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MAY 10, 1967

HANFORD ATOMIC PRODUCTS DEPARTMENT

MONTHLY REPORT

APRIL 1967

GE-AEC BUSINESS RECORD

GENERAL ELECTRIC

RICHLAND, WASHINGTON

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HANFORD ATOMIC PRODUCTS DEPARTMENT

MONTHLY REPORT - APRIL 1967

R. L. Dickeman
General Manager

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

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SUMMARY

During April the N-Reactor operated with a time operated efficiency of 70.8 percent; and a production availability and steam availability to Washington Public Power Supply System of 68.5 percent.

No fuel element failures occurred.

A scheduled outage (April 18 to 25) was used to complete planned maintenance, accomplish the modifications required for the 4800 Mw power level probe, and for charging the fourth increment of the coproduct demonstration load.

Fuel element input production was 100 percent of forecast. Output production was 110 percent of forecast, significantly improving finished-fuel inventories and reducing in-process material levels. Calendar-year-to-date output is 102 percent of forecast.

Seventeen special extrusions were completed for the Douglas United Nuclear Corporation.

There are now 844 columns of coproduct fuel in the reactor. All facets of the coproduct demonstration program are proceeding on schedule and according to plan.

Two columns of high exposure, 1.96 percent enriched, coproduct fuel (discharged during March) were examined. These fuels had attained a tube average exposure of 4094 Mwd/t and operated at uranium temperatures of 550 C to 475 C (temperature range predicted for Mark V fuel). Cladding buckling was minimal. The superior performance is attributed to the 800 ppm Al plus 400 ppm Fe uranium alloy.

Flow testing of the prototypic Mark IV fuel continued during the month. Flow split data are being used to redesign the support system to balance the flow in the coolant channels.

The first inner fuels of the Mark IV design were successfully brazed using a chevron end-cap closure method which was developed to limit the stresses in the cap region which occur during irradiation.

The analytical results from the first Pu-238 target column show that the quantities of plutonium isotopes produced and the neptunium consumed are in close agreement with predicted values.

Flow tests were completed which indicate there will be no fretting corrosion problem during the thulium irradiation test in the graphite cooling system.

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OPERATING STATISTICS

In-Reactor residue (Kmw)	234.7
Highest exposure tube (Mwd/t)	3,631
Input production, plutonium (Kmw)	78.4
Input production, coproduct (Kmw)	54.3
Maximum power level attained (Mw)	4,000
Average power level while operating (Mw)	3,697
Time operated efficiency (%)	70.8
Number of shutdowns	4
Scheduled	1
Unscheduled	3
Reactor process data	
Power at which computed (Mw)	4,000
Primary coolant inlet temperature (°F)	378
Primary coolant outlet temperature (°F)	504
Maximum moderator temperature (°F)	1,100
Maximum tube power (kw)	4,982
Primary loop flow (lb/hr)	95.1x10 ⁶
Heat Loss	
Heat to graphite cooling system (Mw)	40.0
Heat to shield cooling system (Mw)	2.3
Heat to rod cooling system (Mw)	0.45
Steam generation temperature (°F)	355
Steam generation pressure (psig)	128
Dump condenser flow (lb/hr)	325,000
Inlet temperature (°F)	48
Outlet temperature (°F)	73
Nuclear steam generating rate	
Maximum (1000's lb/hr)	13,426
Steam furnished to WPPSS	10,514
Maximum steam dumped 1000's lb/hr)	3,845
Water collected from gas atmosphere (gal/day)	Negligible
Demineralized water produced (gal/hr)	86,580
Helium losses (1000's ft ³)	448
Fuel oil usage (bbl)	13,583

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Reactor flattening efficiency

See Reactor Physics Data

Fuel conditions

Heat flux maximum (Btu/hr-ft ²)	582,000
Heat flux average (Btu/hr-ft ²)	472,000

Cladding temperature (Mk I only)

Maximum (°F)	593
Average (°F)	560

Uranium temperature

Maximum (°F)	888
Average (°F)	800

Fuel Failures

0

	Natural 0.71%	0.95%	1.25%	Coproduct 1.96%	Coproduct 2.1%	Total
<u>Fuel Balance (tons)</u>						
Fuel charged	0	0	0	0	44.99	44.99
Fuel discharged	0	59.99	13.79	0	0.63	74.41
Net change	0	-59.99	-13.79	0	+44.99	-29.42
Total In Reactor	0	43.49	14.35	0	185.39	243.23
Total In Basin	1.10	372.88	87.54	6.38	10.45	478.35

<u>Fuel Element Output (Assemblies)</u>	<u>April</u>	<u>March</u>	<u>CYTD Total</u>
0.95% tube-in-tube	-0-	-0-	-0-
0.95% single tube	-0-	-0-	2
1.25%-0.95% tube-in-tube	-0-	-0-	-0-
1.96% coproduct	-0-	-0-	-0-
1.25% coproduct	-0-	-0-	-0-
2.1% Mark E (driver)	3,160	3,412	10,237
2.1% Mark F (driver)	395	648	2,375
Total assemblies	3,555	4,060	12,614
Tons forecast	45.1	56.4	187.1
Total tons	49.8	63.2	191.4
Percent of forecast	110.4	112.1	102.3
Uranium utilization (2.1%) Est.	62.1%	72.4%	63.7%

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REACTOR LOADING MAP

	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	
34																																			
33							G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	M	G							
32						G	G	G	G	G	G	K	K	K	G	G	G	G	G	G	G	G	G	G	G	G	G	G							
31					GM	G	G	G	J	J	J	J	J	J	J	J	J	J	J	J	J	J	J	J	J	S	G	G							
30					G	G	G	J	J	J	J	G	J	G	J	G	J	G	J	G	J	G	J	J	J	G	J	S	G	G					
29					G	G	J	J	J	J	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G		J	J	G	G	G				
28					G	G	J	J	J	G	G	G		G		G	GM	G	G	G	G	G		G		G	G	G	J	J	G	G			
27	G				G	G	SM	G	G	G	D		G	G	G	G	F	G	FM	F	FM	G	Y	G	G	G	G	G	G	J	SM	G	G		
26					G	J	J	S	G	G		G	D	G		G	F	G	G	G	F	G	G		G	G	G	G	G	S	S	J	S	G	
25	M	G			G	J	S	G	G	D	G	G	G	G	G	F	G	G	G	F	G	G		G		G	G	G	G	S	J	J	G	G	
24					G	J	J	S	G	G	G	G	G	G	F	F	G	G	GM	F	F	G	G	G	G	G	G	G	G	S	J	G	G	G	
23					G	J	J	J	G		G	G	G	GE	G	G	F	F	G	G	G	F	F		G	G	G		G	G	J	J	S	G	G
22	G	G			J	J	J	GE		G		G	D	GE	G	F	F	G	Y	Y	F	F	G	G	G	G	G	G	G	S	J	G	G	G	
21	G	G			S	J	S	G	D	G	G	G	D	G	G	G	G	GM	X	GM	X	G	G	G	G	G	G	G	G	J	J	G	G	G	
20	G	G			J	J	J	G		G	D	G	G	D	G	G	G	G	G	G	G		G		G	G		G	G	J	J	S	G	G	
19	G	G			S	J	S	G	D	D	G	D	G	G		G	G	G	Y	G	G	G		G		GM	G	G	G	J	J	G	G	G	
18	G	G			J	J	J	G	G		G	G	G	D	G	G	G	G	GM	GM	GM	G	G	G	G	G	G	G	G	S	J	S	G	G	
17	G	G			J	J	S	G	D		H	D	G	G		G		G	G	G	G	G	G	G	G	G	G	G	G	S	J	S	G	G	
16	G	G			J	J	S	G	K	B	G	GM	G	G	G	G	GM	G	G	GM	G		G	G		G	G	G	G	J	J	S	G	G	
15	G	G			J	J	S	G	D	G	G	H	GM	G	G	G	G	G	G	G	G	G	G	G	G		G	G	G	J	J	G	G	G	
14	G	G			J	J	S	G		G	G		G		G	G		G	G	G	G	G	G	G	G	G	G	G	G	S	J	S	G	G	
13					G	J	J	S	G	G		G	G	G	G		G	G		G		G	G		G	G	G	G	G	S	J	G	G	G	
12					G	J	J	J	G	G	D		G		G	G	G		G	G	G	G	G	G	G	G	G	G	G	J	J	J	G	G	
11					G	J	J	S	G	D	G	G	G	G	G	G	G	G	G		G		G	G	G	G	G	G	G	S	J	G	G	G	
10					G	S	J	S	G	G	G	G	G		G	G		G	G	G	G	G	G	G	G	SM	G	G	G	S	J	S	G		
09					G	S	J	S	G	G	G	G	G	G	G	G	G	G	G	G	SM	G	SM	G	G	G	G	LA	S	S	G	S	G		
08	G				G	G	J	J	D	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	LA	LA	J	J	G	G	G	G	
07					G	G	J	J	J	G	G	G	G	G	G		G	G	G	G	G		G	G	G	G	J	S	SM	G	G				
06					G	G	G	G	SM	G	J	G	G	G	G	G	G	G		G	G		G	G	G	G	J	J	J	G	G				
05					G		G	G	J	J	SM	JD	J	P	J	J	J	J	J	J	J	J	J	J		J	J	J	S	G	G	G			
04					G	G	G	G	J	J	J	J	J	J	J	J	J	J	J	J	J	J	J	J	J	J	S	G	G		G				
03					G	G	G	G	G	G	G	K	K	K	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G						
02					G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G							
01																																			
TIME																																			
	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	

April 30, 1967

MARK I FUEL

CODE	NO. OF TUBES	DESCRIPTION
	88	Regular
D	4	PT-32 (1 Coupon)
E	3	PT-32 (2 Coupons)
K	7	PT-76 (3 per 8" Fuel)
IA	3	PT-78 (MK IC) + PT-33 (T/C's)
M	2	PT-4*
P	1	PT-64 (Zr. Sample)
	0	MK IA + PT-32 (1 Coupon)
S	43	MK IA (Spikes)
SM	8	MK IA (Spikes) + PT-4*
	159	

*Monitor

MARK II FUEL

CODE	NO. OF TUBES	DESCRIPTION
X	2	PT-6 + PT-33 (T/C's) + PT-4*
Y	4	PT-75 + PT-33 (T/C's) + PT-4*
GD	16	PT-75 + PT-32 (1 Coupon)
FM	2	PT-66 + PT-4*
F	20	PT-66
GM	13	PT-75 + PT-4*
G	632	PT-75
J	149	PT-75 (CP Spikes)
GE	3	PT-75 + PT-32 (2 Coupons)
JD	1	PT-74 + PT-32 (1 Coupon)
H	2	PT-75
	844	

BLANK CHANNEL

B 1 Graphite Samples

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REACTOR PLANT OPERATION

OPERATING HISTORY

N-Reactor plant operated 24 days of the month for a time operated efficiency and production availability of 70.8 percent. Steam availability was 68.5 percent. Level operating efficiency (total plant hours minus hours of unscheduled outages expressed as percent of the total) was 92.2 percent.

The plant was operating at maximum authorized power level at the beginning of the month. The revised specific power limit was controlling the power level at month end at about 96.5 percent of maximum authorized levels. The reactor was shut down four times during the month. One outage resulted from an operator error, one was caused by a 125-volt DC ground in the Reactor Building, one was a scheduled outage, and one resulted from a zone temperature monitor resistance temperature detector failure. Outage time totalled 210 hours.

A total of 203 tubes were recharged.

Washington Public Power Supply System maintained a nominal 600 Mwe output throughout the month.

