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PROGRESS REPORT

April 1959

WORKS TECHNICAL DEPARTMENT

E. I. du Pont de Nemours and Company
Explosives Department - Atomic Energy Division
Savannah River Plant

DELETED VERSION

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PROGRESS REPORT

April 1959

Works Technical Department

May 18, 1959

E. I. du Pont de Nemours and Company
Explosives Department - Atomic Energy Division
Savannah River Plant

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. . . Separations Technology Summary

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227 THE RECOVERY ANION EXCHANGE TEST FACILITY operated satisfactorily on slag and crucible solution during the first runs.

229 PREPARATIONS FOR PROCESSING ENRICHED URANIUM IN H AREA are nearing completion. Necessary equipment changes, equipment cleanout, and solvent changeout are almost completed.

229 THE FIRST COLD DISSOLVING TEST AND COLD URANIUM RUNS in solvent extraction are in progress.

238 MARK VI, MARK VI-J, AND CONTROL ROD SLUGS were processed in Building 232-H at an average rate of 1.3 B/D.

239 THE USE OF CURRENT-QUALITY MARK VI-J SLUGS appears to cause an MBR increase from the Line 2 normal of to about The stack stripper system recovered per run from Mark VI-J batches, compared to a normal per run.

238 THE ISOTOPIC PURITY of diffusion column feed has dropped from to as a result of Mark VI-J processing.

238 NINETY-FIVE PERCENT OF THE GAMMA ACTIVITY in an extraction furnace was removed by 1% nitric acid flushing, in preparation for furnace repair.

239 THE NORTH FURNACE OPERATION WAS DISRUPTED by failure of additional bottom heaters.

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243 STACK LOSSES averaged per run. Unusually large contributions were made by the furnace discharging and charging operation, and decanning of Mark VI-J hollow slugs.

245 MULTISTAGE DIFFUSER CAPACITY for greater than predicted from data at lower feed compositions.

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321	AN AUTOMATIC FILM BADGE READER is undergoing evaluation.
325	A PULSE-TYPE READER FOR POCKET DOSIMETERS offers several advantages over the string electrometers now in use.
328	CONTAINERS FOR USE IN IRRADIATING COBALT PELLETS have been designed.
328	NOZZLES THAT PRODUCE SOLID SQUARE SPRAY PATTERNS have been tested for use in flooding the reactor room floor to cool a dropped fuel element.
330	AN ASSEMBLY OF FOUR STEEL SLEEVES on 3" centers appears satisfactory for use in canning mechanization in Building 313-M.
330	MARK VII-A SLUGS FAILED at lower than expected internal pressures in hydraulic burst tests.
332	IRRADIATED SLUGS WITH COEXTRUDED MARK VII-A CORES showed large length decrease and warp increase as compared to control slugs with rolled cores.
332	THIRTEEN MARK VII-A SLUGS FAILED during April.
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Technical Assistance

Reactor Performance Summary

	R Area	P Area	L Area	K Area	C Area
Fuel (No. of Elements and Mark Number)					
Maximum Power Attained, %	86	86	85	85	87
Innage, %	91.4	82.1	73.2	84.0	80.4
Shutdowns, No.					
Charge-Discharge	1*	0	1*	1	0
Unscheduled	1	4	6	2	5
Effective Tubes, avg No.	474	470	472	464	524
Moderator					
Activity, 1000 c/m/ml	220	322	235	104	240
Conductivity, micromho/cm	0.60**	0.06	0.10	0.15	0.03
Deuterium Generation, scf/day	16.8	28	33	36	45
Isotopic Purity, mol % D_2O	99.67	99.65	99.68	99.66	99.80
H_2O Removed (Distillation), lb	219	123	51	80	58
H_2O Added (Drums), lb	22	22	20	61.5	20
Turbidity, ppm	100	350	115	37	10
HX Data (avg of all)					
Power at which Computed, %	84	79	80	84	77
Process Water Temp, °C					
HX Inlet	86.2	85.5	85.7	86.3	81.9
HX Outlet	37.4	39.5	40.0	37.1	38.3
Cooling Water Temp, °C					
HX Outlet	71.7	72.3	71.7	70.6	67.9
HX Inlet	15.8	18.8	19.4	15.8	17.9
Heat Transfer Coefficient, pcu/(hr)(ft^2)(°C)	560	575	546	543	516
Heat Removed by Shields, %	0.25	0.25	0.33	0.27	0.12
Plenum Pressure, psig	96.4	99.0	96.3	100.0	87.0
Cooling Water Flow, %	964	975	990	989	980
Total Process Water Flow, %	1020	1011	1015	1003	1065
Fuel Flow, %	958	949	955	942	998
Avg Channel or Element Flow, %					
Zone I	0.445	0.448	0.454	0.447	0.439
Zone II	.362	.373	.367	.359	0.341
Zone III	.291	.289	.293	.288	-
Zone IV	.269	.210	.270	.208	-
Zone V	.213	.175	.211	0.193	-
Zone VI	0.176	0.267	0.175	-	-
Spikes	1.58	1.58	1.58	1.62	1.47
Sparger-Jets	2.67	2.67	2.67	2.67	2.62
Special VI-J	-	-	1.60	-	-

* Charge-discharge was initiated following a fuel failure.

** High value associated with R-6 startup.

R AREA

OPERATING SUMMARY

Operation of R-5 cycle continued at power levels of 81 to 86% until the cycle was terminated by a Mark VII-A fuel element failure on April 21 at an exposure of 35.5%. Reactor operation was also interrupted by a Mark VII-A fuel failure on April 18. The subsequent reactor shutdown lasted 31 hours.

Initial criticality for the R-6 cycle was attained on April 25, and a power level of 76% was achieved on April 29. The R-6 initial power ascension is continuing.

Operating power levels during the April portion of the R-5 cycle were limited by approach to the pump shaft break and elbow cavitation temperature limits. Typical and maximum operating conditions are compared with operating limits in the following table.

Condition	Limit	Observed Values	
		Typical*	Maximum
Bulk Moderator, °C	101	96	98
Pump Shaft Break, °C	110	110	110
Reactor Effluent, °C	90.5	88.4	89.2
Channel or Quadrant Effluent, °C			
Mark VII-A	105	99.2	100.3
Mark VI-S	103	92.2	95.8
Mark I	105	98.2	99.2
Burnout Safety Factor			
Mark VII-A	1.4	2.7	2.0
Mark VI-S	1.4	3.0	2.9
Sheath Temperature, °C			
Mark VII-A	180**	130	137
Mark VI-S	160	129	131
Slug Power, %			
Mark VII-A	0.00366	0.00302	0.00321
Mark I	0.00224	0.00167	0.00185
Number of Effective Tubes	-	474	-
Power, %	-	83.5	-
Exposure, %	-	23.9	-
River Water Temperature, °C	-	17.5	-

* R-5 cycle, April 6.
** Or local boiling, 150°-170°C.

The axial flux distribution in the control zone of the R-6 reactor charge is being maintained with a rooftop ratio of 1.10 ± 0.03 to reduce sheath temperature in the lower half of the Mark VII-A fuel columns. (This rooftop ratio will be adjusted as the cycle progresses to correct for fuel burnup.) This action has been taken in an effort to reduce the fuel failure rate. A 5° to 10°C reduction in the maximum Mark VII-A sheath temperature is being attained with this method of operation.

ASSEMBLY

The following table lists components charged and discharged during the R:5-6 outage and lists components recycled in the R-6 charge.

Components	Charged	Discharged	Recycled
Fuel Elements			
Mark VII-A			
Mark VI-S			
Mark I			
Mark V (TA 1-723)			
Mark V-A (TA 1-707)			
SPR (TA 1-744)			
Mark V (Zr Clad) (TA 1-735)			
Total →			
Control Rods			
1.0%, 17-Slug			
3.5%, 16-Slug			
1.5%, 8-Slug			
1.5%, 6-Slug			
3.5%, 6-Slug			
Total →			
Miscellaneous			
Target Thimbles (TA 1-654)			
Insulated Slugs (TA 1-687)			
4.9%, 10-Slug Shadow Rods			

* This Mark VII-A element was charged on April 19 to replace a failed fuel element.

** Fuel and targets autoclaved 40 hours and steam dried 9 minutes.

MODERATOR

Laboratory analyses of moderator isotopic purity increased abruptly on April 13 from 99.64 to 99.68 mol %. Laboratory results for the 15-day period prior to the increase were almost constant at 99.64 mol %. Analyses remained at approximately 99.68 mol % until April 25. No explanation is afforded to account for this indicated change in purity. Abrupt increases were not experienced in other laboratory analyses of isotopic purity using the same instrumentation, and no known change has occurred

in the normal sampling techniques. A program has been initiated to begin more meticulous control of sampling and sample handling during May to determine the cause of these abrupt changes in the indicated isotopic purity.

FUEL ELEMENT FAILURES

Two Mark VII-A fuel failures occurred during the April portion of the R-5 cycle. Pertinent data concerning the failures are summarized in the following table.

R-5 Fuel Element Failures

Failure Number →	52	55
Date	4/18	4/21
Coordinates (X,Y)	39,21	19,21
Channel	A	D
Type Fuel Element	VII-A-SRP Ingot	VII-A-SRP Ingot
Reactor Exposure, %	34.6	35.5
Max Slug Power in Channel, %	0.00300	0.00308
ΔT Increase, °C	2.7	4.4*
Flow Decrease, Segmental Recorder, %	1.75	2.0*
Activity Decreases, %		
Low Energy Gamma	87	40
Blanket Gas	276	50
Seal Head Tank	173	0
Single Channel Gamma	8	0
Delayed Neutron	0	0
Ten-Minute Delay Line	Inoperative	Inoperative
Gas Chromatograph (TA 1-692) (Krypton peak)	>710	60
Blanket Gas Beta (TA 1-433)	Inoperative	Inoperative
Planchet, c/min/ml	112	-
Instrument to Give First Indication of Failure	Blanket Gas Act. Monitor	Flow and Temp Monitors
Time of Initial Indication, hours before shutdown	2.47	1.15
Time of Initial Detection, hours before shutdown	1.88	0.07

* Approximately half of these changes occurred during the last ten minutes of reactor operation.

The vertical locations of the failures have not been determined. Both elements were stored in failed fuel element containers.

DISASSEMBLY - DECONTAMINATION FACILITY

The following tabulation describes the current status of all fuel failures which have occurred in R-Area since the long shutdown (July 15, 1958).

Status of Fuel Element Failures
(Dates are 1958 and 1959)

Failure No.	Date of Failure and Type*	Status
30	10/13/58 (Ingot)	Q-tube in isolation tank contains 19 slugs. End fitting has not been removed.
31	10/21/58 (Dingot)	Q-tube in isolation tank contains 24 slugs. End fitting has not been removed.
32	10/24/58 (Dingot)	Q-tube in isolation tank contains 28 slugs. End fitting has not been removed.
33	10/31/58 (Ingot)	Canned for shipment to high level cave.
34	12/31/58 (Dingot)	Shipped to high level cave.
36	2/6/59 (Dingot)	Q-tube in isolation tank contains 50 slugs. The Q-tube was dropped and is damaged to the extent that the remaining slugs cannot be removed by standard methods. End fitting has been removed.
37	2/9/59 (Dingot)	Shipped to high level cave.
38	2/21/59 (Dingot)	Q-tube in isolation tank contains 5 slugs. All slugs are stuck. End fitting has been removed.
39	2/26/59 (Dingot)	Shipped to high level cave.
-	3/10/59 SPR	Q-tube in harp in disassembly basin.
52	4/18/59 (Ingot)	Q-tube in isolation tank contains 16 slugs. End fitting has not been removed.
55	4/21/59 (Ingot)	Q-tube in harp in disassembly basin.

* All failures were Mark VII-A-SRP except for the failure of March 10 which was a Technical Division Power tube.

An air-water mixture was flushed through the decontamination facility via the filter vessel (bypassing the anion and cation deionizers) on April 7 in an effort to reduce the radiation levels at the D-3 valve, the filter inlet line and the discharge line. Results of the purge are shown in the following table.

Location	Radiation Intensity R/hr at 3" Before	After
D-3 Valve	50	12
Discharge Line	12	1.2
Filter Inlet Line	10	3

The following table summarizes the equipment changes which have been made in the decontamination facility during April.

Equipment Changes

Date	Component	Type Change	Radiation Intensity
4/2	Anion	Regenerated	-
4/7	Filter	Replaced	800 mr/hr at 3" 18 mr/hr at 6'
4/7	Cation	Replaced	10 r/hr at 3" 100 mr/hr at 10'
4/21	Filter	Replaced	1.5 r/hr at 3" 8 mr/hr at 6'

implies that flow bypassing prob-

een by replacing the thin (0.032-inch)

will be of this improved design.

EQUIPMENT IMPROVEMENTS AND SYSTEM MODIFICATIONS

The wing equipment improvements and system modifications were complet

r system was modified to improve system relia-
tion of a high gain preamplifier. The Atomic

- Installation of individual count rate meters and alarms in the cooling water gamma monitor system was completed.

P AREA

OPERATING SUMMARY

The reactor was operated at power levels of 79 to 86% during April. The cumulative exposure of the P-6 charge was 37.70% at the end of April. Shutdowns occurred on April 3, April 16, April 21, and April 30 because of Mark VII-A fuel failures. Reactor power was limited by pump shaft break and Mark VII-A channel effluent temperatures. Operating limits and observed values for April were as follows:

Condition	Limit	Observed Values	
		Typical*	Maximum
Bulk Moderator, °C	101	95.2	99.0
Pump Shaft Break, °C	110	108.8	110.0
Reactor Effluent, °C	92	88.0	90.0
Channel or Quadrant Effluent, °C			
Mark VII-A	100	98.7	100.0
Mark I	105	98.6	100.1
Mark VI-S	103	88.4	92.6
Burnout Safety Factor			
Mark VII-A	1.4	2.6	2.4
Mark VI-S	1.4	3.1	2.3
Sheath Temperature, °C			
Mark VII-A	180**	135	140
Mark VI-S	160	125	133
Slug Power, %			
Mark VII-A	0.00366	0.00319	0.00341
Mark I	0.00224	0.00160	0.00179
Number of Effective Tubes	-	471	-
Power, %	-	83	-
Exposure, %	-	28.1	-
River Water Temperature, °C	-	16.3	-

* P-6 cycle, April 15.
** Limit is local boiling if it occurs below 180°C.

FUEL ELEMENT FAILURES

Four Mark VII-A fuel failures occurred during April. Information pertaining to the failed elements is presented in the following table.

P-6 Fuel Element Failures

Failure Number →	49	51	53	59
Date	4/3	4/16	4/21	4/30
Coordinates (X,Y)	34,54	35,51	30,42	15,45
Channel	D	D	C	D
Slug Position from Top of Channel	15	Unknown*	17	Unknown**
Type Fuel Element	Mark VII-A Ingot	Mark VII-A Ingot	Mark VII-A Ingot	Mark VII-A Ingot
Reactor Exposure, %	19.8	28.7	31.5	37.7
Slug Power at Time of Failure	0.00285	0.00256 (Max)	0.00253	0.00273 (Max)
ΔT Increase, °C†	4.4	1.5	2.8	3.0
Flow Decrease, Segmental Recorder, %	0	0.8	1.0	0
Activity Increases, %				
Low Energy Gamma	0	0	12	17
Blanket Gas	5	25	16	24
Seal Head Tank	3	14	19	14
Overflow Tank	0	6	0	3
Ten-Minute Delay Line	3	4	11	3
Single Channel Gamma	4	0	0	4
Delayed Neutron	0	0	0	0
Cyclone Gamma	Inoperative	Inoperative	10	6
Drip Pan	Inoperative††	21	†	††

* Quatrefoil in fuel failure container.
** Quatrefoil will be put in a fuel failure container.
† Based on comparison with previous 4-point temperature data.
†† A small particle collected in the drip pan showed an activity of 2.5 r/hr.
† Response obtained with moderator but insignificant response from failure because the instrument was on an insensitive scale.
†† Element has not been discharged from reactor.

