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HANFORD ATOMIC PRODUCTS DEPARTMENT

MONTHLY REPORT

MARCH 1967

GE-AEC BUSINESS RECORD

[redacted]

GENERAL ELECTRIC

RICHLAND, WASHINGTON

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HANFORD ATOMIC PRODUCTS DEPARTMENT

MONTHLY REPORT - MARCH 1967

OFFSITE DISTRIBUTION
Approved By:
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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 March N-Reactor operated with a time operated efficiency of 62.4 percent, a production availability and a steam availability to Washington Public Power Supply System of 56.7 percent. Level operating efficiency reached a new high of 92.5 percent for the month of March.

Two fuel element failures occurred. The first was caused by fretting corrosion where a buggy spring contacted the outer element of a 12-inch piece at the downstream end of a spike column. The second occurred in an early prototype of the Mark II driver which was undergoing high-exposure testing.

The third increment of the coproduct load was charged. All elements of the coproduct program are proceeding according to schedule and 643 columns of Mark II fuel are now in the reactor.

Examination of prototypical coproduct fuel, which has reached an exposure of 2100 Mwd/t, shows no evidence of deterioration, fretting, chattering, or other undesirable performance. Eight additional columns have been scheduled for discharge at intervals up to 5500 Mwd/t in a program which is designed to probe higher exposure level potentials.

A parametric study to evaluate the design changes proposed for the Mark IV fuel element (designed for achievement of power levels of 4800 Mw) has advanced to the point where billets can be ordered. Design features that may be tested in reactor include narrower supports, W-spring-stop inner supports, reduced cladding thickness, modified end-closures, revised charging techniques to permit elimination of the upper supports on the outer tube, and two uranium alloys. Initial charging of test fuel is scheduled for early August.

A semi-production scale plan for converting Hanford neptunium to Pu-238 has been prepared and discussed with the Commission. This plan could be accomplished without expenditure of capital monies. Basic information for this program is summarized in the special report section of this document (pages 56 through 67).

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OPERATING STATISTICS

In-reactor residue (Kmwd)	305.6
Highest exposure tube (Mwd/t)	3,846
Input production, plutonium (Kmwd)	71.5
Input production, coproduct (Kmwd)	38.1
Maximum power level attained (Mw)	4,000
Average power level while operating (Mw)	3,706
Time operated efficiency (%)	62.4
Number of shutdowns	6
Scheduled	1
Unscheduled	5
Reactor process data	
Power at which computed (Mw)	4,000
Primary coolant inlet temperature (°F)	384
Primary coolant outlet temperature (°F)	511
Maximum moderator temperature (°F)	1,100
Maximum tube power (kw)	5,024
Primary loop flow (lb/hr)	95.1×10^6
Heat Loss	
Heat to graphite cooling system (Mw)	39.6
Heat to shield cooling system (Mw)	2.25
Heat to rod cooling system (Mw)	0.42
Steam generation temperature (°F)	354
Steam generation pressure (psig)	127
Dump condenser flow (lb/hr)	325,000
Inlet temperature (°F)	47
Outlet temperature (°F)	72
Nuclear steam generating rate	
Maximum (1000's lb/hr)	13,360
Steam furnished to WPPSS (1000's lb/hr)	10,715
Maximum steam dumped (1000's lb/hr)	5,774
Water collected from gas atmosphere (gal/hr)	Negligible
Demineralized water produced (gal/hr)	98,640
Helium losses (1000's ft^3)	761
Fuel oil usage (bbl)	21,854

Reactor flattening efficiency (%)	See Reactor Physics Data
Fuel conditions	
Heat flux maximum (Btu/hr-ft^2)	587,000
Heat flux average (Btu/hr-ft^2)	472,000
Cladding temperature	
Maximum (^{o}F)	601
Average (^{o}F)	559
Uranium temperature	
Maximum (^{o}F)	894
Average (^{o}F)	799
Fuel Failures	2

Fuel Balance (tons)	Natural 0.71%	0.95%	1.25%	Coproduct 1.96%	Coproduct 2.1%	Total
Fuel charged	0	.78	1.38	0	49.71	51.87
Fuel discharged	0.37	52.93	24.54	0.68	3.26	81.78
Net change	-0.37	-52.15	-23.16	-0.68	+46.45	-29.91
Total In Reactor	0	103.48	28.14	0	141.03	272.64
Total In Basin	1.10	331.26	73.75	6.38	9.82	422.31

Fuel Element Output (Assemblies)	March	February	CYTD Total
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)	3,412	2,693	7,077
2.1% Mark F (driver)	648	108	1,980
Total assemblies	4,060	2,801	9,059
Tons forecast	56.4	42.8	142.0
Total tons	63.2	43.5	141.6
Percent of forecast	112.1%	101.6%	99.7%
Uranium utilization (2.1%) Est.	75%	78.6%	79%

32.8 tons of .95% scrap shipped during month.

[redacted]

REACTOR FRONT FACE 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	M	G						
32							G	G		G	G	G	K	K	K	G	G	G	G	G	G	G	G	G		G	G	G	G					
31					GM	G	G	G	J	J	J	J	J	S	J	J	J	J	J	J	J	J	J	J	J	S	G	G						
30					G		G	J	J	S	J	G	S	G	S	G	J	G	S	G	S	G	J	S	S	G	J	S	G					
29					G	S	S	J	J	G	G	G	G	G	G	G	G	G	G	G	G	G	G		G		J	S	G	G	G			
28				G	S	J	J	G	G	G				G	GM	G	G	G	G	G			G		G	G	G	J	S	G	G			
27			G	G	SM	G	G	G	GD		G		G		F	FM	FM	F	FM			G	G	G	G	G	G	G	J	SM	G			
26			G	S	J	S	G	G		G	D	G		G	F	G	G	G	F	G	G		G	G	G	G	G	G	S	S	S	S	G	
25	M		G	S	S	G	GD	G		G			G	E	F	G	G	G	F	G	G	G				G	G	G	G	S	S	J		
24		G	S	J	S	G	G			G	G	G	G	E	F	G	G	GM	F	F	G					G	G	G	G	S	J	G	G	
23			S	J	J	G		G	G	G	GE	G	G	F	F	G	G	G	E	F		G	G	G			G	G	G	J	J	S		
22		G	J	J	J	GE		G		G	D	GE	G	E	F	G	GM A	GM A	E	F	G	G		G	G	G	G	G	G	S	J	G		
21		G	S	J	S	GD	G		G	GD	G	G			G	GM	FM A	GM	FM B	G	G	G	G	G		G	G	G	G	J	J	G	G	
20			S	J	J	G		GD	G	G	GD	D	G	G						G				G	G			G	G	J	J	S	G	
19		G	S	J	S	G	GD	D	GD	G	GD	G	G		G					G	G		G		GM	G	G	G	G	J	J	G	G	
18			S	J	S	G	G	G	G	G	GD		G			G	GM	GM	GM	G	G	G	G	G	G	G	G	G	G	S	J	S	G	G
17		G	J	J	S	G	GD			D		G				G			G	G	G	G	G	G	G	G	G	G	G	S	J	S		
16		G		J	S	G	K	B	G	GM	G	G		G		GM	G	G	GM	G		G			G		G	G	G	J	J	S		
15	G	G		J	S		D	G	G		GM		G	G		G	G		G	G	G	G	G	G		G	G	G	G	J	J	G	G	
14			J	J	S	G		G					G				G	G	G	G	G			G	G	G	G	G	G	S	J	S	G	
13			S	S	S	G	G		G	G		G		G	G	G					G	G		G	G	G	G	G	G	S	J	G	G	G
12		G	S	J	J	G		D						G				G	G		G		G	G	G		G	G	G	J	S	J		
11			J	J	S	G	GD	G	G	G	G			G	G	G						G	G	G	G	G	G	G	G	S	S	G	G	
10			S	J	S	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	SM	G	SM	G	G	G	G	LA	S	S	G	S	G	
08				G	S	S	GD	G	G	G	G	G	G	K				G	G	G	G	G	G	G	G	G	LA	LA	J	J	G	G	G	
07				G	G	S	J	J	G	G	G	G	G	G	G		G	G	G	G	G	G		G	G	G	S	S	SM	G				
06			G		G	G	SM	G	S	G	G	G	G	G	G	G	G							G		S	J	J	G					
05							G	S	S	SM	SD	J	P	S	J	S	J	S	S	J	S	S	J		S	J	J	S	G	G	G			
04					G	G	G	G	S	J	S	J	S	J	J	S	J	J	J	S	J	S	J	S	J	S	G	G		G				
03						G	G	G	G	G			K	K	K	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						
01																																		

Date: 3-31-67

A - PT-33 Thermocouples
B - Blank
C - PT-57, Ceramic Target
D - PT-32, 1 Coupon
E - PT-32, 2 Coupons
F - PT-66, 2.1% Coproduct (13 pcs.)
G - Mark II, 2.1% Coproduct
H - PT-68 Np Target
J - Mark IIA Coproduct Spike
K - PT-76
L - PT-78
M - Monitor Column
N - 0.712% Natural Fuel
P - PT-64
R - PT-13, 1.96% Coproduct
S - Spike, 1.25% Enriched
T - Indicates tube has special thermocouples

[redacted]

REACTOR PLANT OPERATION

OPERATING HISTORY

N-Reactor Plant operated 22 days of the month for a time operated efficiency of 62.4 percent. Steam availability to Washington Public Power Supply System and production availability were 56.7 percent.

Level operating efficiency (total plant hours minus hours of unscheduled outages expressed as percent of total) for a recent 5-week period was as follows:

Week Ending	Actual, %	13-Week Rolling Average, %
2-19-67	80.5	88.0
2-26-67	100.0	89.9
3-05-67	100.0	91.6
3-12-67	88.9	91.8
3-19-67	90.5	92.2

Level operating efficiency for the month of March was 92.5 percent. This represents the Plant's best performance to date.

The plant operated at maximum authorized power level at the beginning and at the end of the month. The reactor was shut down twice during the month because of fuel element failure indications and once for a scheduled outage for fuel replacement and maintenance work. Outage time totaled 280 hours for the month. The fuel failures are discussed in the Technical Activities section under paragraph 1.5.

A total of 231 tubes were charged into the reactor in March.

Washington Public Power Supply System maintained a nominal production of 600 Mwe throughout the month, except for load changes required for No. 2 unit warranty tests.

Reactor Physics Data

Date:	March 31, 1967
Time:	0800
Reactor Power Level:	4000 Mw
Reactivity in Rods:	6.80 mk
Highest Graphite Temperature:	1082°F

	Fuel Types		
	Mark I	Mark IA	Mark II
Maximum Tube Power:	4892 kw	5025 kw	4898 kw
Specific Power:	189 kw/ft	194 kw/ft	194 kw/ft
Flattening Efficiency	82.0%	80.0%	82.0%

PLANT PERFORMANCE

Operating Continuity

Maximum authorized power level was maintained from the beginning of the month until March 8 when the reactor was manually scrammed upon indications of a fuel failure in process tube 2562. Tube 1261 was discharged as a rupture suspect. Startup was effected on March 9 and continued until shutdown for a scheduled charge-discharge and maintenance outage on March 12. Operation was resumed on March 20, concluding an outage of 236.3 hours.

Three automatic flow monitor scrams were experienced on March 22 during startup from the scheduled outage. A second manual scram occurred on March 24 following indication of a fuel failure in process tube 1644. Operation was resumed on March 25. Maximum authorized power level was regained on March 29 and held through month end.

Fuel Loading

The third increment of Production Test-75 fuel was loaded during the March 12 scheduled outage. On March 24 tube 1644 (1.96% U-235) was discharged because of a fuel failure. Tube 1844 (another 1.96% fuel) was also discharged as a failure potential, along with six additional high-exposure tubes, to reduce the probability of additional failures.

Steam Generators

Leaks in steam generators 3A, 3B, and 4B were repaired during the March 12 outage. Leak rates at month end were as follows:

Steam Generator	Leakage Rate Gallons per Hour
1B	1.0
2B	7.0
3A	22.0
3B	80.0
4B	1.0

Flow Monitor

Three flow-monitor scrams occurred during the startup on March 20. All were attributed to inlet connector valve position variations. The inlet connector valve problem was corrected by proper adjustment of the inlet valve manifold pressure.

Pressurizer

The pressurizer level control system malfunctioned during the scheduled outage when the reference legs drained during calibration work. Upgrading of the system is in process.

Equipment Modification

Modification work completed during March included work on the following items of equipment:

a. Pressurizer Vessel Temperature Monitor

b. Diversion Header Pressure Control

c. No. 6 Pony Motor Vent Fan

d. Supply Transformers for Emergency Power Panel EE-184

e. D-Elevator Travel and Charging Machine Interlock

f. C-Elevator Shielding Doors

g. Gas Moisture Analyzer

Cell Isolation Valves

The installation of gate-type isolation valves in the steam headers from Cell 4 was completed. Vent valves were installed in the lines for Cells 3 and 4. These modifications are designed to provide for isolation of the cells which will allow inspection and maintenance while the reactor is operating.

Production Testing

A total of 50.9 hours of outage time was used for production test work. Among production testing activities were physics tests for reactor criticality which required dropping 81 hoppers of balls during the March scheduled outage.

Exposure Reduction Program

A program to reduce personnel radiation exposure is in progress.

One concept involves eliminating the C-elevator charging bridges. Tests performed on a charging mock-up showed that an unsupported end of a loaded monotube will deflect downward about 3.5 inches. An automatic centering and lifting device was designed, fabricated, and successfully demonstrated. The device permits remote handling of the monotubes, thus eliminating the need for personnel on the C-elevator except for initial setup.

C-elevator shielding doors were installed to reduce radiation exposure to charging crews.

Deadlegs in the primary system piping were flushed, and the drain header in the right rupture monitor room valve rack was decontaminated. Equipment to decontaminate rear connectors was demonstrated satisfactorily.

Dump Condensers

A leaking tube was plugged in each of dump condensers Nos. 4 and 7. No other leaks were noted.

Traveling Wire Flux Monitor

Traveling wire flux monitor repairs continued. Eight of the nine wires are now operable.

Standby Boiler

During the startup on March 25, a hot spot was observed on the left side skin of the auxiliary plant boiler. Plans are being made to correct the problem prior to the April scheduled outage.

OUTAGE EXPERIENCE

Date	Outage Type	Duration	Reason
3-8-67	Manual Scram	18.7	Fuel Failure - Tube 2562
3-12-67	Scheduled Scram	223.6	Charge-Discharge and Maintenance
3-22-67	Auto Scram	0.6	Flow Monitor Trip (V-11)
3-22-67	Auto Scram	0.8	Flow Monitor Trip (V-11)
3-22-67	Auto Scram	11.3	Flow Monitor Trip (V-11)
3-24-67	Manual Scram	25.0	Fuel Failure - Tube 1644
Total Outage Hours		280.0	

Production losses were ascribed to:

Reason	Hours
Fuel Failures	43.7
Charge-Discharge	107.5
Scheduled Work	104.8
Startup Preparation	24.0
Total	280.0

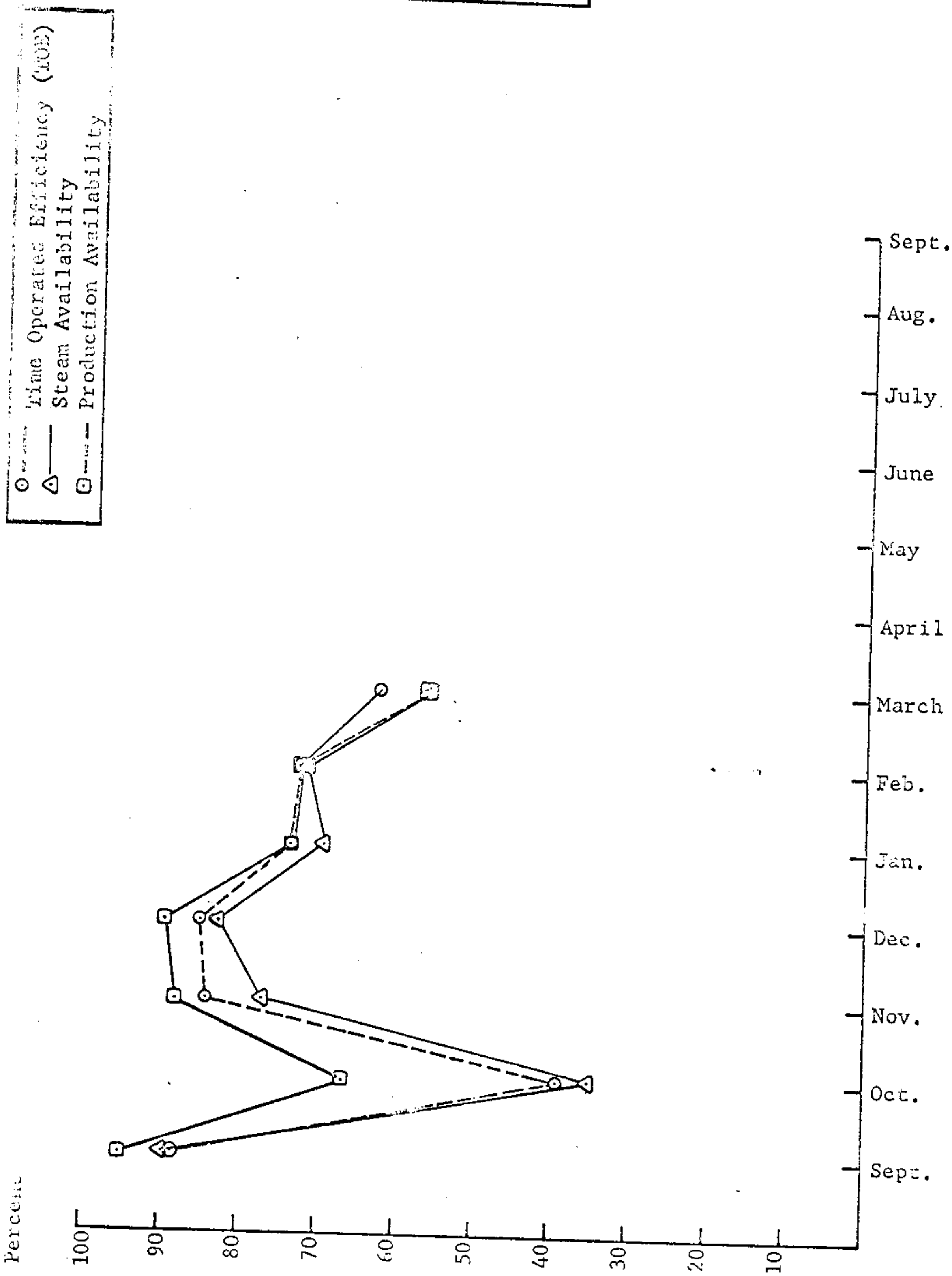
Figure 1 compares time operated efficiency, steam availability, and production availability over the past several months.