Reactor Physics Data

Date:	April 30, 1967
Time:	0800
Reactor Power Level:	3855 Mw
Reactivity in Rods:	9.20 mk
Maximum Graphite Temperature:	1076 F

	<u>Fuel Types</u>		
	<u>Mark I</u>	<u>Mark IA</u>	<u>Mark II</u>
Maximum Tube Power:	4630 kw	4690 kw	4580 kw
Maximum Specific:	184 kw/ft	169 kw/ft	189 kw/ft
Flattening Efficiency:	83.2%	82.2%	84.2%
Loading Changes:	None		

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PLANT PERFORMANCE

Operating Continuity

Maximum authorized power level was maintained during 12 days of the month. The reactor was scrammed on April 2 when an operator error closed a primary coolant supply line valve. Operation was resumed in a little more than two hours. The reactor was shut down on April 8 because of a 125 volt DC ground. Startup was made on April 9. A scheduled outage was initiated on April 18 and continued until startup on April 25. During this outage planned maintenance was accomplished, scheduled plant modifications preparatory to the 4800 Mw probe were completed, and the fourth increment of the coproduct load was charged into the reactor. A shutdown resulted from the failure of a zone temperature monitor resistance temperature detector. Operation was resumed later the same day and continued through month's end.

Coproduct Fuel Loading

The fourth increment of the Production Test-NR-75 material was loaded during the April 18 outage. The in-reactor inventory of coproduct material is now 844 columns and all aspects of the program are proceeding on schedule.

Steam Generators

Preparation of procedures and equipment necessary to retube Cell 3 steam generators is proceeding satisfactorily. The target date for taking Cell 3 out of service remains July 1. Extensive inspections of steam generators in Cells 2 and 3 were performed during the April outage and repairs were completed.

Fog Spray Nozzles Replaced

All of the original fog spray nozzles were removed and new non-plugging nozzles were installed.

Auxiliary Plant Boiler

The hot spot previously observed on the left side of the auxiliary plant boiler was investigated during the scheduled outage and repairs effected by applying castable material to block openings that permitted hot gases to pass too close to the skin of the boiler.

125 Volt DC System

A number of 125 volt DC grounds were experienced during the month. The grounds were mostly caused by moisture in electrical junction boxes. All physical circuits are being checked and upgraded as necessary. Drain holes were provided at the low points in the affected boxes and electrical components were sprayed with an anti-wetting agent to prevent further grounds.

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Resistance Temperature Detectors (RTD's)

The first fifty Rosemount insert RTD's were received two weeks ahead of schedule. Fifty insert RTD's and 400 strap-on RTD's are still on order and delivery is scheduled during May.

Nine Rosemount RTD's are still undergoing tests in the 189-D out-of-pile loop. To date no problem has been encountered with thermal cycling and vibration tests. One of the Rosemount's RTD's initially installed on the reactor has been inspected and does not appear to be affected by the environmental conditions of the thermal barrier. Forty insert units were installed during the April outage.

Process Tube Inspections

Four process tubes were inspected with the TV monitoring equipment during the recent outage. These included two tubes that contained goal exposure coproduct fuel columns. All four tubes were in excellent condition. Charge-discharge scratches and fretting marks from downstream spacers were present but were not severe. Examination of a column of new spacers with increased support contact area showed no fretting damage on the supports.

Reducing Exposure to Personnel

In addition to current modifications to eliminate the charging bridge and increase the effectiveness of the C-elevator shielding doors, the following concepts have been developed for evaluation in the program to reduce personnel radiation exposure.

- a. Portable shielding for use in the diversion valve area,
- b. Portable water tanks that can be used to build temporary shielding walls,
- c. Partial body shields which would take advantage of the different tolerances to radiation by different parts of the body,
- d. A directional radiation monitor to assist in determining the most effective location of shielding,
- e. Wing shields alongside the reactor face, top to bottom, to reduce the radiation emanating from the pipe spaces,
- f. Shield plates on the front or rear face of the reactor,
- g. A movable "cold zone" on the C-elevator.

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OUTAGE EXPERIENCE

<u>Date</u>	<u>Outage Type</u>	<u>Duration</u>	<u>Reason</u>
4-2-67	Scram (Auto)	13.6	Operator error
4-8-67	All Rod Insertion (Manual)	13.6	125 V DC Ground 105 Bldg.
4-18-67	Scheduled	176.0	Charge-discharge and main- tenance
4-25-67	All Rod Insertion	<u>6.8</u>	ZTM-RTD Failure
Total Outage Hours		210.0	

Production losses were ascribed to:

<u>Reason</u>	<u>Hours</u>
Unscheduled Outages	34.0
Charge-Discharge	90.0
Scheduled Work	67.1
Startup Preparation	<u>18.9</u>
Total	210.0

Figure 1 compares time operated efficiency, steam availability, and production availability over the past several months.

% TIME OPERATED EFFICIENCY (TOE) = $\frac{\text{Time Reactor Is Critical During Month}}{\text{Total Time Elapsed During The Month}} \times 100$

% STEAM AVAILABILITY = $\frac{\text{Time Reactor Operates Above 500 Mwt During Month}}{\text{Total Time Elapsed During The Month}} \times 100$

% PRODUCTION AVAILABILITY = Actual TOE plus TOE losses from external causes¹

¹Production availability is the percentage of the time the reactor operates within internally controlled limitations. It excludes time losses resulting from events such as BPA failure and shutdowns required by WPPSS.

Figure 1. Time Operated Efficiency, Steam Availability, and Production Availability

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IN-PLANT TESTS

The status of production tests is as follows:

<u>Production Test</u>	<u>Status</u>
PT-NR-2, Routine Graphite Sample Irradiations (HW-80369)	Continuing. Four graphite boats being irradiated in rod channel 74. Plans are to leave ball channel No. 60 empty until after the PT-NR-75 Physics tests in May.
PT-NR-3, Routine Monitoring of Graphite Oxidation in N-Reactor (HW-81478)	Continuing. Approximately 20 graphite oxidation samples will be charged as soon as outage time can be scheduled following the use of this channel during the PT-NR-75 Physics tests.
PT-NR-4, Eval. of Monitor Column Fuel Elements in N-Reactor (RL-NRJ-218)	Continuing. Ten columns of fuel are presently being irradiated under authorization of this production test. Two columns of Mark I and 3 columns of Mark IA are scheduled for discharge in May. The remaining 5 columns of Mark IA will remain in Reactor during PT-NR-75.
SUPA-2, Authorization for Change in Discharge Schedule	Continuing. This change authorized use of some PT-NR-4 columns for production of non-defense plutonium.
SUPB, Eval. of Spike Ring Monitor Column Fuel Elements in N-Reactor	Continuing. Extended monitoring to Mark IA fuel.
SUPB-1, Eval. of Spike Ring Monitor Column Fuel Elements in N-Reactor.	Continuing. Authorized extension of monitoring to Mark IA fuel.
PT-NR-6, Isotope Producing Rods in N-Reactor (HW-81339)	Continuing. Discharge and recharge of targets from Rod 23 were accomplished on 10-6-66. Target inspection is scheduled to evaluate recommended residence time for future charges.
PT-NR-8, Coproducer Demonstration Test (1.25) (HW-81327)	Final report in preparation.

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<u>Production Test</u>	<u>Status</u>
PT-NR-10, N-Reactor Corrosion Monitoring (HW-82136)	Continuing. Samples are being exposed in two secondary system facilities; additional facilities are planned.
PT-NR-12, Copper Alloys in the N-Reactor Primary System (HW-82241)	Continuing program.
PT-NR-13, Coproducer Demonstration Test (1.96) (RL-NRD-299)	The last two columns of fuel (containing iron-aluminum alloy fuel) being irradiated under this test were discharged on March 24.
PT-NR-14, Irradiation of Th-U Crud Monitor Elements in N-Reactor (HW-82385)	Final report in preparation.
PT-NR-27, Steam Generator Moisture Carryover Detection (HW-84238)	Final report in preparation.
PT-NR-30, Eval. of Ammonia for Controlling Oxygen in the Graphite Cooling System (HW-84232)	Final report in preparation.
PT-NR-32, Exposure of Corrosion Coupons in N-Reactor Process Tubes (HW-84367)	Continuing with periodic examinations. The replacement of coupons in three columns was completed during April.
PT-NR-33, In-Reactor Enthalpy Imbalance Measurements (HW-84401)	Continuing. Nine thermocouple trains are now charged. Two trains were charged under PT-NR-75 in April. A thermocouple train is scheduled for installation under PT-NR-78 during May.
PT-NR-36, Simulated Driver-Target Element (RL-NRD-55)	Final report in preparation.
PT-NR-41, Eval. of the Direct Addition of Ammonia to the Graphite Cooling System (RL-NRD-297)	Final report in preparation.
PT-NR-46, Primary Drive Turbine Speed Reduction Tests (RL-NRD-427)	Final report in preparation.

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<u>Production Test</u>	<u>Status</u>
PT-NR-48, Corrosion Test Samples in End-Caps of the Graphite Cooling System (RL-NRD-466)	Continuing. Sample removal is being delayed until July 1.
PT-NR-57, Coproducer Target Element Evaluation (RL-NRD-657)	Final report in preparation.
PT-NR-62, Eval. of Single-Pass Filtered Water Cooling (RL-NRD-668)	Approved. This test should be initiated any time N-Reactor transfers to single-pass filtered water cooling
PT-NR-63, On-Reactor Tests for New Resistance Temp. Detectors (RL-NRD-695)	Continuing. Fifteen integral connector RTD's are presently being evaluated on-reactor. Ten are operating within the all-tube monitor system and five are connected into the zone temperature monitor system. Three of the five thermocouples in the zone temperature monitor system now have low resistance to ground. Off-Reactor testing of RTD's is continuing. Routine installation of integral connectors in the zone temperature monitor system has been initiated.
PT-NR-64, Exposure of Zircaloy Coupons in N-Reactor Process Tubes (RL-NRD-716)	Continuing. One column of fuel was charged on 10-4-66. This column is to be discharged in May.
PT-NR-65, A Test to Demonstrate the Operability of the Rupture Monitor System with the Helical Sample Chamber Design (RL-NRD-726)	Approved. Test performance delayed to permit demonstration on a small scale basis prior to changing over a complete turret.
PT-NR-66, Coproducer Demonstration Test (2.1) (RL-NRD-830)	Continuing. One column of fuel with large W-spring supports and one column of fuel with small W-spring supports were discharged for examination in April. No fretting corrosion was detected. The process tube containing these two columns of fuel was examined using the TV monitor. No unusual tube damage was detected.
SUP3, Expansion of Coproduct Block	Continuing. Authorized increase in test size and updated test requirements.

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<u>Production Test</u>	<u>Status</u>
SUP7, Spring-Stop Support	Continuing. Authorized use of coproducer fuel utilizing spring-stop supports on targets. Also updated test requirements.
SUP11, Production Test Change Authorization No. 7	Continuing. This change authorized use of normal Mark II (PT-75) thermocouple trains in PT-66 charges.
PT-NR-67, Cobalt-60 Production in N-Reactor (RL-GEN-948)	Complete.
PT-NR-68, Pu-238 Demonstration Program (RL-NRD-793)	Complete.
PT-NR-69, Irradiation of Tubular Zr-2 Samples in N-Reactor Gas Atmosphere (RL-NRD-833)	Complete.
PT-NR-72, Process Tube Rear Nozzle and Connector Decontamination (RL-GEN-1489)	Continuing. Initial test completed on 3-19-67. A supplement to authorize continued testing is presently being prepared.
PT-NR-75, Coproduct Two Load Demonstration (RL-GEN-1065)	Continuing. Charging of first increment was completed early in January, charging of second increment was completed early in February, charging of the third increment was completed in March, and charging of fourth increment was completed in April.
SUP1, Reactor Operating Data Requirements	Continuing. Base data for first, second, third, and fourth increments have been taken.
SUP2, Mark II Physics Tests-Part 1, Verification of Control Rod and Ball Safety System Shutdown Margins	Continuing. First, second, third, and fourth increment control rod system shutdown margins have been obtained.
SUP5, All-Tube Temperature Monitor Calibration	Approved. Scheduled for completion in May.
SUP8, Radiological Controls	Continuing. Routine samples are being taken. There has not been any indication of tritium in the primary system coolant.