Quatrefoils containing fuel failures 49 and 53 have been disassembled in the emergency disassembly basin. Failure 49 was disassembled by removing the slugs from the quatrefoil with the slug removal tool. The quatrefoil containing failure 53 could not be disassembled in this manner. The top 12 slugs were removed with the slug removal tool from the channel containing the failure without difficulty, but the 13th slug stuck. The quatrefoil was inverted and bumped on the basin floor. Four more slugs were removed from channel C in this manner, but considerable contamination was released in the emergency basin. The bottom end fitting was cut off and 3 slugs were removed from the bottom of channel C with the slug removal tool. The remaining slug (17th from the top of the column) could not be removed, therefore, a 14-inch section of the quatrefoil containing the failed slug was cut out and put in a failure can.

Failure 51 was placed in a failed fuel element container when it was discharged from the reactor on April 17.

ASSEMBLY

Small pin holes were discovered in the top end fitting welds of four quatrefoils when water squirted from the holes when the assemblies were hydraulically tested for individual channel flow. The quatrefoils have been rejected for reactor use but will be saved for metallurgical examination. These quatrefoils were being assembled for irradiation in the P-7 cycle.

L AREA

OPERATING SUMMARY

The L-4 cycle was terminated on April 1 at an exposure of 33.61% when a Mark VII-A fuel failure occurred.

Initial criticality for the L-5 cycle was attained on April 5. The reactor was operated at power levels of 80 to 85% during the month. Reactor power was limited by the quadrant effluent temperature limit and the pump shaft break limit. Typical and maximum operating conditions experienced during April are compared with the operating limits in the following table.

Condition	Limit	Observed Values	
		Typical*	Maximum
Bulk Moderator, °C	101	90.6	94.7
Pump Shaft Break, °C	110	107.7	109.1
Reactor Effluent, °C	90.5	85.6	88.4
Channel or Quadrant Effluent, °C			
Mark VII-A	100	96.8	99.9
Mark I	105	96.6	99.4
Mark VI-S	103	96.5	99.5
Mark VI-J	103	88.5	91
Burnout Safety Factor			
Mark VII-A	1.4	4.0	3.3
Mark VI-S	1.4	3.3	2.4
Mark VI-J	1.4	3.6	3.2
Sheath Temperature, °C			
Mark VII-A	180**	131	142
Mark VI-S	160	123	139
Mark VI-J	160	119	125
Slug Power, %			
Mark VII-A	0.00366	0.00316	0.00349
Mark I	0.00224	0.00180	0.00184
Number of Effective Tubes		472	
Power, %		80	
Exposure, %		4.32	
River Water Temperature, °C		17.6	

* L-5, April 13.
** Or local boiling, 150° to 170°C.

Six unscheduled shutdowns occurred during the month. The cause of each shutdown and the equivalent lost time at nominal power are listed in the following table.

Reactor Shutdowns

Date	Time	Power Level Before Shutdown, %	Cause	Time Lost, hours
4/6	9:08 AM	43	S-Foil Header Pressure - Low. Instrument mechanic inadvertently bumped supply valve control, and the valve closed.	8.2
	5:20 PM	65	S-Foil Header Pressure - Low. Possibly hydraulic noise in S-foil header pressure instrumentation.	5.7
4/7	12:05 AM	65	S-Foil Header Pressure - Low. Possibly hydraulic noise in S-foil header pressure instrumentation.	31.25
4/21	4:00 PM	82	Heat exchanger failure*	81.8
4/24	3:25 PM	2	Neutron Flux - 110%. High Level Flux Monitor spike - cause unknown.	3.2
4/26	1:10 PM	80	Mark VII-A fuel failure**	45.0

* See Heat Exchanger Leak section of this report.
** See Fuel Failures section of this report.

The cause of the "S-Foil Header Pressure - Low" scrams is being investigated. The pressure variations in the S-foil header have been measured with a strain gage pressure cell and high-speed recorder. The average pressure at 78% power is 78 psi with variations between 71 psi and 81 psi. The frequency of the variations is about two cycles per second. The set point for the Scram II circuit is currently 15% below the average header pressure rather than the normal setting of 10%. (See Engineering Studies section of this report for further details.)

ASSEMBLY

One hundred and four quatrefoils with perforated bottom end fittings were charged to the reactor for irradiation during the L-5 cycle (TA 1-732). Test station and reactor performance data are discussed in the Engineering Studies section of this report.

DELETED VERSION

The following table lists the components charged and discharged during the L:4-5 shutdown, and the components to be recycled in L-5.

Components	Charged	Discharged	Recycled
Fuel Elements			
Mark VI-S			
Mark VI-J (TA 1-714)			
Mark VII-A			
Mark I			
Total -			
Control Rods			
1.0% 16-Slug			
3.5% 16-Slug			
14.4S 16-Slug			
1.5% 8-Slug			
1.5% 6-Slug			
Miscellaneous			
4.9% 10-Slug Shadow Rods			
IGFM Instrument Rod (TA 1-704)			
Septifoil**, 2-Piece, Type B			

* Includes 104 with perforated bottom end fittings (TA 1-732).

** Replaced because the septifoil contained a broken control rod tip.

HEAT EXCHANGER LEAK

The reactor was shut down at 4:00 PM on April 21 because of a process water leak (three gallons per day) in heat exchanger 4A.

The first indication of the leak was received at 6:55 AM on April 20 when the System 4A cooling water gamma monitor recorder reading increased rapidly from 23% to 29% of full scale and initiated an alarm. The cooling water gamma monitor recorder continued to indicate between 30% and 40% of full scale until the shutdown at 4:00 PM on April 21.

The following three tests were performed to confirm the leak.

1. The 4A and 4B CWGM probes were interchanged and the high activity remained with System 4A.

2. Hourly samples of effluent cooling water from heat exchangers 4A, 4B, and 3A were analyzed for tritium with the Packard Tri-Carb instrument. Laboratory results on all samples taken between 7:15 AM and 5:15 PM on April 21 showed the tritium content to be less than 0.5 microcurie per liter (the lowest value the laboratory will report). However, the number of counts (5-minute counting time) registered by the instrument for samples from the leaking heat exchanger was approximately double the value for the control samples for nearly every set of samples (see following table.

DELETED VERSION

Tritium Content of Cooling Water Samples

Heat Exchanger →		3A		4A*		4B	
Date	Time	μc/ℓ	Counts**	μc/ℓ	Counts**	μc/ℓ	Counts**
4/20	7:15 AM	<0.5	-	<0.5	-	<0.5	-
	8:15	<0.5	309	<0.5	650	<0.5	246
	9:15	<0.5	276	<0.5	601	<0.5	257
	10:15	<0.5	-	<0.5	-	<0.5	-
	11:15	<0.5	286	<0.5	641	<0.5	260
	12:15 PM	<0.5	-	<0.5	-	<0.5	-
	1:15	<0.5	277	<0.5	584	<0.5	295
	2:15	<0.5	298	<0.5	699	<0.5	280
	3:15	<0.5	280	<0.5	744	<0.5	365
	4:15	<0.5	609	<0.5	712	<0.5	270
	5:15	<0.5	561	<0.5	741	<0.5	402
	6:15	<0.5	254	0.51	774	<0.5	369
	7:15	<0.5	447	<0.5	750	<0.5	648
	8:15	<0.5	316	<0.5	750	<0.5	464
	9:15	<0.5	448	0.68	916	<0.5	318
	10:15	<0.5	270	0.57	816	<0.5	248
	11:15	<0.5	256	<0.5	754	<0.5	254
4/21	12:15 AM	<0.5	290	0.51	1159	<0.5	434
	1:15	<0.5	520	0.55	1202	<0.5	523
	3:15	<0.5	652	<0.5	1122	<0.5	678
	5:15	0.53	791	0.99	1221	0.52	782
	7:15	<0.5	642	0.98	1230	<0.5	598
	8:15	<0.5	508	0.60	1015	<0.5	453
	10:15	<0.5	365	0.80	1045	<0.5	315
	12:15 PM	<0.5	606	0.53	1014	<0.5	686

* Leaking heat exchanger.
** Total counts for a 5-minute period.

3. The cooling water flow to the probe housing was valved off and the resulting decay observed with both the 4A and 4B instrument probes. The possibility of instrument error was eliminated by checking the decay with the 4B probe in the 4B housing. The results are listed in the following table.

Decay of CWGM Activity Signal
(L Area, April 21)

Time After Valve Closure, min	Count Rate, counts/min: 4A Probe In 4A Housing at: 8:45 AM	4A Probe In 4A Housing at: 2:30 PM	4B Probe In 4A Housing at 8:45 AM	4B Probe In 4B Housing at 2:30 PM
0	270	240	175	90
2	225	195	150	-
4	210	180	126	-
6	180	165	108	90
8	165	153	99	-
10	160	153	99	-
12	155	150	-	-
14	155	150	-	-

Examination of the cooling water gamma monitor recorder chart and calculations based on laboratory tritium analyses indicated that the heat exchanger had been leaking at a very low rate (approximately 0.8 gallon per day) at least since the startup of the L-5 cycle on April 5. The average leakage rate for the 12 hours before shutdown was 3.1 gallons per day, and the total moderator lost between the time of the alarm and the time of shutdown was calculated to be 3.6 gallons. A leakage rate of 0.8 gallon per day between April 5 and April 20 gives a calculated total moderator loss of 15.6 gallons from this heat exchanger. No loss was detected by moderator inventory.

Heat exchanger 4A (Foster Wheeler OH7832) had been in service since April 1956 and was replaced by a repaired heat exchanger (originally installed in R Area in 1953, and failed in April 1958).

Inspection of the cooling water expansion joint removed from System 4A showed pits 10 to 18 mils deep on one of the two stainless steel flange faces. These pits were similar to but less severe than those observed on the flange faces of the expansion joint removed from System 3B last month. The pits are presumably caused by chlorides present in the asbestos gasket material. Inspection of the flanges and gaskets was made by Engineering Assistance Section and is reported in detail in their report.

FUEL FAILURES

Two Mark VII-A fuel element failures occurred during the month. The first failure (SRP 47) was an SRP dingot slug (less than 3 ppm H_2) located in channel B, position X14,Y66. The reactor was shut down at 10:02 AM, April 1, at a reactor exposure of 33.61%. During the 24-hour period prior to the failure the maximum slug power in the channel was 0.00280%, the maximum sheath temperature was 142°C, and the maximum uranium metal temperature was 300°C. There had been two unscheduled shutdowns during the cycle prior to the failure. The ΔT increase for this failure was very gradual and was not detected until the flow monitor recorder showed a decrease of $\frac{1}{2}$% and the blanket gas activity began to increase.

The second failure (SRP 57) was an SRP ingot slug located in channel D, position X10,Y48. The reactor was shut down at 1:10 PM, April 26 at a reactor exposure of 12.65%. During the 12-hour period before the failure the maximum slug power in the channel was 0.00284%, the maximum sheath temperature was 132°C, and the maximum uranium metal temperature was 280°C. There had been four unscheduled shutdowns prior to the failure. The first three occurred in a 30-hour period immediately after initial critical and before nominal reactor power had been reached.

The first indication of the failure was a sharp increase on the blanket gas activity recorder and the low energy gamma monitor recorder. The location of the failed element was established by surveillance of the temperature and flow recorder charts.

Instrument response on the two failures is summarized in the following table.

Fuel Failure Instrument Response

Instrument	Date of Failure →	Response April 1	Response April 26
Activity Monitors, % increase			
Single-Channel Gamma Monitor		5	7
Delayed Neutron Monitor		0	14
Blanket Gas		34	>38
Overflow Tank Room		0	15
Seal Head Tank		36	40
Low Energy Gamma Monitor		21	>92
10-Minute Delay Line		8*	42
Temperature Monitor			
Channel ΔT Increase, °C		2.7	3.0
Flow Monitor			
Total Flow Decrease, %		1	3

* Initial response occurred just after shutdown.

Both failed elements were placed in failed fuel element containers and moved to positions adjacent to the D&E canal. Gaseous fission products were vented into the process room.

The Kanne chamber indicating stack exhaust activity showed an increase of a factor of 5 on the first failure and a factor of 100 on the second. However, prior to the connection of the vent line on the first failure bubbles from the failed fuel element container caused radiation levels of 60 mrad at one foot above the surface of the basin water. Radiation from the vent pipe on the first container increased to 1000 mrad at 3 inches and to 1500 mrad at 3 inches on the second. Water activity in the D&E canal increased by a factor of 14 on the first failure and by a factor of 4 on the second.

The failed elements have not been disassembled.

INSPECTION OF FAILED HEAT EXCHANGER

The heat exchanger (Foster Wheeler OH7835) that failed in System 3B on March 24 has been repaired. The leak occurred 17½ inches from the process water inlet end (midway between the inner tube sheet and the first baffle plate) of tube B6123 as viewed from the process water outlet end of the heat exchanger. This tube is in the bottom row of tubes but it is not one of the corner positions which has been susceptible to vibrational failures in the past.

The failed tube was located by a "Freon" leak detector while the shell side was pressurized with a mixture of "Freon" and nitrogen. The position of the leak in the tube was determined by passing a rod with a plunger on the end through the tube while using the "Freon" leak detector to indicate when the plunger passed the point of leak.

Visual inspection of the failed tube along its entire length revealed that the failure was not caused by vibration of the tube against a baffle or by vibration of debris against the tube as was the case in all of the past heat exchanger tube failures. Visual inspection and dye penetrant inspection of the bottom surface of the failed tube in the vicinity of the leak did not reveal a hole or crack in the tube. It is presumed therefore that the leak occurred in the top surface of the tube which cannot be inspected visually without removing the tube. The section of tube will be removed for metallurgical examination to determine the cause of failure.

K AREA

OPERATING SUMMARY

The reactor operated at a maximum power level of 85% during April. K-3 cycle was interrupted on April 1 when a shutdown was initiated to discharge a failed Mark VII-A fuel element (see Fuel Element Failure section). The return to power was delayed for replacement of the thrust seal in Bingham pump number 5.

A second Mark VII-A fuel element failure was detected at 17.6% power during the power ascension. The reactor was shut down by a "Very Low Flow" signal and the failed element was discharged.

The K-3 cycle exposure at the time of these failures was 28.3%. The total lost time at 100% nominal power, due to both shutdowns, was approximately 97 hours.

A scheduled shutdown terminated the K-3 cycle from a power level of 82% on April 20 at an exposure of 39.9%.

Power levels in the K-3 cycle during April were limited by the pump shaft break temperature limit. Typical and maximum operating conditions for the K-3 cycle are compared with operating limits in the following table.

Condition	Limit	Observed Values	
		Typical*	Maximum
Bulk Moderator, °C	101	96.5	98.0
Pump Shaft Break, °C	110	108.7	110.0
Reactor Effluent, °C	91	85.5	86.7
Channel or Quadrant Effluent, °C			
Mark VII-A	100	98.0	99.7
Mark VI-S	103	85.6	87.4
Mark I	105	98.5	99.9
Burnout Safety Factor			
Mark VII-A	1.4	3.15	2.48
Mark VI-S	1.4	3.11	3.11
Sheath Temperature, °C			
Mark VII-A	180**	140	148
Mark VI-S	160	121	122
Slug Power, %			
Mark VII-A	0.00366	0.00301	0.00312
Mark I	0.00224	0.00145	0.00152
Number of Effective Tubes	-	464	-
Power, %	-	82	-
Exposure, %	-	34.141	-
River Water, °C	-	17.1	-

* K-3 cycle, April 13.
** Or local boiling, 160°-175°C.

Initial critical for the K-4 cycle was attained on April 23 with the following control rod configuration:

	Gang		
	I	II	III
Full Rod, vu	2402	2402	2402
Partial Rod, vu	0833	0833	0770

All clusters contained a single 1.5 wt % partial rod.

Operation during the K-4 cycle was limited by the quadrant effluent temperature limit in the Mark I fuel elements and by the Mark VII-A fuel element sheath temperature limit. The Mark VII-A fuel element sheath temperature limit was changed from 180°C to 135°C for K-4 and subsequent cycles. (For details refer to the Engineering Studies section of this report.)

FLOW MONITOR

The transducers were recalibrated for 285" H_2O for K-4 cycle as a result of experience in K-3 cycle (see February report, DPSP 59-1-2, pp 81, 118-123). The average cold ΔP of the K-3 and K-4 charges were identical. However, because of a lower power coefficient for the K-4 charge, the average hot ΔP in K-4 is 270" H_2O, 15" less than the K-3 average. It is believed that the power coefficient is lower because the quadrant divider in the quatrefoil bottom end fitting was raised 1/8" for the K-4 cycle to prevent interference between the quadrant divider and the monitor pin (see February report, DPSP 59-1-2, page 85; and March report, DPSP 59-1-3, pp 69, 89-96).