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

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

% PRODUCTION AVAILABILITY = Actual TOE plus TOE losses from external causes[1]

[1] Production 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.



Time Operated Efficiency, Steam Availability, and Production Availability

IN-PLANT TESTS

The status of production tests is as follows:

Production Test	Status
PT-NR-2, Routine Graphite Sample Irradiations (HW-80369)	Continuing. Four graphite boats being irradiated in rod channel 74. Ten graphite samples (blocks) were removed from ball channel 60 during February and the remaining 19 samples were removed in March. Plans are to leave the channel 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 in Channel 1648 as soon as outage time can be scheduled following use of this channel during the PT-NR-75 Physics tests.
PT-NR-4, Eval. of Monitor Column Fuel Elements in N-Reactor (RL-NRD-218)	Continuing. Twenty-one columns of fuel were discharged in March and five columns of Mark IA fuel (spike) were charged in March. These five columns will remain in reactor during the full coproduct mode of operation.
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.
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.

Production Test	Status
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)	Complete. The last two columns of fuel (containing iron-aluminum alloy fuel) irradiated under this test were discharge 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 seven columns was completed during March.
PT-NR-33, In-Reactor Enthalpy Imbalance Measurements (HW-84401)	Continuing. Seven thermocouple trains are now charged. Three trains were charged in March. Also, eight thermocouple trains were discharged in March. One thermocouple train is scheduled for installation during April.
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.

Production Test	Status
PT-NR-48, Corrosion Test Samples in End-Caps of the Graphite Cooling System (RL-NRD-466)	Continuing. Sample removal is anticipated in May.
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. Off-Reactor testing of RTD's is continuing.
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. Fifteen columns of fuel with small W-spring supports obtained goal exposure early in March and were discharged. Also, one column of fuel with large W-spring supports was discharged. No fretting corrosion was detected.
SUP3, Expansion of Coproduct Block	Continuing. Authorized increase in test size and updated test requirements.
SUP4, Target Support Variation	Complete. Authorized use of coproducer fuel which contain targets using either hardsized or tandem interference-fit target supports. Five columns of fuel with tandem interference-fit targets obtained goal exposure in February.

Production Test	Status
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. Discharge of the cobalt target was accomplished on 10-5-66. Examination of the tube for possible fretting was completed on 11-28-66. No fretting was detected. Targets have been cut and sections are being examined.
PT-NR-68, Pu-238 Demonstration Program (RL-NRD-793)	Complete. The initial column of test fuel and its supporting spike were discharged during the first week in October. The second column of test fuel and its supporting spike were discharged early in February with the last column with its supporting spike discharged in March.
PT-NR-69, Irradiation of Tubular Zr-2 Samples in N-Reactor Gas Atmosphere (RL-NRD-833)	Complete. Samples charged on 8-25-66 experienced higher than anticipated temp. These samples were removed during the February 6 outage.
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, and charging of the third increment was completed in March.
SUP1, Reactor Operating Data Requirements	Continuing. Base data for first, second, and third increments have been taken.

Production Test	Status
SUP2, Mark II Physics Tests-Part 1, Verification of Control Rod and Ball Safety System Shut-down Margins	Continuing. First, second, and third control rod system shutdown margins have been obtained.
SUP6, Authorization to Irradiate Fuel Charged Under PT-4 during PT-75	Completed.
SUP8, Radiological Controls	Continuing. Routine samples are being taken. There has not been any indication of tritium in the primary system coolant.
SUP11, Ball Functional Test	Continuing. First half of this test was completed on 3-19-67.
SUP12, Authorization to Irradiate Fuel Under PT-13 During PT-75	Completed. Last two columns of PT-NR-13 fuel discharged on 3-24-67.
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. One additional column is scheduled for discharge in April.
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	Continuing. 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)	Continuing. Two columns of fuel authorized by this production test were charged during February, and one column was charged in March.
PT-NR-82, Modified Rupture Monitor Differential Alarm Module Test (RL-GEN-1413)	Continuing. Circuit installed on 3-17-67 but removed because of sensitivity to DC power supply. Further analysis and off-reactor testing is in progress.

EQUIPMENT MAINTENANCE AND PLANT MODIFICATIONS

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

Title	Cost	Funds Assigned to GE	Percent Funds Expended	Status
CGN-182 Replacement 5 KV Cable, 100-N	$165,000	$ 77,700	75	All 25 circuits involved have been tested. Of these, 14 have been replaced, and the replacements are completed.
CGN-185 Steam Gen. Isolation Valves	268,000 (20 valves)	268,000	57	Twenty valves on order; four to be funded by WPPSS. Twelve valves received. Four have been installed for Cell 4. Two more valves are in transit.
GCP-400 - Backup Boiler Facility	865,000	30,000	10*	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. Comb. Engineering is apparent low bidder on boilers at $395,765. Letter of intent issued, permitting GE to proceed.
GAP-401 - Upgrading Fire Protection, 100-N	150,000	2,600	39*	AEC managed. Bid package by HES Vitro 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; 45 on total est. basis	Total of 1231 canisters, two portable and one permanent tritium monitors and 150 canister guides received to date. Monitors are in use. 100 canister guides have been installed. Remaining 876 canister guides have been shipped. Testing of target separator completed.

* GE's Portion

[redacted]

EQUIPMENT MAINTENANCE AND PLANT MODIFICATIONS (continued)

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

Title	Cost	Funds Assigned to GE	Percent Funds Expended	Status
GCP-404 - Fuel Spacer Disposal System and Refuse Cask, 100-N	$ 77,000	$ 77,000	20	Work based on two fuel spacer disposal pits and one refuse cask was approved for construction by RL-AEC 3-7-67. Pit design approved for construction.
GCE-405 - N-Reactor Temp. Monitoring System Improvements	189,000	189,000	--	Project authorized by RL-AEC 3-8-67.
GCP-406 - Safety Platforms and Accesses	300,000	--	--	Project proposal submitted to RL-AEC 2-10-67. Deferred pending availability of funds.
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.
Corrosion Test Facility - 109-N	45,000 (rev.)	43,224	95	Loops 2, 3, 4, and 5 completed. Loop 1 requires one tie-in for completion.
ERW Tank By-Pass and System Improvements	165,000	--	--	Under consideration by RL-AEC.

[redacted]

FUEL MANUFACTURE

PRODUCTION STATISTICS

Input (Extrusions)	March	February	CYTD Total
0.95% Mark A (outer)	-0-	-0-	-0-
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)	206	227	662
2.1% Mark F (driver)	-0-	-0-	-0-
Total Extrusions	206	227	662
Tons Forecast	62.4	47.4	157.2
Total Tons	42.2	47.2	137.0
Percent of Forecast	67.7	99.6	87.2

Output (Finished Production)	March	February	CYTD Total
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)	3,412	2,693	7,077
2.1% Mark F (driver)	648	108	1,980
Total Assemblies	4,060	2,801	9,059
Tons Forecast	56.4	42.8	142.0
Total Tons	63.2	43.5	141.6
Percent of Forecast	112.1	101.6	99.7
Uranium Utilization (2.1%) Estimate:	75%	78.6%	79%

PRODUCTION SUMMARY

The below-forecast (68 percent) input production resulted from temporary component shortages (outer cladding shell Zircaloy) combined with an unusually high requirement for special extrusion work. A total of 102 special extrusions were made during the month.

The above-forecast (112 percent) output production significantly improves finished fuel inventory position, reduced the in-process inventory to more favorable levels and increased CYTD performance to 99.8 percent of forecast. A major gain was made in reducing scrap inventories. A total of 32.8 tons of uranium scrap was shipped during the month. This increased shipping rate was made possible by recent changes in shipping procedures which have made available considerably more shipping capacity for this purpose.

Mark IC Program

Mark IC has been adopted as the designation for the 0.95 w/o enriched tube-in-tube fuel elements (currently in finished storage) that will be charged in the reactor immediately after the coproduct demonstration load. The outer support circle diameter of these fuels will be increased to 2.690-inch $\pm$ 0.005-inch by replacing the mild steel shoes with a thicker steel shoe. Initial reactor tests showed that the increased support circle diameter balanced the flow between top and bottom in the outer annulus, but that it would be necessary to force more water through the inner annulus by restricting the flow in the outer annulus. A steel shoe with tabs on one end was developed to provide this flow resistance. It was anticipated that an eight percent reduction of flow in the outer annulus would be achieved with the modified shoes. The actual flow change was about half this amount. Plans are now to develop a shoe with tabs on both ends and to prepare fuels for charging machine testing.

Mark IV

Billet and component designs and specifications have been prepared for a 100-column reactor test of the Mark IV designs in two different enrichments, three different cladding ratios, and two different alloys. Materials ordered for this test are equivalent to about an 85-ton input to the fuels shop (or what would normally be considered about a 2-month production period). Schedules include the production of five reactor columns of each test group by September.

Special Extrusion Activities

A total of 102 extrusions were made to produce the following materials: Mark II inner cladding components, Mark IV components, support system stock, outer cladding billet shells, I&E fuel element cores and aluminum cladding for Douglas United Nuclear, and material for long billet tests. The latter material will be used to test a longer billet design for improving material utilization.

[redacted]

Target Reactivity Control

The average Li^6 content for all targets in the reactor (of a given $LiAlO_2$ case) must be within specified limits (± 2 percent). The specified limits and accumulated Li^6 content of material produced to date are well within production specifications.

$LiAlO_2$ powder is tested in the 305 test reactor to determine the $Li^6/LiAlO_2$ ratio. Test capsules of powder are compared to a standard powder capsule to determine reactivity relative to the standard. Periodic checks by mass spectrographic methods are also made. To ensure the desired control, reactivity testing has been established for the following process locations: (a) prior to input into the process; (b) after mixing (if Li^6 content adjustments are required); and (c) after sintering. The after-sintering results are used to establish Li^6 values for finished target production.

Fluorine Contamination, $LiAlO_2$

A number of $LiAlO_2$ powder lots have contained excessive amounts of fluorine in the as-received condition. This powder was introduced into the process on the premise that the fluorine content of pellets would be reduced to acceptable levels during sintering. Subsequent analyses have shown that fluorine content is reduced approximately 90 percent during sintering. Therefore, no fluorine problem exists or is expected in the sintered product.

Flame Sprayed Ceramic Product

A technique of flame-spray deposition of $LiAlO_2$ was initiated during the month. This effort is to develop a process to control the isotope level in the finished target by applying a layer of lithium aluminate at a controlled thickness. This layer thickness would be regulated by the selection of various diameters of undersized rods of alumina or similar material. After spraying a calculated excess deposition of the lithium aluminate onto the core, the product is put through a centerless grinder to reduce it to the proper controlled diameter. Work on this process in continuing.

An attempt to spray a mixture of lithium carbonate and alumina powders to react to $LiAlO_2$ in the flame was not successful. The powder was not free-flowing and clogged the metering valves in the spray gun. Coarser and more granular carbonate will be obtained, and this method of deposition will be further evaluated. The problem of securing better adhesion to alumina calls for activating the surface of the alumina with a fluxing agent. There are numerous fluxes compatible with the end use of the targets, including several lithium salts. Investigations of these fluxes will also be made.

CONVERSION

Backup Boiler Facility

The following types of detail design drawings by Burns and Roe on the backup boiler facility are being reviewed for conformity with the design criteria, as well as for safety, maintenance, and operating considerations:

Site plan

Foundation plans and details

Architectural plans and details

Equipment Arrangements

Piping flow diagrams, isometrics, and layouts

Electrical one-line diagrams

Instrument control schematics and panel layouts

Phase II Power Generation

A study has been made to identify conditions which would provide for optimum Phase II electric power generation, and a method devised for computing power capabilities. The conditions and parameters investigated were the in-plant steam usages, dry steam effect, enthalpy imbalance, pressurizer pressure, pressure allowance, and zone temperature monitor (ZTM) limitations as applied to the maximum steam pressure attainable in the steam generators. Power capabilities will be computed based on those factors.

TECHNICAL ACTIVITIES

1. BASIC PRODUCTION MISSION

1.1 Mark IV Fuel Development

An initial parametric analysis of the economic incentive (limited to potential materials cost savings in fuel production) for improved fuel design was prepared to assist in the design of the Mark IV fuel element. Potential cost savings per ton of finished fuel varied as a function of fuel weight (lb/ft), cladding thickness (heavy, intermediate, and thin) and uranium utilization (output/input) assuming current basic materials prices and maintenance of the current number of fuel blanks per extrusion. The relationships obtained are plotted in Figure 2. The savings for a 24 lb/ft intermediate cladding thickness design compared to the Mark I fuel amount to $1000 per ton of finished fuel, a 20 percent reduction, assuming no decline in uranium utilization.

Design studies have been made to obtain near-prototypical dimensions of Mark IV fuel for development and production testing. These studies explored the potential benefits in product cost that might accrue from increases in fuel length, decreases in cladding thicknesses and decreases in effective width of supports.