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<u>Production Test</u>	<u>Status</u>
SUP11, Ball Functional Test	Continuing. First half of this test was completed on 3-19-67. Interim report on test results has been issued.
SUP13, Production Test Change Authorization No. 5, Irradiation Exposure and Charging Schedule Revision	Continuing. Updated program schedule.
PT-NR-76, Irradiation of 8-Inch Fuel Elements for Subsequent Fuel Failure Testing (RL-GEN-1018)	Continuing. The initial column of the three columns of fuel charged on 9-16-66, was discharged on 2-6-67. Six additional columns of fuel (using 8-inch outers only) were charged on 2-6-67. A second column was discharged in April with the third column scheduled for discharge in May.
SUP3, Irradiation of 8-Inch Fuel Elements for Subsequent Fuel Failure Testing	Approved. This test supplement authorizes irradiation of Mark I fuel during PT-NR-75.
SUP4, Irradiation of 8-Inch Fuel Elements for Subsequent Fuel Failure Testing	Approved. This test supplement authorizes use of 8-inch outers only fuel. Six columns were charged in February.
PT-NR-78, Eval. of Mark IC Monitor Columns in N-Reactor (RL-GEN-1372)	Approved. Two columns of fuel authorized by this production test were charged during February and one column was charged in March. A fourth column is scheduled for charging in May.
PT-NR-82, Modified Rupture Monitor Differential Alarm Module Test (RL-GEN-1413)	Continuing. Test issued 3-14-67. Circuit installed on 3-17-67 but removed because of sensitivity to DC Power Supply. Further analysis and off-reactor testing is in progress.
PT-NR-84, Removal of Activated Films from Connectors by Reducing Coolant pH (RL-GEN-1569)	Approved. Initiation of test scheduled for May.
PT-NR-85, Zone Temperature Monitor and Burnout Flux Limits (RL-GEN-1540)	Approved. This test will be completed as soon as it can appropriately be scheduled.

EQUIPMENT MAINTENANCE AND PLANT MODIFICATIONS

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

<u>Title</u>	<u>Cost</u>	<u>Funds Assigned to GE</u>	<u>Percent Funds Expended</u>	<u>Status</u>
CGN-182 Replacement	\$165,000	\$ 77,700	75	The 25 circuits involved have been tested. Of these, 13 have been replaced to complete the project. N-plant accepted the project 3-17-67. Total indicated underrun is \$62,560.
CGN-185 Steam Gen. Isolation Valves	268,000 (20 valves)	268,000	73	Twenty valves on order; four to be funded by WPPSS. Fourteen valves received. Four have been installed for Cell 4.
GCP-400 - Backup Boiler Facility	865,000	30,000	13*	Funding by WPPSS; Burns & Roe to perform A-E work. Functional and technical requirements issued by HAPD, GE, who will provide engineering services on procurement and on the installation work. The boilers will be manufactured by Comb. Engineering.
GAP-401 - Upgrading Fire Protection, 100-N	150,000	2,600	39*	AEC managed. Bid package by HES is about 90% complete.
GCE-402 - Coproduct Demon. Prog., Phase II, Irrad. Fuel-Target Handling, Storage & Shipping Equipment	609,000	165,000	94 on author. basis.	1231 canisters, two portable and one permanent tritium monitors, 150 canister guides and 277 cubicle covers received to date. Target separator installed.

* GE's Portion

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EQUIPMENT MAINTENANCE AND PLANT MODIFICATIONS (continued)

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

<u>Title</u>	<u>Cost</u>	<u>Funds Assigned to GE</u>	<u>Percent Funds Expended</u>	<u>Status</u>
GCP-404 - Fuel Spacer Disposal System and Refuse Cask, 100-N	\$ 77,000	\$ 77,000	20	Design started 2-10-67. Pit design approved for construction. Start of construction stalled because liner plate was delayed three weeks in transit.
GCE-405 - N-Reactor Temp. Monitoring System Improvements	189,000	189,000	--	Project authorized by RL-AEC 3-8-67. Fourteen of 18 junction boxes installed top of thermo barrier.
GCP-406 - Safety Platforms and Accesses	300,000	--	--	Project proposal originally submitted to RL-AEC 2-10-67 as a 59-a-5 funded item. Deferred to FY-1968.
GCE-407 - Fast Index, Rupture Monitoring System	50,000	--	--	Project proposal submitted to RL-AEC 2-17-67. Deferred pending availability of funds.
GCE-408 - W, C, D Elevator Safety	90,000	--	--	Project proposal submitted to RL-AEC 2-27-67. Deferred pending availability of funds.
GCP-409 - Fuel Handling Improvements	285,000	--	--	Project proposal submitted to RL-AEC 3-30-67. Approved by Review Board; directive in process.
GAP-410 - Decontamination Waste Loadout	65,000	--	--	Project proposal submitted to RL-AEC 4-13-67.
Corrosion Test Facility 109-N	45,000	43,224	95	Loops 2, 3, 4, and 5 completed. Loop 1 requires one tie-in for completion.

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FUEL MANUFACTURE

PRODUCTION STATISTICS

<u>Input (Extrusions)</u>	<u>April</u>	<u>March</u>	<u>CYTD Total</u>
0.95% Mark A (outer)	22	-0-	22
0.95% Mark N (inner)	-0-	-0-	-0-
0.95% Mark D (single tube)	-0-	-0-	-0-
1.25% Mark A (outer, spike)	-0-	-0-	-0-
2.1% Mark E (driver)	227	206	289
2.1% Mark F (driver)	-0-	-0-	-0-
Total Extrusions	249	206	911
Tons Forecast	53.2	62.4	210.4
Total Tons	53.1	42.2	190.1
Percent of Forecast	99.8	67.7	90.4

Output (Finished Production)

0.95% Tube-in-Tube	-0-	-0-	-0-
0.95% Single Tube	-0-	-0-	2
1.25%-0.95% Tube-in-Tube	-0-	-0-	-0-
1.96% Coproduct	-0-	-0-	-0-
1.25% Coproduct	-0-	-0-	-0-
2.1% Mark E (driver)	3,160	3,412	10,357
2.1% Mark F (driver)	295	648	2,375
Total Assemblies	3,555	4,060	12,614
Tons Forecast	45.1	56.4	187.1
Total Tons	49.8	63.2	191.4
Percent of Forecast	110.4	112.1	102.3
Uranium Utilization (2.1%) Estimate:	62.1%	72.4%	63.7%

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PRODUCTION SUMMARY

Input production was 100 percent of forecast. A total of 17 special extrusions were completed during April.

Output production was 110 percent of forecast. This significantly improved finished fuel inventories and reduced in-process inventories. Output performance was 102 percent of forecast for the calendar year to date.

PROCESS DEVELOPMENT

Mark IV End-Closure Brazing

The first inner fuel elements of the Mark IV design were successfully brazed with existing supplies of braze rings utilizing a chevron cap. The elements were recessed to a flat bottom, and chemically milled; recessed to a chevron and chemically milled again to a final depth of 0.463 inch $\pm$ 0.015 inch. The braze rings were Mark-N rings with two tack-welded sections added. Nineteen test fuel elements were end-brazed by this method and only three rejects occurred - one outer diameter nonflow, one inner diameter slopover, and one controller malfunction. Zyglo examination and polishing of samples indicated no other defects. The nonflow reject will be corrected in production runs by using a larger outer diameter braze ring. The braze ring weight will also be lessened to reduce a tendency to over-fill the closure area during the brazing operation. The chevron end cap was developed to distribute the stresses which occur in the end cap region during irradiation. Tests indicate that this design will significantly reduce the shear forces which exist at the cap-core interface in the standard flat bottom cap design. This modification is therefore expected to improve irradiation capability by limiting the number of end failures.

Uranium Billet Quality

Fabrication of the 2.1 w/o enriched uranium billets for the coproduct demonstration program is complete. During production of the billets, carbon contamination was found to be higher than expected near the outer diameter of the ingot at the mid-length position. Carbon levels were controlled between 300 to 750 ppm at the normal specified top-of-ingot position, but values as high as 1650 ppm carbon were found at the mid-length position near the outer diameter. Fuel elements were not accepted for reactor service if the ingot mid-length carbon level exceeded 1100 ppm.

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Tabless Rigid Support Development

The currently used rigid support for the coproduct target is fabricated with welding tabs of 0.040-inch nominal thickness. These tabs are used to enable weld projections to be formed by the conventional "dimpling" method. An undesirable feature of this tab is that the rigid support is raised slightly off the cladding surface by these dimples and this permits flexing under nominal loads.

A sample lot of rigid supports was fabricated without the normal weld tabs. Projections were formed directly on the bottom of the support body. Four projections per support were used instead of eight as on a normal rigid support.

These samples of the tabless support were welded on Zircaloy cladding, using the spike power welder and the standard process for the tab-type rigid supports but using a special electrode. No major problems occurred or are anticipated in the production welding of tabless supports since these tests showed that weld penetration can be controlled and that shear strength is adequate. Further development and testing of this design modification is in progress.

Braze Ring Quality

Inquiries have been submitted to three potential braze ring suppliers for 4000 braze ring sets which will be fabricated according to revised product specifications. Fabrication of these rings will be on a best effort basis until proven limits can be established from experience. The most significant modifications in the specification are changes in impurity limits and the use of a fluidity test instead of a melt test.

Each of the braze ring suppliers has a problem with at least one of the impurity limits. The purpose of the development work in this area is to direct the suppliers' efforts toward improving their product in those specific impurity areas peculiar to their fabrication process.

Special Extrusion Activities

Sixteen special extrusions were made for Douglas United Nuclear Corporation. These extrusions were:

- a. I & E Bare Core Extrusions - Ten extrusions (part of a program to fabricate 1000 bare I & E cores for Douglas United Nuclear) were completed to date. The remaining 250 cores will be extruded during May. Over-all extrusion yield was 82 percent. The

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purpose of this program is to demonstrate an on-site capability for fabricating I & E cores by extrusion rather than by the conventional rolling process used by the feed site facilities.

- b. Zircaloy Cladding Extrusions - Two extrusions were completed to provide inner and outer cladding components for the hot die sizing development.
- c. Aluminum Rod Extrusions - Forty-nine aluminum rod extrusions were completed to provide stock for the manufacture of self-supported aluminum dummies. A number of these extrusions exhibited surface galling that required machining to provide acceptable surfaces. During April, two more billets were extruded using a different liner under conditions similar to those used in the first campaign. A major improvement in surface quality was found, indicating that the liner condition was responsible for galling in the previous run.
- d. Reactor Tube Extrusions - Two extrusions were completed on a continuing program to develop extrusion techniques for fabricating K-Reactor process tubing. Neither of the extrusions was successful. The first billets stalled the press at the planned reduction ratio of 25:1. The second billet was designed to be extruded at a reduction ratio of 20:1. This extrusion proceeded halfway, at which time the mandrel failed. Further tests are planned.

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CONVERSION

Power Generation

Production of steam from N-Reactor and resultant electrical generation surpassed all other nuclear plants for the months of March and April. Total megawatt hours of electrical energy produced by the Washington Public Power Supply System (WPPSS) plant was 275,800 for March and 293,900 for April. In second place for the month of March was Indian Point with 194,750 Mwh. Indian Point has a rated capacity of 292 Mw compared with 800 Mw for the N-Reactor complex.

Backup Boiler Facility

Copies of the package boiler installation specification were received from Burns and Roe for review and comment. Bid drawings accompanied the specification. These defined the scope of the proposed contract for installation of the package boilers - including the furnishing and installation of mechanical and electrical equipment, instrumentation and controls, and supporting services and structures.

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TECHNICAL ACTIVITIES

1. BASIC PRODUCTION MISSION

1.1 Mark IC Development Testing

During the March outage, an additional column of Mark IC fuel was charged as authorized in Production Test-NR-78. This fuel was modified by installation of special shoes for outer cylinder supports. The shoes were provided with "wings" designed to increase their projected area perpendicular to the flow stream of the outer annulus, thereby forcing a larger percentage of flow into the other two subchannels. Thermocouple measurements indicated about a 5 percent improvement in the imbalance among enthalpy rises in the three subchannels, with the center channel running approximately 8 percent hotter than tube bulk (in enthalpy rise). Preparations are under way to charge at least one more column of Mark IC fuel. This column will have shoes on outer cylinders with wings on both ends and should operate with nearly equal temperature rises along the bottom of the outer annulus and the center channel, thus optimizing the fuel performance from a thermal hydraulics standpoint.

