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RL - GEN - 888 6

JULY 10, 1966

HANFORD ATOMIC PRODUCTS DEPARTMENT
MONTHLY REPORT

JUNE 1966

GE-AEC BUSINESS RECORD

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By Jerri Maly, 5-10-94
Verified By DK Schmitt, 5-11-94

HANFORD ATOMIC PRODUCTS DEPARTMENT

MONTHLY REPORT - JUNE 1966

R. L. Dickeman

R. L. Dickeman
General Manager

Work performed under Contract No. AT (45-1) - 1350 between
the Atomic Energy Commission and General Electric Company.

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SUMMARY

The programmed long outage which started May 1 was concluded on June 6. During this outage, the sixth heat exchanger cell was tied in, the auxiliary steam boiler was completely overhauled, four steam generators were equipped with isolation valves, extensive replacements of the process tube resistance temperature detectors were made, the reactor was re-orificed, a leaking reactor gas seal was repaired, extensive preventive maintenance was done, and numerous smaller pieces of work were accomplished.

After startup, testing of the WPPSS generating station equipment was resumed. Electric power up to 320 Mw was generated, generator trip-offs were executed, and data on the reactor plant response were taken. Both the generating station and the reactor plant behaved excellently. The sixth cell installations performed as designed. Electrical power generation is now limited to about 400 Mw by the unavailability of a step down transformer on the BPA system at Vantage. Elevation of the generated power level to the rated 800 Mw will be possible in the early autumn when transformer delivery is expected by BPA.

Three scrams from relatively low power levels occurred during the June 6 startup, which may be regarded as being associated with returning the plant to service after a long outage. Later in the month, there was an additional scheduled scram (from 3000 Mw, to check 6-cell operation) and six other unscheduled scrams due to other causes, as follows:

- a. A BPA power surge, induced by a local electrical storm
- b. A fuel element rupture (No. 20 since plant startup)
- c. Loss of coolant to a control rod
- d. A fuel element rupture (No. 21)
- e. Manual scram due to indicated loss of flow monitor master logic
- f. Loss of No. 4 circulating raw water pump due to failure of motor excitation

Production availability was 71.6%; time operated efficiency was 52.3%.

Preparations for the two-load coproduct demonstration continued on the schedule which calls for loading to start in September 1966. In-pile testing of coproduct assemblies has shown a tendency of some of the target supports to cause fretting corrosion of the driver cladding. A number of corrective measures are being tested.

About 4 kg of Np^{237} are under irradiation as a demonstration of Pu^{238} production in N-Reactor. The test is proceeding uneventfully.

Plant operations were interrupted on June 2 and 3 by a labor dispute over manning of the auxiliary boiler plant electrical control console. Operation was resumed quickly, but settlement of the dispute will require additional time. Preparations were made also to cope with another possible interruption in July, expected to arise from disputes between the Hanford Atomic Metal Trades Council and other AEC contractors at Richland.

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OPERATING STATISTICS

In-reactor KMWD residue	573
Highest exposure tube (Mwd/t)	4,015
Input production Pu (Kmw)	44,941
Input production coproduct (Kmw)	1.28 (Equivalent)
Maximum power level attained (Mw)	4,000
Average power level while operating (Mw)	2,862
Time operated efficiency (%)	52.3
Number of shutdowns	10
Scheduled	1*
Unscheduled	9

*3,000 Mw scram test N-4.

Reactor process data

Power at which computed (Mw)	3,800
Primary coolant inlet temp. (°F)	387
Primary coolant outlet temp. (°F)	509
Maximum moderator temperature (°F)	1,120
Maximum tube power (Kw) Tube #2245	5,000
Heat loss	
Heat to graphite cooling system (Mw)	147.7
Heat to shield cooling system (Mw)	2.285
Heat to rod cooling system	.426
Steam generation temperature (°F)	357
Main steam header pressure (psig)	132

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Dump condenser flow (gpm)	325,000						
Inlet temperature (°F)	56						
Outlet temperature (°F)	99.5						
Nuclear steam generating rate							
Maximum (1000's lb/hr)	12.8 X 10 ³						
Average while operating (1000's lb/hr)	9.9 X 10 ³						
Water collected from gas atmosphere	Neg.						
Demineralized water produced (gal/hr)	82,200						
Helium losses (cu. ft.)	888,604						
Fuel oil usage (bbls)	23,242						
Reactor flattening efficiency (%)	80						
Fuel conditions							
Heat flux maximum (BTU/hr/ft ²)	685,000						
Heat flux average (BTU/hr/ft ²)	475,000						
Cladding temperature							
Maximum (°F)	608						
Average (°F)	563						
Uranium temperature							
Maximum (°F)	898						
Average (°F)	803						
Fuel Failures	2						
Fuel Balance	0.95%	0.95%	1.25% Spike	Coproduct 1.96% CP 2.10% CP		0.71% Natural	Total
Fuel charged (tons)	0	+1.47	+ .37	0	0	0	+1.84
Fuel discharged (tons)	0	0	+ .37	0	-.79	0	+1.16
Net change (tons)	<u>0</u>	<u>+1.47</u>	<u>0</u>	<u>0</u>	<u>-.79</u>	<u>0</u>	<u>+.68</u>
Total in reactor at month end (tons)	<u>.73</u>	<u>291.57</u>	<u>64.94</u>	<u>2.03</u>	<u>3.45</u>	<u>1.09</u>	<u>363.81</u>

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Fuel element output (assemblies)	<u>June</u>	<u>May</u>	<u>CYTD</u> <u>Total</u>
0.95% tube-in-tube	153	2446	9094
0.95% single tube	-	-3*	33
1.25%-0.95% tube-in-tube	1299	72	2991
1.96% coproduct	0	0	0
1.25% coproduct	0	-10*	-10*
2.1% Mark E (outer, driver)	13	40	300
Total assemblies	1465	2445	12,408
Total tons	32.6	45.1	243.2
% of forecast	78.9	90.2	78.4
Uranium Utilization, Percent	80.1	74.9	70.3

*Fuels return for rework included some accepted in previous months.

2 6-30-66 1

		2A, 3A		1A, 4A		5A, 6A																												
100	95	90	85	80	75	70	65	60	55	50	45	40	35	30	25	20	15	10	5	EVEN														
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	41	42	43	44	45	46	47	48	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64	65	66	67	68	69	70	71	72	73	74
ODD	20	25	30	35	40	45	50	55	60	65	70	75	80	85	90	95	100																	
		2A, 3A		1A, 4A		5A, 6A																												

C = PT-57 Ceramic Target
S = Spike 1.25 % Enriched
F = PT-66 2.1 Coproduct (13 pcs)
G = PT-66 2.1 Coproduct (11 pcs)
H = PT-68 Np Target
R = 1.96% Coproduct
N = .712% Natural Fuel

B = Blank
M = Monitor Column
A = PT-NR-33 Thermocouples
D = PT-NR-32 1 Coupon
E = PT-NR-32 2 Coupon
T = Indicates tube has special T/C's
□ = Not filled in = Mark I Standard

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REACTOR PLANT OPERATION

OPERATING HISTORY

N-Reactor Plant operated with a Time Operated Efficiency (TOE) of 52.3 percent and a Production Availability of 71.6 percent.

The conversion outage and sixth cell tie-in outage begun in May was completed on June 6. During the startup from this outage, 3 scrams occurred due to failure to bypass a low level flux monitor, flow monitor leaking toggle valve, and drive turbine trip due to temporary loss of vacuum. These scrams accounted for 6.8 hours of outage time during the startup.

Seven additional outages occurred during the month. One was a planned scram from 3000 Mw to test the system's response with the sixth cell in operation; one scram was caused by a BPA power surge during a lightning storm; two outages were caused by fuel failures, one in a spike charge and the other in a coproduct charge. Scrams also occurred when cooling water was lost to a horizontal rod; when the flow monitor indicated a failure in the master trip logic (manual); and from the loss of No. 4 Circulating Raw Water pump due to failure of exciter current. Two fuel charges, each containing a rupture, were replaced during the month.

Converted plant testing was successfully conducted during the month to bring the WPPSS No. 1 turbine generator on the line and to evaluate the operation of the plant under Phase II conditions at a generating plant output of 400 Mw (electrical).

A labor dispute over manning of the plant service boiler building electrical control console resulted in a strike on June 2 and 3. A court injunction was obtained on June 3, permitting resumption of startup preparations.

The reactor and the converted plant operated successfully under Phase II conditions following the conclusion of the sixth loop tie-in outage on June 6. All testing of the six loop operation of the reactor and the converted plant was completed without difficulty up through the generation of 400 Mw (electrical) by the Supply System. The No. 1 turbine generator was successfully brought into operation and simultaneous operation of both machines has been initiated. Tests to demonstrate the effect of generator trips on the reactor as well as to demonstrate the effect of reactor scrams on the generator yielded expected results. One test in the current series remains to be completed, i.e., to demonstrate the effect on the plant of shedding the 400 Mw (electrical) load with the reactor at a reduced power level.

In addition to the scrams during startup and the planned scram on June 7 to test the effect of a scram on six loop operation, six additional unscheduled outages occurred during the month. On June 10, a scram was caused by a BPA power surge which resulted from the momentary loss of a line due to a

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lightning storm. The surge triggered a discontinuous action relay in the primary pump control circuit causing a flow monitor trip. A 25.5 hour outage resulted. A rupture occurred in tube 0846 on June 13 due to failure of a 1.25 enriched fuel element. Startup from this outage was delayed 16.5 hours to replace a broken stem in the cell 4 steam generator bypass valve. The total outage time was 44.5 hours. On June 16 a cooling water hose on control rod 47 became pinched and shut off coolant to the rod. Efforts to remove the rod from the reactor within the two minute time limit available were unsuccessful and a scram resulted. Some damage has occurred to the rod tip, and this rod is out of service pending replacement. On June 24 a rupture occurred in tube 2059 which contained a coproduct charge. An outage of 24.1 hours resulted. A manual scram was initiated on June 28 when an indicated failure of the master logic of the process tube flow monitor occurred. The failure was traced to the self check circuit. The reactor was shut down for 24.8 hours. A scram occurred on June 30 when erratic current to the rotor of No. 4 circulating raw water pump motor caused by pitted slip-rings caused the pump to shut down. The reactor was shut down for 5.1 hours and remained down at month end.