The designs incorporate the following variables:

a. Outer support width: 0.125 inch

b. Inner support width: 0.125 inch

c. Inner support type, W-spring + stops

d. Column length, 35 feet and 31 feet

e. Fuel length, 26.5 inches and 23.2 inches

f. Reduced cladding thicknesses on both inner and outer elements as shown below:

Cladding Thicknesses of Fuel Elements

	Outer Element		Inner Element	
Case	Outer Clad	Inner Clad	Outer Clad	Inner Clad
1	30	25	40	25
2 (base)	25	20	30	20
3	20	15	20	15

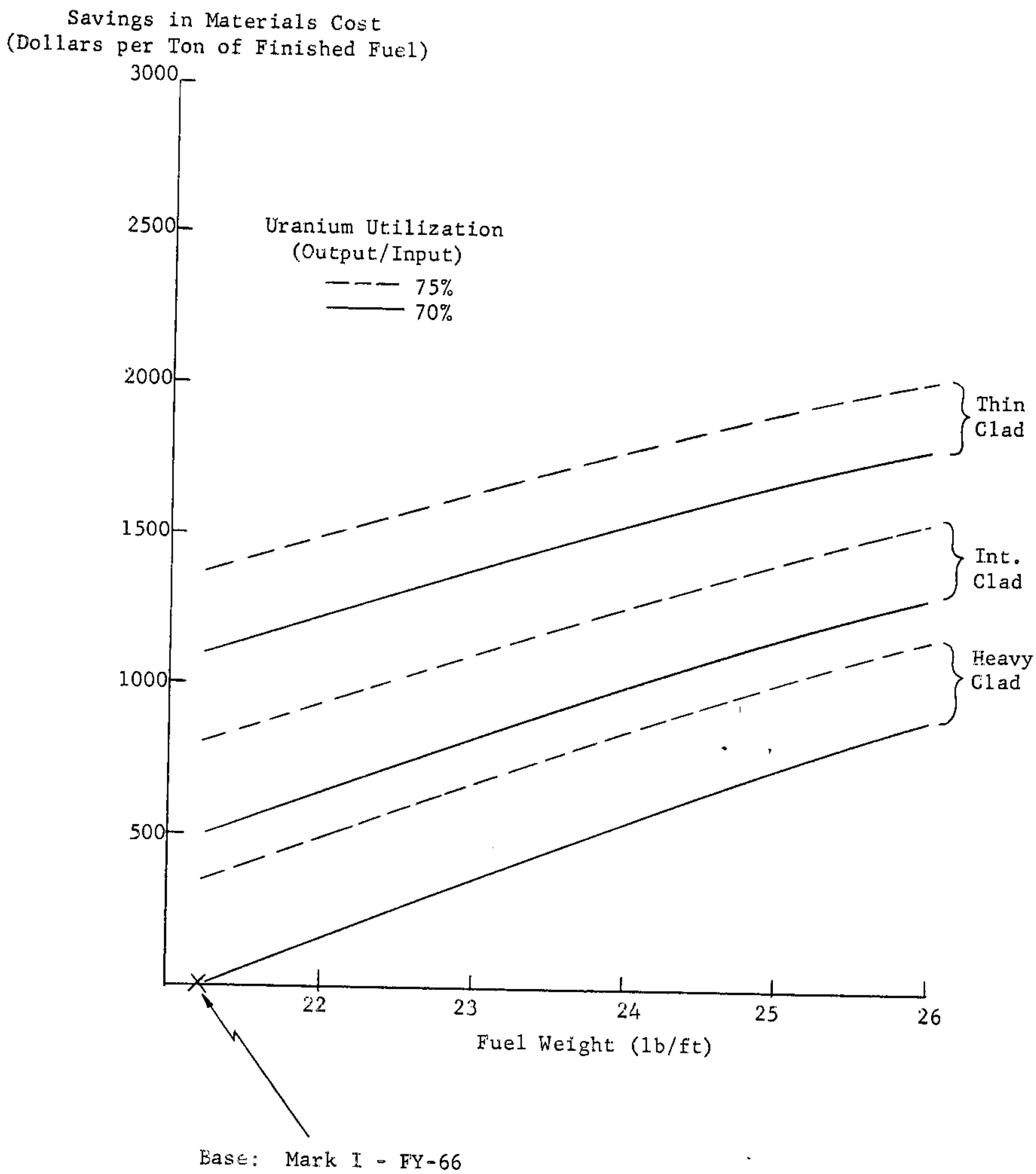


Figure 2. Cost Savings (Materials) as a Function of Fuel Weight, Cladding Thickness (Flex) and Uranium Utilization.

Billets are being ordered to fabricate about 60 columns of fuel elements for a production test that is to be charged into the reactor in August 1967. A column of fuel elements with Mark IV outside dimensions was made from available components on site and was charged into the 189-D test loop. The main purpose of this test is to determine the performance of 0.125-inch W-springs and rigid supports with a tube-in-tube element.

Six other fuel element design variables will be studied in this program:

a. An alternate composition - 250 ppm Fe + 250 ppm Si will be compared with the Mark II composition of 800 ppm Al + 400 ppm Fe.

b. A spike design that will operate at a specific power about equivalent to 4800 Mw reactor operation when the reactor is operating at 4000 Mw.

c. Shaped end closures - two alternate candidates for the standard V-bottomed end cap are:

    (1) A tapered design where the fuel element wall thickness decreases at the cap-to-core junction.

    (2) A design where the claddings are thickened at the junction of the cap-to-core but the element wall thickness remains constant.

d. The use of a spline (during charge-discharge) that would allow fuel elements to be charged into the reactor without having supports contact the process tube, thereby eliminating the supports positioned at 10 and 2 o'clock.

e. An end-spider support for the inner fuel element. This concept may include an integral support for the assembly.

f. Minimum thicknesses of cladding. An attempt will be made to manufacture fuel elements having only 10 mils of cladding.

The principal value of production test results will be a demonstration of satisfactory thermal hydraulic performance and in proving the feasibility of a number of mechanical design modifications. This will require that the test include a significant number of fuel elements, and fabrication of 60 columns of fuel is therefore planned.

1.2 Mark IC Development

Flow tests were completed of Mark IC fuels having tabs on the shoes on their outer supports to divert flow from the outer annulus to the other two subchannels. Results were as follows:

a. No deleterious effects resulted from ex-reactor loop testing period of about 480 hours at 130 percent normal flow and at 570 F.

b. Flow testing in a glass tube with tabs on only one side of the shoes showed that the tabs created only local disturbances in flow and had no discernable effect on the flow direction of the main stream downstream from supports. The results indicated that more efficient vanes would be required to swirl water in the outer annulus to offset the eccentricity effects which product an imbalance in the temperature rise in the outer annulus.

c. Charging tests performed in the ex-reactor mockup indicated that the shoes with tabs had no greater potential for tube scratching than standard shoes.

These tests indicated that it would be safe to charge one column of Mark I fuel with modified shoes, and such a column (with thermocouples) was charged during the outage which started March 12.

1.3 Mark I Fuel Element Performance

Mark I fuel elements (from monitor column 2855) that reached a tube average exposure of 5305 Mwd/t (more than two and one half times normal goal exposure) are undergoing a thorough examination. As noted in last month's record report, considerable cladding wrinkles were observed at this high exposure level.

Recent examinations also showed that nine inner fuel elements are longer than their matching outer elements. Four outer elements are shorter than the specified length, but none of the inner fuel element lengths were below specified values. All elements are manufactured with the outer element about 0.60-inch longer. Thus it appears that the effects of irradiation will cause outer fuel elements to shorten during long-term exposures. The shortening appears to be occurring preferentially in the outer elements in the downstream three-quarters of the charge. Examinations are continuing.

1.4 Failure Mechanism Studies

In last month's record report, research to determine the cause of end-associated failures was reported. The primary work then discussed was measurement of the change in shape of the end closure region of a Mark I inner fuel element as it was heated from 20 C to 500 C. The key conclusions derived from that study were:

a. Zircaloy end caps provide restraint to the uranium and/or the uranium has a different preferred orientation over the last six-tenths of an inch than the remainder of the fuel element.

b. There is no sharp line of strain demarcation at the cap-core interface.

A brazed closure of an inner fuel element was made recently with a thin sheet of mica between the end cap and the uranium core. The mica created an unbonded condition for study. When this closure was heated from room temperature to 500 C, a sharper line of strain demarcation was found at the cap-core interface. This indicates an increased tendency for shear failure of the outer and inner claddings at the base of the end cap.

Other studies to characterize irradiated brazed closures further were also carried out. Using the end cap profilometer in the N-Reactor fuel examination facility, the shapes of closures were measured using Mark I outer fuel elements from a high exposure monitor column (No. 2855 at 5305 Mwd/t tube average exposure). Two important findings were made:

a. The shape of the outer cladding over the endmost one inch of the element has been defined, and

b. The fuel tube wall thickness variation of the endmost one inch has been identified as a function of fuel element position.

The wall thickness data that were obtained are shown in Figure 3. Each element is represented by a line segment with an "X" marking its ends. The X's represent the amount in mils by which the wall thickness of the fuel element exceeds the thickness of the end cap measured at a section through the weld bead. To construct this plot, it was assumed that (a) the thickness of an end cap at a section through its weld bead is a constant, and (b) the wall thickness of the upstream end of the upstream element did not change during irradiation.

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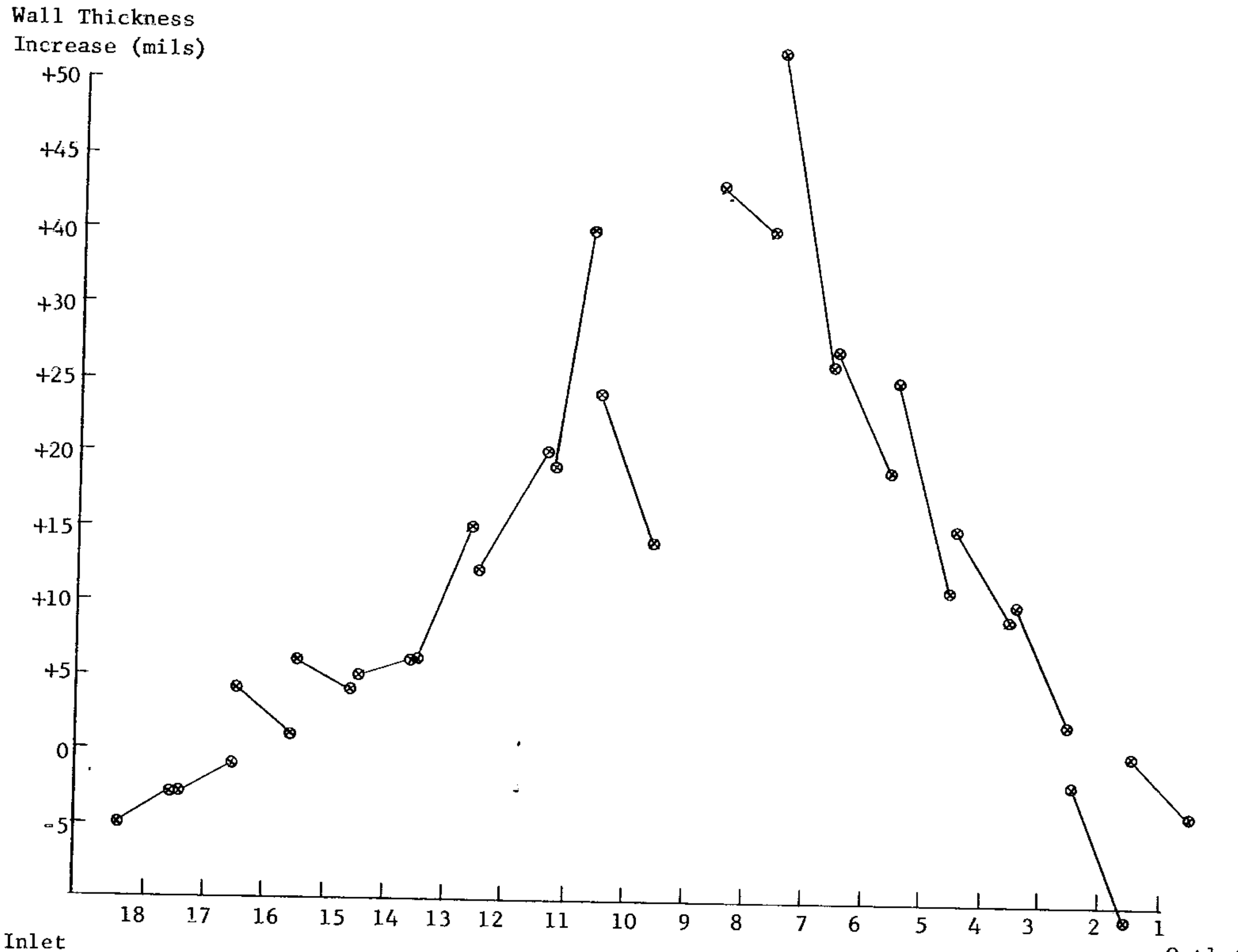

Figure 3. Increase of Thickness of Fuel Wall Compared to the End Cap at a Section Through the Weld Bead. - Mark I Outers, Column 2855, Exposure 5305 Mwd/t.

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Figure 4 shows outside cladding profiles of the endmost inch of outer fuel elements. Each curve is typical of four curves that were traced for each fuel element. The abscissa scale is four times actual size and the ordinate scale is 100 times. Two important conclusions may be drawn from these curves:

a. End caps appear to be bending at high exposure.

b. Fuel element diameters at the center of the column have increased from 22 to 30 mils.

According to the data in Figure 3, wall thickness increased from 30 to 50 mils on the average. Thus, inner diameter decreases of from 38 to 70 mils are computed. This conclusion is in agreement with dimensional change data that were observed in the K-Reactor test loop irradiations. It is not compatible with data that were reported in last month's report which indicated that both inner and outer radii may increase during irradiation, with wall thickness decreasing.

A further analytical effort that remains to be completed in this study of end closure performance is that of combining the "As-irradiated shape" of an end closure with its "high-temperature shape." This will show the shape of a high-exposure fuel element at in-reactor operating conditions.

1.5 Analysis of Failed Fuel Elements

Two fuel elements failed during the March report period. One was a 12-inch Mark I outer element in the downstream position. The mechanism was fretting corrosion caused by the buggy-spring supports on the inner elements reacting with the inner cladding on the outer element. The second failure occurred in an outer tube (driver) element of an early coproduct fuel prototype. This rupture was in a high-exposure PT-13 column. This column contained all Mark II iron-aluminum alloy drivers, 1.96 enriched, with lithium-aluminum targets. The column average exposure was 4094 Mwd/t; the failed piece was the second element from the upstream end with an estimated 3243 Mwd/t exposure. The upstream six inches of the element was broken from the other 22 inches, but was held in place by the target element. The end cap on the end nearest the apparent failure was almost severed from the element. A small wire was caught on the outer support of the failed element, but no evidence connecting it with the failure has been found. The mechanism of this failure is not completely understood at this time. All the other elements in the column were separated, examined, weighed and measured with the end cap profilometer. Light-to-medium wrinkling (2-mil high bumps), and incipient buggy-spring fretting (less than 1 mil), were observed on the inner bores. Swelling measurements were highly erratic with a maximum of 3.8 percent, and a few negative values near the ends of the column. Further examination of the failed fuel is planned.

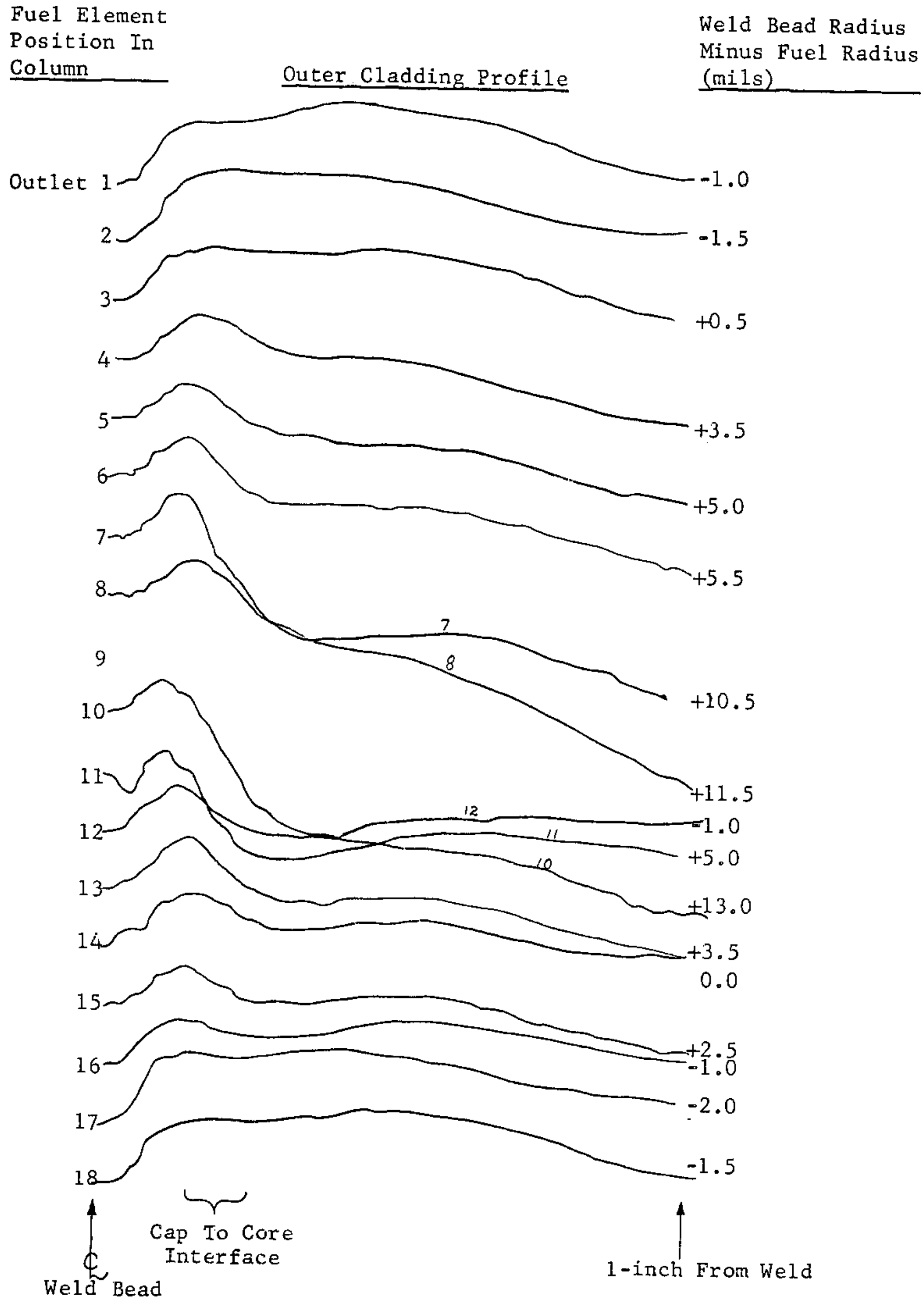

Figure 4. Shape of the Outer Cladding of Mark I outer Fuel Elements - Column 2855 (Tube Average Exposure - 5305 Mwd/t)

Failure Number	Tube Number	Date of Failure	Power (kw)	Exposure Tube (Mwd/t)	Exposure Element (Mwd/t)	Fuel Type	Type of Failure	Hour Loss	Prod. Loss
37	2562	3-8-67	4920	1282	180	Mark I	Outer Fretting	18.7	3117
38	1644	3-24-67	4017	4094	3243	Mark II	Exam. In Progress	25.0	3167

Radiometallurgy work to determine the cause of three Mark I inner fuel element end-associated failures (failure numbers 26, 27, and 28) is essentially complete. The findings point to cracks and/or inclusions in the braze metal as being the most significant departures from the quality and integrity which was expected in these fuel elements; however, in no case was evidence found that indicated water entry through the braze metal to the uranium.