Twenty-four deformed leaf springs were repaired during the K:3-4 shutdown. Since the transducers were calibrated for 285" H_2O and the ΔP was 270" H_2O, the VLF and HΔP switches were set for the K-4 cycle without difficulty.

FUEL ELEMENT FAILURES

Two Mark VII-A fuel failures occurred during April. Pertinent data on the failures are shown in the following table.

Fuel Element Failures During April (K-3 Cycle)

Failure No. →	48	50**
Date	4/1	4/4
Coordinates (X,Y)	08,54	15,21
Channel	B	A
Type Fuel Element	VII-A SRP Dingot	VII-A SRP Ingot
Reactor Exposure, %	28.3	28.3
Slug Power at Time of Failure, %*	0.00285	0.00063
Maximum Sheath Temperature, °C*	141.0	-
ΔT Increase in Offending Channel, °C	4.0	1.1
Flow Decrease, Segmental Recorder, %	2.0	4.5
Activity Increases, %		
Blanket Gas	13.6	-
Overflow Tank	0	-
Seal Head Tank	0	-
Ten-minute Delay Gamma	4.5	-
Delayed Neutron	0	-
Single-Channel Gamma	0	-
Low Energy Gamma	Not installed	Not installed

* Maximum for the channel.
** Failure occurred at low power level during ascension following first failure. No activity increases were observed.

A "Low Flow" alarm was received from the failed element position one hour prior to the shutdown for discharge of the failure on April 4. The reactor was being returned to power operation following the failure of April 1 and the power level was 17.6% at this time. A flow reduction of 3.5% existed as determined by comparison of measured and calculated ΔP's. During the next hour the "A" channel ΔT increased by 1°C and the flow decreased an additional 1%. Shutdown was initiated by a "Very Low Flow" alarm from the failed element.

A comparison of the observed ΔT's prior to the shutdown with the previous four-point temperature data is presented below for the failure that occurred on April 4.

Channel	4-Point Data (March 27) ΔT, °C	Auxiliary Temperature Recorder Data (April 4) ΔT, °C 9:00 PM	10:00 PM
A	53.6	13.6	14.9
B	53.2	12.2	12.6
C	51.0	12.0	12.3
D	50.7	11.9	12.2
Channel A minus D	5.7%	14.3%	22.1%

ASSEMBLY

The following table summarizes the component changes during the K:3-4 shutdown and the components recycled in K-4.

Components	Charged	Discharged	Recycled
Fuel Elements			
Mark VII-A			
Mark I			
Mark VI-S			
Total →			
Control Rods			
1.0%, 16-Slug			
3.5%, 16-Slug			
14.4S 16-Slug			
1.5%, 8-Slug			
1.5%, 6-Slug			
Miscellaneous			
4.9%, 10-Slug Shadow Rods			

* Four neptunium oxide rods (TA 1-742) are being irradiated in the other sparjets.

The neptunium oxide rods were charged during the shutdown for the fuel failure on April 1. Three of the assemblies contain six oxide slugs and one contains five oxide slugs. The assemblies replace the shadow rod and support tube normally used in the sparjets.

NUCLEAR INSTRUMENTATION

A modified high level flux monitor was installed in the shutdown circuit during the K:3-4 shutdown (TA 1-601). This modified instrument differs from the standard high level flux monitor in that the three safety circuits are actuated by separate magnetic amplifier relays. In the standard high level flux monitor only the fast shutdown circuit is activated by a magnetic amplifier relay, while the slow shutdown and rod reversal circuits are activated by mercury switches in the associated recorder. The modified instrument was connected to a spare ion chamber during the K-3 cycle and no spurious shutdown signals were received during that period.

The Westinghouse compensated ion chamber (position D-2) associated with high level flux monitor No. 3 responded erratically during the startup on April 4. Spurious alarms had been received on December 31, 1958 and on February 22, 1959 from this system, but these were attributed to malfunction of electronic amplifier tubes in the circuit. Subsequent circuit tests have identified the ion chamber as the cause of this erratic behavior. This ion chamber has been in service since October 1954. The D-2 chamber will be removed for inspection during the next scheduled charge-discharge shutdown.

PUMP SEAL FAILURE

The thrust seal on the Bingham pump No. 5 leaked excessively during hydraulic startup April 3. Attempts to reseat the seal to stop the leakage by alternating the pump speed with the DC and AC motors were unsuccessful. This seal had been in service for 20 months.

Replacement of the seal delayed reactor startup for 27 hours. The cause of failure was determined by inspection to be abrasive wear of the stellite and carbon seal faces by particulate material. Moderator turbidity is the most probable source of abrasive. The O-rings were in good condition.

DISASSEMBLY

The three neptunium oxide rods (TA 1-683) were disassembled during April with an underwater rod breaker. One slug from the first rod had jammed previously in a different rod breaker. Thirty-eight of the thirty-nine slugs were shipped to SRL. The breaker assembly with the stuck slug will be shipped to SRL for processing when a large cask is available.

MOTION MEASUREMENT PENDULUM

The thermal shield was heated and cooled throughout a range of 25°C to 80°C during the K:3-4 shutdown to determine if mechanical restrictions occurred between the pendulum assembly and external members at operating temperatures, and to obtain a curve of pendulum motion versus shield temperature for determining the zero power reading (or stress free reference) of the pendulum motion measurement instrument. The pendulum and beam data obtained during this test are shown in figure R-1.

Theoretically, as the shield is heated, the beam instrument should measure vertical expansion of the shield tank and the pendulum instrument should show no change in motion. Actually, as the shield was heated from 25°C to 53°C, the beam motion remained almost constant and the pendulum motion increased approximately 25 mils, thereby indicating that binding existed between the beam and some external member. The large change in both pendulum and beam motion between 53° and 62.5°C indicated that the beam restriction was momentarily relieved, however binding again occurred as the shield was heated from 73° to 77°C.

This test demonstrated that the K-Area pendulum is not operating properly and that it cannot be used to determine thermal shield staybolt stress until the binding is corrected. Shield heat load data will be used to control thermal shield stress until the pendulum is repaired.

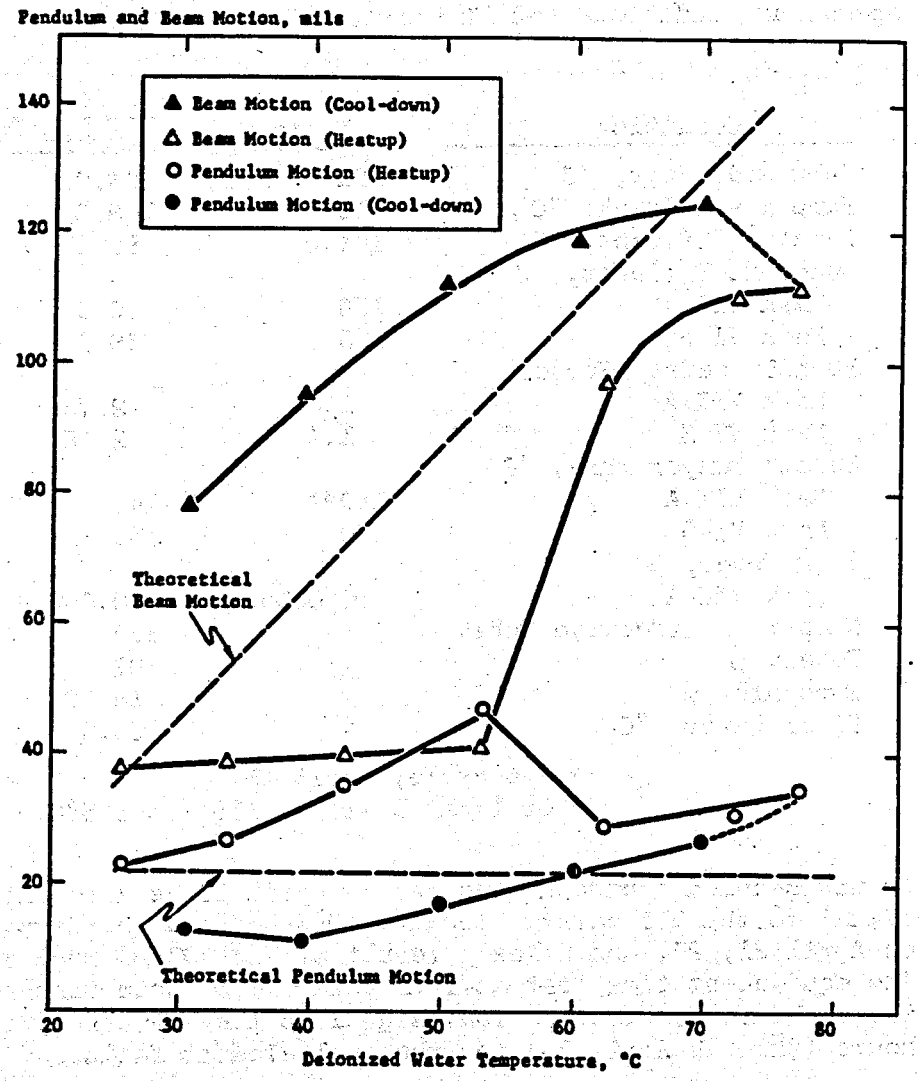


Figure R-1. Pendulum and Beam Motion versus Thermal Shield Deionized Water Temperature

C AREA

OPERATING SUMMARY

The reactor operated at a maximum power level of 87% during April. The operating conditions and limits at this power level were as follows.

Condition	Limit	Observed Values	
		Typical*	Maximum
Bulk Moderator, °C	101	94.9	98.0
Pump Shaft Break, °C	110	108.8	109.8
Reactor Effluent, °C	104.2	88.2	89.9
Quadrant Effluent, °C			
Mark VII-A	100	99.3	100.0
Mark VI-S	103	99.2	102.8
Burnout Safety Factor			
Mark VII-A	1.4	2.78	2.73
Mark VI-S	1.4	2.35	2.20
Sheath Temperature, °C			
Mark VII-A	180**	140	146
Mark VI-S	160	136	144
Slug Power, %			
Mark VII-A	0.00366	0.00294	0.00308
Number of Effective Tubes	-	529	-
Power, %	-	87	-
Exposure, %	-	18.58	-
River Water, °C	-	15.0	-

* C-4 cycle, April 15.
** Or local boiling, 115° to 165°C.

An unscheduled shutdown occurred on April 19 as a result of a spurious signal to the VLF safety circuit. Unscheduled shutdowns also occurred on April 21, 23, and 29 as a result of Mark VII-A fuel element failures. The equivalent time lost at full power from these unscheduled shutdowns was 41 hours for the spurious signal to the VLF safety circuit and 100 hours (through April 30) for the fuel element failures.

FUEL ELEMENT FAILURES

Mark VII-A fuel element failures occurred on April 21, 23, and 29. The first failure was detected after attaining a power level of 82% following the shutdown on April 19 from a spurious signal to the VLF safety circuit. The second failure was detected at a power level of 52% during power ascension following the discharge of the first failure. There was no evidence of the second failure (flow or temperature changes) at the time the reactor was shut down to discharge the first failure. The third failure was detected after approximately 4 days of operation following the second failure.

Information pertaining to the failures is given in the following table:

Fuel Element Failures, C-4 Cycle (April)

Failure Number →	54	56	58
Date of Failure	4/21	4/23	4/29
Time of Shutdown	6:05 PM	4:20 AM	2:04 PM
Coordinates (X,Y)	44,78	11,45	22,12
Channel	A	A	B
Type Fuel Slugs	SRP Ingot	SRP Ingot	SRP Ingot
Reactor Exposure, %	22.049	22.135	25.940
Max Slug Power in Channel,* %	0.00242	0.00245	0.00293
ΔT Increase, °C	4.0**	2.5†	1.5
Flow Decrease, %	1.5	2.0††	1.8
Initial Indicator	Flow and Temp Seg Recorders	VLF Safety Circuit‡	Activity Monitors
Activity Increases, %			
Single Channel Gamma Mon 1	0	0	0
Single Channel Gamma Mon 5	0	0	0
Delayed Neutron	0	0	21
Ten-Minute Delay Line	0	0	13.5
Blanket Gas	0	0	10.3
Overflow Tank Activity	0	0	0
Seal Head Tank Activity	0	0	14.9

* Maximum slug powers for failures 54 and 58 were calculated from data obtained at reactor powers of 82% and 79%, respectively, just prior to the failures. The slug power for failure 56 was calculated from data obtained on April 16 at a reactor power of 87%. No further slug power determinations were available for this position before the failure was detected on April 22 at a reactor power of 52%.

** The total increase in "A" channel ΔT was determined by comparison with channel ΔT's obtained on April 16.

† The "A" channel ΔT was 2.5°C higher than the next highest channel at a reactor power of 28% (the average of the other three channels was approximately 15.5°C). On April 16 the "A" channel ΔT had been the lowest of the four.

†† A 2% decrease in flow was indicated on the flow monitor segmental recorder chart. The ΔP measured after shutdown was 229" H_2O compared with 246" H_2O measured prior to the C-4 startup. This difference in ΔP is equivalent to a 3.5% flow decrease.

‡ A fast shutdown occurred at 12:15 AM from the VLF safety circuit. Criticality was then attained and the reactor operated at a power level of 28% for approximately 2 hours before the decision was made to shut down and discharge the failed element.

DISASSEMBLY

The status of disassembly operations on the seven failed fuel elements stored in the disassembly basin is presented in the following table. The first four failures occurred during the C-3 cycle and are stored in the monitor basin with the stop logs in place.

Disassembly of Failed Fuel Elements

SRP Failure Number	Status
42	The failed slug, the 14th slug from the top of the "D" channel, is stuck in a short section of the quatrefoil which has been placed in a closed container to prevent further contamination of the basin.
43	One slug is stuck in a short section of the quatrefoil which has been placed in a closed container. The slug position and channel are unknown, although the temperature monitor indicated the failure to be in "D" channel. A second slug, location unknown, had a dark streak down one side and was also placed in a slug container.
44	Has not been disassembled.
46	One or two slugs are stuck in each of two adjacent channels.
54	The failed element is stored in a failed fuel element container in the main basin.
56	The failed element is stored on a hanger in the main basin. There appears to be a small protuberance on the quatrefoil, with a slight depression immediately below it, approximately one third of the distance below the top of the channel containing the failure. The cause of the protuberance is unknown.
58	The failed element is stored in a failed fuel element container in the main basin.

The disassembly basin weir activity during April averaged 0.84×10^{-4} μc/cc. A maximum weir activity of 2.0×10^{-4} μc/cc occurred on April 23 following discharge of failure number 56.

The basin activity at various places near failure 56, which is stored on a hanger in the main basin, is given in the following table, for April 30.

Position of Sample	Activity, μc/cc
Surface of Water	0.96×10^{-4}
Top of Element	11.0×10^{-4}
Bottom of Element	1.2×10^{-4}

The monitor basin activity varied from a minimum of 0.15×10^{-2} μc/cc to a maximum of 3.4×10^{-2} μc/cc during April and averaged approximately 1.0×10^{-2} μc/cc. However, pumping to the seepage basin was intermittent because of the limited capacity of the seepage basin.

Engineering Studies

MARK VII-A FUEL FAILURES

Failure History

Thirteen Mark VII-A failures have occurred since last month (DPSP 59-1-3, pp 78-86). This brings the total number of Mark VII-A failures to 33. The history of each Mark VII-A failure is shown in the table on pages 80, 81, and 82. Similar data for previous failures (Mark I and Mark VII) are not included, but may be found in the January monthly report (DPSP 59-1-1, pp 93-96).

Failure Frequency. Failures per million slugs irradiated, and production loss from the various types of Mark VII-A slug failures are given in the table below. The failure rate for SRP ingot slugs in recent cycles is considerably higher than the over-all rate of 38.4.