PLANT PERFORMANCE

The reactor operated at design power level or on restricted limits imposed by the WPPSS generator test program during the month.

The sixth cell has performed as expected during the month with no significant difficulties being experienced. Reactor process tube orificing to accommodate six loop operation is not yet complete and data taken during the month indicates that additional changes are in order to improve reactor coolant distribution among the process tubes.

Rear Face Gas Seal Repair

Repairs were made to the rear face gas seal during the May outage in an attempt to reduce helium gas losses. Pressure decay tests on the reactor after the repairs indicated a significant improvement; however, at about 50 percent power level following the startup, the gas leakage rate was noted to have increased substantially. Inspection after the subsequent shutdown revealed that a vertical expansion joint section had buckled and torn and a sealing joint in the shield had re-cracked. An additional crack was found in the rear face steel in the vicinity of the buckled expansion joint section. Differential thermal expansion was the apparent cause of the failure. Temporary repairs were made to reduce helium gas losses. Additional thermocouples were installed to determine the temperature gradients at the edges of the rear face and across the gas seal.

Changes have been made in the distribution and flow rates of shield cooling water to even out the outlet primary shield temperatures.

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Steam Generators

A complete inspection of all steam generators was completed. Repairs were made to 3A steam generator.

Plant Service Boiler

Following the completion of boiler modifications two small pinhole tube leaks occurred on June 3 along the right water wall in the firebox section as a result of additional thermal expansion in that area. The removal of the auxiliary burner from the right side caused this area to run hotter than it had before. These leaks occurred where the tubes were welded to the buck-stays. These tubes were repaired without difficulty and boiler operation has continued without further difficulty.

Horizontal Control Rods

The pinching of the exit cooling water hose on rod 47, previously referred to, deprived this rod of its normal coolant flow and, according to various indications, considerable overheating of the rod took place. Externally, the rod shows no indication of damage other than some slight warping of the nose of the rod and possibly a small hole where the nose is welded to the active portion of the rod. It is also thought that the O-ring seals between the rod tip and the inactive section of the rod have been blown out. The pinching of the hose resulted when slack due to hose elongation allowed the hose to sag into a sprocket. All other hoses have been inspected and measures taken to assure that a similar situation does not develop on any other rod.

The rod itself is out of service. After a period for radioactive decay, the rod will be inspected more closely, repaired, and returned to service.

Secondary System Controls

A rate action unit was installed on the main steam header pressure control system. Its purpose is to decrease the undershoot of the setpoint as the pressure returns to normal level following a turbine trip-off.

Flow Monitor Studies

Special instrumentation checks on tubes where flow indications were above or below the normal variation (without confirming temperature indications) were made during the May outage. In these cases, and in previous cases where similar checks have been made, transducer malfunctions were found. These investigations showed that air had not been adequately vented from many of

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the transducers originally installed. During subsequent pressure-depressure cycles, primary loop water surged back and forth in the sensing lines and carried foreign material into the assemblies. This interferes with the response and calibration of the flow monitor. Further inspection and the completion of any required maintenance or cleanup is expected to significantly improve flow-monitor operation.

OUTAGE EXPERIENCE

<u>Date</u>	<u>Outage Type</u>	<u>Duration - Hrs.</u>	<u>Reason</u>
6-1	Scheduled	122.1	Completion of the 6th cell outage.
6-6			Startup delayed by strike, and boiler leak repair.
6-6	Scram	0.5	During startup, when intermediate range flux monitor was not removed from service.
6-6	Manual	0.9	Loss of turbine during startup due to low vacuum, required shutdown to restore turbine to service.
6-6	Scram	5.5	Flow monitor trip due to toggle valve leak.
6-7	Scheduled	13.4	Scram test from 3000 Mw to check 6 loop operation.
6-10	Scram	25.5	BPA power surge.
6-13	Scram	44.5	Rupture, tube 0846, 1.25 enriched. Replaced stem in cell 4 steam generator bypass valve.
6-16	Scram	76.9	Loss of coolant to control rod 47.
6-24	Scram	24.1	Rupture, tube 2059, 2.1 enriched coproduct.
6-28	Scram	24.8	Manual scram with indicated loss of flow monitor master logic.
6-30	Scram	5.1	Loss of No. 4 CRW pump.

Figure 1 compares production availability and time operated efficiency over the last several months.

Production availability measures the extent to which N-Reactor approaches full-time production within internally controllable limitations. It excludes external conditions such as BPA failure, shutdown required by Washington Public Power Supply System, and other external causes.

Time Operated Efficiency (TOE) measures the extent to which the N-Reactor approaches full-time production regardless of shutdown causes.

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Figure 1. Production Availability and Time Operated Efficiency

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PROCESS PHYSICS

At 0830 on June 30, reactor power level was 3800 Mw; reactivity held in the rods was 9.84 mk (projecting to 6.3 mk at equilibrium); maximum graphite temperature was 1103 F; flattening efficiency was 81 percent. Maximum power level during the period was 4000 Mw.

Reactor Flattening and Loading Changes

During the recent 30-day operating period, the reactor flattening dropped off slightly to about 77 percent, due to fuel exposure buildup in the fringe channels. Thirty-five additional spike columns have been charged to correct this trend; and a flattening efficiency well in excess of 80 percent is expected under equilibrium conditions. Two fuel charges, each containing a rupture, were replaced during the month.

EQUIPMENT MAINTENANCE AND PLANT MODIFICATION

A summary of the major equipment and construction programs in progress or scheduled follows:

<u>Title</u>	<u>Est. Cost</u>	<u>Funds Assigned To G.E.</u>	<u>% Funds Expended</u>	<u>Status</u>
Proj. CGN-149 - Spec. Hand. & Exam. Equip. For Irrad. R&D Fuel	\$459,000	\$459,000	93	96% physically complete. The discharge area equipment is installed. Some electrical work is to be completed. Directive completion date 6-30-66.
Proj. CAN-168 - Ammonium Hydroxide Injection System - 184-N Building	45,000	1,500 (liaison only)	98	N-Reactor Plant accepted the project 6-2-66. Physical completion notices (AEC and GE) were issued.
Proj. CGN-182 - Replace- ment of 5 Kv Cable, 100-N	165,000	77,700 Capital 87,300 Expense	21	Extension of time to 3-31-67 approved by AEC. 85% cable received during June. Remainder to be shipped July 15.
Proj. CGN-184 - Phase I Coproduct Fuel-Target Fab. Equipment	203,300	203,300	62	98% physically complete. Beneficial use achieved. All major equipment received from offsite. Directive completion date is 7-15-66.
Proj. GCP-400 - Backup Boilers	865,000	---	--	Project proposal submitted and will be considered by RL00-AEC for FY-1967 period.
Proj. GAP-401 - Upgrading Fire Protection - 100-N	175,000	---	--	Project proposal under consideration by RL00-AEC for FY-1967 period
Proj. GCE-402, Phase II, Part A, Coproduct Equip., 105-N	165,000	165,000	10	Bids for the target stripper received 5-31-66. Review of bids indicated that specifications apparently limited flexibility in vendor's design. Specifications are being revised to provide greater degree of design flexibility and for new bids.
Proj. GCE-403 - Steam Generator Isolation Valves (4)	57,000	---	82	Installation of the four valves completed. Insulation will be completed next outage. Beneficial use attained.

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<u>Title</u>	<u>Est. Cost</u>	<u>Funds Assigned To G.E.</u>	<u>% Funds Expended</u>	<u>Status</u>
Proj. Proposal - Improved Irrad. Fuel Handling Equipment	\$275,000	\$ --	--	Proposal justification is being re-evaluated
CMJA-81 - 184-N Boiler Repair	100,000 (Rev. 1)	100,000	104	All repair work is completed.

Corrosion By-Pass Loops

Test loops are planned for all three of the recirculating aqueous systems. One corrosion test loop in the secondary system has been completed, and samples are being tested. Flow is being maintained at 10 to 12 fps. Additional test loops will be tied into the primary system and the graphite cooling system to support the corrosion monitoring program.

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IN-PLANT TESTS

The status of production tests is as follows:

PT No.	Document No.	Title	STATUS			
			<u>Approved</u>	<u>In Progress</u>	<u>Completed</u>	<u>Final Report</u>
2	HW-80369	Routine Graphite Sample Irradiations		X		
3	HW-84178	Routine Monitoring of Graphite Oxidation in N-Reactor		X		
4	RL-NRD-218	Evaluation of Monitor Column Fuel Elements in N-Reactor		X		
6	HW-81339	Isotope Producing Rods in N-Reactor		X		
8	HW-81327	Coproducer Demonstration Test (1.25)				X
9	HW-81609	Low Goal Irradiation of Fuel Elements with Varying Amounts of White Oxide on Surface				X
10	HW-82136	N-Reactor Corrosion Monitoring		X		
12	HW-82241	Copper Alloys in the N-Reactor Primary System		X		
13	RL-NRD-299	Coproducer Demonstration Test (1.96)		X		
14	HW-82385	Irradiation of Th-U Crud Monitor Elements in N-Reactor				X
14 SUP	HW-82385	Supplement to PT-14, Irradiation of 8.46 Inch Long Fuel Element Assemblies				X
24	HW-84095	Plant Service Boiler Base Load Reduction Test Program		X		

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PT No.	Document No.	Title	STATUS			
			<u>Approved</u>	<u>In Progress</u>	<u>Completed</u>	<u>Final Report</u>
24	RL-NRD-865	PT Change Authorization No. 3		X		
27	HW-84238	Steam Generator Moisture Carryover Detection			X	
27 SUP 1	HW-84238	Supplement 1 to PT-27			X	
30	HW-84232	Eval. of Ammonia for Con- trolling Oxygen in the Graphite Cooling System			X	
32	HW-84367	Exposure of Corrosion Cou- pons in N-Reactor Process Tubes		X		
33	HW-84401	In-Reactor Enthalpy Imbalance Measurements		X		
33 ADD2	HW-84401	Addendum 2 to PT-33, Eccentricity Experiments		X		
36	RL-NRD-55	Simulated Driver-Target Elements			X	
41	RL-NRD-297	Eval. of Direct Addition of Ammonia to Graphite Cooling System			X	
46	RL-NRD-427	Primary Drive Turbine Speed Reduction Tests			X	
48	RL-NRD-466	Corrosion Test Samples in End Caps of Graphite Cooling System		X		
55	RL-NRD-581	Eval. of Flow Nozzle Tap Relocation in the Steam Generator Steam and Feed- water Lines		X		
57	RL-NRD-657	Coproducer Target Element Evaluation		X		
57 SUP	RL-NRD-657	Supplement to PT-57			X	

