrifuged 1WW and one 30-gallon batch of uncentrifuged 1WW were shipped to PNL during the month. Two shipments of Pm-147, 1,250 kci (2.5 year aged) and 550 kci (2 year aged), were shipped to PNL during April. STT casks 46 and 47, loaded with 45,000 and 40,000 curies of Cs-137, respectively, were shipped to ORNL on April 8, 1966. The HAPO IB-1 and HAPO II-2 casks were loaded during the month and will be shipped to ORNL and Quehanna, respectively, early in May.

Fission Product Inventory (kci)

	<u>CR Vault</u>	<u>B-Plant</u>	<u>SSW</u>	<u>Casks</u>	<u>Total</u>
Sr-90	1,580	2,190	1,320	490 ⁽¹⁾	5,580
Pm-147 (unaged)	3,440	13,460	1,170 ⁽²⁾	0	18,070
Pm-147 (aged)	0	0	0	0	0

(1) 340 kci in HAPO IB-1; 150 kci in HAPO II-2.

(2) Decay adjustment of 60 kci and processing loss of 20 kci.

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WASTE MANAGEMENT ACTIVITIES

The following table shows the April 30, 1966, status of waste storage tanks that have leaked or are suspected leakers.

<u>Tank</u>	<u>Lateral</u>	<u>Radiation C/M</u>	<u>Liquid Depth (in)</u>	<u>Sludge Depth (in)</u>	<u>Average Temp (C°)</u>
107 SX	S68°15'E	1.7 x 10 ⁶	149.75	14	136.6
108 SX	S41°50'E S59°18'E S77°15'E	1.2 x 10 ⁶ 96 x 10 ⁴ 96 x 10 ⁴	241	20	138.9
109 SX	S20°45'W S45°0'W S69°15'W	3 x 10 ⁴ 57 x 10 ³	223.5	72	115.6
105 A	No. 3	225 x 10 ³	312	24	110
Test Well	No. 29SX	52 x 10 ⁴			

Individual tank inventories and tank farm inventory measurements indicate no measurable loss of material.

A gradual but sustained temperature increase has been observed in waste tank 103 AX. This tank has been receiving current Purex waste since February 1965. Several thermocouples which measure the sludge temperature have indicated temperatures of 200° C, with one point at 270° C. The calculated supernatant solution sodium molarity is 5.2, and tank contents 73% of the amount scheduled for this tank. Following the temperature increase, additional circulators were placed in service, and the tank contents were diluted as allowed by remaining tank volume. The sodium concentration was reduced to 4.9 molar by this addition. It appears the temperature increase was caused by the settling of sludge, for at month end, after the additional circulators had been placed in operation and the tank contents had been diluted, the temperature increase had been arrested, and a few of the thermocouple readings are trending downward.

In-Tank Solidification

During April, 94,000 gallons of underground waste storage space were recovered by this unit. Total space recovered by the unit to date is 1,911,000 gallons. The Teflon de-entrainer pad installed on the

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WASTE MANAGEMENT ACTIVITIES (continued)

In-Tank Solidification (continued)

condenser outlet is removing about 95% of the particulate matter entrained with the off-gas. High gas velocity through the Teflon pad prevents de-entrained liquid from draining and flushing away accumulated particulate matter. Under these conditions, frequent shutdowns of the ITS unit for de-entrainer flushing have been necessary. To correct this problem a second Teflon pad has been installed in parallel with the first. This has reduced gas velocity and consequently reduced flushing frequency. Valving has been installed so that either pad can be flushed without shutting down the unit. This modification has permitted increased air flow rates with subsequent increased boil-off capacity. It now appears the limiting section of the unit is the air circulator which exhibits a steady back pressure increase. This requires periodic water flushing.

Laboratory studies on the failed low power density prototype immersion electric heater to define Inconel 600 corrosion rates have been completed. These studies indicate that Inconel 600 is a satisfactory sheathing material for these types of heaters. Corrosion rates were determined in both synthetic and actual stored waste solutions. The tests evaluated Inconel 600 as immersed coupons and as a submerged heat transfer surface with and without sparging. Corrosion rates determined indicate equipment life estimates on the order of 10 years. Based on these experiments, it has been determined the GE low power density immersion heater failed at temperatures in excess of 800° C after the draft tube became plugged with salt.

242-T Waste Evaporator

The 242-T evaporator operated at an on-line efficiency of 87% at an average boil-off rate of 3.4 gpm while operating. Lower than normal rates are necessary in order to maintain the temperature of the bottoms tanks within operating specifications. Design of a tank cooling system is essentially complete and installation of equipment is currently under way. During April, this unit recovered 129,000 gallons of storage space for a total to date of 1,151,313 gallons.