Type Slug	No. of Failures	Failures per Million Slugs Irradiated*	Failures per Full Charge*	% Production Loss Due to Failures**
SRP Ingot (EP†)	5	72.0	2.7	9.0
††SRP Ingot (Std)	15	38.4	1.5	5.1
SC Ingot	1	6.0	0.23	0.8
††SRP Dingot	12	165.0	6.3	18.7
All Types	33	47.1	1.8	6.1
Recent SRP Ingot (R-5,P-6,L-4,K-3,C-4)	13	92	3.5	11.3

* Failure frequencies and failures per charge are based on the number of slugs irradiated through the R-5,P-6,L-5,K-4, and C-4 cycles except as noted in the last item under "Type Slug." (The P-6,L-5,K-4 and C-4 cycles are still in progress.)

** % Production Loss = $\frac{\text{Days Lost per Cycle Due to Failure}}{50 + \text{days lost}}$

Basis: 44 hours (1.83 days) average loss in production time per failure, 38,000 Mark VII-A slugs per full reactor charge, and 50 days average cycle length.

† "Early Production" slugs manufactured during March-June 1958, when high AlSi penetration rejects occurred during canning. ("Standard" SRP ingot slugs are those manufactured after June 1958.)

†† Based on the assumption that failures 36 and 44 are dingot slugs.

Types of Slugs Currently Being Irradiated. The types of slugs currently being irradiated are shown in the following table.

Cycle	Shutdown Date	Number of Slugs	
		SRP Ingot	SC Ingot
R-6	6/13/59	35,760	1,520
P-6	5/11/59	14,880	22,560
L-5	5/20/59	34,080	3,360
K-4	6/11/59	37,440	0
C-4	5/13/59	44,640	0

Slug Failure Examination. Metallographic examination of failures 37 and 39 (both dingots) was continued in the High Level Caves; however, the cause of failure has not been determined.

The following table shows the order in which the recent failures will be examined.

Examination Priority	Failure Number	Type Slug	Cycle	Status
1	41	SRP Ingot	P-5	Delivered to High Level Caves
2	49	SRP Ingot	P-6	Ready for Shipment to High Level Caves*
3	43	SRP Dingot	C-3	Ready for Shipment to High Level Caves*
4	42	SRP Ingot	C-3	Ready for Shipment to High Level Caves*
5	53	SRP Ingot	P-6	Ready for Shipment to High Level Caves*
6	56	SRP Ingot	C-4	Slug not Unloaded from Quatrefoil

* These slugs are isolated in the respective disassembly basins and cannot be shipped until the activity in the isolated basins has been reduced. Isolated basin activity should be decreased sufficiently to allow shipment of P-Area slugs by May 8 and C-Area slugs by May 15.

History of Mark VII-A Slug Failures

Failure No.	Cycle	Date	Type Slug*	Power, %** Reactor	Power, %** Per Foot†	Exposure, % Reactor	Exposure, % Per Foot†	Gang No.	X,Y Coordinates	Position	Days Irradiated
27	P-15	6/14/58	SRP▲	69.6	3570	19.5	1175	III	33,75	A††	28
28	P-15	6/21/58	SRP	70.4	3350	23.0	1295	II	13,39	A-11	35
29†	P-15	6/23/58	SRP	60	2640	23.3	1256	II	15,57	B††	37
30	R-2	10/13/58	SRP	68	3210	15.4	644	I	33,51	D††	25
31	R-2	10/21/58	SRP♦	76	3860	19.8	850	II	18,30	D††	32
32	R-2	10/24/58	SRP♦	76	3730	21.2	860	II	14,42	B††	35
33	R-2	10/31/58	SRP	80	3800	24.8	1010	I	20,60	C-12	41
34	R-3	12/31/58	SRP♦	86	2860	34.2	1140	III	42,24	D-18	45
35	P-4	1/1/59	SC	84	4200	22.2	1230	I	26,30	D-16	28
36	R-4	2/6/59	SRP††	88	4800	26.1	1420	III	37,21	C††	34
37†	R-4	2/9/59	SRP♦	88	4800	26.6	1450	I	31,57	B-14	37
38	R-4	2/21/59	SRP♦	87	4500	35.4	1820	III	34,18	A††	49
39	R-4	2/26/59	SRP♦	85	4800	38.3	2140	III	41,21	B-16	54
40	L-4	2/27/59	SRP▲	88	4500	10.4	535	I	28,54	D††	14
41	P-5	3/6/59	SRP	86	4060	37.7	1780	I	37,39	A-15	46
42	C-3	3/12/59	SRP	90	4100	33.4	1560	III	12,72	D-14	40
43†	C-3	3/14/59	SRP♦	90	3900	33.4	1480	I	31,51	D††	40
44†	C-3	3/14/59	SRP♦	90	3840	33.4	1450	I	30,42	Unk	40
45	K-3	3/18/59	SRP♦	87	3950	18.9	860	III	46,66	B††	23
46	C-3	3/20/59	SRP	90	3560	38.5	1490	II	30,18	B††	46
47	L-4	4/1/59	SRP♦	84	4000	33.6	1610	III	14,66	B††	47
48	K-3	4/1/59	SRP♦	85	3920	28.3	1310	III	08,54	B††	39
49	P-6	4/3/59	SRP	85	3440	19.9	817	I	34,54	D-15	25
50	K-3	4/4/59	SRP	85	4040	28.4	1350	III	15,21	A††	39
51	P-6	4/16/59	SRP	83	3650	28.7	1240	I	35,51	D††	38
52	R-5	4/18/59	SRP	82	4200	34.7	1780	III	39,21	A††	44
53	P-6	4/21/59	SRP	81	3620	31.6	1410	I	30,42	C-17	43
54	C-4	4/21/59	SRP	82	3450	22.1	928	BZ	44,78	A††	28
55	R-5	4/21/59	SRP	78	4260	35.5	2100	III	19,21	D††	47
56†	C-4	4/23/59	SRP	82	3480	22.1	940	II	11,45	A††	30
57	L-5	4/26/59	SRP	80	4050	12.7	643	III	10,48	D††	22
58	C-4	4/29/59	SRP	79	4180	25.9	1390	III	22,12	B††	37
59	P-6	4/30/59	SRP	78	3890	37.7	1890	II	15,45	D††	52

* ▲ indicates recanned slug; ♦ indicates dingot slug.

** Reactor power and slug power before shutdown to discharge failure, or before first failure in event that a second failure occurred during startup after discharging a failed element. If failure was detected during a scram recovery, the operating power before the scram is given.

† Multiply all values by 10^{-6}. NOTE: Values are based on maximum flux in channel if failed slug elevation is unknown.

†† Failed slug elevation unknown.

† Failure was detected during startup after having shut down to discharge a failed element.

†† Not known whether failed slug is an ingot or dingot slug.

History of Mark VII-A Slug Failures, contd

Failure No.	Q-foil Flow, %	Temperature, °C* Metal	Temperature, °C* Sheath	No. of Prior Shutdowns	Recorded Flow Decrease, %**	Recorded ΔT Increase, °C**	Type of Failure	Date of Manufacture
27	1.74	273	132	0	0.75	5.0	Axial[1]	3/28/58
28	1.74	258	120	1	0.5	5.0	▶Axial[1]	4/14/58
29	1.74	251†	123†	2	1.5	9.0	Side & Axial[1]	4/11/58
30	1.83	240	116	0	2.25	3.6	Unknown	5/21/58-5/23/58
31	1.83	277	130	1	1.5	3.0	Unknown	8/13/58
32	1.83	267	124	2	3.5	5.4	Unknown	8/14/58
33	1.83	271	122	3	1.0	3.0	Unknown	6/9/58-6/27/58
34	1.81	250	130	4	1.25	3.3	Side	9/8/58-9/11/58
35	1.76	310	141	1	1.0	3.7	Axial	10/57
36	1.80	330	139	0;3[a]	0.75	1.0	Unknown	9/5/58-9/11/58
37	1.80	315†	134†	0;4[a]	2.0	2.0	Side	9/18/58-9/23/58
38	1.80	290	128	2	1.0	3.5	Unknown	9/9/58-9/11/58
39	1.80	265	131	3	1.5	3.6	Side[2]	9/3/58
40	1.83	320	128	0	1.25	3.5	Unknown	12/11/58
41	1.84	301	138	2	0	4.0	Side[3]	10/28/58-10/30/58
42	1.69	307	134	0	1.5	3.4	Unknown	12/8/58
43	1.69	279†	125†	1	2.0	1.5	Unknown	1/5/59-1/6/59
44	1.69	278†	125†	1	2.0	0	Unknown	12/1/58-1/6/59
45	1.80	315	130	1	3.0	10.4	Unknown	12/31/58-1/5/59
46	1.69	310	133	2	2.0	7.0	Unknown	11/26/58-11/28/58
47	1.83	293	133	2	1.2	2.7	Unknown	12/15/58
48	1.79	277	121	2	1.25	4.7	Unknown	12/30/58
49	1.81	270	124	0	0.5	4.3	Axial[4]	1/29/59-2/9/59
50	1.79	282	123	3	4.5	7.3	Unknown	1/7/59-1/9/59
51	1.81	273	120	1	0.75	1.5	Unknown	1/9/59-3/9/59
52	1.81	303	129	1;1[a]	1.75	2.7	Unknown	1/30/59
53	1.81	290	135	2	1.0	2.8	Side[5]	2/11/59-2/12/59
54	1.75	265	117	0;1[a]	1.5	4.0	Unknown	2/20/59
55	1.81	313	133	2;1[a]	2.0	4.4	Unknown	1/20/59
56	1.75	265†	118†	1;1[a]	2.0	2.5	Unknown	2/17/59-2/18/59
57	1.83	290	122	1;3[a]	3.0	6.1	Unknown	2/18/59-3/3/59
58	1.75	310	138	2;1[a]	1.8	1.5	Unknown	2/9/59-2/10/59
59	1.81	283	120	3	0	5.3	Unknown	2/11/59-2/12/59

* Metal and sheath temperatures were calculated without hot-spot factors, from operating conditions which prevailed immediately before the failure, unless otherwise noted. Temperatures are based on maximum flux in channel if failed slug elevation is unknown.

** Recorded immediately before shutdown.

† Failure was detected during startup after having shut down to discharge a failed element; temperatures are based on operating conditions at full power before the first failure.

[a] Number of shutdowns and number of scrams, respectively.

[1] Metallographic examination showed that the failure was due to a cladding defect, probably caused by AlSi penetration.

[2] Can split along ribs; crack between ribs.

[3] Appeared to be a side failure (can split between ribs).

[4] Axial channel appeared to be plugged.

[5] Appeared to be a side failure.

▶ This failure was incorrectly listed as a side failure in last month's report.

History of Mark VII-A Slug Failures, contd

Failure No.	% Increase in Signal on Activity Monitors								
	Blanket Gas	Seal Head Tank	Single-Channel γ	Overflow Tank	Low Energy γ	Delayed Neutron	10-Min Delay γ	Gas Chromatograph	Other Monitors
27	15	15	-	-	-	-	-	-	-
28	10	-	-	-	-	-	-	-	-
29	-	-	-	-	-	-	6	-	-
30	120	50	-	37	-	-	-	-	-
31	10	-	-	-	88	-	-	-	21[a]
32	33	0	-	-	19	-	-	-	-
33	13	6	-	-	22	-	-	-	-
34	254	180	12	8	90	-	-	-	77[b]
35	0	6.5	0	4.2	11	0	-	-	0[cd]
36	20	11	0	0	4	0	5	60	-
37	109	48	7;9*	12	-	0	14	164	-
38	45	15	0	0	20	0	12	44	-
39	37	7	-	-	-	-	4	100	-
40	No increase detectable because of high background activity.								-
41	No activity increases were detected.								
42	No activity increases were detected.								
43	21	580	-	-	-	325	160	-	-
44	21	580	-	-	-	325	160	-	-
45	17	10	10	0	-	0	7	-	-
46	36	106	-	7	-	15	13	-	-
47	34	36	5.2	0	21	0	7.8	-	-
48	14	0	0	0	-	-	5	-	350[e]
49	4.8	3.4	4	-	-	0	2.5	-	-
50	No increase detectable because of high background activity.								10[e]
51	25	14	0	6	-	0	3.9	-	-
52	275	138	0	-	87	0	-	710	112[f]
53	15.7	19.2	0	0	11.8	0	10.8	-	9.7[d]
54	0	0	0	0	-	0	-	-	-
55	50	0	0	-	40	0	-	60	-
56	0	0	0	0	-	0	-	-	-
57	8	40	6.5;3.0*	15.4	>92	14.3	42	-	-
58	10.3	14.9	0	0	-	21	13.5	-	-
59	24.2	14.2	3.4	2.5	16.9	0	3.1	-	5.9[d]

* Monitor No. 1 and monitor No. 5, respectively.

[a] Delayed neutron monitor on Purification line.

[b] Special blanket gas beta monitor under test (TA 1-433).

[c] Thyac.

[d] Cyclone gamma monitor.

[e] Crane drip pan monitor.

[f] Planchet.

Effect of Exposure on SRP Ingot Failure Rate

The Mark VII-A-SRP ingot slug failures have been analyzed to determine the effect of exposure on the failure rate. Figure RE-1 is a compilation of all the Mark VII-A charges that have been irradiated and shows the number of charges that have operated at each exposure level. Figure RE-2 shows the number of equivalent charges of SRP ingot slugs alone, that have operated at each exposure level. An equivalent charge is defined as 38,000 slugs, which is the approximate number of Mark VII-A slugs in an R,P,L, or K natural uranium charge. (For example, a charge that contained 19,000 SRP ingot slugs had 0.5 equivalent charges of SRP ingot slugs.) The number of SRP ingot failures is also shown in figure RE-2 at the reactor exposure at which failure occurred. Figure RE-3 shows the ingot failure frequency per unit of exposure at each exposure level. Figure RE-4 presents the data of figure RE-3 in integrated form. Figure RE-5 presents the number of failures per year per reactor as a function of cycle exposure for SRP ingot slugs.

Figure RE-5 indicates that the number of failures per year increases with exposure and that there is a sharp increase at about 34% exposure. The majority of the failure data upon which the curves are based were obtained at winter power levels during the past winter.

Axial Position of Failures

All of the ten Mark VII-A failures whose axial positions are known were located in the lower half of the fuel element, indicating a significant effect of particular reactor operating conditions on the likelihood of failure. Sheath temperature and bulk coolant temperature are two probably important variables.

Of about 168,000 Mark VII-A-SC slugs irradiated thus far, only one failure has been experienced, for a failure rate of six per million slugs irradiated.

Because this rate is so much lower than that for SRP slugs, it is planned to irradiate SC slugs in all Mark VII-A charges (beginning with R-7 and K-5) in axial positions 12 through 17 of Zone I elements in an effort to decrease the Mark VII-A failure rate.