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PT No.	Document No.	Title	STATUS			
			Approved	In Progress	Com- pleted	Final Report
57 SUP B	RL-NRD-657	Supplement B to PT-57			X	
61	RL-NRD-662	Eval. of Uncompensated Level Control for 109-N Steam Generators		X		
62	RL-NRD-668	Eval. of Single Pass Filtered Water Cooling	X			
63	RL-NRD-695	On-Reactor Tests for New Resistance Temp. Detectors		X		
64	RL-NRD-716	Exposure of Zr Coupons in N-Reactor Process Tubes	X			
65	RL-NRD-726	A Test to Demonstrate the Operability of the Rupture Monitor System with the Helical Sample Chamber Design	X			
66	RL-NRD-830	Coproducer Demonstration Test (2.1)		X		
68	RL-NRD-793	Pu-238 Demonstration Program		X		
69	RL-NRD-813	Irrad. of Tubular Zr-2 Samples in N-Reactor Gas Atmosphere	X			
70	RL-NRD-847	Measurement of Confinement System Filter Efficiency by Use of a Condensation Nuclei Detector	X			
76	RL-GEN-1018	Irradiation of 8" Fuel Ele- ments for Subsequent Fuel Failure Testing	X			

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FUEL MANUFACTURE

PRODUCTION STATISTICS

<u>Input (Extrusions)</u>	<u>June</u>	<u>May</u>	<u>CYTD</u>
0.95% Mark A (outer)	-	25	451
0.95% Mark N (inner)	-	46	526
0.95% Mark D (single tube)	-	0	19
1.25% Mark A (outer, spike)	-	159	258
2.1% Mark E (outer, driver)	176	0	226
Total extrusions	176	230	1480
Total tons	36.6	45.5	278.9
% of forecast	65.5	155.3	82.8
<u>Fuel element output (assemblies)</u>			
0.95% Tube-in-tube	153	2446	9094
0.95% Single tube	-	-3*	33
1.25%-0.95% Tube-in-tube	1299	72	2991
1.96% Coproduct	0	0	0
1.25% Coproduct	0	-10*	-10*
2.1% Mark E (outer, driver)	13	40	300
Total assemblies	1465	2445	12,408
Total tons	32.6	45.1	243.2
% of forecast	78.9	90.2	78.4
Uranium utilization, percent	80.1	74.9	70.3

*Fuels returned for rework included some accepted in previous months.

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FUEL MANUFACTURE

N-FUEL PRODUCTION SUMMARY

Conversion of the production operation to the fabrication of 2.1 percent enriched coproduct driver fuels was accomplished during the month. Although no finished production has been achieved, all processing operations were operating satisfactorily. Choice among alternative target supports is all that prevented some final assembly being made.

The revised nuclear safety requirements have required extensive modification to equipment and procedures throughout the operation. This work has now been completed and all equipment is operating satisfactorily within the revised limitations imposed by the nuclear safety specifications. The initial processing yield data are very encouraging. Of the material presently in progress, it is estimated that the Shop Adjusted Yield will be about 80 percent, and that the 85 percent level will be attainable early in this program.

Startup of all facilities for target fabrication was accomplished during the month. Some duplicate facilities (such as the third sintering furnace, and the second welder) required to meet full production throughputs are still not completed, but process flow is now according to the design plan.

PROCESS DEVELOPMENT

Extrusion Activities

One hundred aluminum billets were extruded into tubing for coproduct target element cladding. It has been necessary to extrude the aluminum oversize and allow approximately 0.050 inches of overstock for machining to ensure the required uniform surface and to correct wall thickness variations that occur during extrusion. Efforts are continuing to obtain the surface finishes and dimensional control needed for extrusion directly to finished size.

Seven Zircaloy extrusions were made to develop the process for fabricating coproduct driver inner Zircaloy cladding components on-site. The extrusions were made from the 7.600-inch liner using billets which were primarily extruded on-site from 11-inch ingots. Results of the test were generally good. Dimensional attributes were as follows:

Outer Diameter	2.708 $\pm$.005	2.785 $\pm$.008
Inner Diameter	1.925 $\pm$.005	1.914 $\pm$.007
Wall Thickness Variation (eccentricity)	.008	.013 $\pm$.013

The outer diameter was designed oversize to permit machining. A machined Zircaloy surface was planned for contact with the uranium during coextrusion. The slightly undersize inner diameter was designed to provide overstock for grit blasting and chemical etching prior to coextrusion.

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A total of 8000 pounds of lithium aluminate is available for production. All of this material will require on-site adjustment of Li^6 content by mixing natural powder and powder depleted in Li^6 . Union Carbide Nuclear Co. has been successful in producing approximately 15,000 pounds of lithium hydroxide at the design Li^6 level. This hydroxide will be shipped for conversion to LiAlO_2 during the first week of July.

Uranium Fabrication Development Activities

Eight N-inner fuel billets produced by the upset-forging process from the standard iron-silicon alloy and 14 N-outer fuel billets produced by direct casting to size were coextruded. Five of the direct cast billets were of standard iron-silicon alloy and nine of these billets were produced from the iron-aluminum alloy. The coextrusion results were all satisfactory. Seven of the upset-forged coextrusions and four of the direct-cast coextrusions were processed in the cladding thickness tester to obtain a measurement of cladding variation, and both processes demonstrated satisfactory control.

Support Fabrication

As noted in a later section of this report, a target support problem has developed in the coproduct program. A key characteristic is the spring-back of the support.

Spring-back measurements on target supports indicated that 100 percent recovery is obtained on both autoclaved and unautoclaved supports after assembly regardless of the source of strip (tubing or sheet-bar) used to fabricate the supports. Work has been initiated on support fabrication tooling to produce supports having a greater as-fabricated height to determine the maximum amount of spring-back which can be achieved and retained. Experience to date with interference-fit type supports indicates that additional spring-back may be made available by increasing as-fabricated height without detrimental galling occurring on the inner diameter of the driver.

Moisture Control in Ceramic Targets

Control of ceramic moisture content has been well within the 150 ppm limit. Moisture sampling to date has been on a worst-case basis, for example, the last piece of each lot to be processed through the controlled atmosphere glovebox was sampled since this piece had maximum time for moisture adsorption. Samples of all production processed through the Pilot Plant controlled atmosphere chamber have averaged 41 ppm with a range of 29 to 81 ppm. Samples of production recently processed through the glovebox chamber in the Target Fabrication Facility have averaged 47 ppm with a range from 13 to 50 ppm. The above experience demonstrates the effectiveness of this method for the control moisture in the ceramic pellet. Efforts to develop hot assembly as a backup process for moisture control have therefore been discontinued.

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Hollow Pellet Fabrication

Capability to fabricate hollow pellets with existing pellet fabrication equipment has been demonstrated. Approximately five pounds of pellets were pressed and sintered to 1.1-inch outer diameter by 0.6-inch inner diameter with a final density of 78 percent of theoretical.

The present facility can, therefore, be used to produce targets requiring higher void volumes if necessary to accommodate increased gas volume (GVR) associated with higher burnup.

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CONVERSION

The plant was started up on June 6, 1966, after being shut down since May 1, 1966, for tie-in of the sixth cell. Prior to startup, all necessary pre-operational testing was accomplished with satisfactory results.

Following the reactor startup, testing of the sixth cell as specified in the N-4 Test Program was begun and, as a part of the testing, the reactor was scrammed from 3000 Mw on June 7. The reactor was restarted on June 8 and, after determination that the results of the sixth cell testing were satisfactory, the initial roll of Supply System turbine-generator Unit No. 1 and pre-load testing was started. To date, the No. 1 turbine-generator unit has been tested to the 200 Mwe level, including a programmed trip and the No. 1 and 2 units, which have been tested in combination to the 320 Mwe level including a non-scheduled trip because of a reactor scram for other reasons.

One test in the current series remains to be done, i.e., to demonstrate the effect on the Plant of shedding the 400 Mwe load with the reactor at reduced power level.

The second phase of the test series will be completed during October or November, 1966, when the Vantage 500 kv autotransformers are scheduled for completion. At that time, N-4 testing will be resumed and carried out through the 800 Mwe level. In the interim, the Generating Plant will be limited to approximately 400 Mwe total output.

Plant system transients during the 320 Mwe trip were well within predicted ranges. The test involving pressure increase to 130 psig in the main steam header and reactor scram test from 3000 Mw produced no problems. Results from the scram were very satisfactory, with only a 7.6 foot drop in pressurizer level and a 4.0 foot drop in surge tank level. Main steam pressure reduction was only 5 psi. These transients were all less severe than predicted by simulation studies.

The requirements for six cell operation were accomplished on the reorificing program during the May outage. The changes that were made were successful from all standpoints. No mechanical problems were encountered, subsequent flow checks were satisfactory, limits were applied as expected, and tube outlet temperatures are in acceptable ranges. A review of the rest of the program based on current operating data was made; no changes in the program appear to be necessary at this time.

TECHNICAL ACTIVITIES

1. BASIC PRODUCTION MISSION

1.1 Fuel Element Performance

a. Fuel Element Failure Number 2

Fuel element failure number 20 occurred on June 13 in process channel number 0846. The failed element was a 15-inch 1.25 enriched spike outer, the fifth element in the column. It had been charged in November, 1965, and had reached 1736 Mwd/t (column average was 1504 Mwd/t). This was the first spike failure at N-Reactor. The element was broken approximately in half. One half was intact, but cracked. It was somewhat swollen and had bumps on the cladding. The inner member was still locked to this half and was sticking out of it. The separated half was broken into several pieces. About 75 percent of them have been recovered. One piece, a triangular sandwich of Zircaloy and uranium, has what appears to be a buggy spring fretting mark along one edge. The pieces of this element will be shipped to Battelle Northwest Radiometallurgy Laboratory for more detailed analysis.