As discussed in last month's record report, fuel element failure number 35 suffered the most extensive loss of core metal and cladding of any failure to date. During Radiometallurgy examination, it was found that the inner cladding of the outer fuel element had a tiny (1/64-inch x 3/16-inch) hole near the point where a buggy spring support on the inner fuel element contacted the inner cladding. Borescope examination of the water side of the inner cladding disclosed a fretting penetration and one of the buggy springs on the inner fuel element showed that a foreign object had lodged against the upstream loop and fretted both the loop and the weld tab. Further examination showed that the outer cladding of the inner fuel element had a severely fretted area at the side of the buggy spring support. The fretting member was not found. Nevertheless, examination has proved that it was present and that it was the cause of failure.

1.6 Reactor Long-Term Reactivity Transients

Reactor reactivity changes with fuel exposures have been deduced from a long history of operating data. The results for Mark I and Mark IA fuels are presented in Figures 5 and 6, where they are compared with values calculated by FLEX and MOFDA computer codes. Inspection indicates that the MOFDA code quite accurately matches the shape of reactivity curve, but is based high by about 18 mk. FLEX code matches the total curve more exactly. Comparisons of this type provide information which will be used to improve computer codes.

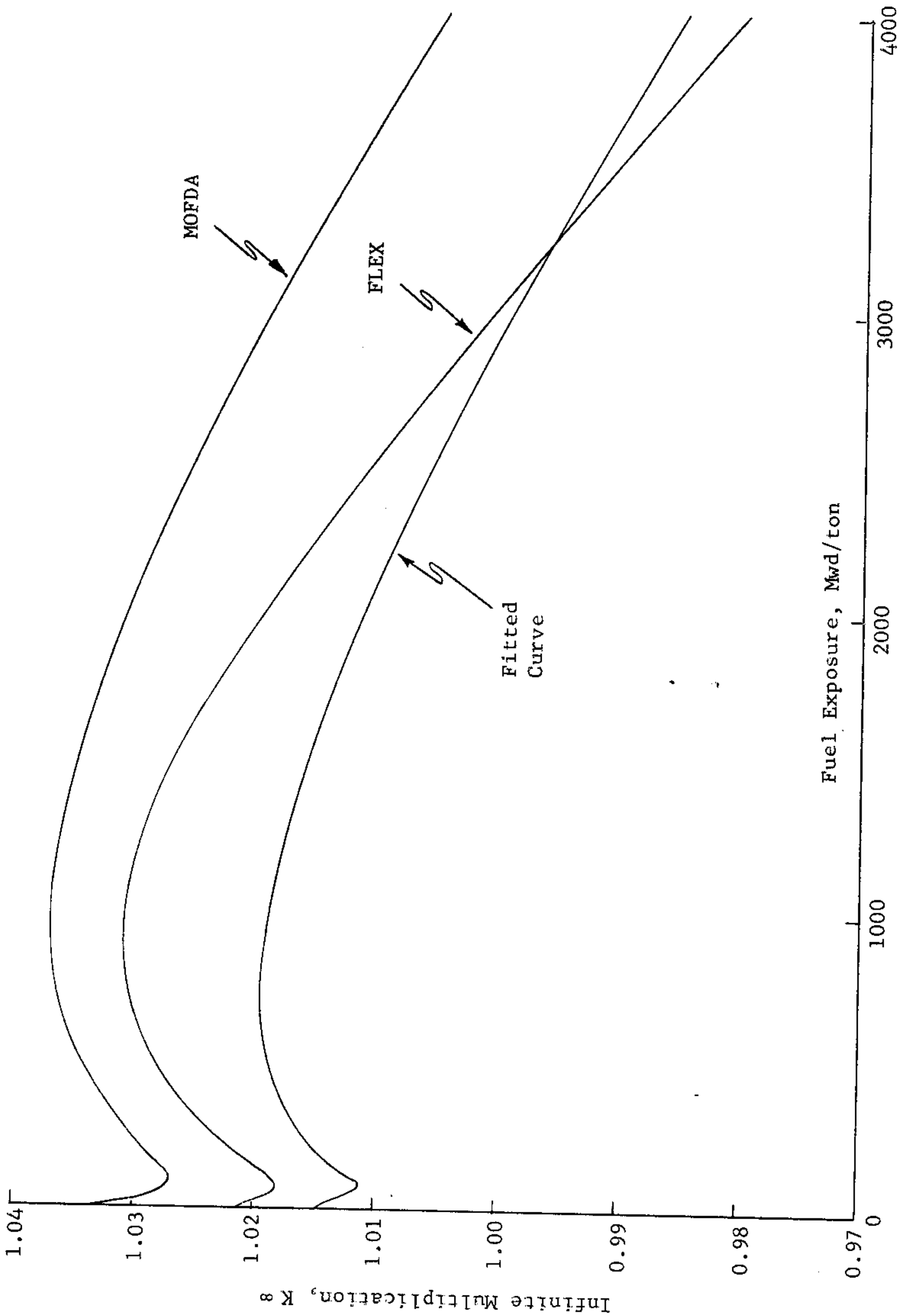

Figure 5. Mark I Reactivity Transient Hot, Operating

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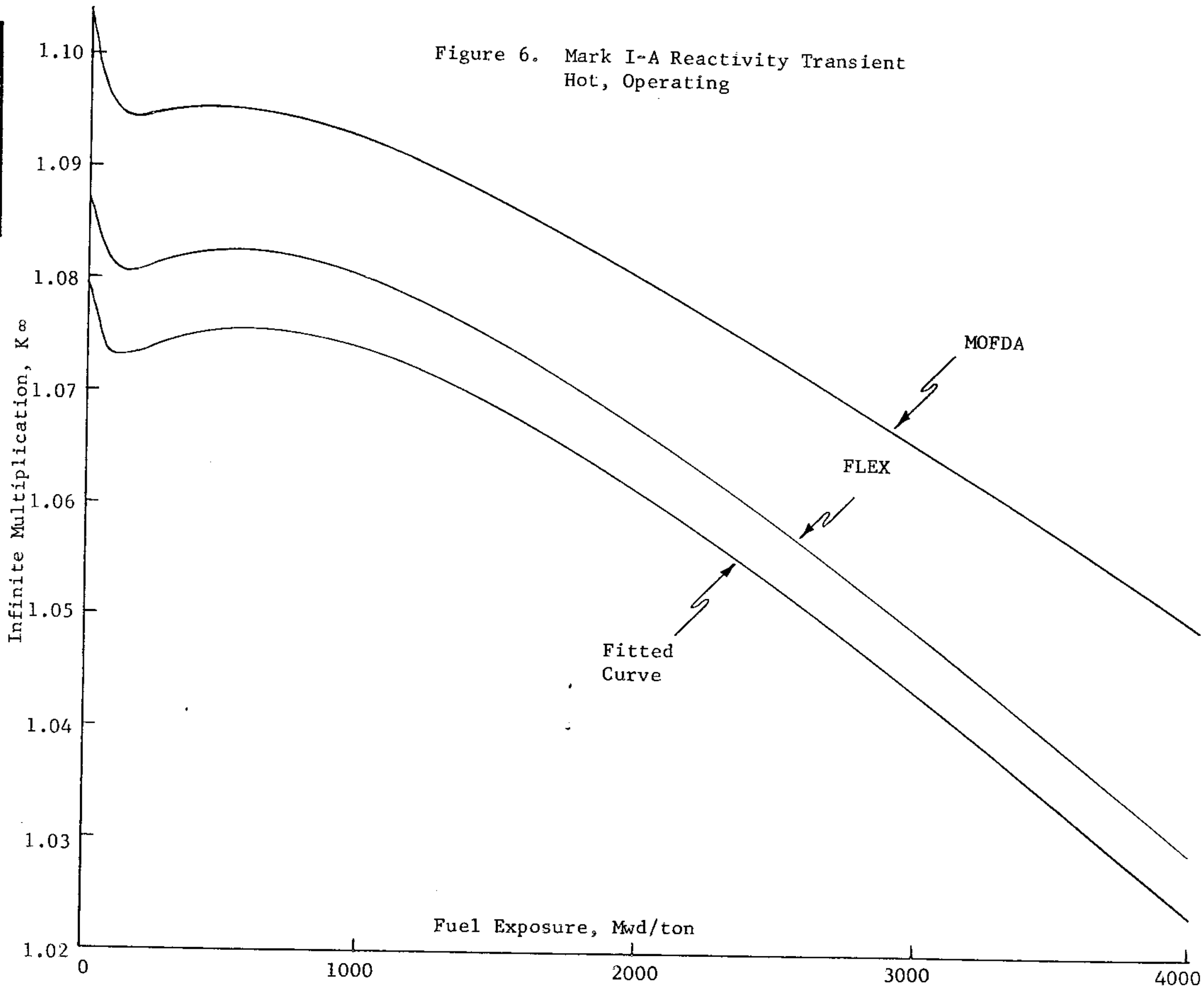
Figure 6. Mark I-A Reactivity Transient
Hot, Operating



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1.7 Graphite Studies

Initial parallel and transverse measurements on graphite bar samples removed from ball channel 60 last month show that the graphite has gone through the initial growth phase predicted and has just started to contract. The small dimensional changes agree quite well with the appropriate isotherm on TSX (graphite) data from General Electric test reactor experiments and data from other Hanford reactor tests. At this low exposure there is no measurable size effect.

1.8 Decontamination

Laboratory and loop demonstration tests to confirm the suitability of the procedure specified for Cell 3 decontamination are underway. Inadequate performance (a decontamination factor of two) of the alkaline permanganate-acid procedure in decontaminating the left rupture monitor room drain headers last month raised concern about the capability of the procedure to decontaminate steam generator tubing. Allowing for the same procedural deficiencies - old caked acid, low flow rate, and poor temperature control - the process was checked on a laboratory scale in the N-Reactor water quality laboratory. A piece of steam generator tubing removed from steam generator 3A in February was given the alkaline permanganate-acid treatment using the same materials used for the rupture monitor line drain header. The activity of the tubing sample was reduced from 24 mr/hr to 900 c/m, a decontamination factor of about 80. It is therefore apparent that the factor which prevented good decontamination of the rupture monitor line drain header does not apply to the steam generator tubing.

1.9 Primary Loop Studies

Primary Piping Monitoring

Procedures are being prepared to establish a program for monitoring the primary piping. Three regions have been identified where testing for weld flaws is desirable. Two are high-stress regions and one is a high-vibration zone. Testing will be done with ultrasonic methods over an area one inch on either side of the welds in those regions. A region of high-flow velocity with change of direction where erosion of the pipe wall could be expected will also be selected for testing. A region of high thermal stress, where make-up water is injected, will also be included in the program. Direct measurements of pipe diameters will also be made to determine any geometric changes. This program will be initiated during April.

Primary Coolant Shrinkage

A study of the primary loop volume response during a reactor scram has been started to define more precisely the present primary loop shrinkage rates. This will provide a basis for comparison with full coproduct operation and higher power level operation (4800 Mw power ascension).

The scheduled outage starting March 12 was initiated by a "scramming down" of the reactor to obtain additional information on the N-Reactor scram transient. The three pressurizer level transmitters were observed during the scram transient to determine if the level indicator in the control circuit was in error. The three level transmitters were all in good agreement with each other and the Control Room instrumentation. The high speed recordings from this scram are now being evaluated to further define the N-Reactor primary loop shrinkage following a scram. The high-pressure injection system flow instrumentation was also calibrated during this outage.

1.10 Analog Computer Studies

N-Reactor Transients

A summary of the first phase for more accurately predicting N-Reactor transients is given below.

The recorded 46-inch header pressure transient following an 800 Mwe load rejection was programmed into the analog computer model, and the primary thermal mixing lag was varied to obtain reactor power ramps close to those observed during the Washington Public Power Supply trip-off (N-4) tests.

During the trip-off tests, the recorded plant transient reponses were less severe than those predicted by the simulated transients (both analog and digital). See Figure 7. Of main concern were the deviations between the simulated and plant responses in the 46-inch steam header pressure. Of primary interest were its decay from its peak value, (reached after the turbine load rejection) to its setpoint; and the effect that this rate of pressure change had on the reactor power rate-of-rise.

Using the recorded 46-inch header pressure transient following an 800 Mwe turbine trip, as shown in Figure 8, the simulated reactor power rates-of-rise as functions of the mixing time constant and different water and metal reactivity coefficients were obtained as given in Figure 9. The simulated power ramps for Curves 2 and 3 are within the observed range of 100-150 megawatts per minute.