1.2 Mark IV Development Testing

Testing was begun, in the 189-D out-of-pile test loop, on a column (18 elements) of Mark IV prototype fuel (designed for operation at 4800 Mw). Test series 1 lasted 204.5 hours and was shut down April 5 for fuel inspection. Initial observations indicate that there is no basic mechanical problem with the new support design. Laboratory apparatus will be in place by early May to measure flow distribution to subchannels. Results from these measurements will determine the dimensional changes in fuel element support which are required to balance temperature rise in subchannels.

1.3 Fuel Spacer Redesign

The study to modify the design of spacers (dummies) to reduce their potential for damage to the inside walls of process tubes has progressed through the design of a load cell for installation on the downstream nozzle plug. By measurement of the fluctuation of forces at that point, the load cell is expected to prove or disprove the postulation that the spacer motion which causes the process tube damage is an unstable buckling action of the spacer column. This theory has been substantiated to the extent that it can be shown that, in moving from a buckled to a perfectly aligned condition, elastic deformation of the spacers will compensate for the tendency to increase the length of the spacer charge. Thus, the lateral motion of the spacers can take place without displacing the fuel column. If this finding is confirmed, an important design requirement will be identified - that of providing a geometry that is stable with respect to both the column action and the hydraulic forces.

1.4 Critical Mass of Exposed Fuel

As indicated last month, critical mass parameters for exposed fuel are being completed. Two factors were evaluated (a) the change in reactivity as plutonium is exchanged for U-235, and (b) the change in reactivity with fission product buildup.

The resulting data indicate that critical mass parameters for unexposed uranium fuel (0.95 percent enriched or greater) will be conservative for fuel after irradiation. The data provide a technical basis for a revision in Process Standards which would greatly simplify the fuel handling standard. Work is continuing.

1.5 Poison Worth of Replaced Ceramic Balls

A supply of replacement balls for the reactor ball safety system was recently purchased. These balls are being utilized as necessary to "top off" low content hoppers. Measurements in the test pile have been made which indicate that the poison strength of the replacement balls is 96 to 97 percent of that of the original loading. This is adequate for safety control purposes. The ball safety system provides back-up capability to the control rods and will automatically shut down the reactor under certain emergency conditions.

1.6 Decontamination Development

A high-temperature recirculating loop test of the decontamination procedure for Cell 3 has been completed. The test duplicated expected temperatures, flow velocity, surface-to-volume ratio, iron-oxide concentration, and carbon steel-to-stainless steel surface ratio. Test results showed:

- a. Corrosion rates in the loop test were five to ten times the rates observed in laboratory beaker tests attempting to simulate the same conditions. The critical parameters appear to be surface-to-volume ratio and flow velocity. A similar discrepancy had been observed during the Plutonium Recycle Test Reactor decontamination.
- b. Application of three complete alkaline-permanganate-acid decontamination cycles to the Cell 3 steam generators should produce 0.6 mils of uniform carbon steel corrosion, 0.01 mils of steam generator tubing corrosion, and 7 to 9 mils of localized attack at carbon steel-stainless steel welds.

Since the only carbon steel-stainless steel welds in the system being decontaminated are those joining the stainless steel tube sheet overlay to the 5.5-inch thick carbon steel tube sheet, this amount of corrosion is not serious.

The previously reported inability to decontaminate the rupture monitor drain headers appears to result from low alkaline permanganate temperature (95 C compared to a desired 100 to 105 C). This hypothesis will be checked by laboratory tests of rupture monitor drain line samples which were removed during the April outage.

Production Test NR-84, "Removal of Activated Films from Connectors by Reducing Coolant pH," was prepared to evaluate the effectiveness of reducing coolant pH from 10 to 7 for 24 hours to remove some of the deposited radionuclides from the front and rear connectors of a diverted process tube during shutdown. The test was initiated during the April outage. Results are not yet available.

1.7 Steam Generator Cell Return During Reactor Operation

Returning a steam generator cell to service during reactor operation involves two critical aspects: (a) flow from the shut-down cell must be raised slowly to avoid excessively perturbing reactor power (coolant from the shut-down cell will be at a lower temperature than from other cells), and (b) some flow through the primary pump must be quickly established to avoid overheating the pump. It had been planned to bring the pump up to minimum speed to increase pump discharge pressure, to provide a positive driving force and then to jog open the cell outlet valve.

During the March outage, a test was made to determine the flow response characteristics of the valve. The valve was arranged for jogging and was jogged from open to the closed position at intervals of several seconds each. Flow was noted at each interval. This valve exercise was performed with pump speeds at 900 and 1400 rpm.

The valve appears to have a steep response curve and goes from a low to nearly full flow with only a small valve travel. As a result, turbine speed will be used primarily to control the amount of initial flow from the isolated cell into the primary loop.

To ensure that the temperature of the water in the isolated cell is near primary loop temperature, the isolated cell's secondary side is to be valved off from the main steam header and allowed to rise in pressure corresponding to the primary loop water temperature which has been allowed to flow back into the cell from the inlet manifold during the "soak" period. The isolated cell water temperature is expected to be within 15 to 20 F of primary loop temperature. This should minimize the power rate-of-rise which is expected when the slightly colder isolated cell water is returned to the rest of the primary loop.

1.8 Variable Orificing

A second prototype nozzle cap orifice has been designed and successfully tested in the out-of-pile (PCE) test loop. This device is designed so that the point of orificing is downstream of the inlet nozzle rolled joint. This configuration is necessary in order to permit orificing in the range of 30 to 40 percent.

This orifice consists of two circular plates (one fixed and one movable) on the end of an extension shaft attached to the cap insert. Each plate contains four 0.75-inch holes on 90-degree centers. Orificing is accomplished by rotating the movable plate which covers up the holes in the fixed plate.

Two orificing spacers consisting of a 15.5-inch spacer with a 2.5-inch diameter circular plate welded on both ends were fabricated. The orificing spacers were installed in a process tube for test and evaluation.

The orificing program is designed to optimize heat transfer and flow conditions under planned operating conditions.

1.9 Fuel Element Motion Detector

A magnetic motion detector was developed and successfully demonstrated. The device is simple, consisting of an Alnico magnet attached to a fuel element foot, with another magnet suspended on the outside of the process tube. Axial movement of the fuel element is accurately indicated by rotation of the suspended magnet.

1.10 In-Tube Fuel Element Load Cell

Design was completed for a special load cell to measure the force of a fuel column against an outlet nozzle end cap during reactor operation. The design employs a temperature-compensated half bridge strain gage circuit. Fabrication of machined components is approximately 50 percent complete.

1.11 Nuclear Health and Safety

Test Wells

The fifth test well was completed April 10. The wells were drilled to an average depth of 75 feet and all are now in use to sample the underground water seepage from the crib.

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Effluent Control Program

The design criteria for the Effluent Control Program was issued as a single encompassing document, RL-GEN-1476, N-REACTOR EFFLUENT CONTROL PROGRAM - COMPOSITE DESIGN CRITERIA.

Decontamination Waste Disposal

The design criteria for decontamination waste disposal, RL-GEN-1353, N-REACTOR DESIGN CRITERIA - DECONTAMINATION WASTE TRUCKING FACILITIES, was approved by General Electric, Isochem and the AEC, and issued.

Environmental Limits

During the last report period no release to the environs of radioactive or chemical effluents in liquid or gaseous form occurred in which the AEC, U. S. Public Health Service, or Washington State Limits were exceeded.

2. COPRODUCT MISSION

2.1 Coproduct Status

The fourth increment of the coproduct load was charged during the April 18 outage and there are now 844 columns of Mark II fuel in the reactor. Reactor performance continues to agree with predictions.

2.2 Fuel Performance

Two columns of 1.96 coproduct fuel were discharged and examined after failure No. 38 occurred in one of them on March 24 (reported last month). The failure occurred in process tube number 1644, which had been charged with the 1.96 percent enriched Mark II prototype on June 15, 1965, and had attained a tube average exposure of 4094 Mwd/t. The failed element was the second fuel element in the column. Failure appears to be end-associated; the end cap is nearly detached; the fuel element is fragmented over the last six inches; the fragmented length is detached from the remainder of the element by a transverse fracture.

All of the other fuel elements in the column were in excellent condition. Even at the high exposure attained, buckling of the inner cladding was minimal. This superior performance is attributed to the 800 ppm Al plus 400 ppm Fe alloy core metal. It should be noted that this column and column 1844, which was also discharged, were operating at 550 and 575 C maximum uranium temperature. This is the temperature range predicted for Mark V fuel.

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Swelling results for the fuel elements from these two columns are shown in Table I. Bore diameters were measured on a few of the drivers and all were within 10 mils of their nominal unirradiated values.

As reported previously, the swelling data on the iron-aluminum-uranium alloy vary widely although the largest values recorded are only about one-half of that for the standard iron-silicon alloy under equivalent irradiation conditions. The reason for the greater variability is being studied.

Radiometallurgical examination of fuel element failure No. 34 has been completed. Like the other end-associated failures, the braze at the failed end showed hot tears in the region from the underside of the weld-recast volume to the uranium interface below the braze layer. This phenomenon has been observed only in failed closures. Unfailed closures on the same elements have not been similarly damaged. A report summarizing the investigations of end-associated failures is being prepared.

Two columns of Mark II fuel that had reached 2500 to 2700 Mwd/t (approximately 125 percent goal) were discharged. Inspection showed them to be in good condition. They are being separated for detailed evaluation. One is a striped column with high carbon uranium drivers.

2.3 Coproduct Transition

The Li-6 content of the base Mark II elements charged during the increment (No. 3) was about 6 percent higher than the first two increments. This resulted in a reactor average increase of 2 percent of the design Li-6 concentration, and a reactivity loss of 0.9 mk from the predicted values for the third increment. If the data for increment No. 3 are extrapolated out to 23 days of operation, the reactivity will be 5.6 mk or 0.8 mk lower than predicted which agrees with the change in Li-6 concentration. If the reactor average Li-6 concentration remains at the same level as the third increment, the reactivity of the reactor after the fifth increment is charged will be 7.5 mk and decrease to 6.1 after 23 days.

2.4 Process Tube Outlet Temperature Limits for PT-NR-75 (Coproduct Two-Load Demonstration Test)

Results from thermocouple measurements in Production Test-NR-66 and Production Test-NR-75 have been applied to establish process tube outlet temperature limits for Production Test-NR-75 during the power level increase probe. Since readings for only two trains are available for a standard full-length fuel column and maximum outer annulus imbalance measured by the two trains covers a 4 percent spread (a normal situation), indirect means were used to estimate the maximum imbalance to be expected with Mark II fuel in a process tube of any diameter.

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TABLE I
FUEL SWELLING IN TWO PT-NR-13 COLUMNS

Position From Rear	Column No. 1644		Column No. 1844	
	Exposure Mwd/t	Swelling % V	Exposure Mwd/t	Swelling % V
1	1013	-0.94	1157	+0.38
2	2432	-0.94	2748	+0.94
3	3467	-0.15	3904	-
4	4499	+1.99	4483	+2.22
5	4783	+2.68	4917	+2.52
6	4702	+1.21	4266	+3.04
7	5270	+2.30	5567	+3.97
8	5229	+2.61	5350	+4.26
9	5188	+3.82	5495	+3.82
10	4864	+1.29	5206	-
11	5188	+3.17	5423	+3.67
12	5351	-3.71	5350	+2.82
13	4621	+2.39	4844	+2.20
14	3243	failure	3326	+1.26
15	1459	+0.48	1591	+0.17

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The formulation comprises the product of independent factors involving fuel dimensions; thus, probability is low that a product would deviate a significant amount from one obtained through use of nominal dimensions. Hence, the analysis considered the dependency of enthalpy imbalance on the relationship of nominal fuel and fuel support diameters to a specific inside diameter of a process tube. The factor used in the limits formulation to accommodate for enthalpy imbalance thus was comprised of one term that accounts for the temperature rise imbalance between the outer annulus and the bulk tube with nominal dimension fuel and another term that accounts for the imbalance caused by eccentricity in the outer annulus of a tube of specific size containing fuel having nominal dimensions. The results indicated that the greatest imbalance would occur in the largest diameter process tube in spite of the off-setting effect of increase in annulus dimension.