Evidence indicates that the extensive damage to the element is a result of the fall into the basin. No evidence of tube damage was observed when the process tube was borescoped.

b. Fuel Element Failure Number 21

Fuel element failure number 21 occurred on June 24 in process channel number 2059. The failed element, a 26.5-inch coproduct driver, was the fourth element in the column. It had been charged during the May outage. The driver was fretted in two places at the unlocked end of the element. One of the hard-sized buggy springs on the corresponding target element was found to be broken. The crown was gone completely. No fretting marks were found on the other 12 elements.

<u>Failure</u> <u>Number</u>	<u>Tube</u> <u>Number</u>	<u>Date of</u> <u>Failure</u>	<u>Tube</u> <u>Power</u> <u>(kw)</u>	<u>Exposure</u> <u>Tube</u> <u>(Mwd/t)</u>	<u>Element</u> <u>(Mwd/t)</u>	<u>Fuel</u> <u>Type</u>	<u>Type of</u> <u>Failure</u>	<u>Prod.</u> <u>Loss</u> <u>Mw</u>	<u>Hour</u> <u>Loss</u>
20	0846	6-13-66	3500	1504	1736	Mark IA	Not Outer Defined	3546	24
21	2059	6-24-66	3815	166	226	Mark II	Inner Driver Cladding Fretted	4240	24.1

Discussion of the behavior of the coproduct target supports appears later in this report.

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1.2 Process Tube Monitoring Program

As reported last month, the TV camera examination of 16 process tubes indicated that fuel element spacer (dummy) chattering had occurred in the downstream portion of the tubes. The damage to the tube wall is not considered significant at this time since it apparently progresses very slowly and ample time for the development of corrective measures is available. Based on previous testing of tube sections of the Plutonium Recycle Test Reactor (PRTR) with deeper defects of similar nature, no appreciable shortening of tube life is anticipated.

Tests have provided conclusive evidence, however, that chattering does occur under typical downstream conditions and have shown that at least one spacer geometry change eliminates chattering at flow rates of interest. Other tests are being formulated to explore experimentally a range of pertinent spacer geometry variables in the Primary Coolant Experimental facility with the intent of establishing acceptable configuration combinations. These include measures to design a spacer with a larger contact area provided on the spacer supports. Further studies and tests seek to incorporate design improvements to induce more efficient flow characteristics and reduce any turbulence that might contribute to spacer instability.

As reported in May, a process tube was removed as part of a planned inspection program. A burst test sample was prepared from the tube section just downstream from the front rolled joint, and initial attempts to burst it failed. The maximum pressure attainable with the existing pump is 20,000 psi. This is equivalent to a stress of 110,000 psi. All previous unirradiated samples have failed between 18,000 and 19,000 psi pressure. The reason for the higher burst strength is under investigation since the amount of radiation received by this section of the tube should not have caused an appreciable increase in ultimate strength.

1.3 Primary Coolant System Strainers

An engineering study has been made of a relatively compact permanent strainer for the N-Reactor primary coolant system which will effectively trap all particles over .008-inch in diameter, including long needle-like pieces. The proposed strainer design is a self-cleaning vertical-flow type using system pressure as the actuating force for a continuous self-cleaning action. The basic filtering unit or cartridge is a well-known commercial assembly with a long-established reputation for reliability. The .008-inch degree of filtration was used in this study because it is the largest spacing commercially available in the desired design. However, a spacing of .012-inch or .015-inch may be made available in the quantities we would require. The larger spacing would reduce the over-all cost of these strainers, as fewer cartridges would be needed for similar flow conditions.

The primary justification for the strainers is to minimize the possibility of fuel failures from fretting corrosion caused by foreign particles in the coolant stream.

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Two possible locations are being considered: (a) in the 24-inch discharge line downstream of each of the six primary pumps, or (b) in the 18-inch reactor inlet header upstream of each of the 16 check valves in the primary coolant supply lines. Costs are being obtained for both approaches.

1.4 Increased Power Level

Studies analyzing the potential for increasing the power level of the N-Reactor indicate the following two areas where emphasis should be given:

- a. increased reliability and testability of the fog spray system, and
- b. improved techniques for the control and monitoring of pile gas.

Improvements in the first area would be made to accommodate the increases in specific fuel power. Improvements in the second area would be made to accommodate the higher graphite and process tube temperatures which have been calculated. Current planning is directed towards a power level increase starting early in 1967.

1.5 Physics Studies - Conversion Ratio Data

The results of product yield and isotopic ratio measurements on the first campaign Mark I fuel (2100 Mwd/t) have been received from Isochem. The data are listed below, and are compared to values calculated by the FLEX and MOFDA computer codes:

Conversion Ratio Data - Mark I Fuel

<u>Isotope</u>	<u>Analytical Results</u>	<u>MOFDA Code</u>	<u>FLEX-3R Code</u>	<u>Data In CAGE</u>
Pu-238	0.0552	0.450	0.0338	
Pu-239	85.834	85.791	84.132	
Pu-240	11.538	11.430	13.720	12.0
Pu-241	2.435	2.550	1.950	
Pu-242	0.193	0.184	0.164	
Np-237	15.45 gm/ton	12.60	10.60	19.32
U-235	0.736	0.732	0.735	0.740
U-236	0.0400	0.0375	0.0369	0.047
U-238	99.217	99.224	99.221	

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The differences between observed and calculated data are significant, and the MOFDA code seems to be more accurate, except for Pu²³⁸. The changes in effective cross sections which would cause FLEX to match the observed data and the causes of the differences in the codes are being evaluated. A comparison of the codes used for various types of reactor lattices, from Bradwell (gas-cooled) to the K-Reactor, to N-Reactor, indicates changes in relative cross sections which are apparently attributable to the presence of water coolant; the discrepancy between the codes becomes most pronounced in the N-Reactor lattice, and is almost insignificant in the water-free lattices. The method of computing neutron temperatures and the effect of the reactor coolant on cross-section values is being reviewed in order to improve the computer code models and the inputs for CAGE and production forecasts. The conversion ratio assumptions used in CAGE are evidently in reasonable agreement with the analyses except for Np²³⁷, in which the CAGE assumption is about 1.25 times the experimentally found value. For the present, no change in the CAGE assumption is planned until a better understanding of the origin of the apparent discrepancy can be developed.

1.6 Nuclear Health and Safety

A study has been completed on modifications required to ensure an unattended 21-day operational cool-down period by once-through emergency reactor cooling. Changes would include increasing the capacity of the lubricating oil reservoirs of seven diesel engines, modification of the emergency diesel generator engine to provide larger air filters and additional gasket cooling water. Additional provisions would be made to provide additional emergency raw water through self-cleaning strainers in the event that filtered water is not available. Additional diesel fuel tank capacity would provide fuel for all necessary engines for the 21-day period.

1.7 Ground Disposal of Radioactive Waste

Test well drilling along the river bank has been completed. After initially encountering difficult drilling conditions, very good drilling conditions were found. Seven test wells from 20 to 100 feet deep were drilled.

A heavy clay layer was found at about 52 feet. The clay is believed to effectively block flow of the crib effluent water below the 52 foot level. Sampling of these test holes has begun. The samples are being taken as part of the program to determine more precisely the average I-131 concentration in N-Reactor effluent water entering the Columbia River.

1.8 Environmental Effects

No release to the environment of radioactive or chemical effluents in liquid or gaseous form occurred during the past report period in which AEC, U.S. Public Health Service, or Washington State limits were exceeded.

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2. COPRODUCT MISSION

2.1 Program Management and Programmatic Progress

The implementation of the two reactor load demonstration of coproduct fuel in N-Reactor is proceeding on schedule. The following accomplishments were made in June.

- a. Uranium billets from the National Lead of Ohio (NLO) pilot plant were received.
- b. Driver fabrication was initiated.
- c. Additional lithium aluminate spike material was received.
- d. Oak Ridge has started isotopic adjustment on the lithium hydroxide for the large order of lithium aluminate scheduled to start being received at Hanford the latter part of June.
- e. Beneficial use of the target fabrication building has been established.
- f. Thirteen more tubes of the 20-tube R&D test were charged into the reactor (during May), and one tube was discharged during June because of a fuel failure. There is now a total of 21 tubes of the test block in the reactor.
- g. Necessary revisions to the reactor operating safety limits to accommodate the coproduct mode have been transmitted to the Atomic Energy Commission.
- h. Laboratory tests to establish the boiling burnout limit of the 14.6 lb/ft driver element have been started.
- i. Design of the reactor basin equipment (Part A of the project) is about 70 percent complete, versus 63 percent scheduled for completion.

A study was completed to evaluate the potential for producing sizable quantities of plutonium containing 12 percent Pu-240 during the coproduct two-load demonstration program. It was concluded that the second reactor load of fuel, if extended to an exposure of approximately 3700 Mwd/t, would produce plutonium containing 12 percent Pu-240. Prior to committing the second load to this high goal exposure, it will be necessary to perform tests on pilot quantities of fuel.

To permit measurement of the conversion ratio for plutonium and tritium during the coproduct demonstration prior to July 1967, it is planned to expand the current 20-tube pilot test to a 45-tube test block. Fuel irradiated in the center of this block during the demonstration will arrive at goal exposure sufficiently early to permit measurement of the conversion ratios by May 1967.

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Since the Purex dissolver inserts would not be available for processing the 2.1 percent U-235 Mark II drivers, it will be necessary to establish critically safe batch sizes. Calculations indicate that an unmodified dissolver will be capable of processing one fuel column per batch.

2.2 Reusable Canisters

In compliance with the Isochem processing requirement that the present aluminum canisters for production fuel cannot be dissolved in the Purex plant without incurring a severe penalty, a reusable canister of stainless steel has been designed and tested at both plants. A purchase specification was issued for bid, and a subsequent order for 600 reusable canisters has been placed.

2.3 Mark II Fuel Performance

A problem in the coproduct fuel design involves establishment of the right amount of elastic recovery in the supports which center the target in the driver. Theoretically, the amount of elastic recovery or "spring-back" required to accommodate the expansion in fuel element annulus (which occurs at maximum fuel operating temperatures due to thermal expansion differences between the driver and target) is 0.006 inch (radial).

Two different support concepts have been used on fuel assemblies fabricated for Production Test-66 (the current 20-tube test of coproduct fuel). On a majority of the Production Test-66 targets, the support system uses six 0.315-inch wide buggy spring supports which were sized (compressed in place) to increase the spring recovery constant. This is the standard method used for the present tube-in-tube fuel design. This support system provides an average elastic recovery of 0.009 inch (radial).