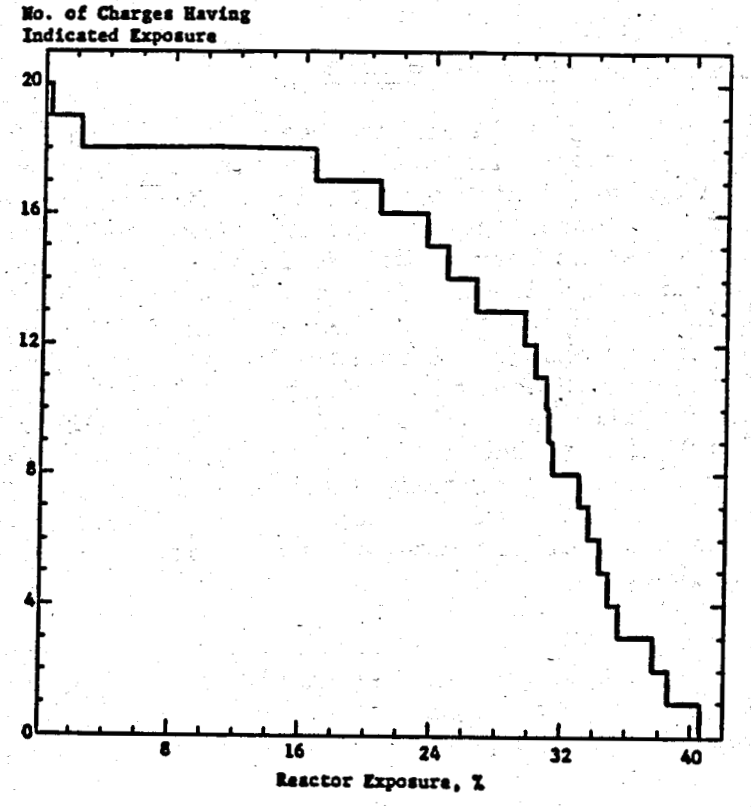


Figure RE-1. Number of Mark VII-A Charges versus Exposure (as of April 27, 1959)

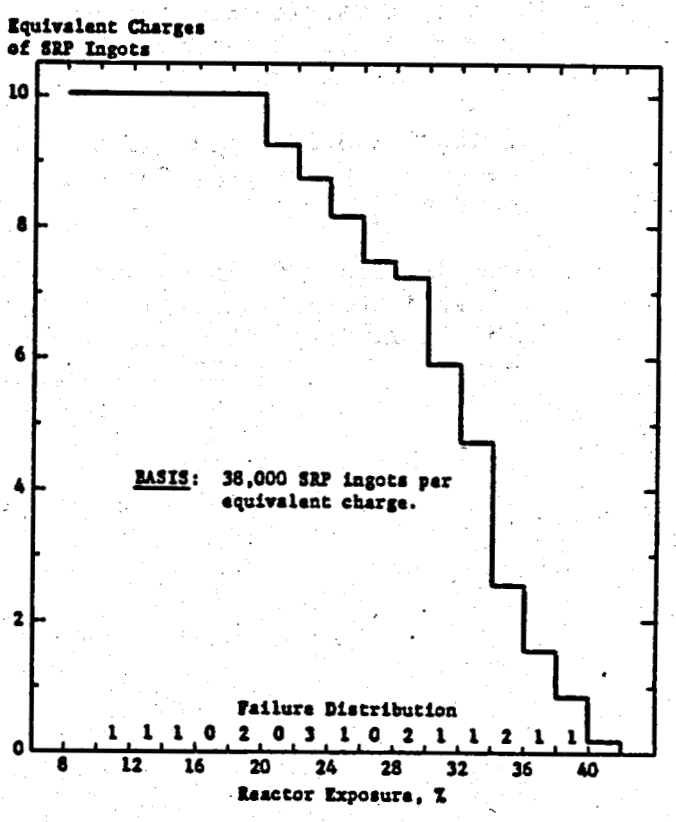


Figure RE-2. Number of Equivalent Charges of SRP Ingots versus Exposure

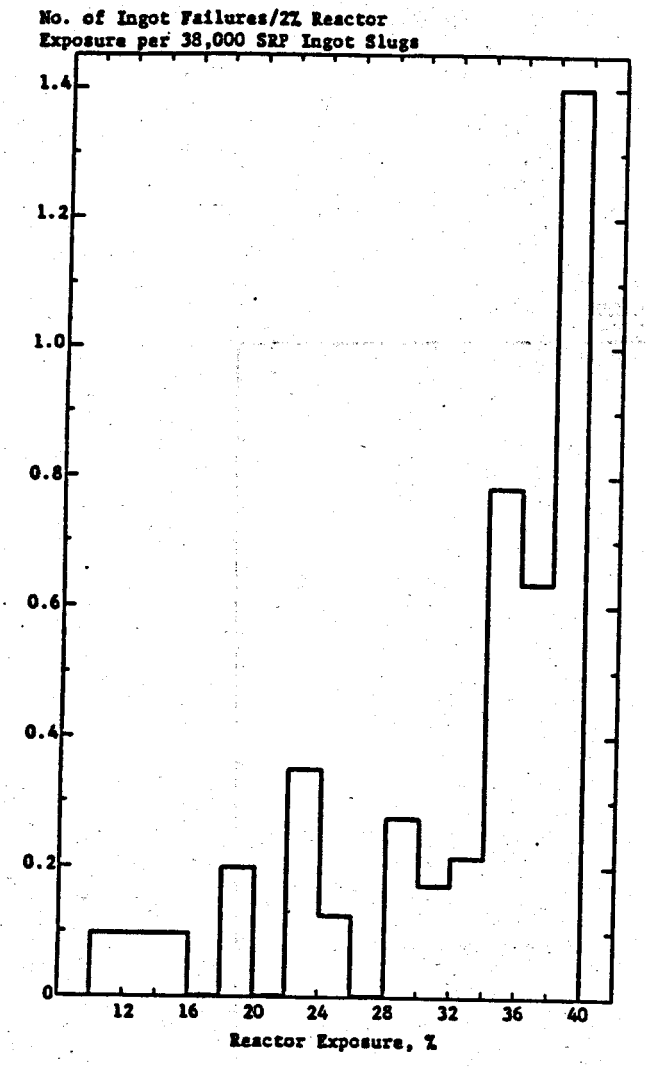


Figure RE-3. Failures/2% Reactor Exposure per 38,000 SRP Ingots versus Exposure (as of April 27, 1959)

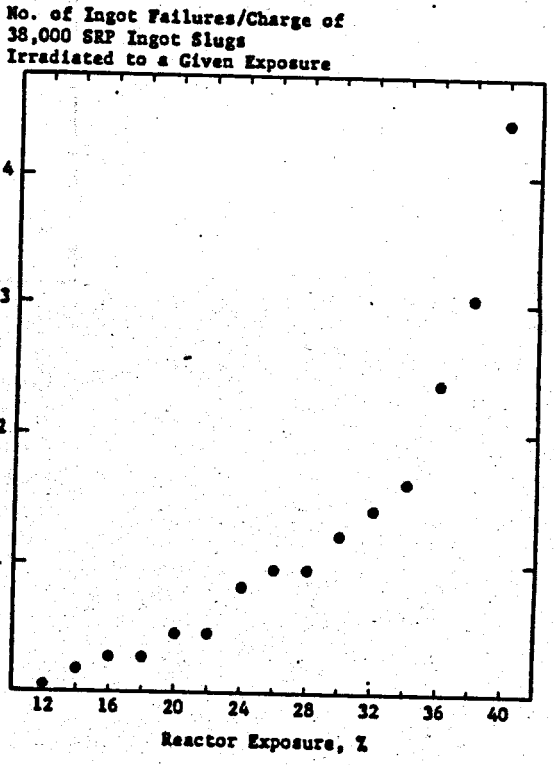


Figure RE-4. Ingot Failures per Charge versus Exposure (as of April 27, 1959)

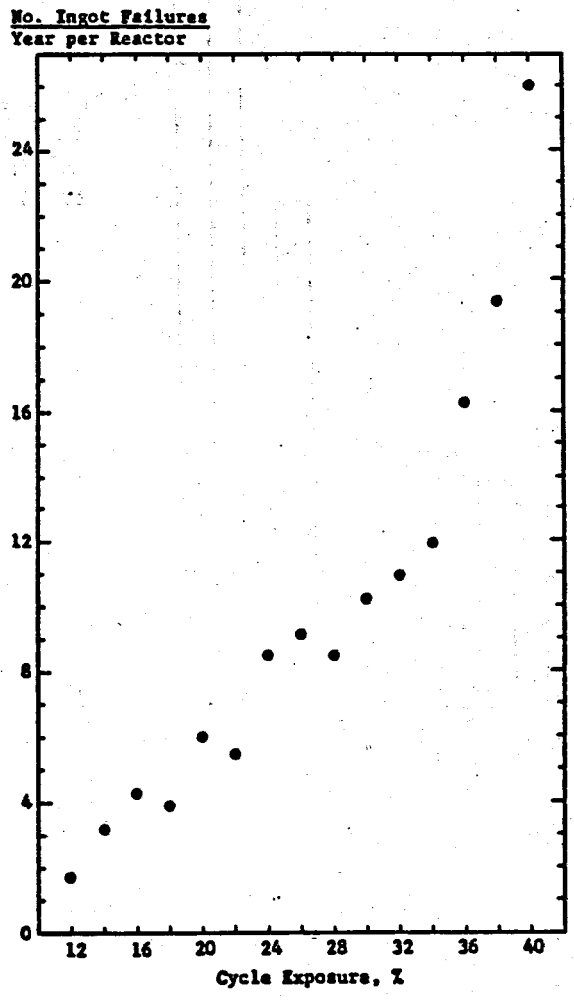


Figure RE-5. Annual Rate of Ingot Failures per Reactor versus Cycle Exposure. [For operation with full charges (38,000) of SRP ingot slugs.]

Reactor Operating Conditions

Of the ten failed Mark VII-A slugs whose axial position is known, all have occurred in the bottom half of the slug column, the region of highest sheath temperature. Annular and axial sheath temperatures of these slugs are shown in the following table.

Slug Sheath Temperature and Location of Known Mark VII-A Failures

Failure Number	Slug Position	Flow Zone	Slug Type	Sheath Temp, °C	
				Annular	Axial
28	11	I	SRP Ingot	118	120
33	12	I	SRP Ingot	121	122
34	18	I	SRP Dingot	130	126
35	16	I	SC Ingot	141	140
37	14	I	SRP Dingot	134	134
39	16	I	SRP Dingot	131	127
41	15	I	SRP Ingot	138	136
42	14	I	SRP Ingot	134	134
49	15	I	SRP Ingot	123	124
53	17	I	SRP Ingot	135	133

Six of these failures have occurred in slug positions 14, 15 and 16, the positions having maximum sheath temperature. No failures are known to have occurred below the 18th or above the 11th slug, where sheath temperatures are 8° to 15°C below the maximum values (see figure RE-6). Examination of irradiated slugs has shown the largest amount of pitting on slugs in positions 12 through 17.

The Mark VII slugs operated at lower sheath temperatures, heat fluxes and coolant velocities than the Mark VII-A slugs, as shown in the following table. The four Mark VII-SRP slug failures occurred at axial positions 4, 6, 13, and 14, indicating no strong effect of local reactor operating conditions on the failure rate.

Comparative Operating Conditions of Various Fuel Types

(No Hot-Spot Factors)

Fuel Type (Mark)	Max Observed Sheath Temp, °C				Coolant Velocity, ft/sec		Max Heat Flux, pcu/(ft²)(hr)	
	Summer Operation		Winter Operation					
	Axial	Annular	Axial	Annular	Axial	Annular	Axial	Annular
VII-A								
VII								
I								
VIII-A								
VI								

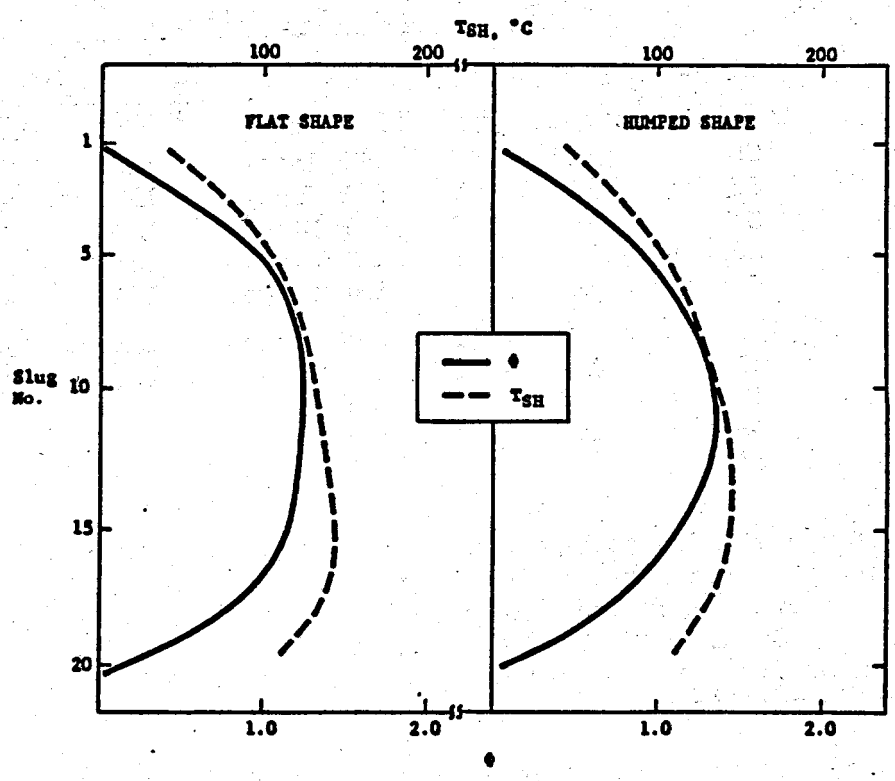


Figure RE-6. Mark VII-A Axial Sheath Temperatures at 82.5% Power and 20°C River Temperature

River Temp, °C	Sheath Temp, °C		
	Max	10th Slug	19th Slug
10	146	135	137
20	143	132	134
30	140	129	132

River Temp, °C	Sheath Temp, °C		
	Max	10th Slug	19th Slug
10	148	141	130
20	144	137	128
30	140	134	127

The R-6 charge is being operated with a rooftopped axial flux shape (rooftop ratio = 1.20) to test the effectiveness of this method of operation on reducing sheath temperatures and possibly fuel failures. With normally flat shapes, a reduction of about 5°C is expected in the maximum sheath temperature. The reduction will be smaller for the humped axial flux shapes encountered during the period of minimum reactivity (see figure RE-7).

Figure RE-8 presents observed maximum sheath temperatures for the five reactors during the past month. The temperatures have been normalized to a river temperature of 20°C. To reduce the sheath temperature by 10°C at a constant river temperature, the required power reduction would be about 8% (code).

Radial Distribution of Failures

Figure RE-9 is a histogram showing Mark VII-A-SRP failure rates, plotted as failures per million slugs irradiated in each hexagonal ring outward from the central septifoil. The plot shows a peak near the center of Gang I, indicating that the failure rate at that location is approximately four times the over-all average failure rate for SRP slugs. The calculated failure rates assume present flow-zoning patterns of Mark VII-A charges and radially uniform irradiation of SC slugs[1] (which were subtracted from the total number of Mark VII-A slugs irradiated in each hexagonal ring), and are hence subject to some inaccuracy; however, it is not expected that the errors significantly affect the general trend of the plot.

Several facts concerning the channels and the ΔT's of the elements containing failed slugs have been noted (based on failures 27 through 61):

- Of 32 channels containing failed slugs (the channel locations of failures 43, 44, and 50 are questionable), 20 (or 62%) had ΔT's prior to failure detection which were the highest ΔT's in the element (a random distribution would have given a value of 25%). Four-point temperature data were reviewed from the beginning of the cycles to the time of the failure for failures 52 through 61. The data disclosed that the normal ΔT's of the failed channels for seven of these ten failures were the highest in the element at the beginning of the cycle as well as at the time of failure.
- Of the 24 channels surrounding each septifoil, 8 are closer to the septifoil than the remaining 16 (see figure RE-10).

[1] Such is not the case for the SC plugs in P-6 and L-5, for example, but was true for earlier P charges in which the majority of SC slugs have been irradiated.