Waste Management Computer Program

A computer code to be used in the Waste Management Program to account for fission products, waste tank transfers, sludge formation, tank levels, volumes, and concentrations is near completion. A sample run has been successfully completed with very satisfactory results. Work is now continuing on selecting and evaluating the types of input data that is felt will be useful when the program is placed in actual operation.

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CHEMICAL RESEARCH AND DEVELOPMENT

Thoria and U-233

When thoria is sintered at 1050 C, the resulting pellet density is dependent both on the sodium concentration and on the density of the cold pressed pellet. There appears to be a reaction between the Na_2O and the ThO_2 during sintering. Sodium content appears to affect both the physical character of the powder and the sintering reaction, while the method of calcination (spray versus pot, etc.) affects only the physical characteristics.

Distribution coefficients for thorium were measured under higher acid concentrations than previously studied for the imminent Purex Plant Thorex campaign. Other flowsheet testing confirmed that residual thorium can be removed from the U-233 stream by cation exchange, but the absorbed thorium will not readily elute if allowed to remain long on the resin.

The first denitration of a thorium nitrate solution was performed in the four-inch diameter agitated trough calciner. Eight pounds of thorium oxide were produced and the oxide had good dispersion characteristics when treated by the standard Sol-Gel process.

Bismuth-Polonium Separation

Exploratory tests were made on the chemical stability of dibutyl carbitol (DBC). Storage of DBC containing extracted HNO_3 of 0.5 to 2.0 M showed no loss of polonium extraction capability after ten hours at 50° C. However, after 24 hours at this temperature, polonium distribution ratios had decreased by a factor of about 20. Other solutions of DBC containing HNO_3 , Po and Bi showed formation of unidentified solids. Studies continue to identify the reactions involved and purification of the DBC.

In the reduction of Bi_2O_3 to metallic bismuth by carbon monoxide, it was found the Cr_2O_3 and Fe_2O_3 were not reduced to the metal and were concentrating at the top of the Bi column. Some of the molten Bi filtered through the quartz frit. Nickel oxide was found to act similar to the chromium and iron oxides, but in the case of NiO , approximately half was reduced to metal.

Extraction of Neptunium and Plutonium from Purex Acid Waste

The use of 25% dibutylbutyl phosphonate in normal paraffin hydrocarbon and 0.1 M oxalic acid as strip solution are being studied for recovery of Np and Pu from Purex process wastes. Tests were made on the possible effects of 6-8% of the Zr-Nb-95 and the 10-14%

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CHEMICAL RESEARCH AND DEVELOPMENT (continued)

Extraction of Neptunium and Plutonium from Purex Acid Waste (continued)

of the Ce-Pr-144 activity which follow the Np and Pu in this process. Irradiation of the solvent to as much as 10^6 rad did not increase fission product extraction. Tracer level experiments with simulated 1W indicated most of the Ce-Pr activity was removed from the aqueous phase by scrubbing with HNO_3 while less than 1% of the Np and Pu were removed. It was also shown that all of the Th-134 was extracted, but did not strip with the Np and Pu into the 0.1 M oxalic acid. This thorium was removed from the solvent by contact with 10% sodium carbonate solution.

Rhodium and Palladium Recovery

Forty gallons of Purex tank farm supernate were moved to the 325-A Building hot cell to determine the ultimate capacity of a resin for Rh and Pd during cycles of repeated loading and elution of technetium. Two technetium cycles have been completed. After the third cycle, the resin is to be removed and analyzed for Rh and Pd.

Failure of In-Tank Solidification Immersion Units

The failure of immersion heating elements under test in the in-tank solidification program raised questions on the use of Inconel 600 for sheathing. In some areas, the sheathing had melted; in other areas, severe pitting had occurred. The melting was explained by the formation of solids preventing circulation of the liquid past the heating element. Corrosion tests made on additional samples of Inconel-600 under various corrosive conditions confirmed previous tests that this alloy should be satisfactory. The maximum corrosion rate under the various conditions tested ranged from .5 to 4 mils per month.

Americium Purification

The D2EHPA solvent extraction flowsheet was further tested for application to americium purification for the Am-241 recovered from the Z-Plant waste treatment process. One series of experiments was performed to define the stripping behavior of cerium, americium, iron, and magnesium as affected by the extraction pH and whether pH adjustment was with NaOH or NH_4OH .

In laboratory work on a liter scale, the purification extraction step was successfully demonstrated. Product concentration was satisfactory at a factor of 3, but when concentrated by a factor of 5, the solution was syrupy and unmanageable. The demonstration included precipitation as the oxalate, filtration, ignition at 800 C for five hours, and impurity analysis which indicated product purity of greater than 99%.

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