A second (uncompressed) support system has been developed in which the supports must be deflected in order to insert the target in the tube. These "interference fit" supports will be tested on at least four column loadings of PT-66. This system makes use of nine 0.250-inch wide buggy spring supports.

Since in-place compression (hard-sizing) tends to decrease elastic recovery, the interference fit supports will not be compressed after target insertion. The tension provided by the uncompressed buggy springs will more readily adjust to expansion under irradiation and reduce the potential for fretting corrosion.

Tests so far have shown that the "hard sized" support is satisfactory--does not lead to fretting--in all but a small fraction of the elements tested. Thus it appears that chance variations in manufacture are probably the cause of the fretting which has been encountered. The interference fit support is subject to better manufacturing control and should therefore obviate this difficulty. However, early tests on it indicate that hydraulic forces may set the target into oscillation leading to vibration fatigue failure. Various approaches

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to solution of this problem are being followed, such as increasing the strength of the buggy spring, increasing the number of buggy springs, and spring loading the target against a solid support.

An out-of-pile test loop in which accelerated testing is being conducted has been activated and should provide reliable information from short term tests by which to predict long term performance.

2.4 Tritium Separation Studies

The results of laboratory scale ceramic target extraction studies show that approximately 99 percent of the tritium is evolved as a condensable material (T_2O). Studies of the effect of $CO+CO_2$ gases on the palladium diffuser have led to the conclusion that these gases have no detrimental effect on the palladium or the hydrogen diffusion rate. Restriction of the carbon content to 250 ppm in the reactants will eliminate the need for incorporating a Hopcalite-zeolite bed between the furnace vessel and the uranium bed decomposer unit at the Savannah River Plant.

The presence of substantial quantities of free gas in the aluminum-clad targets may require the installation of a surge tank between the furnace vessel and the uranium bed decomposer unit at Savannah River.

Fourteen irradiated targets from Production Test-57 were shipped to Savannah River for plant extraction studies. This shipment included targets with higher gas volume ratio (GVR) values (3.5) than previously shipped and lithium aluminate pellets of up to 89 percent of theoretical density.

3. INTEGRATED SITE MISSION

As previously reported, 17 UO_2 tubular fuel elements, designed to operate at the same specific power as Mark I fuel, have been fabricated at Hanford for test irradiation in N-Reactor. The UO_2 fuel elements contain 2.4 percent U^{235} and were vibrationally compacted to 85 percent of theoretical density. Evaluation of the flow splitter and fuel support designs is continuing to ensure attainment of desired irradiation capability.

4. NON-DEFENSE PLUTONIUM

Requirement-supply study is still in progress. No reportable information at this time.

4.1 2.1% Critical Mass Experiments

Experiments to determine the critical mass of 2.1 percent Mark II fuel, with and without integral targets, have been completed.

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5. PLUTONIUM TECHNOLOGY MISSION

As part of a study requested by AEC and to be done in collaboration with Savannah River, work is starting on the development of a model whereby the least-cost method of producing a given Cm-244 requirement can be determined. The model is to take into account the capabilities of the various AEC reactors, as well as side-stream inputs from power reactors.

The model currently envisioned would consist of four or five irradiation steps with recycle of material in the last two steps. Separation of the Am and Cm from the plutonium would be made after each irradiation. The irradiation steps and recycles are:

- a. Irradiate U-238 to high Pu-240 plutonium
- b. First plutonium irradiation to burn out up to 70 percent of the Pu-239
- c. Second plutonium irradiation and recycle to burn out most of the remaining fissile plutonium
- d. Irradiate and recycle the Am in special targets

If the material from step 2 is too low in Pu-242 and consequently too high in fissile plutonium, it may be desirable to put the plutonium recycle into a third plutonium irradiation step which would produce relatively pure Pu-242. Curium from the first and second irradiation steps, being high in Cm-242, should be aged to get the Pu-238 which is formed.

The procedure used at Savannah River in the Cm-II program combines steps 3 and 4 into a single irradiation, with the Cm-244 being separated after step 4, and the remaining plutonium and Am being recycled together. In this program, 200 kg of Pu-239 were reduced in the first plutonium irradiation to 60 kg of Pu-239 which were then irradiated, with recycle, in the high flux reactor to produce 3 kg of Cm-244.

Battelle Northwest representatives have an existing computer code which is applicable to this problem and will require only minor modifications. Basically, the code starts with the desired results and works backwards to determine the required inputs at each stage and the costs involved. An evaluation will weigh the effect of isotopic content of the plutonium at each step upon the quantity of Cm-244 produced, and the value of plutonium at each step.

6. Pu-238 DEMONSTRATION PROGRAM

The program to irradiate 5 kg of neptunium in N-Reactor has progressed as scheduled, and three columns of neptunium-aluminum alloyed target elements are undergoing irradiation in the N-Reactor.

The yield of neptunium targets suitable for irradiation was higher than had been expected from a development program of this type.

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The schedule for the irradiation and separation of the neptunium targets is shown in the following table. Actual completion dates are noted for those items that have been completed to date. Three columns of targets, containing 4 kg of neptunium, have been charged into the reactor. It is estimated that 630 grams of this neptunium will be consumed in producing 595 grams of Pu-238 at 85 percent quality.

TABLE I

Schedule for Neptunium Irradiation

	<u>Scheduled Date</u>	<u>Actual Completion Date</u>
<u>*Conversion of a portion of 2100 grams of Np-237 to Pu-238</u>		
Convert neptunium nitrate to neptunium oxide	January 17	January 12
Start fabrication of targets	January 24	January 21
Charge one column (16 targets)	March 21	March 25
Discharge targets	August	
Cool (20 days)		
Separate Pu-238	August	
Ship neptunium nitrate to Savannah River	October	
Store Pu-238		
<u>*Conversion of a portion of 2900 grams of Np-237 to Pu-238</u>		
Convert neptunium nitrate to neptunium oxide	March 14	March 21
Start fabrication of targets	March 21	March 24
Charge two columns (32 targets)	May 16	May 29
Discharge targets	October	
Cool (20 days)		
Separate Pu-238	October	
Ship neptunium nitrate to Savannah River	December	
Convert Pu-238 to oxide and ship to Mound Laboratory	December	

*The quantities shown include originally anticipated fabrication losses. The quantity actually charged is smaller.

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7. URANIUM-233 MISSION

Not a Hanford Atomic Products Department mission at present.

8. COBALT-60 MISSION

No significant activity on this mission during June.

9. POLONIUM-210 MISSION

See page 35 of this document for special report outlining scope of the polonium-210 program.

10. HIGH-POWER DENSITY FUEL MISSION

Not a Hanford Atomic Products Department mission.

11. SMALL REACTOR MODERNIZATION MISSION

Not a Hanford Atomic Products Department mission.

12. FUNDAMENTAL AND GENERAL STUDIES

Not a Hanford Atomic Products Department mission.

13. NUCLEAR SAFETY MISSION

13.1 Fuel Failure and Fission Release Tests

Revised values for the fractional release of Xe, Kr, and Cs have been calculated. The revised release numbers reported in Table 13-I reflect corrections of analytical errors and a more accurate calculation of fission product inventory. There is still uncertainty about the iodine fraction, and future tests will be required to establish more precisely the magnitude of the I-131 release from failed fuel elements.

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TABLE 13-I

Percentage Release of Fission Products - Fuel Heating-To-Failure Tests

<u>Test Run No.</u>	<u>Xe¹³¹</u>	<u>Kr⁸⁵</u>	<u>I¹³¹*</u>	<u>Cs¹³⁷</u>
R1	3.6	1.0	$< 5 \times 10^{-1}$	0.2
R2	2.1	0.2	1.5×10^{-3}	0.2
R3	2	0.32	< 3	1.6
R4	2	0.2	< 0.6	0.01
R5 (1)	3.4	0.1	1×10^{-4}	
R5 (2)	2.4	0.7	6×10^{-4}	
R5 Total	6	0.8	7×10^{-4}	0.84
R6 (1)	2	0.1	6×10^{-4}	
R6 (2)	2.8	0.34	2×10^{-4}	
R6 Total	5	0.44	8×10^{-4}	4.7

*A large fraction of decay had occurred before the Iodine samples were counted. Counting errors were therefore high. Future tests will determine precisely the Iodine release fraction.

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These results are still considered preliminary, but there is at least one general conclusion which may be drawn. This is that the fractional release of fission products from fuel elements failed below the melting point is much less than the values that have traditionally been used in hazards evaluations. For example, in the past 100 percent release of Xe and Kr have been assumed; 50 percent of I, and 50 percent of Ce. These tests indicate one to two orders of magnitude less release than had been previously assumed.

14. COLUMBIA RIVER MISSION

Not a Hanford Atomic Products Department mission.

15. WASTE MANAGEMENT AND FISSION PRODUCT SEPARATION

Not a Hanford Atomic Products Department mission.

16. OTHER ISOTOPES MISSION

Fission Products Chemistry Operation of Battelle Northwest Laboratories will encapsulate about 100 grams of thulium oxide ($Tm_{2}O_{3}$) in a target of Hanford Atomic Products Department design. A conceptual target design has been developed, and design of the in-reactor test is in progress. Physics, thermal hydraulics, and other irradiation characteristics of this design are being evaluated. These targets will be charged in the graphite coolant cross channels. Upon completion of irradiation, Battelle Northwest Laboratories will measure the heat output, radiation characteristics, and shielding requirements of Tm_{170} generated in N-Reactor. Charging the capsule will be accomplished sometime in August.

17. ENRICHED FUEL MISSION

No work performed in June on this mission.

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PRODUCTION OF POLONIUM IN N-REACTOR

INTRODUCTION

The need for radionuclides to generate space power is a subject of primary interest to government space agencies. A significant market for polonium is considered likely in that polonium-210 as a heat source has two attractive potentials: (a) space capsule auxiliary power, and (b) space capsule propulsion. This attractiveness is founded on the desirable qualities that polonium-210 possesses for space applications, for example, high power density, minimum safety problems, minimum shielding, high melting point, established production technology, and availability. These properties have created a demand for polonium-210 estimated to be as much as 3 kilograms at launch by 1970, provided polonium-210 is economically competitive with other heat possible sources.