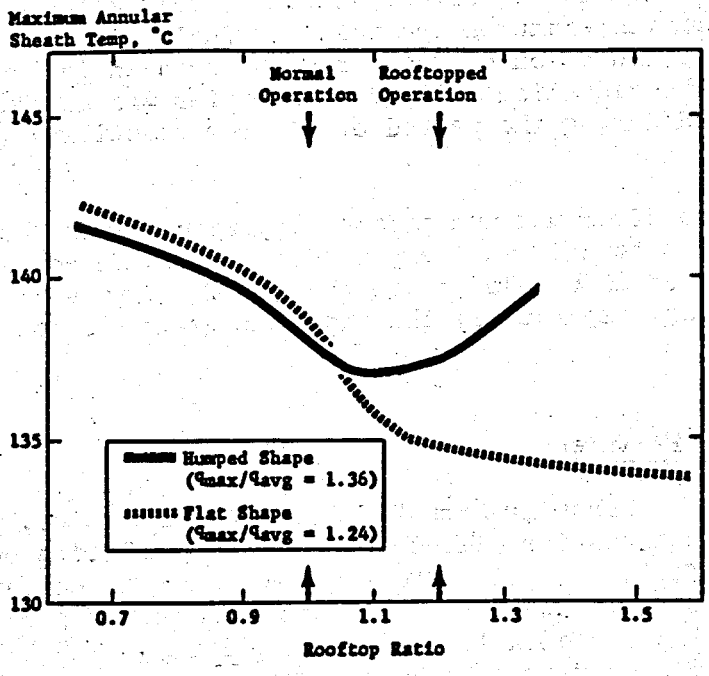


Figure RE-7. Variation of Sheath Temperature With Rooftop Ratio at 83% Power and 18°C River Temperature

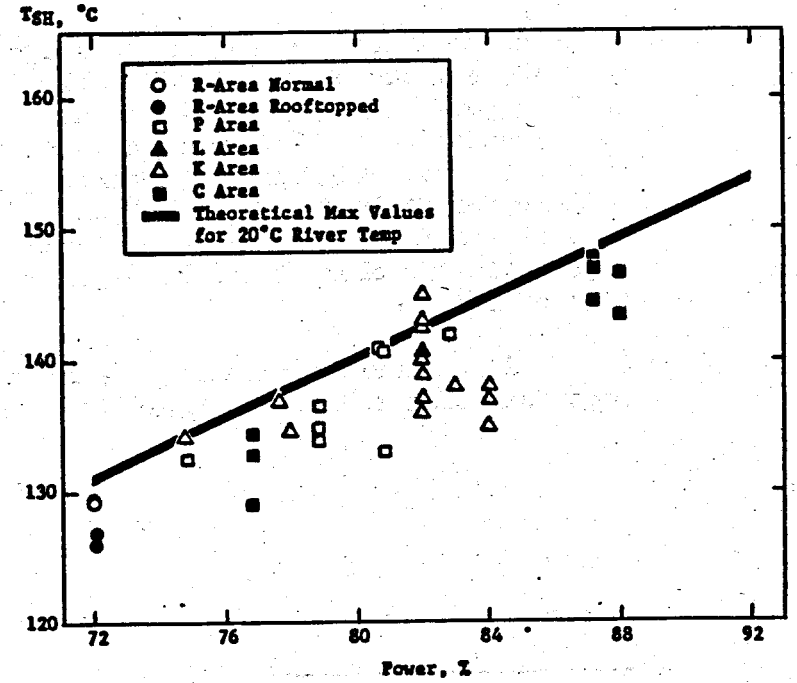


Figure RE-8. Observed Mark VII-A Sheath Temperatures, April 1959

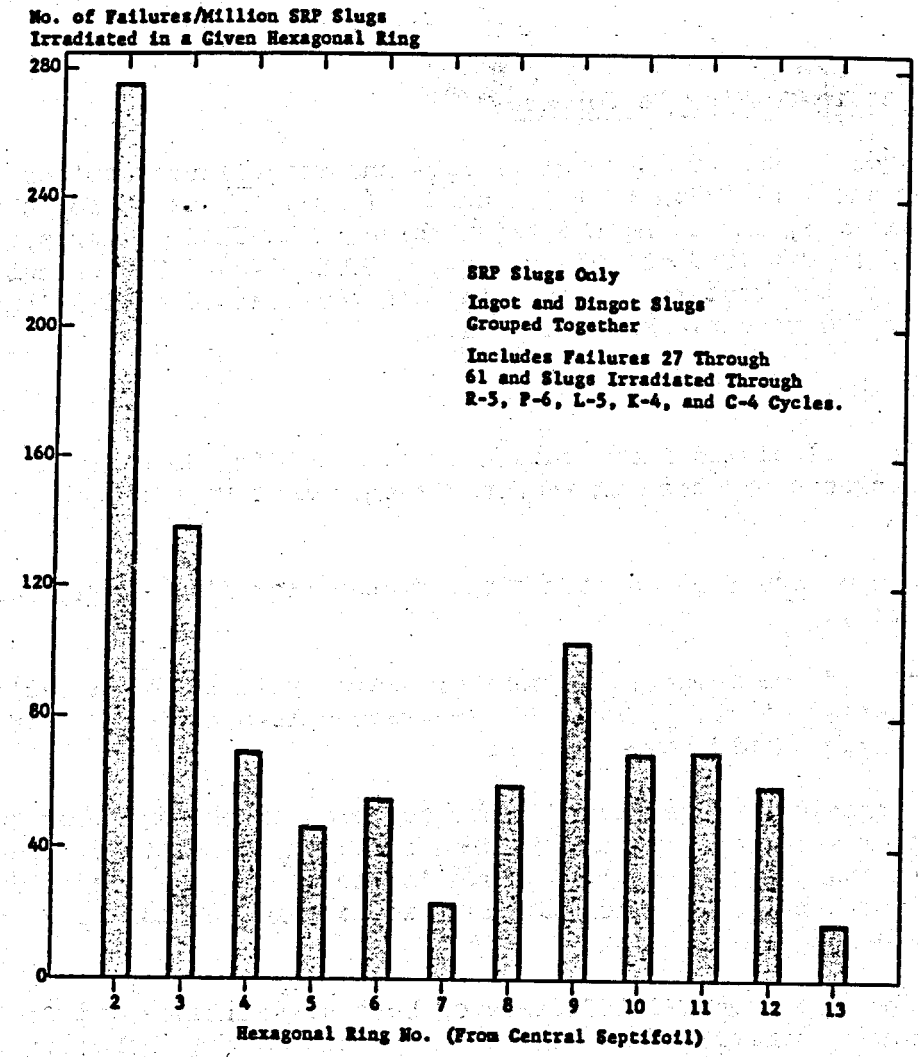
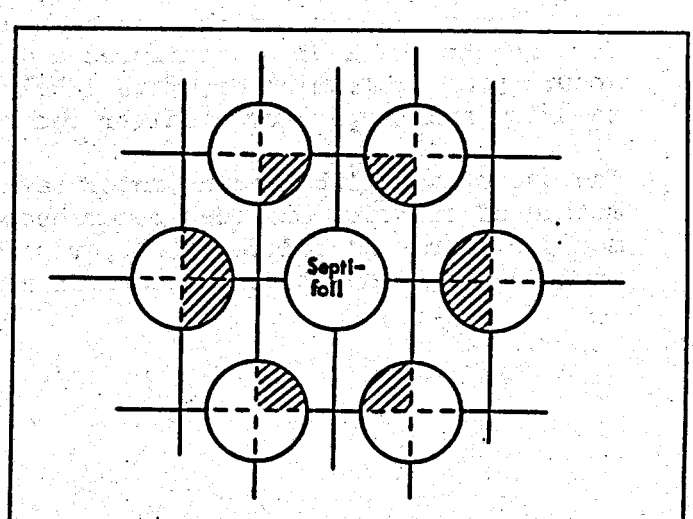


Figure RE-9. Radial Distribution of Mark VII-A Failure Rates



NOTE: Of 32 channels containing failures, 22 (or 69%) were located in one of the shaded channels above (33% would be expected from random phenomena). Of these 22, 19 (or 86%) had ΔT's which were normally the highest ΔT's in the element prior to detection of failure.

Figure RE-10. Failures in Channels Adjacent to a Septifoil

Flow and Temperature Monitor Performance

The table on pages 94 and 95 summarizes flow and temperature monitor performance for the Mark VII-A fuel failures. (A similar table for the first 29 SRP failures, including the first three Mark VII-A failures, was published in the August 1958 monthly report, DPSP 58-1-8). The calculations are based on the most recent CMX flow and temperature monitoring data and reactor thermocouple element data.

The table shows that:

- In general, the calculated flow reduction corresponding to a given measured ΔT increase has been in reasonable agreement with the observed flow reduction.
- The performance of the flow monitor alarm system (low flow alarm) has been satisfactory.
- The performance of the temperature monitor alarm system (VH and VVH) has not been satisfactory. Only 3 VVH rod reversals occurred out of a total of 9 which should have occurred.

Calculations to predict whether the failure occurred in the axial or the annular subchannel were made, using measured flow and temperature changes and known monitoring performance. As shown in the table, the predicted location (axial or annular) was confirmed by actual examination in 9 out of the 10 failures examined to date.

Figure RE-11 shows the combined influence of both surveillance and the flow and temperature monitor alarm and scram systems on the probability of shutting down the reactor for a fuel failure by the time a given increase in fuel channel ΔT has occurred. The curves show that the protection afforded by both surveillance and the flow and temperature monitor alarm and scram systems has been apparently increasing during the more recent (Mark VII-A) failures. This could be a result of increasing accuracy of programing or, more likely, of more rigorous surveillance of the flow and temperature monitor recorder charts.

The use of a null-balancing meter rather than the present oscilloscope method of trimming has been recommended as a means of improving the monitor alarm and rod reversal performance. A schedule for conversion to 2400-pot temperature monitors in all areas is being developed.

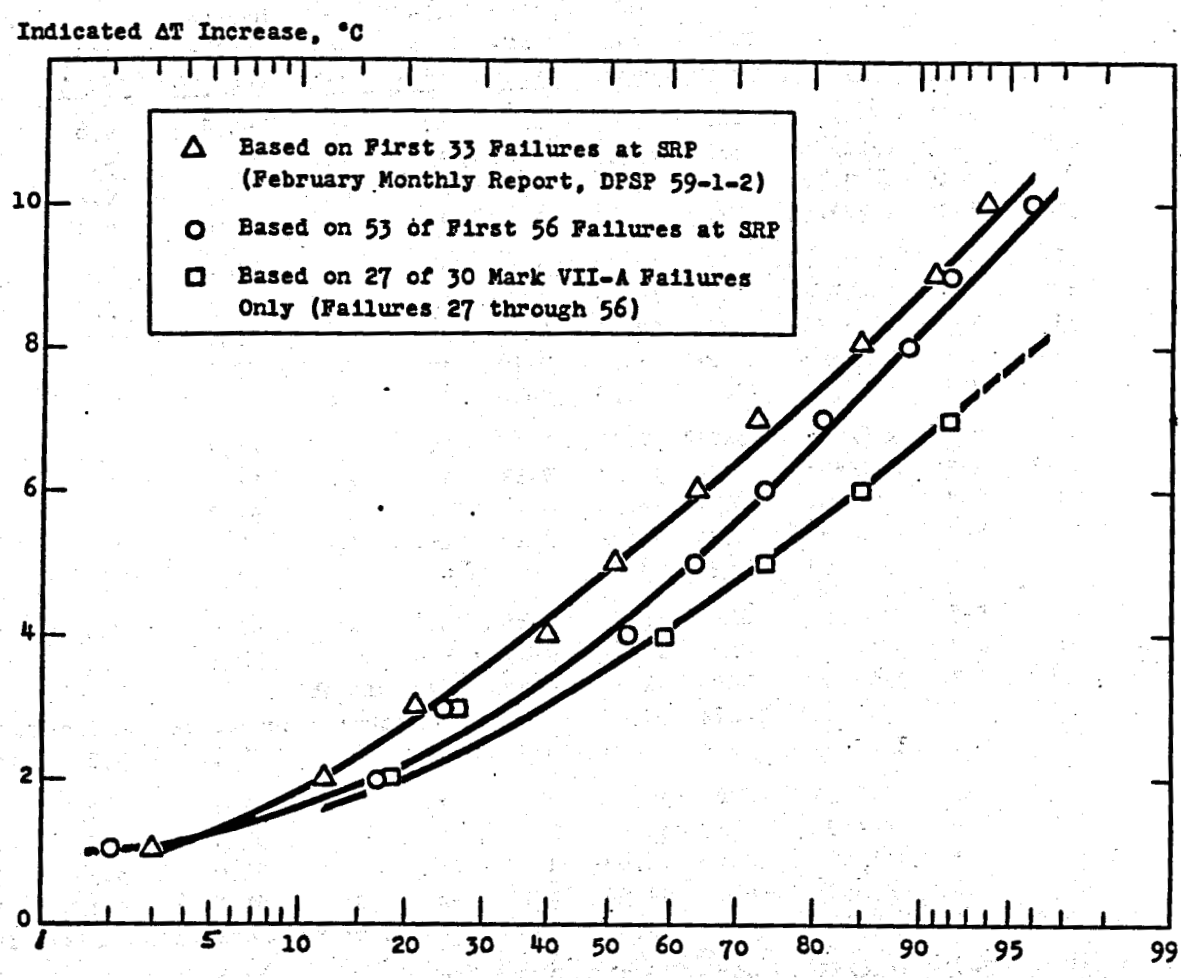


Figure RE-11. Probability of Shutting Down Reactor for a Given ΔT Increase

Mark VII-A Fuel Failure Summary - Monitoring

Failure No.	Cycle	Date	Type Slug*	Slug Location: X,Y Coordinates	Slug Location: Channel & Position†	Slug Location: Subchannel Calculated††	Slug Location: Subchannel Actual	ΔT in Channel Before Failure, °C	ΔT Increase, °C: Indicated	ΔT Increase, °C: Above Highest Normal ΔT†
27	P-15	6/14/58	SRP	33,75	A	Ax	Ax	53.8	4.8	4.8
28	P-15	6/21/58	SRP	13,39	A-11	Ax	Ax	49.9	5.4	5.4
29††	P-15	6/23/58	SRP	15,57	B	Ax	Ax	40.3	9.2	6.6
30	R-2	10/13/58	SRP	33,51	D	Ax		43.0	6.4	6.4
31	R-2	10/21/58	SRP♦	18,30	D	An		50.9	4.9	4.3
32	R-2	10/24/58	SRP♦	14,42	B	An		53.5	5.4	5.4
33	R-2	10/31/58	SRP	20,60	C-12	Ax		51.5	3.0	1.9
34	R-3	12/31/58	SRP♦	42,24	D-18	An	An	57.6	3.1	3.1
35	P-4	1/1/59	SC	26,30	D-16	An	Ax	62.5	3.7	3.7
36	R-4	2/6/59	SRP**	37,21	C	An		61.3	1.7	1.7
37††	R-4	2/9/59	SRP♦	31,57	B-14	An	An	51.6	2.0	2.0
38	R-4	2/21/59	SRP♦	34,18	A	Ax		59.7	3.6	3.3
39	R-4	2/26/59	SRP♦	41,21	B-16	An	An	54.6	3.4	0
40	L-4	2/27/59	SRP	28,54	D	Ax		59.2	3.5	3.5
41	P-5	3/6/59	SRP	37,39	A-15	An	An	60.6	4.0	4.0
42	C-3	3/12/59	SRP	12,72	D-14	Ax		58.3	3.3	3.3
43††	C-3	3/14/59	SRP♦	31,51	Channel questionable.					
44††	C-3	3/14/59	SRP♦	30,42	Channel unknown; recorder out of index.					
45	K-3	3/18/59	SRP♦	46,66	B	Ax		51.3	9.2	7.9
46	C-3	3/20/59	SRP	30,18	B	Ax		54.2	7.0	6.0
47	L-4	4/1/59	SRP♦	14,66	B	An		57.4	1.7	1.7
48	K-3	4/1/59	SRP♦	08,54	B	Ax		52.0	5.3	5.3
49	P-6	4/3/59	SRP	34,54	D-15	Ax	Ax	51.3	4.3	3.9
50††	K-3	4/4/59	SRP	15,21	Channel unknown; recorder chart illegible.					
51	P-6	4/16/59	SRP	35,51	D	An		54.4	1.5	0
52	R-5	4/18/59	SRP	39,21	A	An		56.3	1.6	1.6
53	P-6	4/21/59	SRP	30,42	C-17	An	An	57.0	3.4	3.4
54††	C-4	4/21/59	SRP	44,78	A	Ax		44.7	3.7	2.5
55	R-5	4/21/59	SRP	19,21	D	Ax		55.4	4.5	4.5
56††	C-4	4/23/59	SRP	11,45	A	Ax		15.0	2.8	2.2
57	L-5	4/26/59	SRP	10,48	D	An		51.1	2.8	2.8

* ♦ indicates dingot slug; other slugs are ingots.

** Not known whether this is a dingot or ingot slug.

† Position 1 is the top slug in the channel; position 20 is the bottom slug. If slug position is not given, it has not yet been determined.

†† Ax = axial subchannel; An = uniformly plugged annular subchannel. The "calculated" subchannel for each failure is the one which would have given most nearly the flow and temperature changes actually observed.

† The values given represent the ΔT increase above the highest normal ΔT for the fuel element. When a ΔT increase is 3°C, a VH alarm should have fired; when it is 4°C, a VVH alarm should have fired.

†† Power ascension in progress.