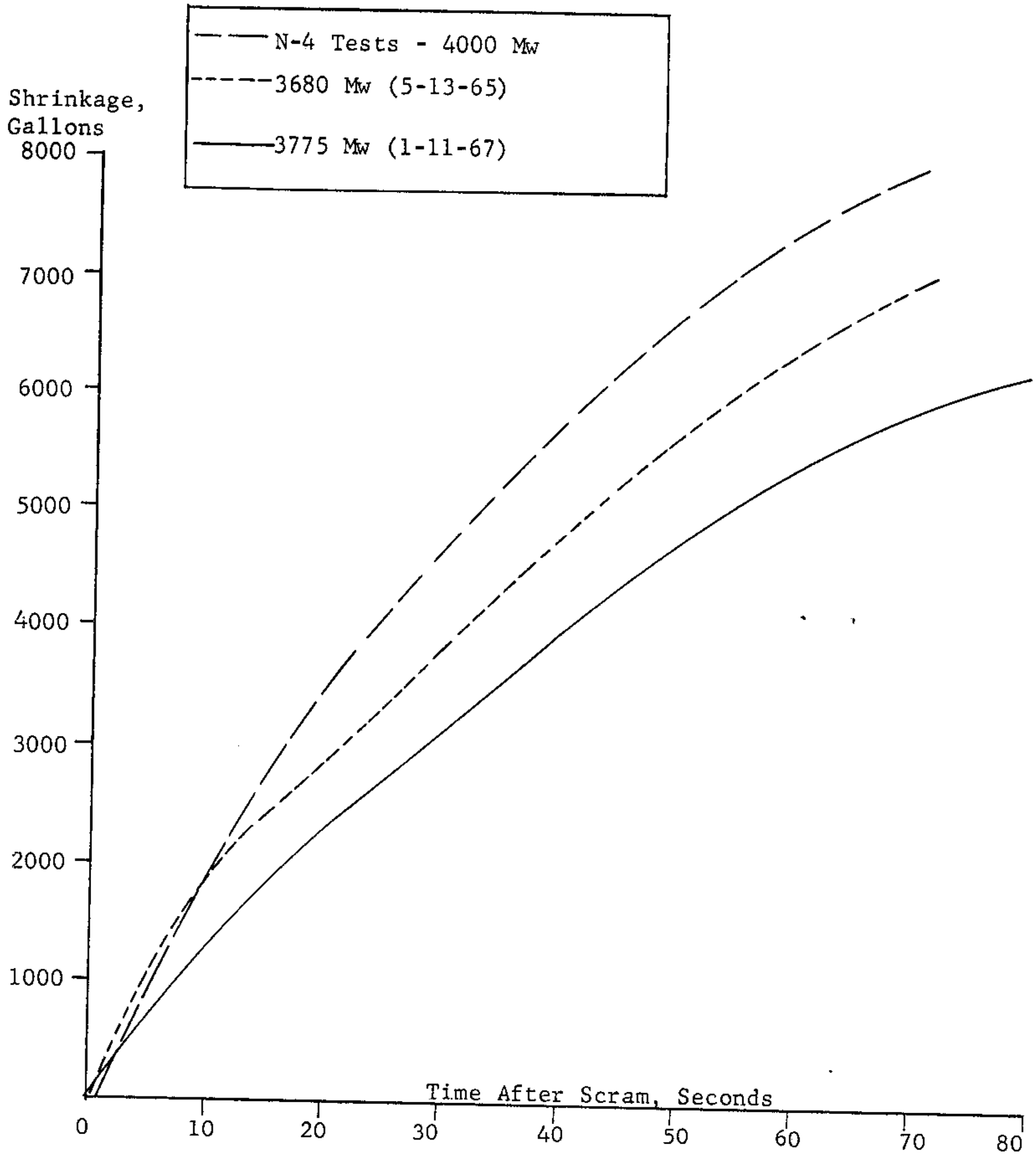


Figure 7. Accumulative Shrinkage Primary Loop

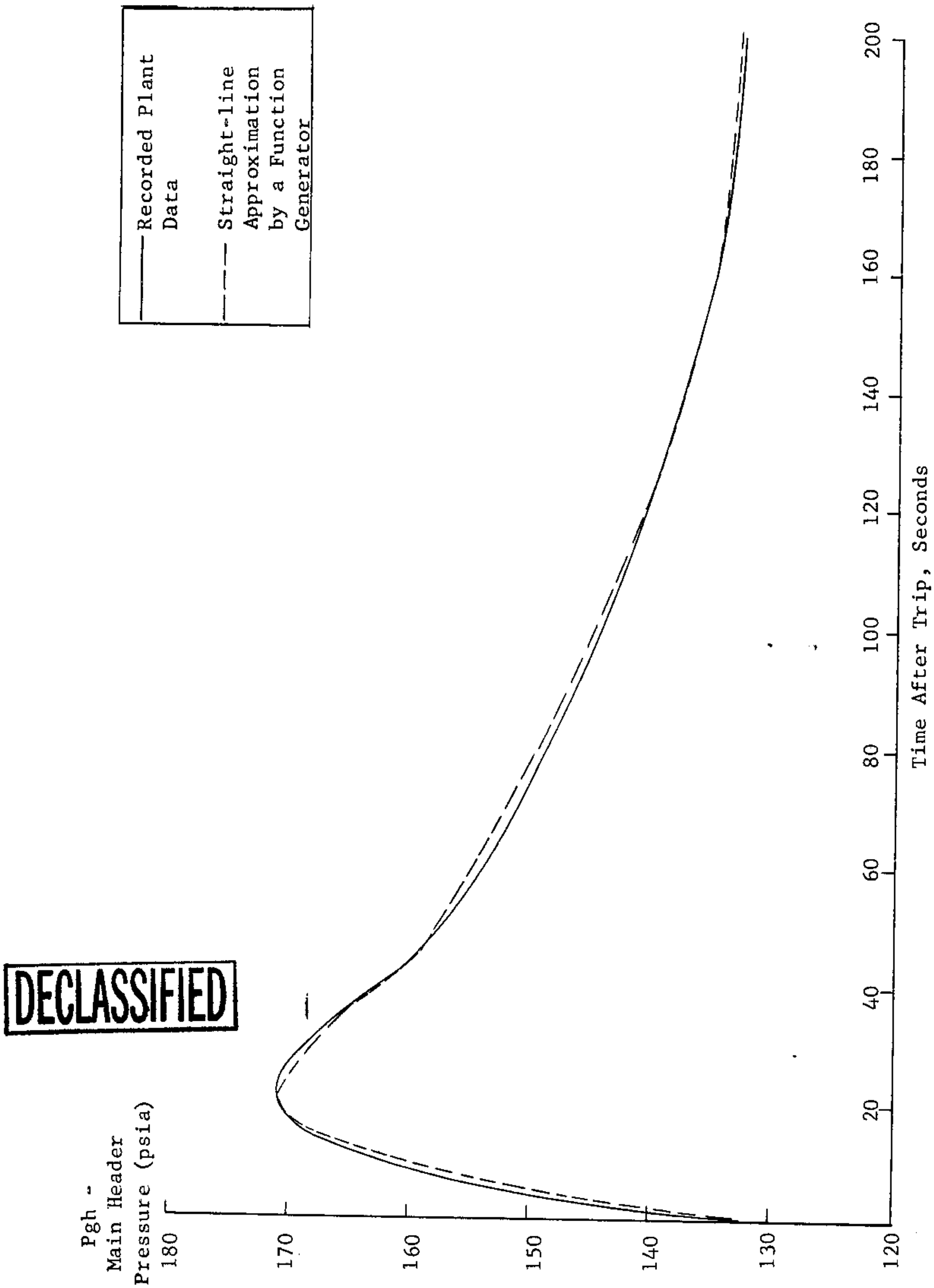
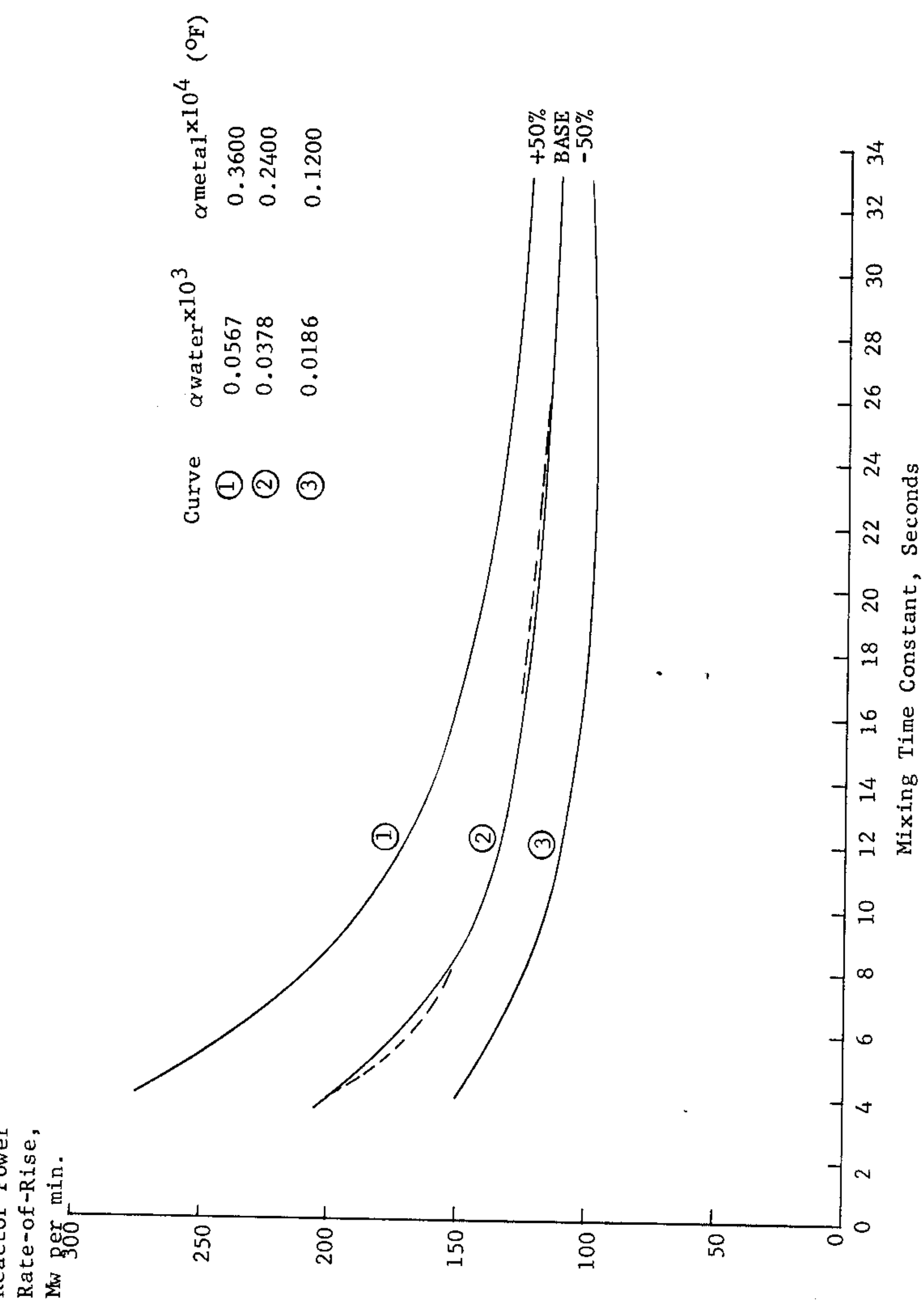


Figure 8. Main Header Pressure Transient Following 800 Mwe Load Rejection

Figure 9. Power Rate-of-Rise vs. Mixing Time Constant



N-Reactor Plant Model

A detailed description of the N-Reactor computer model is contained in HW-82589, ANALOG SIMULATION OF THE HANFORD N-REACTOR PLANT, by CD Swanson, et al, September 1, 1964.

The reactor kinetics model used was one with standard space-independent kinetic equations. Heat balance equations both for the fuel elements and coolants were included along with the reactivity effects from the changes in the water and metal temperatures. Consequently, estimates of the metal-water time constants are considered in the model. The reactivity effects of changes in the graphite temperature were not included.

Both the transport lags (deadtime) and mixing lags were included in primary loop considerations. The transport lags were essentially equivalent to the volume of the cold and hot legs, divided by the nominal total flowrate. The values used were 25 and 17 seconds, respectively. For the transport legs, there is a waiting time after which the output changes in a manner similar to the input. For the mixing leg, assumed to be a first order lag, the output starts changing as soon as the input does, but gradually approaches the final values.

Neither heat losses from, nor the heating and cooling of the piping between the steam generators and the reactor were included in the simulation model. However, the heating and cooling of the 353,000 pounds of metal in twelve steam generators was considered.

As noted previously, the recorded main steam header pressure transient for a dual turbine trip from 800 Mwe on December 8, 1966 was programmed into the simulation model. During the pressure and accompanying temperature transient, the heating and cooling of the steam generator shell metal was not included in the model.

Simulation Results

Figure 10 shows a comparison of the recorded and simulated steam generator coolant outlet temperature changes following the 800 Mwe load rejection. The family of curves represents the simulated temperature response after applying different time constants (0-40 seconds).

As illustrated, after considering a thermal lag (T = 0), the primary coolant changed 16 F. After applying different lags of 10, 20, and 40 seconds, the temperature changed about 14, 12, and 10 degrees, respectively. Note that the simulated temperature begins to match the recorded response when the thermal lag is about 20 seconds - although they are offset by 5-10 seconds.

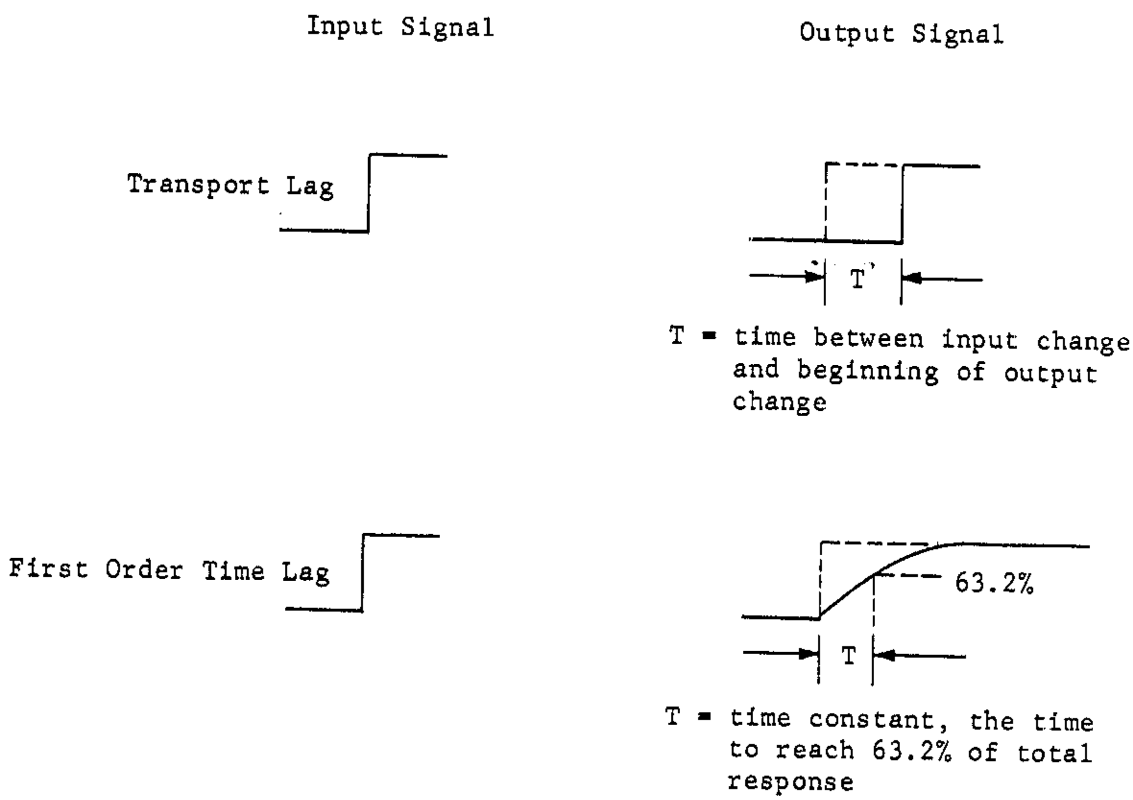


Figure 10. Lag Outputs For a Step Change In Input

The only significant change made to the mathematical model to bring steam generator coolant outlet temperature change more in line with the recorded response was to "refit" the linear equations of steam temperature as a function of pressure, and the steam and water enthalpy as functions of pressure over a narrower pressure range (130 to 180 psia). These changes gave the results shown in Figure 11.

Using the recorded 46-inch header pressure transient and the accompanying primary coolant temperature transient for T = 0 (given in Figure 11), the reactor power rate-of-rise as a function of the mixing time constant was calculated for different water and metal coefficients of reactivity. The results are summarized in Figure 9.

The mixing time constant is calculated by dividing the nominal volumetric flow-rate into the volume of the "mixing tank." In all previous analog runs, this mixing tank was assumed to have the equivalent volume of the inlet header and the inlet vertical risers (not the cross-over lines or the inlet downcomers). At first, the increase in the time constant was thought to be simply a means to bring the simulated power ramps in line with the observed ramps. However, a more detailed analysis of system volumes shows that this time constant has physical meaning which should not be ignored for values up to 25 seconds.

Conclusions

As a result of these comparisons, efforts to improve the simulation models will be directed to duplicate the 46-inch header pressure transient. The results in Figure 8 indicate that the observed physics transients can be duplicated fairly well once the proper steam pressure transients have been more precisely defined.

N-Plant parameters which will be designated for entry on the high speed scram recorder will include primary coolant temperatures from the different steam generators, cells, and inlet risers. These data will give an estimate of the amount of coolant mixing and temperature distribution at the inlet of the reactor following turbine trip-offs. The data will also serve as a basis for other reactor inlet temperature transient studies such as valving in a steam generator cell during reactor operation.

1.11 Nuclear Health and Safety

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.