A document (RL-GEN-1589) has been issued to summarize results and provide recommended formulations for tube outlet temperature limits as calculated from the formulations, the maximum achievable reactor level permitted by tube outlet temperature limits would be 4770 Mw, assuming a main steam header pressure of 130 psi, a maximum tube flow of 97,000 lb/hr and a low pressure trip set at 1450 psia. Reduction of instrument error allowances by calibration of the all-tube temperature monitor (planned during the hot physics tests), together with operation with a lower secondary steam pressure, should permit operation at 4800 Mw without pushing tube outlet temperature limits very hard.

2.5 Fuel Life Testing

Life testing of the Mark II fuel in the out-of-pile test loop (189-D) was completed during the past month. Final charge makeup and accumulated testing times are shown in Table II. With the completion of these tests, life testing of the Mark II fuel will be discontinued in order to permit development of support systems for Mark IV and Mark V fuel designs. The long and successful testing program, together with satisfactory results in the reactor, has proven the soundness of the spring-stop support system of Mark II fuels.

2.6 Extraction Testing

A test with unirradiated targets at Savannah River Plant in their Line 2 furnace results in complete penetration of a 3/8-inch thick wall stainless steel crucible. The corrosion occurred at the liquid metal air interface during the cold test approximately eight inches from the bottom of the crucible. The reason for the penetration has not been established. It is contrary to all earlier laboratory scale tests. Work is continuing at Savannah River Plant to explain the results.

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TABLE II
SUMMARY OF EX-REACTOR LOOP TESTING
DURING MARCH 1967

<u>Series</u>	<u>Fuel Type</u>	<u>Support Type</u>	<u>No. of Elements</u>	<u>Test Conditions</u>	<u>Test Time Accum. hr.</u>	<u>Series Term. Date</u>
22	7.5-Inch Mark I	Spider	4	340 gpm 570 F	570.5	3-21-67
	Standard Mark I*	Buggy Spring	4		570.5	
	Mark II	W-spring (hand formed)	1		2973.75	
	Mark II	W-spring (machine formed)	1		2395.75	
	Mark II	W-spring (production grade)	2		1875.75	
	Mark II	W-spring (production grade)	2		1520.25	
	Mark II	W-spring (offset)	1		853.0	
	Mark II	W-spring (13-inch reworks)	1		853.0	
	Mark II	W-spring (13-inch reworks)	2		368.5	

*Driver supports have shoes with tabs called "winged shoes."

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Additional prototypical cold target subassemblies were shipped to and received by Savannah River Plant for a repeat cold performance test during the week of April 24. The first hot performance run is scheduled for the week of May 8 and will be followed by an analytical run (conversion data) the week of May 15.

2.7 Target Shipments

The preparations necessary for shipment of targets to Savannah River are on schedule. The SCRUP cask, mounted on the Union Pacific railcar, is now at the N-Reactor ready for loading. The Production Test-NR-66 targets for yield determination have been stripped and baskets are available to accommodate the initial shipment. Since two cold runs are still in progress as Savannah River, the initial shipment using the SCRUP cask will not be made until it is possible to include all the remaining targets from Production Test-NR-13 (Coproduct Demonstration Test - 1.96% Enrichment) and any other miscellaneous targets now at the N-Reactor.

2.8 Blowdown Studies - 4800 Mw Operation

A study was made of a primary loop depressurization after an inlet header rupture at 120 percent power operation. A tank model such as those⁽¹⁾ previously applied to this accident was used: major differences were higher initial values⁽²⁾ which correspond to operation at 4800 Mw, and frictional resistance, based on actual operational data⁽³⁾ which are less than the theoretical predictions. With these conditions, the blowdown time was found to be of the same order of magnitude as that predicted for an inlet header rupture at 4000 Mw operation⁽²⁾. The initial hot leg depressurization from 120 percent power was found to occur at a faster rate than from 100 percent power, primarily because of the reduced flow resistance. In 6 seconds after the rupture, the hot leg pressure for both cases is found to be the same. The subsequent depressurization from 120 percent power would be at least as rapid as that calculated for 100 percent power, again because of the reduced flow resistance.

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- (1) D. D. Stepnewski. N-REACTOR PRIMARY LOOP BLOWDOWN FROM MAIN HEADER RUPTURES - PHASE II, RL-NRD-64. November 12, 1964.
 - (2) E. G. Pierick. FUEL REFERENCE DESIGN AND PERTINENT OPERATING DATA FOR COPRODUCT DEMONSTRATION LOAD, RL-NRD-737 REV4. June 8, 1966.
 - (3) R. H. Shoemaker. PRELIMINARY EVALUATION: HYDRAULIC CHARACTERISTICS OF PRIMARY COOLING PUMPS AND EX-REACTOR PIPING, RL-NRD-222. February 3, 1965.

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The tank models previously applied to primary loop blowdowns have considered two phase flow effects in the cold leg; however, no consideration was given to two phase flow phenomena during the isenthalpic expansion of the hot leg fluid. The effect of two phase pressure drop in the hot leg pipe and in the heat exchanger is currently being examined to ensure that the effect on blowdown times is of negligible concern. The study is being performed on the 1107 computer because of the copious repetitive calculations required for solution of the two phase flow conditions.

2.9 Mark V Fuel Design Studies

Initial studies are complete for the purpose of supplying thermal hydraulics parameters for design of Mark V fuel. These studies considered parametrically the minimum process tube flow rate (thus maximum fuel weight) that could be tolerated without occurrence of bulk boiling in fuel subchannels, within the capabilities of the process tube flow monitor system. Design assumptions and results are shown in Table III.

The tube flow rate obtained in these studies is about 3.5 percent greater than that used in Mark IV design designated for use in the first advanced technology studies perviously reported (RL-GEN-900). Since by itself this result would require a reduction in fuel weight of about two pounds per foot (uranium), it will be necessary to follow the same procedure taken in the design of Mark IV fuel - namely, to decrease the hydraulic resistance of the supports. A projected decrease in fuel column length from 35 to 31 feet will help to improve flow characteristics.

2.10 Zone Temperature Monitor Limits

An analysis has been made to determine if the zone temperature monitor (ZTM) system will limit reactor power or steam pressure during the latter stages of the coproduct demonstration program. Complications were anticipated when the reactor is operated with a mixed fuel load of Mark II and Mark IC fuel, at a time that Cell 3 might be out of service for retubing. Reactor operating conditions would be complex under those conditions inasmuch as an unbalanced flow pattern would result from the mixed load. The most critical fuel element mixture, as related to zone temperature monitor limits, would be a small percentage of Mark II fuel and a large percentage of Mark IC fuel since this will result in the lowest flow rate and highest outlet temperature in the Mark II columns. The analysis indicated that the reactor could operate at 4000 Mw and 150 psia steam pressure with no interference from ZTM limits. An operating margin of 14 F below ZTM limits was calculated.

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TABLE III

DESIGN ASSUMPTIONS AND RESULTS FOR MARK V FUEL

Assumptions:

Reactor power	4800 Mwt, 800 Mwe
Maximum tube power	5650 kw
Radial flattening efficiency	85 percent
Orificing efficiency	89 percent
Low pressure trip level	1450 psia
Subchannel enthalpy imbalance	10 percent
Instrument error in temperature measurement	5 percent
Operating margin in flow	6 percent

Results:

Tube flow rate	96,030 lb/hr
Reactor inlet temperature	389.2 F
Maximum tube outlet temperature	560.0 F

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3. TRANSPLUTONIUM TECHNOLOGY

Results accomplished in April not at a reportable stage.

4. PLUTONIUM-238

4.1 Target Column Evaluation

Analytical results from the first (0856) target column discharged have been evaluated and are detailed in Figures 2 to 6. The quantity of plutonium produced and the neptunium consumed is in good agreement with predicted values. The purity (i.e., Pu-238 in total plutonium) is significantly better than predicted. The reasons for this are not yet obvious and further analysis is required. Since the total production is about as predicted, the higher purity may be due to the use of Pu-238 resonance or absorption cross sections which are too large. In Figure 3 the measured neutron flux is plotted utilizing the weasel⁽¹⁾ data to determine the flux shape. The peak flux measured on element 7 was 8.57×10^{13} neutrons/cm²-sec. This flux was calculated using the bare cobalt-aluminum wire data and a 2200 m/sec cross section.

The production parameters for each target or group of targets along the column were calculated and plotted as a function of the accumulated exposure. The average flux over the column was 6.94×10^{13} n/cm²-sec of 4.5×10^{20} . These data are plotted in Figures 3 to 6. The conversion efficiency was calculated as:

$$\text{Efficiency of conversion to Pu-238} = \frac{\text{Pu-238}}{\text{Np Consumed}} = \frac{\text{Pu-238}}{\text{Pu (total)} + \text{Cs}}$$

The measured neptunium in the unirradiated targets was assumed to be most correct since the chemical analysis was not complicated by the presence of large amounts of plutonium. The neptunium analysis of the irradiated targets seems to be inconsistent and therefore the total plutonium and cesium were used to determine the target burnout. The quantity of Pu-238 in the plutonium is plotted as a function of exposure in Figure 4 and is self-explanatory. The parameters alpha and (1-beta) are plotted in Figures 5 and 6, respectively.

(1) Weasel data - flux assessments obtained from direct radiation measurements of discharged fuel.

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Figure 2. 0850 Target Column Flux Data

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Figure 3. Efficiency of Conversion of Np-237 to Pu-238

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Figure 4. Product Purity

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Figure 5. α : Grams Pu-238 Produced per Gram Np-237 Charged

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Figure 6. 1- β Target Burnout

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The data from the second (0858) test column are being evaluated; however, at this time the production parameters are not available. These data will be plotted on the same graphs as discussed here to show the dependency on exposure and neptunium concentration. The neutron flux on the second column has been calculated. The maximum flux point (element 6) yielded a value of 1.07×10^{14} n/cm²-sec.

The results of the experimental measurements of spike-to-target column ratios during the recently completed neptunium irradiations have been evaluated. Corrections in the measurements have been made for (a) the short-column effect - water segments at the ends of the columns act as poison; (b) exposure differences, measurements were made with green or fully exposed spike charges - both have been corrected to average exposure; and (c) base column type - the reactivity effect of Mark I-A (spike) is different when compared to Mark I than when compared to Mark II. The results of these corrections indicate a spike-to-target ratio of 5.3:1 for 70 grams of neptunium per foot and 4.2:1 for 45 grams of neptunium per foot.

The theoretical ratio of spikes to target has now been computed by three independent methods: (a) MOFDA, (b) four-group diffusion theory - HFN -, and (c) four-group transport theory, including both cylindricized and actual supercell representation. All models yield values of 4:1 and 3:1 for the 70 and 45 grams per foot columns respectively. These values are acceptably close to the experimentally derived values.

If the theoretical calculations are extrapolated to the proposed production target design of 15 percent neptunium in aluminum (approximately 130 grams per foot), a ratio of 5.6 to 1 is obtained. Correcting this to correspond to the experimental observations yields a ratio between 7.3 to 1 and 7.6 to 1.

4.2 Neptunium Production in Mark I Fuel

Uncertainties in the calculations have led to a lack of confidence in the neptunium production predictions as now contained in the CAGE program; therefore, a proposal for a production test to experimentally confirm these predictions is being prepared. Specifically, the test is to provide high quality data on the production of neptunium for Mark I type fuels of various U-236 levels irradiated to various exposures. As currently envisioned, groups of four or five columns of fuel would be irradiated to three different exposures. For each exposure level there would be four or five batches of fuel of different U-236 concentration. Statisticians from Battelle Northwest Laboratories will aid in both the design of the test and in the analysis of the results.