This report presents production and cost capabilities that should establish N-Reactor as a competitive producer of polonium-210.

FEASIBILITY

The request (item 1 of Bibliography) of the Atomic Energy Commission for a study of reactor capabilities to produce polonium-210 to meet the projected needs of:

Over 1 kilogram in FY-1968
Over 2 kilograms in FY-1969
Over 3 kilograms in FY-1970 and 1971

is evidence of the continued interest in polonium-210 as a heat source. N-Reactor's contribution to the production study, (item 2 of Bibliography) was 1.8 kilograms of polonium-210 (at reactor discharge) per year.

Based on a projected period of 68 days from reactor discharge to launch, the quantity of polonium present at launch time would be about 0.71 of that present at the time of discharge from the reactor. Consequently, N-Reactor can provide up to 1.3 kilograms of polonium at launch per year.

The 68 spare channels and 100 process tubes were selected as possible locations for irradiation of aluminum-clad, bismuth metal targets in water-cooled Zircaloy process tubes.

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The 68 spare channels have a diameter of 3.156 inches and are located in a high thermal flux region between the core and the graphite reflector.

The conversion of bismuth to polonium in these channels is compatible with the plutonium production load without displacing any fertile material from the reactor core. Neutron flux in the reactor core remains essentially unchanged; therefore, plutonium production also remains unchanged. Since some neutrons that would otherwise be lost from the reactor are captured by the bismuth, the over-all neutron economy of N-Reactor is improved. By using the 68 spare channels, 688 grams (at launch) of polonium per year can be produced in N-Reactor without loss of plutonium production at a cost of \$6.30 per thermal watt. The cost contribution from target fabrication and reactor irradiation accounts for \$1.10 per thermal watt or less than 15 percent of the total unit cost.

Additional reactor space could be allocated to polonium-210 production by diverting 100 or more process tubes from plutonium production. Use of these channels, although attractive, does require the displacement of plutonium-producing fuel. Since the absorption cross section of bismuth is extremely low in relation to the cross section of uranium displaced, the effective cross section of the core is lowered, neutron economy is decreased, and plutonium production is reduced.

By diverting 100 process tubes from plutonium production to polonium production, 603 grams (at launch) of polonium per year can be produced at a cost of \$9.70 per thermal watt.

To achieve the estimated unit costs for polonium-210, there must be a substantial polonium production program in other reactors (approximately 3 kg per year) to support high capacity, low cost target fabrication and chemical processing facilities.

Table I presents a summary of polonium-210 production and unit cost. As a heat source for missions of short durations (approximately 45 days) plutonium-238 at a cost of \$480 per thermal watt becomes second rate compared to polonium-210 at a cost of \$7.90 per thermal watt.

Polonium concentrations can be tailored for cost considerations in the target fabrication and chemical processing operations over a range of 4 to 29 grams of polonium-210 per ton of bismuth through regulation of irradiation time and length of target charge.

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TABLE I

PRODUCTION AND COST SUMMARY

	<u>68 Spare Channels</u>	<u>100 Process Tubes</u>	<u>Combined Locations</u>
Po-210 - grams/yr			
At discharge	935	844	1779
At Launch	668	603	1271
Unit Costs			
Target fabrication and reactor costs - \$/grams	\$ 92	\$ 412	\$ 244
Total unit cost - \$/grams at launch	910	1401	1143
\$/Thermal watt at launch	6.30	9.70	7.90
Capital costs (in thousands)	485	700	1185

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Production capability could be established in N-Reactor 17 months after project authorization. Reactor modification would cost about \$485,000 to place the 68 spare channels into irradiation service and \$700,000 to convert the 100 process channels.

Should the Commission decide to supply polonium-210 to the space program, the 68 spare channels in N-Reactor with their capability to produce 935 grams (at discharge) of polonium-210 per year, could be an integral part of the Commission's program.

PHYSICAL PLANT

N-Reactor was originally designed with a reactor lattice containing 1180 process tubes. By 1960, calculations had been made which indicated that only 1004 tubes were necessary to obtain the design power level. Installation of the additional 176 process tubes was dropped; however, it was decided to leave 34 channels in the graphite at the top and 34 at the bottom of the core in a condition which would permit easy installation of process tubes at a later date.

These 68 spare channels have a diameter of 3.156 inches and are located in a high thermal flux region between the core and the graphite reflector. Because of their location and size, they are ideal for irradiating large quantities of bismuth to obtain polonium-210.

Process for Polonium Production

Polonium is produced by irradiating bismuth in a neutron flux. The reactions are:

$$\text{Bi-210} (t_{1/2} = 2.6 \times 10^6 \text{ yr})$$

Bi-209 + n

Bismuth has a thermal neutron absorption cross section of 0.034 barn of which 0.019 barn contributes to polonium production.

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The process for producing polonium in N-Reactor has not been optimized. Based on survey studies coupled with engineering judgement, two promising methods were selected for producing polonium in N-Reactor. One involves reactor modifications to charge aluminum-clad (2.9 inch outer diameter) bismuth metal target elements in 68 spare channels equipped with thin-wall Zircaloy process tubes to produce an estimated 935 grams at discharge per year. The other requires replacement of uranium fuel with aluminum-clad (2.4 inch outer diameter), bismuth metal target elements in 100 process tubes to produce an estimated 844 grams at discharge per year.

A schematic description of the polonium production process is shown in Figure 1.

Production and Operating Costs

Since bismuth has a small cross section in comparison to other materials in the reactor, a large reactor inventory of bismuth and a high neutron flux are required to make significant quantities of polonium-210.

Bismuth Inventory

The quantity of bismuth that could be placed in the 68 spare channels is limited by the physical volume in the channel that is available. Since these channels would be equipped with thin-wall Zircaloy process tubes, an aluminum-clad, bismuth metal target with a 2.9-inch outside diameter would be permitted. Therefore, the weight of bismuth that could be charged into these 68 spare columns is 33 tons. For the 100 heavy-walled Zircaloy-2 process tubes, the target diameter is 2.4-inch outside diameter. The quantity of bismuth that could be placed in these process tubes located in the reactor core is limited by the space available. However, the number of process tubes that could be removed from plutonium production without reducing maximum reactor power level is limited. At the limiting point, additional polonium production can be produced by a trade-off with reactor power level or by redesigning the fuel elements at some loss of plutonium conversion. Without fuel redesign, it is estimated that diversion of 100 process tubes to polonium production could limit maximum reactor power level to about 4300 Mw. Consequently, 500 Mw are traded for polonium production in these process tubes. Steam requirements at 150 psi can be met. A total of about 33 tons of bismuth could be charged in 100 process tubes.

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Figure 1. Flow Chart of Po-210 Irradiation and Separation

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Neutron Flux

At a reactor power level of 4800 Mw, the flux in the 68 spare channels is about 8.1×10^{13} n/cm²-sec. All flux levels provided herein are equivalent fluxes at a neutron velocity of 2200 meters per second. Further improvement in the flux can be realized by increasing power in the process tubes near the 68 spare channels. Replacement of 40 columns of the present 0.947 percent uranium-235 standard tube-in-tube fuel (Mark I) with fuel enriched to about 1.25 percent uranium-235 (Mark I-A) increases the flux to 9.0×10^{13} n/cm²-sec.

At a power level of 4300 Mw, the flux in the 100 process tubes is calculated to be about 8.1×10^{13} n/cm²-sec. These process tubes would be within the flat zone of the reactor. The uranium fuel columns supporting the 68 spare channels are sufficiently isolated from the 100 process tubes to permit the power level in these columns to remain essentially unchanged. Therefore, the flux in the 68 spare channels would not be changed by the diversion of 100 process tubes to polonium production.

The production of polonium-210 per ton of bismuth as a function of irradiation time for two flux conditions in N-Reactor is illustrated in Figure 2.

Alternate Proposal Costs

Annual production and operating costs for polonium using the 68 spare channels and 100 process tubes are shown in Table II. To calculate production and costs, a 35-foot charge of aluminum-clad bismuth metal targets and a 150-day irradiation time were assumed. Technological advances and optimization of process variables in N-Reactor should lead to unit costs calculated in Table II. This table includes a detailed cost breakdown and the key assumptions for the breakdown.

A particularly low cost of \$6.30 per thermal watt at launch can be obtained using the 68 spare channels. This attractive cost for 668 grams (at launch) of polonium per year, is possible because displacement of plutonium-producing fuel is not required. Plutonium production remains essentially unchanged and reactor cost assignable to polonium is simply the cost of the few neutrons absorbed by the bismuth. Although the cost was based on the neutrons absorbed, some of these neutrons would, in reality, be free because they would normally leak from the reactor; total

Grams of
Po-210 per Ton
of Bismuth

Flux is Expressed in Units
of 10^{13} n/cm²-sec

Figure 2. Po-210 Production As a Function Of Flux and Time

TABLE II

N-REACTOR Po-210 PRODUCTION AND COSTS

	68 Spare Channels				100 Process Tubes				68 Spare Channels Plus 100 Process Tubes			
	FY-68	FY-69	FY-70	FY-71	FY-68	FY-69	FY-70	FY-71	FY-68	FY-69	FY-70	FY-71
Po-210 at discharge, grams	870	935	935	935	785	844	844	844	1655	1779	1779	1779
Po-210 at launch, grams	544	623	668	668	497	563	603	603	1041	1186	1271	1271
Bi-209 throughput, tons	61	66	66	66	61	66	66	66	122	132	132	132
Costs, \$ x 10 ⁻³												
Conversion												
Uranium burnout	14	15	15	15	14	14	14	14	28	29	29	29
Bi loss (burnout + recovery loss)*	25	26	26	26	25	26	26	26	50	52	52	52
Target fabrication	31	33	33	33	31	33	33	33	62	66	66	66
Onsite and offsite uranium fabrication	10	10	9	9	8	8	8	8	18	18	17	17
Irradiations	4	3	3	3	326	313	283	267	330	316	286	270
Total conversion cost	84	87	86	86	404	394	364	348	488	481	450	434
Chemical processing and encapsulation												
Po-210 extraction	202	180	165	165	202	180	165	165	404	360	330	330
Bi-recovery	95	104	104	104	95	104	104	104	190	208	208	208
Po-210 encapsulation	205	235	253	253	188	213	228	228	393	448	481	481
Total chemical Proc. and encapsulation	502	519	522	522	485	497	497	497	987	1016	1019	1019
Total Cost	586	606	608	608	889	891	861	845	1475	1497	1469	1453
Conversion cost, \$/grams at discharge	97	93	92	92	515	467	431	412	295	270	253	244
Total cost												
\$/grams at discharge	674	648	650	650	1132	1056	1020	1001	891	841	826	817
\$/grams at launch	1077	973	910	910	1789	1583	1428	1401	1417	1262	1156	1143
\$/thermal watt at launch	7.40	6.70	6.30	6.30	12.30	10.90	9.80	9.70	9.80	8.70	8.00	7.90
\$/electrical watt at launch	149	134	126	126	247	218	197	193	195	174	159	158

*Assumes bismuth recycle. Discarding the bismuth following irradiation would increase Po-210 cost by about \$4 per thermal watt at launch.