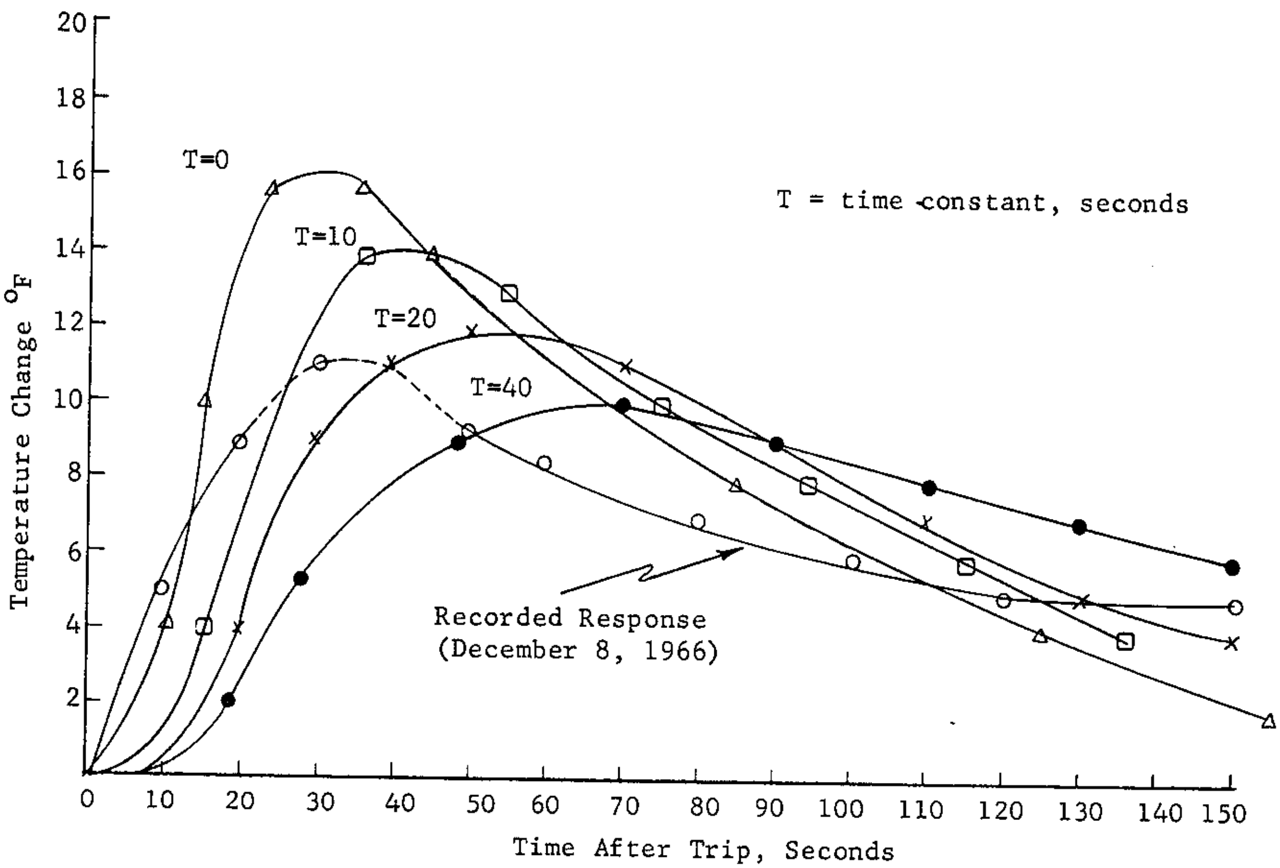


Figure 11. Steam Generator Primary Coolant Outlet Temperature Change Following 800 Mwe Load Rejection.

2. COPRODUCT MISSION

2.1 Production Schedule

The third increment of coproduct fuel was charged into the reactor during the March 12 outage, and eight more columns were charged during the March 24 outage, bringing the total Mark II inventory to 643 columns. All aspects of the program are on schedule and the coproduct demonstration continues to proceed smoothly.

Results of physics behavior during operation with the first two increments have been close to predictions.

As scheduled, a test was run during the March 12 outage to determine how many ball columns would be required to hold the reactor sub-critical with all rods withdrawn. It was found that 81 ball columns would satisfy this requirement. Extrapolation of these data to a prediction of the number of ball columns required to hold the reactor sub-critical when fully loaded with coproduct fuel is underway.

Fifteen columns of Production Test-66 coproduct fuel irradiated to goal exposure of 2100 Mwd/t were discharged for yield determination. Prior to charging the third increment of the coproduct demonstration, three columns of this fuel were visually examined and found to be free of defects.

The remaining Production Test-66 columns will be used to probe higher exposures. In addition to this block of 14 columns (one column was discharged for examination of long W-spring supports), there are eight PT-66 columns scheduled for discharge at intervals between 2100 and 5500 Mwd/t. If irradiation proceeds successfully, the 14-column block still in the reactor may be irradiated to 5000 Mwd/t (12 percent Pu-240) in order to obtain a yield determination on these columns. This would provide a yield determination at a sufficiently high exposure to verify predicted tritium and plutonium conversions as functions of exposure.

Pellets in the bottom layer reached 700 C and in the top layer 600 C. These data indicate 97.5 percent of the tritium in a charge will be recovered in a 50-hour extraction. The crucible is to be sectioned for corrosion measurements.

A projection of the needs for Mark II and Mark IIA fuel indicates that maximum full-load coproduct demonstration objectives will be met by fabricating a total of 1300 Mark II columns, 300 Mark IIA columns, and 50 Mark IIB columns. This split will permit recharging all fuel columns as they reach goal exposure, recharging three increments (in June, July, and August) and starting the transition to Mark IC fuel in September.

2.2 Mark II Fuel Performance

Significant quantities of Mark II fuel have now been discharged from the reactor at goal exposure, and important analyses have been completed. Five columns with tandem interference-fit buggy springs, three with 0.8-inch W-springs and two with 1.2-inch W-springs have been examined in the N-Reactor fuel examination facility. All drivers and targets were found to be in excellent condition. A few fretted areas were found under the interference-fit buggy springs, but none were deeper than 10 mils. What was thought to be shallow depressions were observed under the crowns of the W-springs. Sectioning through one of the worst of these "depressions" showed that it had no measurable depth. Apparently the effect was an optical illusion under basin viewing conditions.

Radiometallurgical examinations of target elements from Production Test-NR-66, irradiated to exposures of 2160 Mwd/t, have produced the following information:

a. No apparent change in warpage has resulted from the irradiations.

b. Free gas volumes collected from the drilled targets indicate that gas concentrations will increase linearly with increased exposure.

c. There are insufficient data available at present to define what the tritium fraction of this free gas would be. It is expected, however, that it too will increase with exposure and will approach 8 to 10 percent of the total tritium content in a high exposure target.

The latest results still indicate that the room temperature internal pressures within the aluminum target cans will be of the order of 20 atmospheres. This produces stresses well below the yield stress of even fully annealed aluminum. During operation, assuming an average target element temperature of 325 C and complete gas release (fission gases, H_2O and CO and CO_2 gases from allowable sintered pellet contents - 150 ppm of the CO and CO_2), it is calculated that a 7000 psi hoop stress will exist in the Zr-2 cladding. This is within the working stress limit (20,000 psi) by more than a factor of 2.

2.3 Coproduct Physics Test Preparations

Preparations are in progress to perform physics testing in May. The initial intention for the hot physics test series was to heat the primary loop by introducing energy through pumping (ΔP losses) as was done during the N-1 test series. Since exposed fuel is present, the need for rupture monitoring resulted in a series of special requirements--special instrumentation to detect fission products in each riser, a method for quickly returning the rupture monitor to service, and methods of quickly diverting all channels on a riser to minimize loop contamination. Recent investigations show that the leak rate from the primary loop is high enough to allow energy losses due to leaks, radiation, and partial flow in the rupture monitor which might limit the test to below 400 F. A study is in progress to determine whether the steam generators can be utilized as condensers to heat the loop. Up to three times as much energy can be introduced by this method. Normal usage of the rupture monitor could be made under these conditions, and leakage energy losses would be compensated for. Test temperatures would be in the range of 420 to 450 F.

2.4 Tritium Monitoring

The continuous monitor for tritium in the storage basin atmosphere was placed in operation during the month and stripping of the lithium aluminum targets (Production Test-NR-13) was initiated.

2.5 Shipment of Targets to Savannah River

Preparations for shipment of targets (Production Test-NR-13 and Production Test-NR-66) to Savannah River Operation continued during the month. The rail car to be used with the SCRUP cast for shipments to Savannah River Operations has arrived on plant. The vehicle, a Union Pacific car of 70-ton capacity, has been inspected and studies of ways to attach the cask to the car are under way.

2.6 Storage Basin Corrosion of Aluminum-Jacketed Target Elements

The continuing test of the corrosion of aluminum-jacketed target elements in N-Reactor storage basin water was concluded February 4 after element exposures of 5 to 10 months in storage basin water. The conclusions from the test were:

a. Elements that had been canned in Zr-2 jackets by standard procedures and dejacketed with the stripper showed greater pitting than pieces that had not been canned in Zr-2. No pits were observed on the unjacketed elements after an exposure of 10 months while shallow pits from 2 to 15 mils deep occurred on the jacketed and stripped elements after eight months exposure.

b. Deliberate surface abnormalities, such as embedded Zr-2 particles in the aluminum surface and unacceptably high tungsten inclusions in the welds, showed no indications of attack after five months of exposure.

c. Localized penetration of stripped target elements will not occur during storage in basin water for six months.

These tests indicate that corrosion of aluminum-clad targets will not present a problem for the planned periods of basin storage.

2.8 Flow Loop Tests

Life testing of Mark II fuel elements continued during the past month in the out-of-pile (189-D) test loop. The most recent series is summarized in Table 2-I.

Tests were conducted to determine the pressure drop across a column of 13.25-inch Mark II fuel elements in order to establish the effect of having more supports in the flow stream. Results show very little difference in over-all pressure drop between the 13.25 and 26.5-inch elements. Flow split tests will also be performed using these elements.

2.9 Increased Reactor Power Level Program

The basic objective of the proposed increased power demonstration program is to permit realization of N-Reactor's maximum power capability on a timely schedule. This demonstration represents the first step in a more comprehensive program to utilize the reactor's "stretch" capacity. The incentives for higher power level operation are many. One of the most important incentives is the increased product flexibility which may be derived from higher power operation. In addition, higher power operation will result in lower product unit costs and higher total production.

The demonstration will have several specific objectives which are related to the over-all power increase program. The demonstration will permit early verification of system and equipment capacities and responses. The data on response and capacities will be directly applicable to subsequent higher power operation with both coproduct fuel and plutonium-only fuel.

a. Verification of heat removal system capacities will be completed with enough lead time to permit any modifications or changes necessary with no sacrifice in potential higher power operation.

TABLE 2-I

Series	Fuel Type	Support Type	No. of Element	Test Cond	Test Time	Test Time Accum.	W-Spring Force (pounds)
21	7.5-Inch (Mark I)	Spider	4	340 gpm 570 F	484.5 hours	484.5**	
	Standard Mark I*	Buggy Spring	4			484.5	
	Mark II	W-Spring (hand formed)	1			2887.75	
	Mark II	W-Spring (machine formed)	1			2309.75	68
	Mark II	W-Spring (production grade)	2			1789.75	74, 74
	Mark II	W-Spring (production grade)	2			1434.25	83, 73
	Mark II	W-Spring	1			767.0	
	Mark II	W-Spring (13-Inch reworks)	1			767.0	

*Outer supports have shoes with tabs.

**Test completed March 3, 1967.

b. Verification of reactor transient response will be made to allow system tuning as required prior to continuous higher power operation. Of these transient responses, the most important will be the response of the reactor to a scram at increased power levels.

Because of the limited duration of the current coproduct testing program, this increased power demonstration will provide the only opportunity to obtain data and experience with the Mark II fuel at higher power levels. This would, of course, be highly desirable for future development of coproduct fuel should the reactor eventually go to sustained coproduct operation. A portion of the benefit for coproduct fuel has already been realized as a result of the intensive engineering investigation which was required for the hazards analysis. This analysis has provided a framework from which future studies will be based.

A document, RL-GEN-1493, describing increased power program objectives has been issued. The document gives the schedule, expected operating parameters, and necessary testing and data requirements.

3. TRANSPLUTONIUM TECHNOLOGY

Results accomplished in March not at a reportable stage.

4. PLUTONIUM-238

The chemical separation of the second column of neptunium targets is in progress. To date, six elements (12 pieces) have been processed with the dissolution and separation of the remainder of the column continuing. There are no analytical results to date.

The mechanical separation progressed as scheduled with only slight difficulty encountered during removal of the graphite plug and sleeve. The flux monitor wires were sent to the Battelle Northwest Analytical Chemistry Laboratory for analysis.

The processing of the first six elements of the third column for the Douglas United Nuclear fast turnaround demonstration is scheduled so program continuity will be retained. The targets were discharged on March 17 and transferred to the radiometallurgy laboratory on March 20. Following mechanical separation, the elements were transferred to the chemical separation facilities on March 24 and dissolution commenced on March 27. Approximately 400 grams of neptunium nitrate were available for use by Douglas United Nuclear April 1, 1967.

Results of measurements of the initial U-236 content in Mark I fuel indicate concentration of 60 ppm. Feeding this value into our calculations, rather than zero ppm as before, requires that the U-236 resonance integral be reduced to about 400 barns, rather than 480 barns, in order to match the experimental yields of Np-237 reported by Isochem. The value of 400 barns is a commonly accepted value, being at the low edge of measured data rather than the high. These adjustments also result in calculated neptunium yields compatible with those calculated by Douglas United Nuclear. Revisions to the N-Reactor conversion ratio tables will be initiated.

A semi-production scheme for converting Hanford neptunium to Pu-238 has been prepared and discussed with the Commission. The basic text of the report is contained as a special section of this document.

5. OTHER ISOTOPES

Isotope Production Study

The productivity of the 68 fringe channels has been calculated for several isotopes. In all cases it was assumed that each target column would be supported by three spike columns and that the target blackness would be adjusted to a net reactivity effect of zero. The production amounts, product quality, irradiation time, and target throughput are given in Table 5-I.

Because each column is supported by three spike columns, the loss of standard product per year is 19.3 kg of plutonium at 12 percent Pu-240 for the plutonium-only mode. Assuming a $500,000 capital investment, a 5-year pay-off period, and a 7.5 percent interest rate, capital charges have been assigned as shown in Table 5-II. Also shown are irradiation charges.

6. ENRICHED FUEL PROCESSING

Results accomplished in March not at a reportable stage.

7. TARGET SPACE ENHANCEMENT

Not a Hanford Atomic Products Department program at present.

TABLE 5-I

PRODUCTIVITY OF THE 68 SPARE CHANNELS

Isotope	Annual Production	Quality	Irradiation Time, Mo.	Annual Target Throughput	Target Material
Po-210	.9 kg	-	6	66 tons	Bi
H-3	.28 kg	-	6	40 kg	Nat Li
Pu-238	20 kg	95%	4	720 kg	Np
U-233	26.8 kg	-	6	17,140 kg	Th
Co-60	3.3 MC	17 Ci/g	24	195 kg	Co
Tm-170	4.7 kg	.35 w/g Tm	12	145 kg	Tm

TABLE 5-II

CAPITAL AND OPERATING CHARGES FOR ISOTOPE PRODUCTION IN THE 68 SPARE CHANNELS

Isotope	Capital Expense	Irradiation Expense, $/g	Estimated Market Value
Po-210, $/w			
H-3, $/g			
Pu-238, $/g			
U-233, $/g			
Co-60, $/Ci	0.037	0.146	0.10
Tm-170, $/2	2.19	8.55	10.00

Po-210 the loss of standard production is considerably less than the 19.3 kg for the other isotopes.

8. NUCLEAR SAFETY

8.1 Metal-Water Reaction Studies

Calculations of hydrogen evolution rates and other consequences of a reactor accident involving a primary system dump and emergency raw water system failure have been completely rechecked. A report is being issued giving simplified versions of the uranium and Zr-2 steam corrosion laws, in analytical as well as graphical form. Given the metal surface areas and temperatures for an accident model, one can quickly determine hydrogen production, depth of attack, energy production, and chemical power as a function of time.

A second report will be issued to describe the factors that may affect the behavior of fuel and cladding during an accident giving the results of analytical studies of several models. Several conclusions have been reached as a result of these studies.

a. The protective nature of the oxides found on uranium and Zr-2 during steam corrosion at high temperatures is a prominent aspect of the problem.

   Parabolic reaction kinetics ensure a steadily decreasing rate of Δ hydrogen evolution as the oxide films thicken with time. The rate constants are well known. This conclusion is supported by laboratory data, by engineering test results and, in the case of Zr-2, by industry-wide acceptance.

b. The strength, tenacity and self-healing characteristics of the oxide film should virtually prevent longitudinal movement of molten uranium in a steam environment. This, however, has not yet been demonstrated in prototype tests.

c. The fission gases cause cladding failure below the melting point of uranium (1020-1040 C). Failure of the cladding is by formation of a short longitudinal crack. A solid state U-Xe foam is extruded through the crack. If the presence of the xenon as a cover gas does not protect the uranium from rapid corrosion during foaming (because of increased surface area) then melting and superheating should speed the separation of gas and metal, causing it to settle down to a quiet, oxide-covered pool.

d. Corrosion product-to-metal volume ratios are such that, at about 40 percent of complete reaction, there is no more free space in the process channel. Diffusion of steam through the compatcted corrosion products should become limiting. For accidents that are terminated after several hours, this appears to be a more reasonable limit than complete reaction of fuel, cladding, and tube.

e. If molten uranium should contact the process tube and the temperature not exceed about 1300 C, the process tube should be able to contain the debris for at least several hours.

f. The molten pool model (with the uranium area doubled) appears to be the most realistic model. The logic is not that this is assured as the ultimate configuration but that the initial configurations one would expect near the melting point offer less surface area than this model. If chemical heating should cause the temperature to rise significantly, then failure could spread to the outer tube. If the temperature remains high, the outer uranium may melt and something resembling the molten pool configuration should be the result..