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In the preparation of the fuel, it will be necessary to have manufactured sufficient inner and outer billets at each U-236 level to provide a minimum of 12 full-length columns of Mark I (or Mark IV) assemblies (enough for three exposure levels). Three levels of U-236 are planned - 1000 ppm, 2000 ppm and 3000 ppm. To provide this, it will be necessary to obtain some 100 kg to 200 kg of VI-B tails containing greater than 15 percent U-236. Using this as sweetener, Dresden or Yankee tails from NFS can be blended up to 0.947 w/o U-235. In addition, there are some 10 tons of depleted tails from Shippingport in storage at Isochem. This material can be blended up with VI-B tails to a 0.947 w/o U-235 fuel containing more than 3000 ppm U-236. A tentative schedule indicates that the work could be completed within two years after authorization. It is to be noted that eventually this information would become available under our present fuel recycle mode of operation; however, the proposed test would shorten the time by a factor of two or three.

5. OTHER ISOTOPES

Thulium Target Flow Tests

Tests have been conducted in the Battelle Northwest Thermal Hydraulics Laboratory to determine the vibration or chattering produced by the thulium target in a graphite coolant tube. Results were as follows:

- a. Below approximately 13 gpm very little target motion was detected relative to the tube.
- b. Significant target motion was noted at approximately 14 gpm and vibration increased with flow rate up to 21 gpm which was the maximum flow rate used in the test.

Calculations indicate that the flow rate reduction resulting from charging the graphite cooling tube with 14 spacers and three thulium targets should be approximately 10 percent. Based on the satisfactory experience with the cobalt irradiation, there appears to be no potential for fretting problems in the thulium irradiation.

A cobalt target in the form of a flat bar was also tested with results as follows:

- a. Below approximately 10 gallons per minute the vibration detected indicated very little target motion relative to the tube.
- b. Large vibrations (audible chatter) were noted in flow rate range of approximately 14 to 19 gpm. This motion was independent of target orientation.

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6. ENRICHED FUEL PROCESSING

Results accomplished in April not at a reportable stage.

7. TARGET SPACE ENHANCEMENT

Not a Hanford Atomic Products Department program at present.

8. NUCLEAR SAFETY

8.1 Fuel Thermal Transients on Loss of Coolant

A parametric study of the parameters affecting Mark II coolant-loss thermal transients is complete. The model used is a refinement of that used for studies previously reported. Recent additions and modifications to the model are as follows:

- a. A new power decay curve accounts for variation in heat distribution during power decay.
- b. The Baker and Just parabolic Zircaloy reaction rate constants have been substituted for previously-used values.
- c. Heat of transformation of the Zircaloy alpha-beta phase change has been accounted for.
- d. Gray body radiative heat transfer is now calculated on the basis of oxide thickness dependent emissivity.
- e. Thermal expansion of the fuel is accounted for in the heat transfer calculations.
- f. Credit is taken for turbulent heat transport during the initial blowdown of inlet piping water in applicable accidents.

A list of assumptions is given in Table IV.

The refined model yields a melting time of 151 seconds for fuel normally operating at 235 kw/ft. Curves of melting time versus specific power and degree of heatup versus time are included as Figures 7 and 8. Of particular interest is Table V which shows the effect of varying assumed parameters about the nominal values used in the calculations.

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TABLE IV

ASSUMPTIONS - MARK II MELTING TIMES

HEAT GENERATION

Nuclear	Scram generation decay includes distribution of heat to lattice components
Chemical	Zircaloy oxidation rate computed from parabolic rate law

HEAT STORAGE: Specific Heats and Heats of Fusion Accounted for:

Ceramic	Solid phase heat capacity
Aluminum	Sold, liquid-phase heat capacities. Heat of fusion.
Uranium Alloy	α , β , γ , molten-phase heat capacities. Heats of transformation and fusion.
Zircaloy	α - β heat capacities. Heat of transformation.
Graphite	Solid-phase heat capacity.
Gases	Heat capacity neglected.

HEAT TRANSFER: Transport by conduction in solids and gases and Radiation in gases.

Conduction	Steady state thermal resistances employed. Thermal conductivities of gases evaluated as function of average temperature
Radiation	Gray body theory. Emissivities of reacting zirconium calculated as function of oxide growth from chemical reaction.
Forced Convection	Credit taken for blowdown of water stored in reactor inlet piping (only for applicable accidents).
Free Convection	Determined negligible from Beckman correlation.
Thermal Expansion	Average dimensions during transient used.

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TABLE IV (continued)

INITIAL CONDITIONS: Temperatures

Driver and Target	Volume averaged initial temperature calculated from internal resistance, film resistance, and process water temperature.
Driver	Non-uniform power generation considered.
Target	Uniform power generation assumed.
Process Tube	Initial temperature = process water temperature.
Moderator	Temperature is related to initial power generation and process water temperature. Evaluated from curve fits to steady heat transfer program results.

CRITICAL ASSUMPTIONS

Assumptions of primary importance to evaluation of timing and degree of earliest melting:

<u>Assumption</u>	<u>Source</u>
Decay and distribution of power generation.	Internally calculated. WS Nechodom.
Surface emissivities of reacting zirconium.	CEMP482 (preliminary draft). White, Sjodahl, Bittel. AEC sponsored material properties review. 1967.
Heat capacity of uranium and Zircaloy in various phases.	Reactor Handbook, Materials Volume. AEC Sponsored Technology Review. 1960.
Initial Fuel Temperature.	Flex fuel design and analysis program.
Initial Moderator Temperature.	STHTP steady state heat transfer program.
Uranium alloy melting point.	BNWL-CC-499.
Solidus 1994 F	
Liquidus 2034 F	

Driver
Melting
Time
(Seconds)

Figure 7. Melting Time Versus Specific Power in Mark II Fuel

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Percent of Core
(Weight by Power)

Figure 8. Fraction of Core Molten at 1994 F and at 1900 F Versus Time

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TABLE V
SENSITIVITY OF MELTING TIME TO ASSUMPTIONS AND PARAMETERS

<u>Parameters</u>	<u>Assumed Uncertainty</u>	<u>Effect Upon Allowable Power</u>
Initial Temperatures		
Driver	± 40 F	$\pm 1\%$
Target	± 100 F	$\pm .2\%$
Moderator	± 25 F	$\pm .5\%$
Driver Dimensions	± 10 mils on Nominal I.D. and O.D.	$\pm 1.8\%$
Over-all Heat Capacity (Including Phase Change)		
Driver	$\pm 2.5\%$	$\pm 1.1\%$
Target	$\pm 10\%$	$\pm .06\%$
Moderator	$\pm 5\%$	Negligible
Thermal Conductivities		
Metal (fuel, process tube)	$\pm 5\%$	$\pm .05\%$
Steam	$\pm 2\%$	$\pm .1\%$
Moderator	$\pm 10\%$	$\pm .2\%$
Uniform Emissivity (Not Applicable)	$\pm 10\%$	$\pm 1.7\%$
Initial Oxide Thickness	$\pm 50\%$	$\pm 2.2\%$
Fuel Alloy Melting Point	± 20 F	$\pm .6\%$

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"Melting uranium" model additions to the melting time computer code have been made. This model assumes that, once the uranium reaches the solidus, each temperature increment results in one fortieth of (time increment) x (uranium temperature rise per time increment) x (uranium volume) of the uranium melts, falling to the process tube, forming a new surface on the uranium already there, oxidizing at the molten temperature, then coming to thermal equilibrium with the process tube. In this simplified model, no oxidation or uranium remaining within the jacket or exposed interior jacket surfaces is assumed, and no change in heat transfer coefficient between the fuel and process tube is made even though the puddled uranium touches the fuel.

Establishment of heat transfer coefficients, power decay rates and emissivities is in progress to provide the bases for more accurate calculation of hydrogen release rates. Initial evaluations indicate that the uncontained uranium will about double the hydrogen gas release rate over the contained uranium rate.

In a test of an inner component where failure occurred at 1060 C in steam, the element was oxidized quite rapidly during the cooling portion of the cycle when the temperature dropped below 400 C. This is confirmed by data of Argonne National Laboratories and others. This factor will be considered in the fuel melting analyses which are being made.

Work is also in progress to obtain a better knowledge of the temperature transient for the process tube, especially the maximum temperature and the time required for it to drop to 800 C. This would permit a better assessment of the affect of this factor on the fuel melting mechanism.

The fuel tested to date was irradiated to about 1500 Mwd/t. This is considered to be a good approximation of an "average" fuel element in the hot zone of the reactor. However, cladding failure is apparently caused by internal gas (or foam) pressure and at 3000 Mwd/t the pressure should develop much faster as the temperature rises. The strength of Zr-2 declines with temperature in the range above about 700 C. This will compensate to a degree for the increased amount of gas. Elements irradiated to higher exposure will be included in the test program soon.

8.2 Metal-Water Studies

Metal-water reaction tests on unirradiated fuel were initiated. One test of 75 minutes duration has been completed and a gas sample was obtained for measurement of the hydrogen content.

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8.3 Wahluke Slope Meteorology Study

A computer program has been written to analyze compression of air stream lines over sloping terrain. With computer results from this program, it will be possible to estimate the increase in dose from air compression only. The next step will take into consideration meteorological factors such as cooling of the air streams by condensation.

8.4 Smoke Tests

Several trial runs have been completed in the program to determine wind patterns with smoke releases.

8.5 Pipe Rupture Studies

Earlier studies of a complete inlet riser break⁽¹⁾ were made on the assumption that the emergency raw water (ERW) system would continue to be intact and operable. The results indicated that the fuel would not melt with ERW cooling available. The study has been re-opened to evaluate the effect on fuel melt time if a simultaneous failure of the ERW is assumed. The resulting analysis indicates that all process tubes will be adequately cooled until about 250 seconds after the incident. At 250 seconds, the system pressure is calculated to be about 130 psia and the saturated temperature about 347 F. Since the heat transfer mechanism is forced convection to a 2-phase mixture, the fuel element temperature would be only slightly above the saturation temperature. The heat stored in the intact fuel during the interim between 10 and 40 seconds would be quenched before 250 seconds. At that time, only the decay heat being generated must be removed. On this basis it was determined that fuel melting would not occur before 300 seconds, which is the confinement vent closure time.

Calculations were also made to determine the transport time from the reactor outlet to the inlet header through the heat exchanger cells following a partial riser break. The transport times ranged from 160 to 180 seconds for periods of time between 100 and 150 seconds after the break. Although ERW cooling was assumed to be unavailable to the reactor core, the effect on transport time if ERW were available was small.

(1) RL-NRD-683, INLET RISER BREAK DURING COPRODUCT OPERATION, December 15, 1965.

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9. **WASTE MANAGEMENT**

Not a Hanford Atomic Products Department program at present.

10. **COLUMBIA RIVER STUDIES**

Not a Hanford Atomic Products Department program at present.

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INCREASED POWER LEVEL DEMONSTRATION PROGRAM

There are several incentives which justify early efforts to demonstrate the capabilities of operating the N-Reactor at a 4800 Mw power level. These include:

- a. An indicated production increase of 20 percent,
- b. A decrease in unit cost of 10 percent in the products produced,
- c. An improved flexibility in maintaining a multi-product program.

Improved flexibility accrues from the capability of diverting a portion of the process channels to the production of other desired isotopes while maintaining a reactor power level of at least 4000 Mw by operating the fueled channels at limits equivalent to 4800 Mw operation.

The Mark II (coproduct) fuel was designed to operate at a reactor power level of 4800 Mw. It is not possible, however, to operate the reactor at 4800 Mw with the Mark I, plutonium-only, fuel element in place because of the enthalpy imbalance between the coolant subchannels of the current plutonium-only fuel elements. A new plutonium-only fuel for 4800 Mw operation is therefore being designed but will not be ready for reactor use until late in 1968.

Early identification of problem areas associated with 4800 Mw operation is extremely desirable to permit correction of any system deficiencies prior to loading the reactor with this improved fuel element. Limitations that might be identified include the capacity of key systems, the response dynamics of key systems, and operating margins and fuel design parameters. Because the Mark II fuel was designed for 4800 Mw operation, the program provides an excellent opportunity to probe higher power levels, and to safely evaluate the capacities and responses of the key reactor systems during the coproduct demonstration test.