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reactor neutron economy is improved. Conversion cost, for example, target fabrication and reactor irradiation contributes less than 15 percent to the total cost of \$86,000 per year which represents \$1.10 per thermal watt at launch.

A competitive polonium cost of \$9.70 per thermal watt at launch can be achieved using the 100 process tubes. Polonium produced in the 100 process tubes must absorb a much greater portion of reactor cost than polonium produced in the 68 spare channels because plutonium-producing fuel is displaced and over-all reactor neutron economy is decreased. The 100 process tubes would produce 603 grams (at launch) of polonium per year with a loss of 117 kilograms of plutonium.

Production of 1.3 kilograms of polonium at launch for a cost of \$7.90 per thermal watt can be obtained from the combined productions of the 68 spare channels and the 100 process tubes.

As a heat source for missions of short duration (approximately 45 days), polonium-210 at a cost of \$7.90 per thermal watt has a large cost superiority over plutonium-238 at \$480 per thermal watt (\$270 per gram).

Polonium production and unit cost are functions of thermal neutron flux, bismuth irradiation time, and reactor inventory of bismuth. These parameters, together with a polonium demand schedule, and/or polonium value, determine the optimum operating method. Some of these parameters are discussed below.

Polonium concentration per ton of bismuth is directly proportional to thermal neutron flux. Doubling the flux, therefore, doubles the polonium produced and halves the unit cost of chemical processing and bismuth purchase. One method of increasing the effective neutron flux is to shorten the target column length. Table III lists production and unit costs for various target column lengths in the 68 spare channels. Decreasing the target column length increases the polonium concentration, decreases unit cost and decreases total production.

Increasing irradiation cycle time increases polonium concentration per ton of bismuth at a decreasing rate (see Figure 3), thereby reducing unit cost for chemical processing target fabrication, and bismuth purchase; and decreasing total polonium production. Production and unit costs for several irradiation cycle times using full-length (35 feet) target columns in the 68 spare channels are listed in Table IV.

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TABLE III

PRODUCTION AND COST
FOR VARIOUS TARGET COLUMN LENGTHS
IN THE 68 SPARE CHANNELS
FOR A 150-DAY IRRADIATION TIME

Target Column Length*	Flux, 10^{13} n/cm ² -sec	Po-210 Concentration	Annual Production Grams at Launch	Unit Cost, \$/Thermal Watt at Launch	Percent Decrease In Unit Cost
		Grams Po-210/ton Bismuth			
Full	9.0	14.2	668	\$ 6.30	0
3/4	10.7	16.9	600	5.70	9
1/2	12.2	19.2	446	5.40	14
1/4	12.9	20.4	239	5.20	17

*Full length is 35 feet.

TABLE IV

PRODUCTION AND COST
FOR VARIOUS IRRADIATION TIMES
IN THE 68 SPARE CHANNELS

Irradiation Time, Days	Po-210 Concentration,	Annual Production, Grams at Launch	Unit Cost, \$/Thermal Watt at Launch	Percent Decrease in Unit Cost
	Grams Po-210/ton Bismuth			
100	10.6	749	\$ 7.30	-16
150	14.2	668	6.30	0
200	16.9	597	5.90	6
255	19.2	532	5.70	9
305	20.4	474	5.60	11

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Figure 3. Incremental Po-210 Production
As A Function Of Time For A
Flux Of 9.0×10^{13} n/cm²-sec

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Data in Table IV illustrate the trade-off of polonium production for lower unit costs. If the irradiation days were increased from 100 days to 305 days, the bismuth throughput would be decreased by a factor of 3; polonium produced would decrease from 749 grams to 474 grams, and unit cost would decrease by \$1.70 per thermal watt.

Combining irradiation cycle time with target column length would provide additional trade-offs. A 3/4-length target column combined with a 200-day irradiation cycle reduces unit cost to \$5.40 per thermal watt at launch for an annual production rate of 533 grams at launch. As shown in Table III, this unit cost is the same as for a 1/2-length target column; however, production is improved by 87 grams.

Technological Advances

After bismuth irradiation experience has been gained, efforts would be made to increase production efficiency by irradiating targets of various designs to find the optimum water-to-bismuth ratio in the process tube. As the ratio increases, the flux in turn increases; however, the bismuth inventory decreases. A balancing of these effects might lead to a higher production efficiency than those derived from the chosen target design. Also, detailed studies would be made to compare the effects of short loading of target columns versus longer irradiation time on production, operating costs and unit costs. Attractive features from each would be combined to provide a more favorable process.

Reactor Modifications

Both process methods as described in Section B, "Process for Polonium Production," require a separate low pressure, filtered water cooling system. To be compatible with the aluminum cladding, the coolant would be adjusted to a low pH and dichromate added.

The 68 spare channels would be equipped with commercially available thin-walled Zircaloy process tubes and a low temperature auxiliary coolant supply to prevent melting of the bismuth metal. Engineering analyses indicate that there is a sufficient excess of cooling water in the shield-cooling, 10-inch supply header to supply 600 gpm of low temperature, filtered water to the 68 channels. From the shield-cooling supply header, a 6-inch line would furnish water to two 3-inch cross headers, one at the top and one at the bottom of the front face of the reactor. The 3-inch header would supply

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the 68 channels by means of 3/4-inch stainless steel connectors. The 6-inch and 3-inch piping would be carbon steel. A control valve and flow orifice would be installed in the 6-inch supply. An existing spare 4-inch pipe penetration into Zone 1 would be used to link the supply piping between Zone 1 and Zone 2. No additional penetrations would be required. The coolant would be discharged into a process drain or radioactive crib storage facility. Instrumentation would be provided to ensure continuous cooling to these channels.

The combined coolant requirements for 68 spare channels and 100 process tubes would be beyond the excess capacity of the shield cooling system. An independent system from the 18-inch filtered water header would supply low temperature, filtered water to the 100 process tubes. The existing heavy-walled process tubes and tube connectors would be used.

From the filtered water header, a 12-inch line would furnish water to one 8-inch cross header located at the top of the reactor. Six 4-inch risers would supply the 100 process tubes through the presently-existing connectors. The coolant could be discharged into a process drain or radioactive crib storage facility. Presently-installed instrumentation could be used with modifications to ensure continuous cooling to these process tubes.

In both cases, the aluminum-clad, bismuth metal targets could be charged into these process channels using the production fuel charging machine with modified monotubes.

Dollar Requirements

Modifications to N-Reactor to irradiate bismuth elements in 68 spare channels and 100 process tubes are estimated at \$485,000 and \$700,000, respectively.

A breakdown of the total project estimated costs is shown in Table V.

Research and Development

No serious technical problems are foreseen in supporting bismuth irradiations in the 68 spare channels of N-Reactor. Use of 100 process tubes in the core of the reactor would require (a) a study of the effects of target loadings on reactor physics in terms of safety systems, transients, flattening, and reactivity coefficient changes, and (b) a hazards analysis to ensure that safety criteria for total control, speed of control, approach to burnout, and confinement are met.

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TABLE V
PROJECT COST ESTIMATES

	<u>68 Spare Channels</u>	<u>100 Process Tubes</u>	<u>Total</u>
Zircaloy-2 process tube and nozzle assembly	\$ 175,000	\$ --	\$ 175,000
Cooling piping	16,000	350,000	366,000
Connector assembly	12,000	--	12,000
Removal and replacement	--	<u>30,000</u>	<u>30,000</u>
Subtotal	\$ 203,000	\$ 380,000	\$ 583,000
Overtime allowance	11,000	15,000	26,000
Support craft	10,000	15,000	25,000
Special work permit conditions	40,000	40,000	80,000
Equipment usage	6,000	5,000	11,000
Construction overheads	<u>50,000</u>	<u>60,000</u>	<u>110,000</u>
Total cost plus fixed fee construction cost	\$ 320,000	\$ 515,000	\$ 835,000
Charging machine modifications	20,000	--	20,000
Instrumentation and electrical	15,000	25,000	40,000
Chemical feed pump	<u>2,000</u>	<u>2,000</u>	<u>4,000</u>
Total direct construction cost	\$ 357,000	\$ 542,000	\$ 899,000
Engineering	35,000	40,000	75,000
Plant force work	2,000	5,000	7,000
Supervision of construction (Title III)	15,000	20,000	35,000
Contingency	<u>76,000</u>	<u>93,000</u>	<u>169,000</u>
Total Project Cost	\$ 485,000	\$ 700,000	\$1,185,000

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Implementation Schedule

N-Reactor capability with modifications to produce 1.3 kilograms of polonium-210 (at launch) per year could be established within 17 months after project authorization.

This schedule allows ample time for the installation of a once-through cooling system and associated hardware to accommodate bismuth irradiations. In both cases, reactor outages are required for equipment "tie-in", i.e., two weeks for the 68 spare channels and six weeks for the 100 process tubes. These outages are the most critical factor in achievement of the implementation schedule. Equipment "tie-in" could be delayed if the times for reactor outages were not in harmony with plutonium and/or steam requirements.

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APPENDIX

ASSUMPTIONS AND BASES

1. Bismuth is assumed to cost \$4 per pound.
2. The bismuth targets would be fabricated in Douglas United Nuclear's facility at a cost of 25 cents per pound of bismuth.
3. Irradiation cost is allocated to polonium by the formula.
$$(\text{Neutron absorption} + \text{reactor space})/2$$
4. All costs include overhead and AEC assessment charges.
5. Costs are on an as-generated basis in the same manner as the 1964 Refined Reactor Cost Studies.
6. Bismuth losses are 5 percent per cycle. For costing purposes, the bismuth cost during recycle is only the cost of the material lost; that is, the cost is calculated in the same manner as uranium burnout and does not include inventory cost (the required inventory and implementation of bismuth recycle is detailed in item 3 of Bibliography).