At this point the correct temperature to assume has not been defined. There is reason to believe, however, that the whole mass would cool to the process tube temperature - leading to a low rate of hydrogen production. Even if the temperature remains at 1200 C, the rate of hydrogen production is not very great. Only if it should substantially exceed 1200 C would large hydrogen volumes be predicted.

Figure 12 portrays a range of possibilities covered in the model studies. The volumes of hydrogen shown are for one column only. One can extend this to various size accidents by appropriate multiplication of the ordinate. The curve labeled "Steam Flow Limit - ∞ Rate" refers to use of maximum inner annulus and central hole water flow rate in a damaged fuel column (0.055 lb/sec/tube) and assuming an infinite corrosion rate. The curve labeled "Zr-2 Corrosion Only" refers to the model in which it is assumed that the fuel and tube reach the maximum temperatures (inner-1100 C, outer-1000 C, and tube-900 C) without failure of the cladding. For the molten pool model, the breadth of the band indicates the effect of doubling the uranium area. Effort will be continued to define and establish conditions which would prevail under accident assumptions with the objective of limiting the model to the most realistic cases.

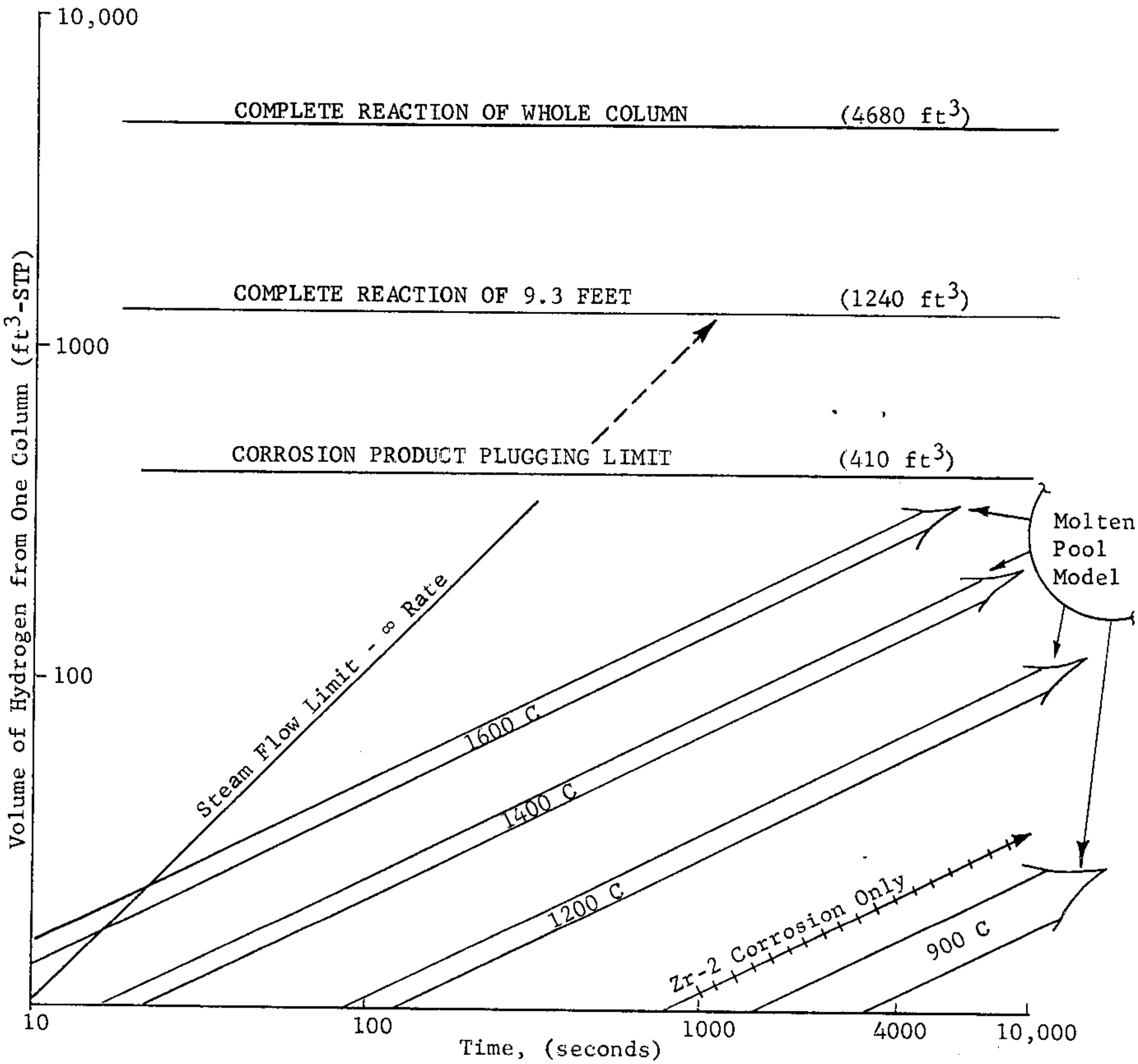


Figure 12. Accident Model Studies, Hydrogen Production

8.2 Primary Loop Blowdown After Pipe Rupture

The study of the transient primary coolant depressurization following a pipe rupture is currently being modeled for solution on the Dynamics System Analyzer (Dyna Sar IV). The model is constructed with the conventional partial differential equation needed for the solution. The loop will be considered in appropriate sections and the partial derivations with respect to position will be described algebraically; the resulting ordinary differential equation will be integrated by Dyna Sar.

The model will provide estimates of coolant flow, pressure, and temperature profiles; steam release from the break; and transport time from reactor exit to break location for various break sizes. For the initial efforts, only inlet header breaks upstream of the check valves are being considered.

8.3 Fission Product Release Tests

As reported in the February Record Report, one column of Production Test-76 fuel (which included three 8-inch fuel assemblies) was discharged in preparation for resumption of fission-product release tests. The fuel must cool 40 days but not more than 90 days for the fission product inventory to be within allowable limits. The tests are to be resumed in mid-April following hydrogen evaluation tests with unirradiated fuel, which are now underway.

8.4 Results from Soil Sampling Wells (N-Reactor Crib)

Several soil-sampling wells have been drilled in the vicinity of the N-Reactor to determine the distribution of isotopes in the soil and to record their movement towards the river. Three walls have been completed and a fourth is being drilled in the vicinity of the N-Reactor crib. Results for Strontium-90 and Cesium-137 tests approximated predicted values and are well within established limits.

8.5 Pipe Rupture Study

A study of the process tube connectors and the primary risers was made to determine if a single process pipe break can proliferate into multiple failures. In a previous study for HAPD, General Electric consultant Dr. W.J. Love examined the process tubes for the same purpose, using both analytical and experimental techniques. In the current analysis, conservative assumptions were used which demonstrated that the primary system piping is adequately designed to prevent proliferation of a single failure.

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.

PRODUCTION PROGRAM
FOR IRRADIATING ALL HANFORD-PRODUCED NEPTUNIUM

INTRODUCTION

Current space technology needs and projected future demands for industrial uses have created a strong interest in expanding the production and reducing the cost of plutonium-238. A manifestation of this interest was the creation of a program by the AEC in FY-1966 to evaluate Hanford's capability to produce Pu-238 on a continuing basis. The program entailed irradiating three three-quarter length columns containing 44 target elements of neptunium and aluminum alloy (4000 grams of neptunium) in the N-Reactor neutron flux field and individual targets in K-Reactor (400 grams of neptunium). The demonstration program has been largely completed, and an advantageous economic and production potential has been identified. The large-scale program at N-Reactor has established a sound base to derive a complete production program by providing information on:

Production potential

Chemical separation technology

Target fabrication technology

Program continuity

The purpose of this document is to present the basis for and a general outline of a process scheme, initiated within the calendar year, to produce in N-Reactor significant quantities of Pu-238 by irradiating all Hanford-generated neptunium. The complete target fabrication and product separation to support this production plan will be performed in previously demonstrated facilities without capital expenditure.

The production scheme presented and the program being proposed for K-Reactor will demonstrate the full production capability at Hanford and will provide a firm basis for expanding the program should the anticipated post-1970 market develop.

SUMMARY

The irradiation of neptunium elements for the production of Pu-238 will again demonstrate the multiproduct capability of the N-Reactor plant and will enhance the diversification effort for marketable reactor products.

TABLE I

NEPTUNIUM RECYCLE PROGRAM

BY QUARTER*

	CY-1967			CY-1968			CY-1969			CY-1970			CY-1971		
Quarter	S	F	I	S	F	I	S	F	I	S	F	I	S	F	I
1					8	24	11	18	57	18	26	83	27	29	105
2				8	11	35	11	20	64	20	29	90	29	31	105
3		8	8	8	17	38	17	25	67	26	32	94	33	21	104
4		8	16	8	17	47	18	26	75	26	37	103	29	25	85

*NOTE: The kilogram values for Separation (S) and Fabrication (F) are totals, however, the value for Irradiation (I) is the maximum quantity in the reactor at any one time during the quarter.

DECLASSIFIED

[redacted]

The production plan proposed for N-Reactor is to:

a. Divert the neptunium production stream from the Isochem Purex facility to the fabrication of 15 w/o neptunium-aluminum alloy target elements.

b. Expose to the N-Reactor flux for 250 days at 4000 Mw.

c. Chemical processing of irradiated elements in existing pilot-scale facilities.

d. Recycle to produce target elements for further irradiation.

Each phase of the production plan has been demonstrated, and the proposal can be accomplished within the calendar year with a minimum research and development effort and no capital expenditure. The proposed plan can produce up to 15 percent more Pu-238 than the present process for the 1968 to 1970 period

Using targets containing 15 percent neptunium in a 10-month irradiation and a 3-month ex-reactor processing cycle, the capacity of all needed facilities is adequate to process the entire quantity of neptunium from Purex, as well as the recycled material through calendar year 1970. Should the market for Pu-238 develop as projected and with the availability of neptunium from commercial power reactors, the need for expanded production facilities within the AEC complex would occur during calendar year 1971. The experience and detailed engineering data gained by this interim production scheme would provide a solid base for the design and construction of such facilities.

The quantity of neptunium at each phase of the program on a quarterly basis is tabulated in Table I. The yearly operating cost is shown in Table II.

BASIS

The technological basis for the Pu-238 production proposal is concluded from the 4000-gram neptunium irradiation test initiated at N-Reactor early in calendar year 1966. The demonstration program utilized 14 cast alloy target elements containing 9.2 w/o neptunium in aluminum and 30 cast alloy elements of 6.5 w/o neptunium in aluminum. The charge makeup and the exposure history for the 44 target elements are detailed in Table III.

The process flowsheet derived for the total production plan and demonstrated during the test period is shown in Figure 13, along with an alternate plan. Each step was completed with good success and continued routine operation is basically proven.

[redacted]

[redacted]

TABLE II

Pu-238 PRODUCTION PROPOSAL

OPERATING COST

	Unit Cost $ / Element	Operating Budget X 1000 1967	1968	1969	1970
Conversion					
Fuel Fabrication					
Scheme 1 (Iso)	185	16	52	84	114
Scheme 2 (BNW)	238	21	67	108	147
Chemical Separation					
Decladding	219		28	65	102
Irradiation(1)					
TOTAL					
Scheme 1	1555	41	246	507	779
Scheme 2	1608	46	261	531	812
Average Unit Cost ($/g)					
Scheme 1					
Scheme 2					

(1) each element contains ~187 grams neptunium at 22.5% conversion yields 42 grams plutonium per element to full exposure.

[redacted]

TABLE III

CHARGE DETAIL AND EXPOSURE HISTORY
Pu-238 DEMONSTRATION PROGRAM

Column	Charge Makeup	Charge Date	Discharge Date	Days at 4000 Mw	Target Burnup
1	14 elements at 9.2 w/o Np	3-25-66	10-3-66	74	13.8
2	15 elements at 6.5 w/o Np	5-29-66	2-6-67	112	17.0*
3	15 elements at 6.5 w/o Np	5-29-66	3-12-67	141	22.0*

*Calculated Values

[redacted]

Scheme 1		Scheme 2
Isochem	Withdraw $Np(NO_3)_4$	Isochem
Isochem	Convert to NpO_2	Isochem
Isochem	Form Master Alloy	BNW
Isochem	Fabricate Aluminum-Clad Elements	BNW
GE	Zr-2 Clad and Inspection	GE
GE	N-Reactor Irradiation	GE
GE	Declad N-Basin	GE
BNW	Separations	BNW

Figure 13. Pu-238 Production Proposal Flowsheet

[redacted]

The rate and efficiency of neptunium conversion and the annual production levels being forecast are based on measured results recorded from the experimental program. These data show, for the 9.2 w/o neptunium-aluminum target exposed for 74 full power days, 9.0 percent of the neptunium atoms charged are converted to plutonium of which 94.4 percent are Pu-238.

For the production plan, a 15 percent neptunium target element will be used, the objective being to reduce the total number of elements required and increase the exposure time to reach a given product purity and hence increase the total number of Np-237 atoms converted to Pu-238. The 250 full power day exposure time to be utilized for the proposed production mission will yield a product purity of 80 percent at a conversion efficiency of 86 percent, as compared to the 80 percent purity and 82 percent conversion efficiency of the present process.

The data for the 9.2 w/o and 15 w/o neptunium in aluminum target elements are tabulated in Table IV. Using the results for the latter case, the annual recycle production level through 1970 is calculated and tabulated in Table V.

PRODUCTION PROGRAM

The pilot-scale production program demonstrated at N-Reactor requires minimal research and development effort and no capital expenditures to expand to a routine operation involving all available Hanford-produced neptunium. The major operational steps in the process flow are detailed below.

Fabrication

The target element to be used for the Pu-238 program will be an improved version of the cast neptunium-aluminum alloyed element used in the demonstration program, the major difference being an approximate 65 percent increase in the neptunium concentration (i.e., 15 w/o as compared to 9.2 w/o for the first column of the demonstration load) in the alloy.

The target element fabrication facility, based on experience to date, can produce usable target elements at the rate of approximately 150 elements (29 kg neptunium) or ten columns per month on an operating schedule of one shift per day. This rate is an estimated equilibrium figure and should be treated as a maximum quantity; however, the quarterly fabrication rate requirement is not expected to exceed 30 kg of neptunium during any one quarter before 1971. The process flow, rate of production, and manpower are detailed in Table VI.

[redacted]

TABLE IV

EXPERIMENTAL Pu-238 DATA

(Column Average)

Target	Exposure	Np per Column		Pu per Column		α[1]	$1-\beta$[1]	ϵ
w/o	fpd	Irradiated grams	Remaining grams	Total grams	Pu-238 %	%	%	%
9.2	74	1612.1	1455	145.7	94.4	9.02	9.8	94.8[2]
15[3]	250	2790.0	2051	636	80.0	22.8	26.5	86.0

(1) α - Atom of plutonium produced per atom of neptunium charged, expressed as a percent.

$(1-\beta)$ - Fraction of neptunium consumed per cycle, expressed as a percent.

(2) $$\frac{\text{Pu Total}}{\text{Pu Total + Grams Fissioned}} = 94.8$$

(3) These data for the 15 w/o target are calculated values.

[redacted]

[redacted]

TABLE V

YEARLY PRODUCTION
USING HANFORD NEPTUNIUM

Year	Neptunium Available kg	Maximum No. Target Columns During Year	kg Np-237 Throughput	kg Plutonium Delivered
1967[1]	24	6	- -	-
1968	45	17	24 (126)[2]	5.4
1969	50	27	57 (300)	12.8
1970	58	36	89 (467)	20.1
1971[3]	23	36	119 (625)	27.0

(1) Last three quarters of 1967. All neptunium quantities based on revised conversion ratio data.

(2) Number of target elements fabricated from given neptunium quantity.

(3) Neptunium addition in 1971 equivalent to burnout in 1970.

[redacted]

TABLE VI

FABRICATION DETAIL

Process Flow	Rate of Production	Manpower
Form master alloy	1.6 elements per shift	1 engineer, 3 technicians
Form and cast aluminum-neptunium alloy	8 elements per shift	1 engineer, 3 technicians
Machine assembly to specifications	6.5 elements per shift	1 engineer, 3 technicians
Clean aluminum tube cladding	15 elements per shift	2 technicians
Insert alloy in outer cladding; insert graphite rod; weld aluminum end closure	15 elements per shift	1 engineer, 3 technicians
Insert into Zr-2 shroud; weld Zr-2 end closure	12 elements per shift	2 technicians
Weld Zr-2 supports; leak check; etch	15 elements per shift	1 engineer 3 technicians
Autoclave*	15 elements per autoclave cycle plus 2 days	2 technicians
Leak check	15 elements per half shift	1 technician
Attach support covers; final inspection	15 elements per half shift	1 engineer 2 technicians

*For a cycle of 16 hours, the throughput would be approximately 60 elements per week with two autoclaves.