Areas discussed in this presentation include improvements which will contribute to increased power levels, and additional safety factors. The higher power levels will be obtained from:

- a. Primary system pressure increase
- b. Improvement in Zone Temperature Monitor system
- c. Calibration of all-tube temperature monitoring system
- d. Gas atmosphere control improvements
- e. Traveling wire flux monitor improvements

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- f. Process tube orificing
- g. Circulating raw water weir modifications
- h. Completion of enthalpy studies to establish tube outlet temperatures which provide operating conditions which ensure that bulk boiling does not occur.

Additional safety factors will be introduced by the following improvements.

- a. Addition of an emergency raw water confiner trip signal
- b. Fog spray improvement
- c. Transient analyses which have verified capability of pressurizer tank to handle transients from higher power levels.

The following chart depicts the schedule of the planned probe for higher power level operation of the N-Reactor. Major points of the plan include:

- a. Raising the reactor power level to 4400 Mw, evaluating the reactor operating characteristics, followed by a scram test to evaluate the transient effects of a scram from 4400 Mw.
- b. A stepwise increase in reactor power level above 4400 Mw until prudent application of operating margins or safety limits decree that the maximum power level has been reached or the power level has attained 4800 Mw. A scram test would be performed from 4800 Mw to establish the transient effects that accompany such a scram.
- c. The WPPSS electrical generating plant will be shut down for a Class A inspection during the test.

- 1. Achieve Full Load Mk II
- 2. Hot Physics Tests
- 3. Calibrate Rods & Balls
- 4. Replace 200 Cols Mk II

- 5. Probe 4400 Mw
- 6. Scram Test
- 7. Probe 4800 Mw
- 8. Scram Test

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KEY ACTION PLAN

The key action plan for the Power Level Increase program is shown below. Progress of the reactor loading program has incurred no delays. Activities prerequisite to performing the test should be completed in ample time to permit the test to be accomplished during June.

1967

Safeguards Summary Report	February
Safeguards Review	March
ACRS Review	April
Operating Limits	April
Achieve Full Load	May
Reactor Modifications	May
Probe 4800 Mw	June

ENTHALPY IMBALANCE STUDIES

The Mark II fuel configuration lends itself to channeling of the coolant flow through the column length. Therefore, fuel dimensions and orientation within the process tube will affect the relative enthalpy increase along the channel length. Since current safety philosophy requires that there be no bulk boiling in any channel, the maximum tube outlet temperature permitted must be low enough to compensate for any potential enthalpy imbalance along the flow channels in a process tube. The enthalpy study is designed to establish the tube outlet temperature limits which provide the desired operations under conditions which prevent bulk boiling.

MARK II

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Schedule - Enthalpy Studies

Install four thermocouple trains on fuel column loaded with Mark II fuel.

Utilize data to determine tube outlet temperature limits.

Expected imbalance ~10 percent.

Gain in power ~400 Mw.

SYSTEM IMPROVEMENTS

System improvements, or changes, which are planned or in progress are discussed in the following sections. As previously noted these will provide for the safe and effective evaluation of reactor systems capacities under higher power level conditions.

Primary System Pressure Increase

Additional power potential will be obtained by increasing the minimum ensured primary system pressure.

Operating limits for tube outlet temperatures are calculated, based on a minimum pressure at the outlet risers of the reactor. This minimum pressure is ensured by the low-pressure scram trips which are located in four of the outlet risers. These trips are currently set at 1450 psig, and accuracy is considered to be ± 30 psi. The maximum tube outlet temperature is determined at the minimum potential pressure which is 1450-30 or 1420 psig.

New high-low pressure trips of a static O-ring design have been installed and are expected to have an accuracy of ± 10 psi. If operating data verify the accuracy of these trips, the low pressure trip point can be increased which will increase the minimum saturation temperature and power potential. Currently an increase in temperature limits equivalent to 80 Mw is expected.

ZONE TEMPERATURE MONITOR IMPROVEMENT

The zone temperature monitor system (ZTM) has been a significant maintenance problem as evidenced by the high failure rate of the resistance temperature detectors (RTD's). The repair of these RTD's has resulted in large expenditures in manpower and radiation exposure. All process limits for 4000 Mw operation can be met with a substantial number of these RTD's out of service, but as the number of operable RTD's increases, the allowable process tube operating temperature and power decreases. Improved prototype

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RTD's have been tested which will provide improved service and reduced installation costs. These RTD's will be installed to establish the system at full operability. This increased operability will result in a potential improvement in ZTM limits equivalent to 85 Mw power.

Other changes unrelated to the plant will be made which will result in temperature limits which will permit operation at 4800 Mw. These include an improved method of limits application.

CALIBRATION OF ATTM's

The all-tube temperature monitoring system (ATTM) is used to monitor the process tube powers, and outlet temperature, and to provide a secondary method to calculate total reactor thermal power. For this reason, the accuracy of system is extremely important.

Subsequent to obtaining a full reactor load of coproduct fuel, and prior to the demonstration, a hot physics test will be completed which will permit calibration of the ATTM system. The test will be completed at a temperature of approximately 450 F. During this test the heat generation of the reactor will be negligible and the primary system coolant will be a uniform temperature. This will allow precise determination of the accuracy of all the temperature monitors in the primary system and will provide an added margin of safety for temperature and power monitoring.

GAS ATMOSPHERE CONTROL IMPROVEMENTS

Gas atmosphere moisture control will be improved so that the more stringent moisture control requirements of higher power operation can be maintained.

Higher power operation will result in higher graphite moderator temperatures and higher process tube temperatures. As the process tube temperature increases, the required moisture content in the gas system increases in order to maintain a protective oxide coating on the zirconium process tubes. This oxide coating is required to prevent absorption of hydrogen by the zirconium tube and subsequent embrittlement. The graphite moderator imposes a contrasting upper limit on the moisture content of the gas system. As the moderator temperature increases, the maximum moisture content in the gas system must decrease to prevent excessive oxidation of the graphite.

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The result of these two requirements is a narrowing band of allowable moisture composition at higher power levels. The moisture composition of the gas is presently controlled by use of two silica gel drying towers. The rate of moisture removal is controlled by controlling the moisture in the drying towers and by proper switching from one dryer to the other. This method of control is rather coarse and does not permit the precise control which is desirable for higher power operation.

To improve the moisture controlling capability, a bypass with a modulating capability will be installed around the dryers. This modification is scheduled in May 1967.

TRAVELING WIRE FLUX MONITOR IMPROVEMENTS

The traveling wire flux monitor (TWFM) system will be returned to maximum operability to provide maximum monitoring of specific power limits.

The nine traveling wire flux monitors are used to measure front-to-rear flux profile which is an indication of specific power. This information is also used to calibrate the in-core flux monitors which are especially important during reactor startups to monitor specific power limits.

The TWFM system has been plagued with mechanical problems which have resulted in poor operability of the monitors. With this reduced operability, added conservatism has been required for specific power potential of the reactor.

The TWFM system will be returned to maximum operability prior to the demonstration so that specific powers can be determined with maximum accuracy. A maintenance program for the system was completed in April and all nine of the monitors are operable.

PROCESS TUBE ORIFICING

Process tube orificing may be increased in order to increase central zone tube flow and decrease the maximum tube outlet temperatures.

Fringe process tubes are presently orificed to reduce the coolant flow rate to these low power tubes with a resulting increase in the flow to the higher power central zone tubes. The degree of orificing is less than the maximum for six-cell operations so that the reactor can be operated with five cells without orifice changes. In addition, the application of temperature limits as related to orificing becomes simpler as the degree of orificing decreases. For these reasons the present degree of orificing or orificing efficiency is not at the maximum.

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New orificing methods are being developed which will permit simpler limits of application and easier installation. Special nozzle cap orifices and upstream dummy orifices have been developed.

These orificing methods will be available for use during the demonstration and the orificing pattern will be adjusted as needed to provide acceptable outlet temperatures. It is technically feasible to obtain an additional 4 percent flow increase to the central zone tubes above the flow rate which will be obtained by the present orificing pattern. Orificing changes could therefore increase the power potential as related to tube outlet temperatures by approximately 150 Mw.

CIRCULATING RAW WATER SYSTEM WEIR MODIFICATIONS

Additional heat removal capacity has been obtained for the circulating raw water (CRW) system by increasing the system pressure and thereby increasing the saturation temperature.

The heat removal capacity of the CRW system is a function of total flow rate, inlet temperature and operating pressure. The flow rate and inlet temperature are controlled by the river conditions; however the operating pressure at the heat exchangers is controlled almost exclusively by the weir in the discharge line.

Engineering studies have shown that the weir height is not optimum for maximum power removal. An increase in the height of the weir would result in significantly higher system pressure and therefore, saturation temperature, but only a slight decrease in total flow rate. The net result is a higher heat removal capacity as shown on the following illustration.

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A small increase of three feet has been added to the height of the weir and system pressure measurement will be taken to establish the operating pressure accurately. It is estimated that an additional 150 Mw capacity has been added to the system. This system currently has ample capacity for 120 percent power operation in June.

EMERGENCY RAW WATER CONFINER TRIP

An additional emergency raw water system (ERWS) actuation signal will be added to ensure adequate cooling following a postulated critical-sized inlet riser piping break accident.

The emergency cooling system has been designed to trip on the coincidence of any two of the following signals:

- Very high coolant temperature
- Very low primary pump speed
- Very low coolant flow
- Very low reactor outlet pressure
- Very high reactor outlet pressure

Investigations have shown this logic to be adequate for all accidents except a certain sized inlet riser break. For this accident a partial break upstream of the check valve is assumed which is just large enough to prevent flow past the check valve to the tubes supplied by that riser. In addition, this break is small enough to prevent rapid depressurization of the primary loop and thus would prevent early entry of emergency cooling water. The probability of this accident is so low that it might well be judged an acceptable risk. However, protection will be provided.

The addition of a high confiner pressure trip will forestall, and an increase in the very low-pressure limit will result in earlier actuation of the ERW system. After the ERW system has been actuated, the primary system pressure decay rate will be increased as a result of a blowdown through the ERW dump line. As a result, the ERW coolant enters the system at an earlier time and adequate fuel cooling is maintained.

This change is beneficial for operation at 4000 Mw as well as at 4800 Mw.

DECLASSIFIED**FOG SPRAY IMPROVEMENT**

The existing fog spray nozzles could become plugged by silt and vegetation in the spray water. These nozzles are being replaced with a design which has self-cleaning characteristics. Tests have adequately demonstrated the efficiency of this improvement, and installation of the new nozzles was completed during April.

OLD **NEW** **TRANSIENT ANALYSIS**

Transient analyses using digital analog computer techniques have been made to verify the systems ability to handle anticipated transients such as a reactor scram and key transient information is summarized on the following curves.

The primary area of interest within the primary loop is the capability of the pressurizer and high-pressure injection pumps to prevent drainage of the pressurizer both with and without loss of one electrical power source. The results show an increased outsurge of approximately five feet from that observed from 4000 Mw and this is within the capability of the system.

The main steam header pressure drop during a scram is relatively independent of reactor power while the surge tank water level is almost directly dependent. The surge tank water level is expected to drop an additional two feet relative to a scram from 4000 Mw and this is also within the capability of the system.

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PRIMARY SYSTEM TRANSIENTS FROM 4800 Mw

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SECONDARY SYSTEM TRANSIENTS FROM 4800 Mw

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SUMMARY

The following chart summarizes the effect of the various plant tune-up actions which have been described. The plant will have a capacity for operation up to about 4000 Mw if not limited to a lower level by the fuel specific power limit. The reactor power level attainable at a given specific power is dependent upon the power flattening which can be achieved, which is in turn dependent upon such factors as control rod insertion patterns, green/ripe fuel distribution, normal/spike fuel distribution and effect of graphite temperature on reactivity at higher temperatures. Final evaluation of these factors can only be made within the reactor as it approaches full power. Hence, the extent to which the fuel power density limit will restrict the attainable power level cannot be stated unequivocally at this time. However, it is not expected to limit power below 4800 Mw.