7. The ratios of polonium-210 at discharge to polonium-210 available at launch are as follows:

	<u>FY-1968</u>	<u>FY-1969</u>	<u>FY-1970</u>	<u>FY-1971</u>
Ratio	1.6	1.5	1.4	1.4
Time from discharge to launch, days	96	82	68	68

8. There are 145 thermal watts per gram of polonium-210 and the efficiency of conversion to electrical watts is 0.05.
9. Chemical processing and encapsulation costs are detailed in item 3 of Bibliography.

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BIBLIOGRAPHY

1. Memorandum, Po-210 TOTAL COSTS AT HANFORD
M.R. Schneller, December 16, 1965 (Secret-Rough Draft).
2. ISO-62, Po-210 PRODUCTION CAPABILITY AT HANFORD
compiled by O.F. Hill, February 15, 1966 (Secret).
3. RL-SEP-911, Po-210 COSTS AT HANFORD
O.F. Hill, December 29, 1965 (Secret).
4. RL-NRD-792, PRODUCTION OF POLONIUM IN N-REACTOR
E.G. Pierick, R.D. Shimer, March 1, 1966 (Secret).

ADMINISTRATIVE SUMMARY

JUNE 1966

EMPLOYMENT

SECTION	Permanent		Temporary		Total		Change		Total
	E	NE	E	NE	E	NE	E	NE	
General	2	2	-	-	2	2	-	-	4
Closeout Group	18	31	-	-	18	31	-3	-7	49
Employee Relations	5	4	8*	1	13	5	+4	+1	18
Finance	21	16	-	2	21	18	-1	+2	39
Fuels	42	120	1**	2	43	122	-1	+4	165
Plant	60	216	-	3	60	219	-	+5	279
Project	61	16	1	1	62	17	-	+1	79
Research & Engineering	<u>52</u>	<u>21</u>	<u>+3</u>	<u>+2</u>	<u>55</u>	<u>23</u>	<u>+3</u>	<u>+2</u>	<u>78</u>
Total	261	426	13	11	274	437	+2	+8	711
Department Total					711				
Net Change							+10		

*Tech Grads

**MTP

Safety

Days without a disabling injury 2,617

Hours worked without a disabling injury 6,824,600

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VISITORS

<u>Name</u>	<u>Component & Location</u>	<u>Contact</u>	<u>Date</u>	<u>Purpose</u>
Dr. W.J. Love	R&D Center - GE Schenectady, N.Y.	C.E. Love J.H. Fastabend	6-14-to 6-17-66	Steam generator and equipment problems.
Dr. E.O. Bergman	Consulting Engineer San Marino, Calif.	C.E. Love	6-15-to 6-16-66	Steam generator problems.
Dr. E.L. Knoedler	Sheppard T. Powell & Associates, Baltimore, Maryland	C.E. Love W.D. Bainard	6-15-to 6-17-66	Steam generator problems.
Dr. W.L. Pearl	APED - GE San Jose, Calif.	C.E. Love	6-15-66	Steam generator problems.
R. Marshall	Parker Hannifin Portland, Oregon	R.J. Robinson J.M. Pinkerton	6-22-66	Discuss new tube fittings and V-11 valves.
R.C. Webber	Webber Mfg. Co. Indianapolis, Indiana	R.M. Smithers J.M. Pinkerton	6-27-66	Environmental test chamber discussions.
B.M. Wareham	Hereaus Engelhard, Seattle, Washington	S.G. Forbes	6-23-66	Service and instruct on use of mass spectrometer.
A. D'Angelo	Hereaus Engelhard, Monroeville, Pa.	S.G. Forbes	6-23-66	" "
C. Haverstock	Campbell Industrial Supply	M.J. Veal	6-22-66	Abrasive blade samples and literature.
R.R. Baum	Proto Tool Company Los Angeles, Calif.	C.E. Gruhl	6-23-66	Discuss crimping pliers for buggy springs.
R.R. Baum	Proto Tool Company Los Angeles, Calif.	I.V. Nelson	6-21-66	Fabricating tools to closer locking clips and sizing supports.

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VISITORS

<u>Name</u>	<u>Component & Location</u>	<u>Contact</u>	<u>Date</u>	<u>Purpose</u>
T. Snyder	APED, General Electric San Jose, California	M.C. Leverett	6- 7-to 6-10-66	Technical discussions
F.P. Baranowski	AEC Headquarters Washington, D.C.	R.L. Dickeman	6- 9-to 6-10-66	Technical discussions

- - - - -

VISITS

A.E. Guay D.H. Nyman	GE Research Laboratory Schenectady, N.Y.	B.W. Roberts	6-14 to 6-15-66	Discuss galling of Zr-2
A.E. Guay D.H. Nyman	Canadian GE, Peterborough, Ontario, Canada	Mr. Pawliw	6-16-to 6-17-66	Discuss galling of Zr-2
C.F. Poor	American Nuclear Society Denver, Colorado		6-21-to 6-24-66	Present paper to ANS National Meeting
M.C. Leverett	American Nuclear Society Denver, Colorado		6-21-to 6-24-66	Attend National Meeting and to Chair meeting of ASA Sectional Committee N6
D.H. Curtiss	ASTM Annual Meeting Atlantic City, N.J.		6-27-to 6-30-66	Attend annual meeting
N.R. Miller	Reactor Safety Conference MIT, Boston, Massachusetts		6-27-to 7- 8-66	Attend conference

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VISITS

<u>Name</u>	<u>Component & Location</u>	<u>Contact</u>	<u>Date</u>	<u>Purpose</u>
J.J. Wick	Harvey Aluminum Torrance, California	R. Higgs	6- 7-66	Mark T. cladding
	Coast Metal Craft Culver City, California	G. Reed	6- 8-66	Copper sleeves
	NUMEC Apollo, Pennsylvania	A. Blum	6- 9-66	Braze rings
	Carborundum Metal Climax Co. Akron, N.Y.	R. Lorenzi	6-10-66	Mark T. caps
D.H. Walker	Tempus Corporation Los Angeles, California	K.W. West	6- 6-66	Discuss target stripper
	Harvey Aluminum Torrance, California	R. Higgs	6- 7-66	Discuss Zirconium cladding production
G.R. Klingler	Tempus Corporation Los Angeles, California	K. West R. Walker	6- 8-66	Discuss fabrication problems associated with target stripper

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SIGNIFICANT REPORTS

<u>Report Number</u>		<u>Title</u>	<u>Author</u>	<u>Date</u>
HW-69941 Rev. 1	C	Coextrusion Process Specs	N-Fuels	June 1966
RL-NRD-218 SUP B	S	PT-NR-4 Supplement B, Evaluation of Spike Ring Monitor Column Fuel Elements in N-Reactor	A.E. Guay	March 24, 1966
RL-NRD-500	U	Redundant Level Monitoring System for W, C, and D Elevators	H.L. Pringle	June 13, 1966
RL-NRD-830 SUP 1	C	PT-NR-66, Coproducer Demonstration Test (2.1) Supplement No. 1	R.E. Trumble	April 28, 1966
RL-GEN-963	U	On the Variational Method and Its Application to Reactor Calculations	C.K. Lewe	May 13, 1966
RL-GEN-980	C	Plugging from Recent Graphite Cooling System Dumps	R.E. Spicka	May 16, 1966
RL-GEN-1014	S	Non-Defense Plutonium During Coproduct Demonstration	R.L. Dickeman	June 1, 1966
RL-GEN-1018	S	PT-NR-76, Irradiation of 8-Inch Fuel Elements for Subsequent Fuel Failure Testing	M.M. Hendrickson, L.L. Sims	June 1, 1966
RL-GEN-1021	C	Summary Report, PT-66 Coproduct Target Element Fabrication Details and Process Development Status	D.I. Brazier J.E. Hansen R.R. Studer D.H. Walker	June 3, 1966
RL-GEN-1022	C	Predicted Parameters for Six Loop Operation	R.E. Trumble	June 7, 1966

SIGNIFICANT REPORTS

<u>Report Number</u>		<u>Title</u>	<u>Author</u>	<u>Date</u>
RL-GEN-1025	U	N-4 Test 5.3, 5.4 and 5.6 Transient Predictions	W.G. Conn	June 6, 1966
RL-GEN-1033	U	Status - Action Plan - Boiler	J.H. Fastabend	June 9, 1966
RL-GEN-1034	U	N-Reactor Conversion Equipment Failure Sequence Analysis	J.A. White	June 8, 1966
RL-GEN-1036	S	Storage Basin - Canister Utilization - Coproduct Program	D. Marinos	June 10, 1966
RL-GEN-1039	C	NP-237 Irradiation Data, First Demonstration Charge 1	R. Cooperstein	May 31, 1966
RL-GEN-1039	C	NP-237 Irradiation Data, Second Demonstration Charge 2	R. Cooperstein	June 10, 1966
RL-GEN-1042	S	High Exposure Plutonium for ZPPR	R.L. Dickeman	May 26, 1966
RL-GEN-1043	S	Engineering Analysis of the Fuel Spacer and Refuse Disposal	F.E. Hatch	June 13, 1966
RL-GEN-1044	U	Report of 109-N Steam Generator Inspection Program - May 1966 RD	J.P. Sluka	June 15, 1966
RL-GEN-1047	U	Specification for Underwater Radiation Tolerant Television System	K.O. Creek	June 21, 1966
RL-GEN-1056	U	Interim Report, PT-NR-24, Plant Service Boiler Base Load Reduction Program, PT Change Authorization No. 3	A.J. Ebens	June 24, 1966

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SUGGESTION PLAN PARTICIPATION

	<u>June</u>	<u>CY-1966</u>
Number of eligible employees	437	399
Number of suggestions received	38	269
Number of suggestions acted upon	52	380
Number of suggestions adopted	25	131
Net annual savings	\$ 10,147	\$ 21,697
Amount of awards	\$ 974	\$ 2,963.50
Percent of awards to savings	9.6%	13.6%
Average amount of awards	\$ 24.35	\$ 22.63
Adoption rate per 1000 employees	-	657%

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