Mark VII-A Fuel Failure Summary - Monitoring, contd

Failure No.	Flow Decrease, % From ΔT*	From Recorders	From Measured ΔP**	Alarm Response† Flow LF	Flow VLF	Flow HΔP	Temp VH	Temp VVH	Calculated Real ΔT Changes and Maximum Effluent Temperatures, °C, based on:†† Recorded ΔΔT and CMX Monitoring Data† ΔΔT	Eff	Recorded Flow Reduction ΔΔT	Eff	Measured ΔP Change ΔΔT	Eff
27	1.7	1	4.6[a]	0	0	0	0	[0]	10.3	100.2	5.5	95.4	-	-
28	2.2	½	1.5	0	0	0	0	[0]	12.8	98.9	2.3	88.4	8.1	94.2
29	5.4	1½	4.0	●	0	0	0	0[b]	45.0	120.0	6.8	81.8	25.0	100.0
30	3.1	2¼	2.5	0	0	0	●	0[c]	18.4	102.0	11.5	95.1	13.5	97.1
31	3.0	1½	2.5	0	0	0	●	[0]	13.3	110.2	5.8	102.7	10.7	107.6
32	3.1	3½	3.25	●	0	0	●	●	14.6	112.2	17.2	114.8	15.6	113.2
33	1.2	1	-	0	0	0	●	0	6.2	91.5	5.3	90.6	-	-
34	1.7	1¼	1.1	●	0	0	0	0	7.8	104.9	5.7	102.8	5.0	102.1
35	1.9	1	1.9	0	0	0	0	0	9.6	112.9	5.0	108.3	9.6	112.9
36	0.9	¾	0.5	0	0	0	0	0	4.1	103.5	3.3	102.7	2.2	103.6
37	1.3	2	2.9	0	0	0	0	0	5.0	91.3	7.7	94.0	11.2	97.5
38	1.2	1	-	0	0	0	●	0	7.5	97.3	6.3	96.1	-	-
39	2.0	1½	4.4	●	0	0	●	0	8.8	102.6	6.7	100.5	19.5	113.3
40	1.2	1¼	0	0	0	0	0	0	7.5	96.4	8.2	97.1	0	88.9
41	2.1	½[d]	1.8	0	0	0	0	[0]	10.3	111.6	2.1	103.4	8.6	109.9
42	1.0	1	-	●	0	0	0	0	7.3	97.2	7.3	97.2	-	-
43		2	~9.4	0	0	0	0	0						
44		2	~6.4	0	0	0	0	0						
45	4.0	3	3.1	●	0	0	0	[0]	31.8	112.6	20.3	101.1	21.2	102.0
46	3.0	2	1.9	●	0	0	0	[0]	21.0	106.0	12.2	97.2	11.4	96.4
47	1.0	1.2	-	0	0	0	0	0	4.3	103.4	5.3	104.4	-	-
48	2.4	1¼	-	0	0	0	●	●	14.6	96.8	5.9	88.1	-	-
49	1.6	e	0.5	0	0	0	0	0	9.4	93.7	0	84.3	2.3	86.6
50				●	●	0	0	0						
51	0.9	¾	1.9	0	0	0	0	0	3.7	101.1	2.9	100.3	9.3	106.7
52	0.9	1½	-	0	0	0	0	0	3.9	102.7	7.1	105.9	-	-
53	1.9	1	4.0	0	0	0	0	0	8.7	110.0	4.5	105.8	18.0	119.3
54	1.9	1½	2.0	0	0	0	0	0	9.5	89.2	7.1	86.8	10.1	89.8
55	1.6	2	-	0	0	0	●	●	9.8	98.6	12.2	101.0	-	-
56	4.2	1½	3.5	0	●	0	0	0	10.0	49.5	2.4	41.9	7.7	47.2
57	2.1	3	3.1	●	0	0	0	0	8.4	101.0	12.9	105.5	13.4	106.0

* Theoretical flow reductions were obtained from CMX temperature monitoring efficiency data and the ΔT increase observed on the temperature monitor. The values given assume that the flow reduction occurred in the channels indicated in the "Channel and Position" and the "Subchannel, Calculated" columns. In calculating these flow decreases, R-4 (SRP) and P-5 (SC) thermocouple element data were used to determine the true axial and annular ΔT's and flows prior to the failure. It was also assumed that there was no redistribution of fuel heat when flow was reduced in one of the subchannels.

** Dash (-) indicates no data available.

† ● indicates that the alarm fired; 0 indicates that the alarm did not fire. Boxed zero indicates that the alarm should have fired but did not.

†† The values given assume that the flow reduction occurred entirely in the channels indicated in the "Channel and Position" and the "Subchannel, Calculated" columns.

‡ For the affected subchannel.

a The flow reduction based on ΔP's was not correct because gradual indicated flow reductions (ΔP changes) had occurred on many of the fuel elements in the same region prior to the failure.

b Power ascension was in progress and the percent power meter was advanced 5% (approximately 3°C increase in alarm point).

c Percent power meter advanced twice after first VH alarm (total equivalent increase of 3° to 4°C in alarm point).

d Recorded flow decreased ½% during 16-hour period preceding failure detection, then behaved erratically after failure detection while ΔP measurements were being made.

● Recorded flow decrease masked by 2% recorder noise.

MARK VII-A REDESIGN

Temperature measurements from Mark VII-A special thermocouple tubes have indicated that at the beginning of the cycle the annular ΔT's are about 12 to 15% higher than axial ΔT's. (See February monthly report, DPSP 59-1-2, p 100.) Reactor safety would be improved if annular and axial ΔT's could be equalized.

A larger Mark VII-A quatrefoil has been designed which has a 0.020-inch-larger channel ID, resulting in equal annular and axial ΔT's. The resulting quatrefoil extrusion is larger and the upper four inches will be rolled down to fit current design tube adapters. The current design tube adapter (ST MDX-4-1241, rev 9) must be used because it is the largest which will fit into the semipermanent sleeve, and it is undesirable to decrease the thin wall section of the adapter (from 0.030 to 0.015 inch) to accommodate the larger extrusion.

The "rolled down" section of the extrusion will not restrict channel flows because the top of the slug column will be 2 to 5 inches below the "rolled down" section (see figure RE-12). Also the slugs should be just as easy, if not easier, to load or unload because the slug column centerline and crossflow sleeve centerline will be closer. All dimensional tolerances on the large Mark VII-A extrusion will be the same as the current Mark VII-A tolerances.

The new Mark VII-A design will result in an 11% increase in annular flow, a 3.2% decrease in axial flow, and a 3.4% increase in total plenum flow. These flow changes will equalize axial and annular ΔT's.

Proposed Mark VII-A extrusion dimensional changes are tabulated in the following table and are given on drawing ST MDX-5-2430, rev 3.

Table of Proposed Dimension Changes

Item	Inches		
	Current	New	Change
Channel ID	1.450	1.470	+0.020
Channel OD	1.530	1.540	+0.010
Wall Thickness	0.040	0.035	-0.005
Common Wall Thickness	0.040	0.035	-0.005
Quatrefoil Extrusion OD	3.638	3.668	+0.030
Rib Circle Eccentricity	0.008	0.009	+0.001
Rib Circle ID	1.235	1.235	0
Slug	Same in All Respects		

The proposed conversion schedule from current type Mark VII-A elements to the new type Mark VII-A elements is given in the table on page 98.

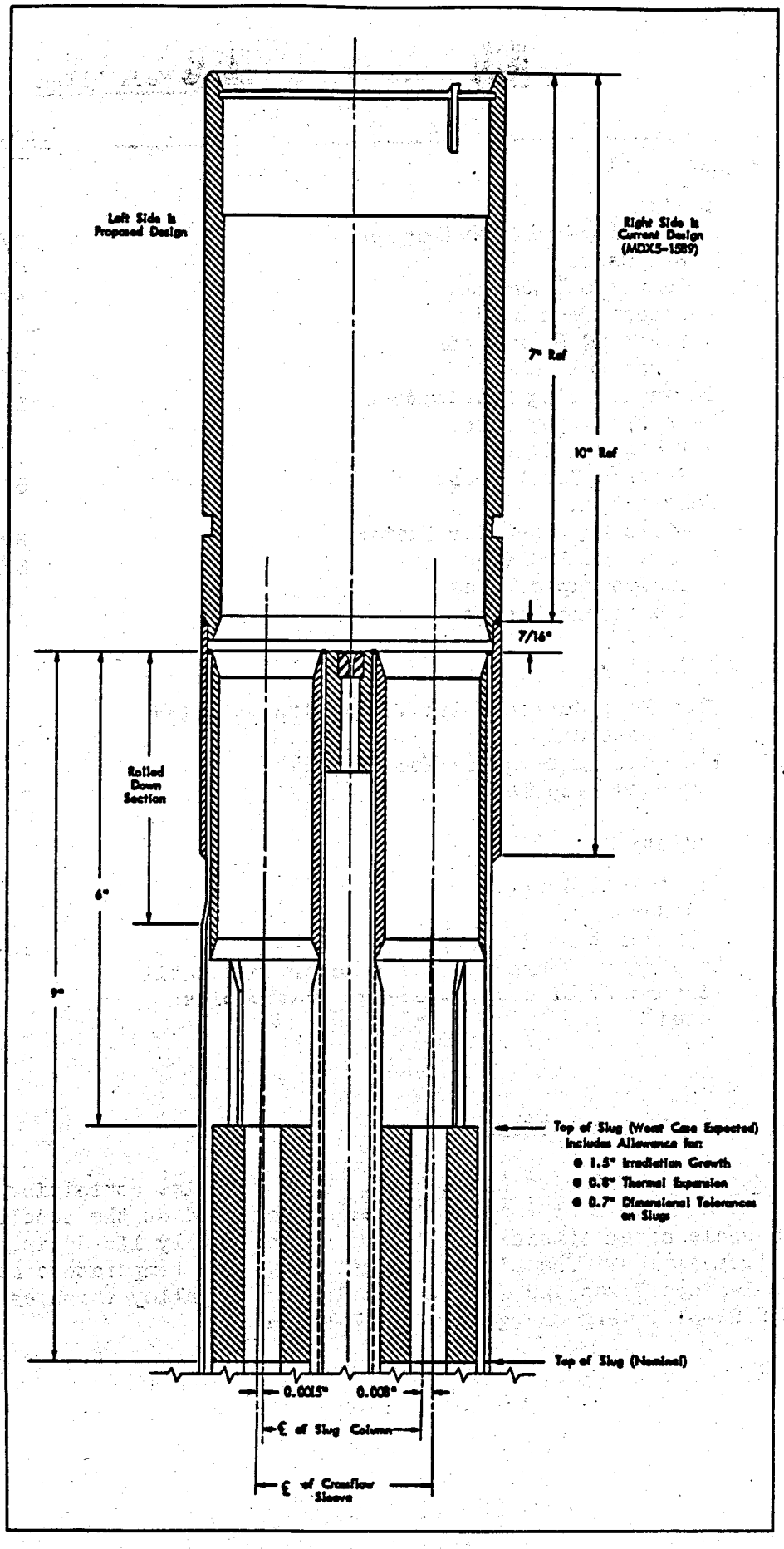


Figure RE-12. Mark VII-A Redesign

Conversion Schedule for Large Mark VII-A

Item	Completion Date
● Quatrefoil	
Design	
Reducing-roll Development	3/2 to 4/6
Extrusion	2/24 to 4/6
Quatrefoil Assembly	4/20 to 5/15
Purchase Requisition	
First 20 Extrusions	4/7
First Full Charge	5/18
TA Authorizing Development	5/13
Development by Alcoa	
Tools and Dies	4/13 to 5/22
Process Development	5/25 to 7/3
Fabrication	
First 20 Tubes for Testing	By 7/10
First Full Charge	8/10 to 8/28
Thermocouple Tubes	7/20 to 8/28
Subsequent Charge	9/1 →
● CMX Testing	
Flow Test Current Mark VII-A with Modified "B" Boattail	5/15 to 6/15
Flow and Temperature Monitoring	7/17 to 8/10
Extended Flow Tests	7/17 to 11/30
● Irradiation	
First Full Charge	
Assembly	9/14 to 10/9
Irradiation P-10 Cycle	10/13
Subsequent Charges - one reactor only until inventory of current design quatrefoils used.	

R-5 SPECIAL IRRADIATIONS

Target Alloy Thimbles (TA 1-654). Five thimbles containing target alloy (as described in DPSP 58-1700) were discharged at the conclusion of the R-5 cycle after attaining a GVR of approximately 115 during three cycles of irradiation. The highest target slug core temperature measured during the R-5 cycle was 129°C. Six similar target alloy thimbles (described in DPST 59-316) were charged to the R-6 cycle.

Insulated Slugs (TA 1-687). Eight insulated slug assemblies were irradiated in the R-5 cycle, four for the first cycle and four for the third cycle. All insulated slugs were recycled into the R-6 cycle. Typical operating data are shown in the following table.

Insulated Slugs

Position X	Position Y	Assembly Number	Slug Elevation, inches below reactor midplane	Reactor Power, %	Maximum Slug Power, %/ft	Average BOSF
51	65	3	64	86	0.00107	2.3
45	79	6	68	86	.00095	2.6
04	58	8	68	86	.00095	2.6
25	89	9	71	86	.00098	2.5
43	11	11	63	86	.00077	3.2
08	74	12	69	86	.00085	2.9
25	05	13	69	86	.00088	2.8
02	44	14	63	86	.00075	3.3
Predicted by TA		-	-	86	0.00086	2.8

Mark V-A (TA 1-707). Three Mark V-A assemblies, each having twelve subchannel effluent temperature thermocouples, were charged to the R-5 cycle and irradiated for one cycle. Typical operating data are shown in the following table. No further Mark V-A irradiations are planned before R-8.

Mark V-A Fuel Element

Position →	X27,Y45	X27,Y51	X26,Y48	Predicted By TA
TFEN Number	60	61	62	-
Reactor Power	86	86	86	86
Plenum Inlet Temp, °C	37.0	37.0	37.0	-
Average ΔT, °C	54.1	53.7	53.0	57.3
Average Effluent Temp, °C	91.1	90.7	90.0	94.3
Average Effluent Temp, °C Measured by Subchannel Thermocouples	89.5	89.1	89.0	-
Assembly Power, %	0.200	0.198	0.196	0.206
Assembly Power Relative to Adjacent Mark VII-A	1.16	1.16	1.16	1.12

Four-Foot Extrusion Clad Mark V (TA 1-723). One 4-foot, Mark V was charged to R-5 and irradiated for one cycle. The natural uranium fuel tube with extrusion-clad aluminum cladding was irradiated to determine the feasibility of extrusion-cladding long tubular natural uranium fuel elements. Typical operating data are shown in the following table.

Mark V Fuel Elements

Type Element →	Zr-Clad Natural Uranium Fuel Tube		4-Foot Long Extrusion-Clad Mark V	
Position	X14,Y66		X12,Y78	
TA Number	1-735		1-723	
Reactor Power, %	86		86	
Plenum Inlet Temp, °C	38.3		38.3	
	Actual	Predicted By TA	Actual	Predicted By TA
Average ΔT, °C	53.6	58.5	15.0*	23.9*
Average Effluent Temp, °C	91.9	96.8	53.3	62.2
Assembly Power, %	0.136	0.146	0.0408	0.0666
Assembly Power Relative to Adjacent Mark VII-A	0.75	0.80	0.35	0.52

* Includes 2.3°C temperature rise due to heating from bulk moderator.

Zirconium-Clad Natural Uranium Fuel Tube (TA 1-735). One zirconium-clad natural uranium fuel tube was charged to the R-5 cycle to determine if this type of fuel tube has greater dimensional stability than U-2% Zr fuel tubes with Zircaloy cladding. Operating data are shown in the above table. The irradiation of this fuel tube is continuing in R-6.

Special Irradiations. A list of the special irradiations in R-6 is given in the following table.