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SUMMARY

TECHNICAL POWER LIMITS

ADMINISTRATIVE SUMMARY

APRIL 1967

EMPLOYMENT

<u>Section</u>	<u>Permanent</u>		<u>Temporary</u>		<u>Total</u>		<u>Change</u>		<u>Total</u>
	<u>E</u>	<u>NE</u>	<u>E</u>	<u>NE</u>	<u>E</u>	<u>NE</u>	<u>E</u>	<u>NE</u>	
General	2	1	0	0	2	1	0	0	3
Closeout Group	12	22	0	0	12	22	0	0	34
Employee Relations	6	4	4*	0	10	4	-1	0	14
Finance	9	12	0	0	9	12	-1	0	21
Fuels	45	140	0	0	45	140	-3	-3	185
Plant	62	211	0	0	62	211	+3	-2	273
Project	54	17	0	0	54	17	-2	0	71
Research and Engineering	<u>52</u>	<u>19</u>	<u>0</u>	<u>0</u>	<u>52</u>	<u>19</u>	<u>0</u>	<u>0</u>	<u>71</u>
Total	242	426	4	0	246	426	-4	-5	672
New Change					672		-9		

*Technical Graduates

SAFETY

Days without a disabling injury	309
Hours worked without a disabling injury	1,082,409

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SIGNIFICANT REPORTS

<u>Report No. & Class.</u>	<u>Title</u>	<u>Author</u>	<u>Date</u>
RL-GEN-1065 SUP4 S	PT-NR-75, SUP4, Evaluation of Mark II Monitor Columns	EE Leitz AE Guay	4-24-67
RL-GEN-1065 SUP5 C	PT-NR-75, SUP5, All Tube Temperature Monitor Calibration	JW Long	4 20-67
RL-GEN-1353 U	N-Reactor Design Criteria - Decontamination Waste Trucking Facilities	WD Bainard	4 12-67
RL-GEN-1355 4 C	Reactor Design Analysis Monthly Record Report, April 1967	DD Stepnewski	4-21-67
RL-GEN-1443 S	Target Yield and Defective Rates	WH Hodgson	3-67
RL-GEN-1454 S	Target Yield and Defective Rates	WH Hodgson	3-67
RL-GEN-1490 U	Project Proposal, "Decontamination Waste Truck Loading and Unloading Facilities, 100-N and 200-E Areas," (Project GAP-410)	AA Janos	4-7-67
RL-GEN-1509 C	A Rationalized Set of Long Term Reactivity Curves for Mark I and Mark IA Fuel	JA Marks	4-10-67
RL-GEN-1533 U	Presentation to Meteorology Subcommittee of ACRS, March 8, 1967	MC Leverett	3-22-67
RL-GEN-1534 S	Average Power Densities for Various Reactors	RP Sharp	3-21-67
RL-GEN-1539 C	Fuel Heat Generation Following Shutdown for Mark I and Mark II Fuels	JA Marks	3 21-67
RL-GEN-1540 C	Production Test PT-NR-85 Zone Temperature Monitor and Burnout Flux Limits	LJ Agee	4-24-67

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SIGNIFICANT REPORTS (continued)

<u>Report No. & Class.</u>	<u>Title</u>	<u>Author</u>	<u>Date</u>
RL-GEN-1542 U	Proposed HCR Cooling System Improvements	PK Telford	3-27-67
RL-GEN-1546 C	Presentations to Hanford Subcommittee of ACRS, April 5, 1967	MC Leverett	4-1-67
RL-GEN-1546 SUP U	Effluent Control Program, Presentation to DRL and ACRS, April 5-7, 1967	RK Robinson	4-1-67
RL-GEN-1548 C	Support Welding on Unbonded Zircaloy Clad Target Elements for N-Reactor	DH Walker	4-20-67
RL-GEN-1552 U	Evaluation of the Tube Manifold Vent Pipe Failure in NPR Steam Generator #3B	JP Sluka	4-10-67
RL-GEN-1567 S	The 68 Spare Channels as a General Isotope Production Facility	JE Jackel	4-12-67
RL-GEN-1569 U	Reduction of Connector Activity by Reducing Reducing Coolant pH PT-NR-84	WK Kratzer	4-4-67
RL-GEN-1573 C	Fluorine Contents of Lithium Aluminate Power and Sintered Pellets	RR Studer	4-10-67
RL-GEN-1574 S	Hanford Atomic Products Department Plant and Capital Equipment Budget Submission for FY-1969 and Revision of FY-1968	AA Janos NE Ingram	4-19-67
RL-GEN-1575 C	PT-NR-83, Variable Process Tube Orificing	LL Sims JL Benson	4-20-67
RL-GEN-1582 S	Hanford Atomic Products Department Research and Development Budget for FY-1969 and Revision of Budget for FY-1968	MC Leverett	4-24-67

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SIGNIFICANT REPORTS (continued)

<u>Report No. & Class.</u>		<u>Title</u>	<u>Author</u>	<u>Date</u>
RL-GEN-1589	C	Tube Outlet Temperature Limits for Increased Power Level Probe	RH Shoemaker	4-17-67
RL-GEN-1590	C	An Analytical Study of Phase II Electrical Power Generation	JA White	4-20 67
RL-GEN-1599	C	Preliminary Results of the First Neptunium Target Column	DW Constable	4-18-67
RL-GEN-1603	U	Summary of PWR-203	JJ Wick	4-19-67
RL-GEN-1611	S	Life Expectancy - N-Reactor	RL Dickeman	4-28-67
RL-GEN-1613	C	Charts Used in N-Reactor Boiling Discussion	MC Leverett	4-24-67
RL-GEN-1618	S	Coproduct Shipments to Savannah River Laboratories	RL Dickeman	4 28-67

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VISITS

<u>Name</u>	<u>Company and Location</u>	<u>Contact</u>	<u>Date</u>	<u>Purpose</u>
RL Dickeman MC Leverett JW Riches TW Evans NR Miller MM Hendrickson WS Nechodom RH Shoemaker RK Robinson	Presentations to ACRS, to Production Division, and to Division of Reactor Licensing, Washington, D. C.		4-5-67 thru 4-7-67	Discuss higher power demonstration, efflu- ent control program, research and develop- ment program
MC Leverett	USASI Offices New York City		5-10-67	Chair meeting of N6 Sectional Committee
MC Leverett	Los Angeles, Calif.		5-27-67	N6 Section Committee Business
DH Curmiss	Washington, D. C.		5-5-67 thru 5-7-67	Presentation of Pu-238 Program
DW Leiby	GE - Cincinnati GE - Schenectady (KAPL & R&DC)	Mr. Brosnan Mr. G. Smallwood Dr. H. Chestnut	4-9-67 to 4-14-67	For Interviews
MM Hendrickson	Savannah River Labs, Savannah River, Ga.	B.C. Rusche	4-10-67	Discuss Meteorology
	Oak Ridge National Laboratories	G.W. Parker	4-11-67	Discuss Fuel Failure
GR Klingler	Tempus Corp. Gardena, Calif.	Ken West	4-10-67 to 4-11-67	To witness the accep- tance of the Mark III stripper.

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VISITS

<u>Name</u>	<u>Company and Location</u>	<u>Contact</u>	<u>Date</u>	<u>Purpose</u>
RL Dickeman MC Leverett JW Riches TW Evans NR Miller MM Hendrickson WS Nechodom RH Shoemaker RK Robinson	Presentations to ACRS, to Production Division, and to Division of Reactor Licensing, Washington, D. C.		4-5-67 thru 4-7-67	Discuss higher power demonstration, efflu- ent control program, research and develop- ment program
MC Leverett	USASI Offices New York City		5-10-67	Chair meeting of N6 Sectional Committee
MC Leverett	Los Angeles, Calif.		5-27-67	N6 Section Committee Business
DH Curtiss	Washington, D. C.		5-5-67 thru 5-7-67	Presentation of Pu-238 Program
DW Leiby	GE - Cincinnati GE - Schenectady (KAPL & R&DC)	Mr. Brosnan Mr. G. Smallwood Dr. H. Chestnut	4-9-67 to 4-14-67	For Interviews
MM Hendrickson	Savannah River Labs, Savannah River, Ga.	B.C. Rusche	4-10-67	Discuss Meteorology
	Oak Ridge National Laboratories	G.W. Parker	4-11-67	Discuss Fuel Failure
GR Klingler	Tempus Corp. Gardena, Calif.	Ken West	4-10-67 to 4-11-67	To witness the accep- tance of the Mark III stripper.

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VISITS (continued)

<u>Name</u>	<u>Company and Location</u>	<u>Contact</u>	<u>Date</u>	<u>Purpose</u>
HJ Bellarts	Tempus Mfg. Co. Los Angeles, Calif.	Ken West	4-10-67 to 4-11-67	Demonstration of Mark III Coproduct Stripper
GA Newell	Wilkins & Associates Tacoma, Washington	J Redding	4-17-67	Discuss fuel spacer fabrication problems
JR Glindmeyer	Wilkins and Associates Tacoma, Washington	J Redding	4-6-67 & 4-14-67	Expedite Material
TP Butterfield	Tempus Corp. Paramount, Calif.	Ken West	4-10-67 & 4-11-67	Approval Tests
	Harvey Aluminum, Inc. Torrance, Calif.	K. Davidson	4-10-67	Material Availability
	Coast Metal Craft Culver City, Calif.	Gene Reed	4-10-67	Material Quality
TD Naylor DH Nyman	National Lead Company of Ohio, Cincinnati	C E Bussert	4-25-67 & 4-26-67	Discuss Uranium Billet Fabrication.
	Reactive Metals, Inc. Ashtabula, Ohio	R Bean	4-27-67	Discuss Development Work on Primary Extrusions of Coextrusion Billets
SG Forbes CE Gruhl	Scientific Supplies Co. Seattle, Washington	D W Girard	4-4-67	Information on eddy current testing

VISITORS

<u>Name</u>	<u>Company and Location</u>	<u>Contact</u>	<u>Date</u>	<u>Purpose</u>
CE Bussert, J Schiltz, PN McCreery	National Lead Company of Ohio Cincinnati, Ohio	TD Naylor	4-5-67 to 4-6-67	To discuss uranium billet fabrication.
DW White WG Moffatt LF Coffin	GE R&D Center Schenetady, NY	M Lewis	4-11-67 to 4-12-67	Consultants on Primary Pipe Monitoring Fuel Performance Fuel Fabrication Development
RA Graham LK Fairchild	Wah Chang Corporation Albany, Oregon	HP Kraemer DW Darsow	4-13-67	Procurement Problems
CA Brown	Rem, Inc. Albany, Oregon	HP Kraemer DW Darsow	4-13-67	Procurement Problems
M Eastly	Charles W. Fowler Co. Bellevue, Washington	RK Bollinger GL Swezea	4-14-67	To discuss in-core ion chambers
RN Confer	F. T. Crowe & Co. Seattle, Washington	NJ Roy PK Telford	4-17-67	To discuss the segregation area bridge cranes and hoist problems
JR Mohondro	Hallidie Machinery Seattle, Washington	NJ Roy PK Telford	4-18-67	To discuss the segregation area bridge cranes and hoist problems
JD James	GE-APED San Jose, California	RK Bollinger GL Swezea	4-24-67	To discuss in-core ion chambers

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SUGGESTION PLAN PARTICIPATION

	<u>April</u>	<u>CY-1967</u>	
Number of Eligible Employees	426	433	(Average)
Number of Suggestions Received	41	180	
Number of Suggestions Acted Upon	39	179	
Number of Suggestions Adopted	10	64	
Net Annual Savings	\$136	\$ 8,921	
Amount of Awards	\$165	\$ 1,460	
Percent of Awards to Savings	121.3	16.6	
Average Amount of Awards	\$ 16.50	\$ 22.81	
Adoption Rate Per 1,000 Employees (Annualized)	--	443	

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G. L. Swezea

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Exhaust Electrostatic
Cleaner

J. V. Hambley

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Hoist or Crane Hook.

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