Irradiation

The irradiation scheme to be utilized for the program will include a target and adjacent driver geometry as demonstrated by the production test. The drivers-to-target ratio will be 8 to 1 with a maximum of 36 target columns and 288 Mark I-A driver columns in the reactor at any one time through calendar year 1970. The maximum number of target columns in the reactor at any one time is listed by year in Table I, along with the total neptunium throughput and the plutonium production rate. The target elements for the production proposal will be discharged after approximately 300 calendar days of exposure (250 full power days). The outer cladding and graphite insert will be removed in the fuel examination facility at N-Reactor and transferred to Chemical Separations, Battelle Northwest. The decladding operation was demonstrated in the Radiometallurgy Laboratory of Battelle Northwest; however, on a full-scale production mode, the decladding operation will be moved to the reactor area.

Chemical Separation

The chemical separation will involve a mercury catalyzed HNO_3 dissolution and two anion exchange cycles as demonstrated by the experimental production program. Since all phases of the separation process are performed in the laboratory hot cell, the manpower requirement of one chemist and one operator per shift is constant. The chemical separation process is graphically displayed in simplified form in Figure 14 with the percentage of separation and the neptunium and plutonium product lost to the waste stream indicated on the flow diagram. The separation process has a maximum throughput controlled by the aluminum content in the dissolver tank and the anion exchange column. The process flow rate as demonstrated is 6 kg of alloy per 85 hours when the process waste stream is held and discharged to a retainer tank at the conclusion of each process cycle. However, if the waste material is discharged on a continual basis, the process cycles can overlap and the throughput potential is 12 kg of alloy per 127 hours. If an A-B-C-D shift and an 80 percent operating efficiency are assumed, the annual throughput will be:

a. Alloy throughput - 662 kg per year

b. Neptunium throughput - 99.3 kg per year.

The annual throughput potential is approximately 50 percent greater than the anticipated annual neptunium throughput to 1971. This throughput could be increased by 25 percent by utilizing a larger dissolver tank. The neptunium nitrate will be isolated and transferred to Isochem's Z-Plant where the oxide conversion will be made and the cycle repeated. Two items necessary for program continuity that require some further schedule work are (1) storage facilities for and transportation of process waste, and (2) transportation of kilogram quantities of neptunium nitrate on a regular schedule.

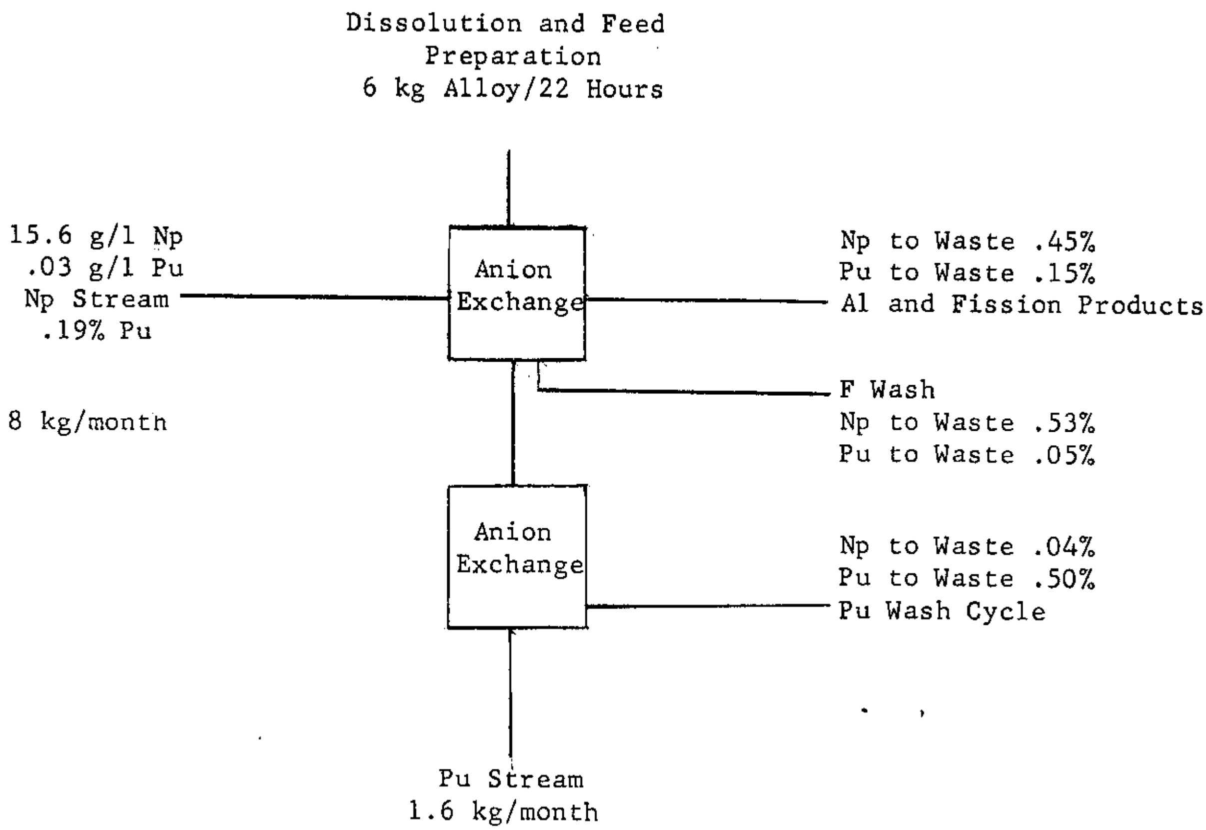


Figure 14. Chemical Separation Process

ADMINISTRATIVE SUMMARY

MARCH 1967

EMPLOYMENT

Section	Permanent		Temporary		Total		Change		Total
	E	NE	E	NE	E	NE	E	NE	
General	2	1	0	0	2	1	0	0	3
Closeout Group	12	22	0	0	12	22	-2	-1	34
Employee Relations	6	4	5*	0	11	4	0	0	15
Finance	10	12	0	0	10	12	0	0	22
Fuels	48	143	0	0	48	143	0	-2	191
Plant	59	213	0	0	59	213	-1	0	272
Project	56	17	0	0	56	17	+1	0	43
Research and Engineering	52	19	0	0	52	19	-1	-1	71
Total	245	431	5	0	250	431	-3	-4	681
New Change					681		-7		

*Technical Graduates

SAFETY

Days without a disabling injury 279

Hours worked without a disabling injury Est. 951,856

VISITS

Name	Company and Location	Contact	Date	Purpose
WD Bainard	General Electric Vallecitos, Calif.	JM Skarpelos	3-13-67	To discuss water quality monitoring
	General Electric San Jose, Calif.	EA Grimm	3-13-67	To discuss chemical cleaning
	NACE, Biltmore Hotel Los Angeles, Calif.	Convention	3-14-67 thru 3-17-67	Attended National Assoc. of Corrosion Engineers Convention
	Turco Products Wilmington, Calif.	K Newman	3-17-67	Discuss decontamination
J Muraoka	Allbrook Laboratory Wash. State Univ. Pullman, Wash.	H Copp R Tinney	3-3-67	Observed Columbia River hydraulic model.
GR Lawrence TP Butterfield RC Knop	Rosemount Engineering Minneapolis, Minn.	L Lofgren B Brown Al Dreis	3-20-67 to 3-21-67	Discuss RTD problems
	Minco Products Minneapolis, Minn.	D Zalinka Ray Dunning Carl Schirr	3-21-67	Discuss RTD problems
WJ Dowis	General Electric Philadelphia, Pa.	RE Trumble	3-20-67	Discuss isotopes
	ORNL, Oak Ridge Tennessee	J Shacter	3-21-67 to 3-22-67	Discuss AEC planning

VISITS (continued)

Name	Company and Location	Contact	Date	Purpose
AC Callen	Reactive Metals, Inc. Ashtabula, Ohio	G Carozza	3-20-67 thru 3-21-67	Observe upset-forging and extrusion-to-size
JJ Wick	General Electric Co. San Jose, Calif.	P Smith	3-12-67	Interview for possible position at APED
JR Glindmeyer	REM, Inc. Portland, Oregon	RH Brown	3-14-67	To expedite material
TP Butterfield	Rosemount Engineering Minneapolis, Minn.	LF Lofgren	3-20-67	Negotiate procurement
	Minco Products Minneapolis, Minn.	R Zelinka	3-20-67	Negotiate procurement
TP Butterfield & SA Fitch	Precision Products Gresham, Oregon	Mr. Wilcox	3-28-67	Negotiate P.O. Changes
	Wah Chang, Inc. Albany, Oregon	R Graham	3-29-67	Negotiate P.O. Changes
	REM, Inc. Albany, Oregon	RH Brown	3-29-67	Negotiate P.O. Changes
JW Nickolaus & HP Kraemer	Carborundum Metals Akron, Ohio	J Houston	2-27-67	Discuss procurement problem
	Brush Berrylium Cleveland, Ohio	H Piper	2-28-67	Discuss procurement problem
	Reactive Metals, Inc. Niles, Ohio	Wm. Fetter Dick Bean	3-1-67	Discuss procurement problem

VISITS (continued)

Name	Company and Location	Contact	Date	Purpose
JW Nickolaus & HP Kraemer	NUMEC Apollo, Pa.	M Waring	3-2-67	Discuss procurement problem
	National Lead Cincinnati, Ohio	Chas. Bussert Max Cawdrey	3-3-67	Discuss procurement problem
	AEC Hdqtrs. Washington, D.C. (Germantown)	B Ritzman W Kester	3-6-67	Discuss procurement problem
MC Leverett JW Riches MM Hendrickson	ACRS Subcommittee on Meteorology Washington, D.C.	Frank Gifford	3-10-67	Meteorological Calculations
WK Kratzer KL Fowler	GE-Nuclear Eng. Div. San Jose, Calif.	EA Grimm	3-13-67	Discuss chemical cleaning and decontamination
	GE-Vallecitos Atomic Power Lab. Pleasanton	J Skarpelos RS Gilbert	3-13-67	Discuss primary system coolant control, water analysis, radio chemical isotopic analysis
	Turco Products Co. Wilmington, Calif.	K Newman	3-17-67	Discuss Decontamination chemical performance
KL Fowler	Beckman Instruments Fullerton, Calif.	Dave Wieseler	3-14-67	Current problems with our Bechman instruments
KL Fowler	City of Los Angeles Power Los Angeles (Seal Beach)	Robert Bradley	3-15-67	Water analysis and control techniques
WK Kratzer KL Fowler	National Assoc. of Corrosion Engineers meeting Los Angeles, Calif.		3-14-67 thru 3-17-67	Attend meeting

VISITORS

Name	Company and Location	Contact	Date	Purpose
GL Andrews and J Beecher	Drew Chemical Co. Tacoma, Wash New York, N.Y.	WD Bainard	3-1-67	Discuss boiler fuel oil treatments
EL Knoedler	Sheppard T. Powell & Assoc. Baltimore, Md.	WD Bainard	3-2-67 to 3-3-67	Discuss chemical cleaning and water treatment
ST Smith	Ann Arbor Bearing & Manufacturing Co. Ann Arbor, Michigan	GA Newell	3-10-67	Discuss "No-Bak" maintenance problems
M Eastly	Chas. W. Fowler Co. Bellevue, Wash.	RK Bollinger GL Swezea	3-17-67	Discuss in-core flux monitoring
LL Becker RA Cudney DA Doss	ALCAN Aluminum Riverside, Calif.	DW Darsow	3-15-67 3-16-67 3-17-67	Procurement specifications
HC Brassfield	GE-Cincinnati, Ohio	MC Leverett M Lewis	3-13-67	Hazards analysis
JJ Shea	AEC-Division of Reactor Licensing Washington, D. C.	MC Leverett	3-13-67	Technical discussions
Joe Dull	Process Automation Co. Los Angeles, Calif.	CF Poor AJ Grambihler MG Venneberg	3-27-67	Instructions on ISI-609 System

SIGNIFICANT REPORTS

Report No. & Class.		Title	Author	Date
HW-83091 Rev 1	C	Bibliography - Coextrusion Process	MH Fitch	3-7-67
RL-NRD-66 SUPII	S	Budget Study Report, Tritium Extraction Facility, N-Reactor, 100-N Area	CE Love R Cooperstein DD Stepnewski	3-24-67
RL-NRD-726 SUPII	U	Supplement to PT-NR-65: Evaluation of Effect of Annealing After Fabrication Upon Performance of Inconel Helical Sample Chambers	DW Leiby	3-6-67
RL-GEN-1038 7	S	Coproduct Demonstration Schedule	EE Leitz	3-24-67
RL-GEN-1065 11	U	Ball Safety Functional Test	GC Sorensen	3-10-67
RL-GEN-1443	S	Target Yield and Defective Rates	WH Hodgson	2-15-67
RL-GEN-1454	S	Yield and Defective Rates	WH Hodgson	2-15-67
RL-GEN-1456	C	Increased Orificing Program	JL Benson	3-10-67
RL-GEN-1468	S	Schedule for Testing Fuel Irradiation	EE Leitz	3-1-67
RL-GEN-1471	U	Description and Operation of the Oil-Fired Boiler and Auxiliary System	AJ Ebens	3-16-67
RL-GEN-1477	S	Coproduct Program Technical Criteria	MC Leverett	3-20-67
RL-GEN-1478	U	A Summary of Plant, Metal-Water Reaction Tests (A Nuclear Safety, Mission 8, Research and Development Program)	MM Hendrickson	3-28-67
RL-GEN-1482	S	Production Program for Irradiating all Hanford-Produced Neptunium	DW Constable	3-22-67

SIGNIFICANT REPORTS (continued)

Report No. & Class.		Title	Author	Date
RL-GEN-1487	U	N-4 Test 5.4, Part 2, Detailed Results from 600 Mwe Load Rejection Test of November 28, 1967	DL Renberger	3-16-67
RL-GEN-1488	S	Cost Improvement, End Test Sensitivity Limits	HD Bell	3-3-67
RL-GEN-1489	U	PT-NR-72, Process Tube Rear Nozzle and Connector Decontamination	LL Sims	3-10-67
RL-GEN-1491	S	Project Proposal, Improved Irradiated Fuel Handling System, 105-N Building, (Project GCP-409)	AA Janos	3-20-67
RL-GEN-1493	C	A Summary of the Coproduct Increased Power Demonstration Objectives	AJ Ebens	3-8-67
RL-GEN-1494	C	Mark IV Fuel Data (Preliminary Design)	MC Leverett	3-9-67
RL-GEN-1499	S	Fuels Planning Data-Advanced Fuel Designs	TD Naylor	3-13-67
RL-GEN-1512	C	Coextrusion Billet Design Data for Three Alternate Mark IV Fuel Geometries	RH Scanlon	3-16-67
RL-GEN-1513	C	Operating Safety Limits Revisions for Increased Power Operation	RL Dickeman	3-29-67
RL-GEN-1532	C	Engineering Study - In-Core Flux Monitors	CA Mansius	3-20-67
RL-GEN-1535	U	Alternate Formulations of the Rate Laws for the High Temperature Metal-Water Reactions Involving Uranium and Zircaloy-2	TW Evans	3-23-67
RL-GEN-1541	U	Metal-Water Reactions During a Loss of Coolant Accident in N-Reactor	TW Evans	3-23-67

SIGNIFICANT REPORTS (continued)

Report No. & Class.		Title	Author	Date
RL-GEN-1544	S	Request to Ship Unirradiated Aluminum Clad Coproduct Target Assemblies	MC Leverett	3-24-67
RL-GEN-1545	C	Request for Chemical Analysis of Drivers from 2.1 Coproduct Block (PT-66)	MC Leverett	3-27-67
RL-GEN-1547	U	Notes on Calculation of Metal-Water Reaction for Mark II Fuel	NR Miller	3-24-67
RL-GEN-1549	C	Confiner Release Rates Following Failure of a Primary Loop Pipe	NR Miller	3-29-67
RL-GEN-1550	C	Fuel Heating Following Loss of Cooling	NR Miller	3-29-67

SUGGESTION PLAN PARTICIPATION

	March	CY-1967
Number of Eligible Employees	431	435 (Average)
Number of Suggestions Received	37	139
Number of Suggestions Acted Upon	32	225
Number of Suggestions Adopted	16	54
Net Annual Savings	$644	$8,785
Amount of Awards	$215	$1,295
Percent of Awards to Savings	33.4	14.7
Average Amount of Awards	$ 13.44	$ 23.98
Adoption Rate Per 1,000 Employees (annualized)	---	495

INVENTIONS

Name	Title
R. Cooperstein	Extraction of Tritium from Lithium Aluminate Targets
R. R. Studer	Methods of Fabrication of Lithium-Bearing Targets for Irradiation

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