Special Irradiations in R-6

TA Number	Type Element	Element No.	Position X	Position Y	Remarks
1-687	Insulated Slugs	IS-3	51	65	Charged to R-3
		6	45	79	Charged to R-3
		8	04	58	Charged to R-3
		9	25	89	Charged to R-3
		11	43	11	Charged to R-5
		12	08	74	Charged to R-5
		13	25	05	Charged to R-5
		14	02	44	Charged to R-5
1-735	Zr Clad Mark V	TFEN 64	14	66	Charged to R-5
1-744	Power Reactor Fuel Tube	SPR-6	30	48	Charged to R-6
1-654	Target Alloy Thimbles	LA-7	21	41	Five 14.4S Slugs, charged to R-6
		8	23	65	Five 14.4S Slugs, charged to R-6
		9	27	29	Five 14.4S slugs, charged to R-6
		10	28	62	Five 14.4S Slugs, charged to R-6
		11	33	59	Four 11.7S Slugs, charged to R-6
		12	35	41	Five 7.8S slugs, charged to R-6

PERFORATED END FITTINGS

A test group of 104 Zone I, Mark VII-A elements with perforated end fittings was charged in L-5 cycle. The perforated end fitting, drilled with many small orifice holes rather than a few large holes, was designed in an attempt to eliminate the difference in power coefficient of flow from fuel element to fuel element in the reactor. Figure RE-13 shows the location of the perforated end fitting elements in L-5 cycle.

Figure RE-14 compares the power coefficients of flow for the perforated end fittings with the power coefficients of standard Mark VII-A elements in similar positions in L-5 cycle. The data indicate that negligible benefit was obtained from the perforated end fittings in reducing the power coefficient spread.

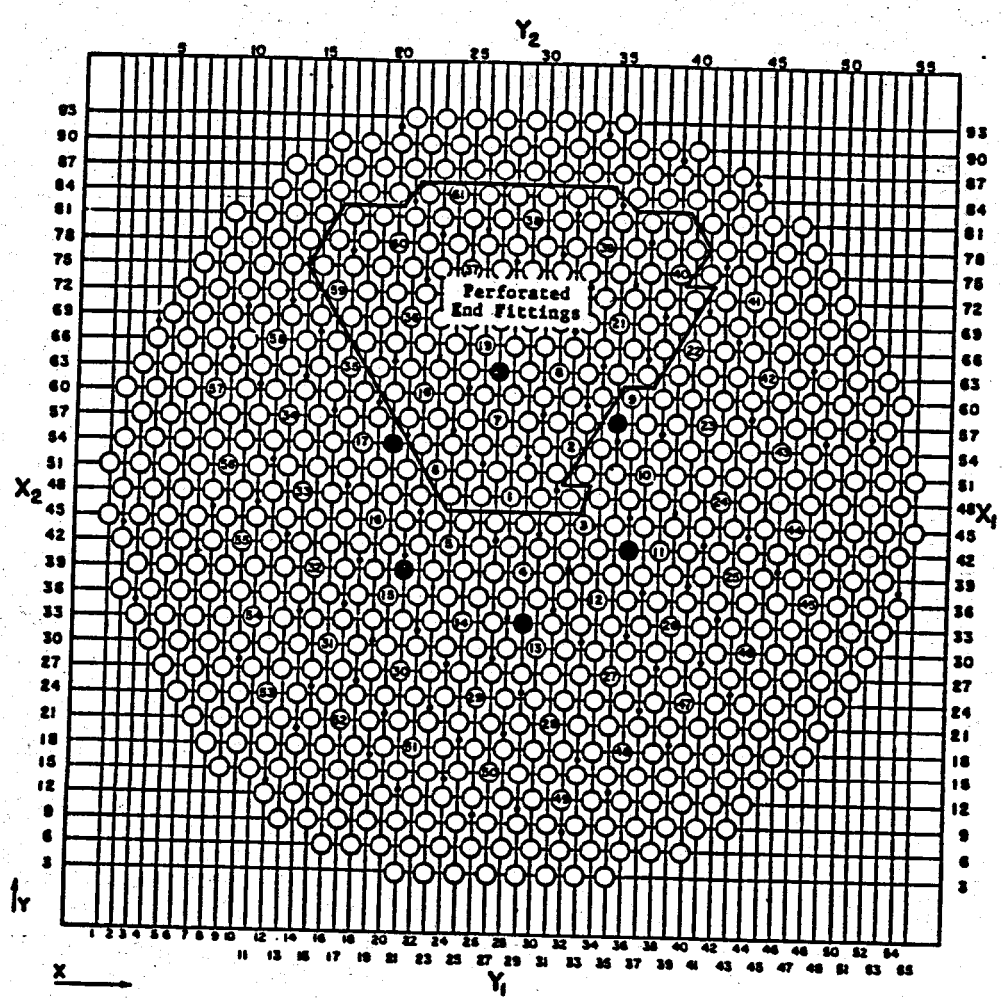


Figure RE-13. Mark VII-A Perforated End Fitting Locations, L-5 Cycle

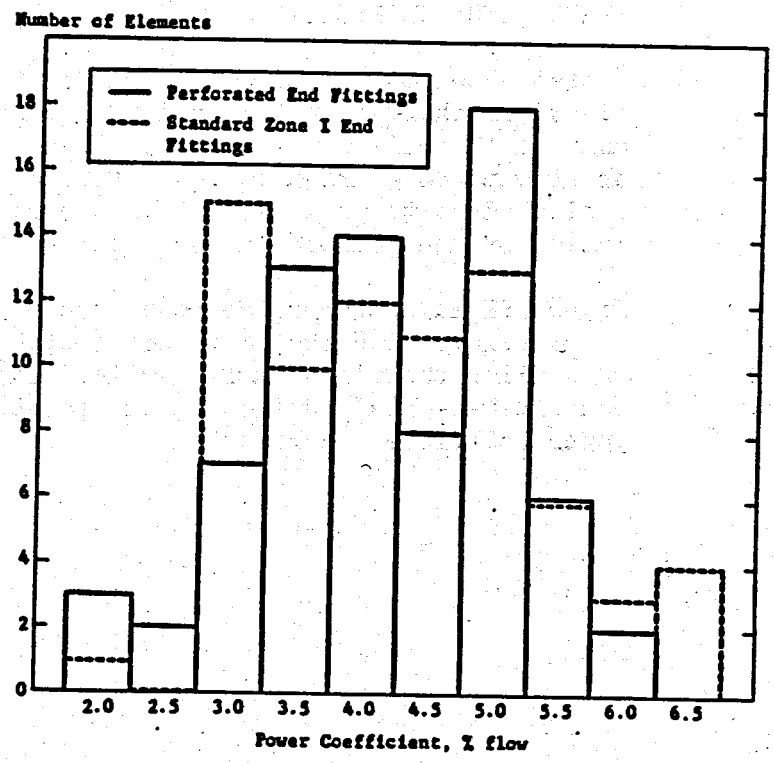


Figure RE-14. Zone I Power Coefficients of Flow, L-5 Cycle

The fact that no large benefit was obtained from the perforated end fittings in reducing the power coefficient spread may be the result of a large mating distance effect for these elements. In addition to the increase in end fitting ΔP as a result of a real flow increase from cold to hot reactor conditions, the end fitting ΔP is also affected by mating distance changes (distance from the bottom of the fuel element to the shoulder of the monitor pin) as a result of thermal expansion of the reactor components. Figure RE-15 shows the mating distance curve for the perforated end fitting elements as determined by CMX and shows the relatively large effect of mating distance on the power coefficient for these elements. CMX has been requested to redesign the perforated end fitting to eliminate the present large mating distance effect and to investigate the power coefficient variations in the crossflow tank.

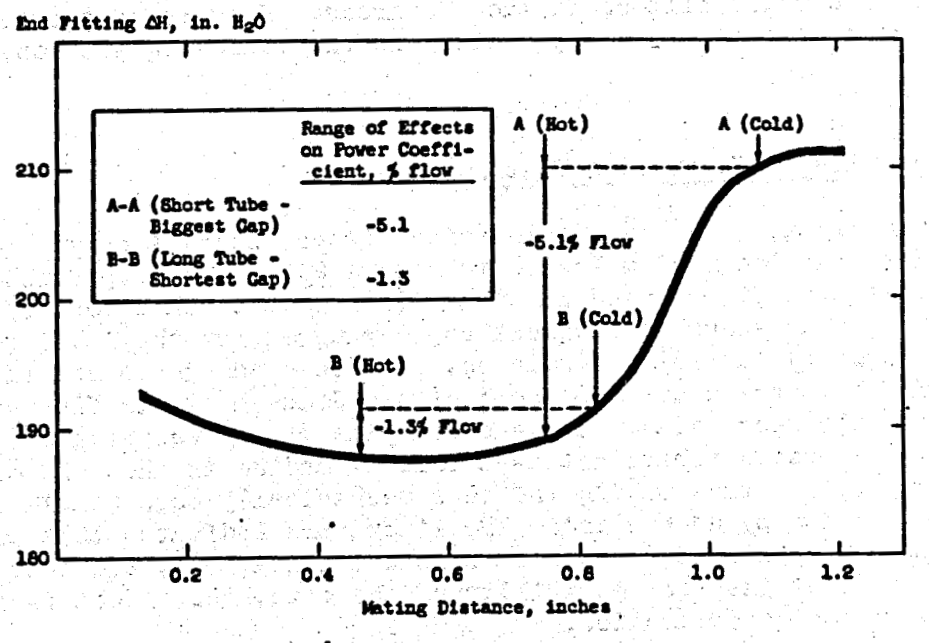


Figure RE-15. L-Reactor Mating Distance Effects, Perforated End Fitting Elements (CMX Data)

MODERATOR FILTRATION

Results from the most recent pilot-scale filtration test in L Area (test No. 9) show that, by utilizing the proper type of filter aid and filter medium, a full-scale moderator filter can be designed which will normally maintain the turbidity level of the reactor moderator below 5 ppm.

As an example, test No. 9 results indicate that a 200 ft^2 filter would operate $2\frac{1}{2}$ months under high turbidity generation rates (3.4 lb turbidity/day) before requiring replacement. Because such a filter would be substantially less expensive to install and operate than evaporators or centrifuges, it has been proposed to proceed with design and installation of moderator filters in the 100 Areas. Details of the test and the scale-up estimate are presented in the following section.

Results of Filtration Test No. 9

The Alsop, 1.11 ft^2, sealed-disc filter with Alsop type 40 cellulose discs as the medium (rated to remove 20-25 micron particles) was the same type filter used in test 8 (see last month's report). The medium is porous enough to prevent rapid pluggage by the small (0.1-5 micron diameter) turbidity particles, but dense enough to retain the filter aid. The filter aid used in the test was Johns Manville Fibra-flo 1C, which is filter-cel diatomaceous earth plus $7\frac{1}{2}\%$ by weight asbestos fibers. The asbestos fibers act as a bonding medium in the cake of diatomite particles, thus making the cake mechanically more stable and resistant to damage by channeling. The literature indicates that asbestos fibers improve particle retention in some applications, and that while asbestos increases the compressibility of the cake, the effect is small for $7\frac{1}{2}\%$ asbestos concentration.

The test was started by establishing flow of turbid (210 ppm) reactor moderator through the filter at 0.201 gpm (0.181 gpm/ft^2) and immediately valving filter aid slurry into the feed stream. During the first hour of operation, 56 grams of filter aid were added to form a 1/16" precoat on the medium to minimize plugging the medium. For the remainder of the test, filter aid was added continuously (as slurry) at a rate of 8.85 g/hr, or 1.26 g of filter aid per g of actual turbidity. The filter flow was adjusted hourly to 0.201 gpm and the ΔP recorded. Correcting for the flow decrease between valve adjustments and flow contributed by the slurry feed, the integrated average flow of turbid moderator to the filter was 0.175 gpm.

Turbidity and ΔP data for the course of the run are presented in figure RE-16 and the following table. The test ran for 60 hours, at which time the ΔP was 21 psi and the filter contained 600 g of filter aid (540 g/ft^2 or 1.19 lb/ft^2) and 119,400 ppm-gal of turbidity (107,600 ppm-gal/ft^2 or 0.834 lb/ft^2). It is estimated that the filter was full of solids at the end of the test, equivalent to a 1/2"-thick cake on each filter disc. Six hundred and fifty-five gallons of filtrate was collected, of which 28.7 gallons, or 4.4%, originated in filter aid slurry. The

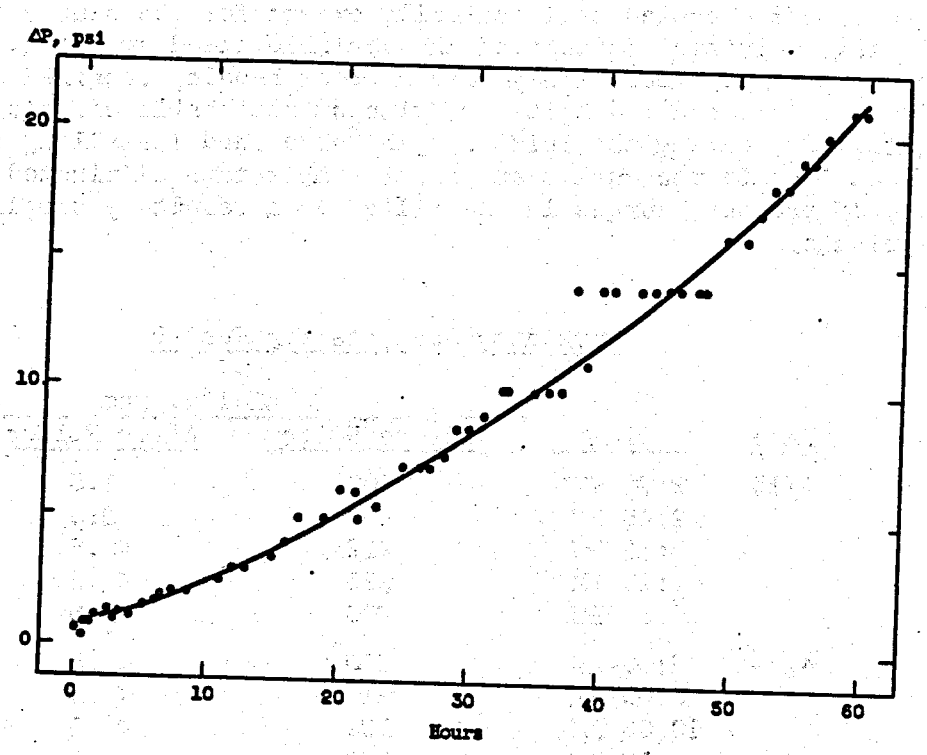


Figure RE-16. ΔP versus Time, Moderator Filtration Test No. 9 (Flow = 0.183 gpm/ft^2)

table shows that except for the first 37 minutes of the precoating period, complete clarification of the moderator was achieved throughout the test.

The rate of ΔP buildup was substantially less in test 9 than in previous tests, as shown in figure RE-17. This may be due to (1) more complete retention of the turbidity particles which reduced or eliminated plugging of the medium, or (2) less tendency of the filter aid particles to plug the medium because of the matting property imparted in the filter aid cake by the asbestos fibers.

The equipment operated satisfactorily except for the slurry tank agitator. Following an initial failure of the variable speed agitator, the slurry tank was stirred successively with a broom handle, repaired agitator, helium bubbler, and new agitator, without noticeable effects on the test results. Relocating the inlet sample valve, and installing restricting orifices in both the outlet and inlet sample taps eliminated the possibility of pressure surges in the filter as a result by sampling manipulations.

Turbidity Analyses for Test 9

Date	Time	Turbidity, ppm	
		Before Filter	After Filter
4/13	2:20 PM*	200	1.5
	2:45 PM	-	2.0
	3:10 PM	212	<0.5
	7:00 PM	220	< .5
	10:30 PM	200	< .5
4/14	2:00 AM	210	< .5
	6:00 AM	210	< .5
	10:05 AM	220	< .5
	2:15 PM	182	< .5
	6:07 PM	200	< .5
	10:25 Pm	167	< .5
4/15	2:00 AM	170	< .5
	6:00 AM	190	< .5
	10:50 AM	174	< .5
	2:00 PM	175	< .5
	6:00 PM	174	< .5
	10:00 PM	162	< .5
4/16	2:00 AM	211	<0.5

* Test started at 2:08 PM, April 13.

Full-Scale Moderator Filter

The approximate size and cycle length of a full-scale moderator filter has been calculated by making the following assumptions:

1. The turbidity generation rate in the reactor is 3.4 lb/day. This was the K-2 rate, which is the highest sustained rate of turbidity generation experienced thus far.

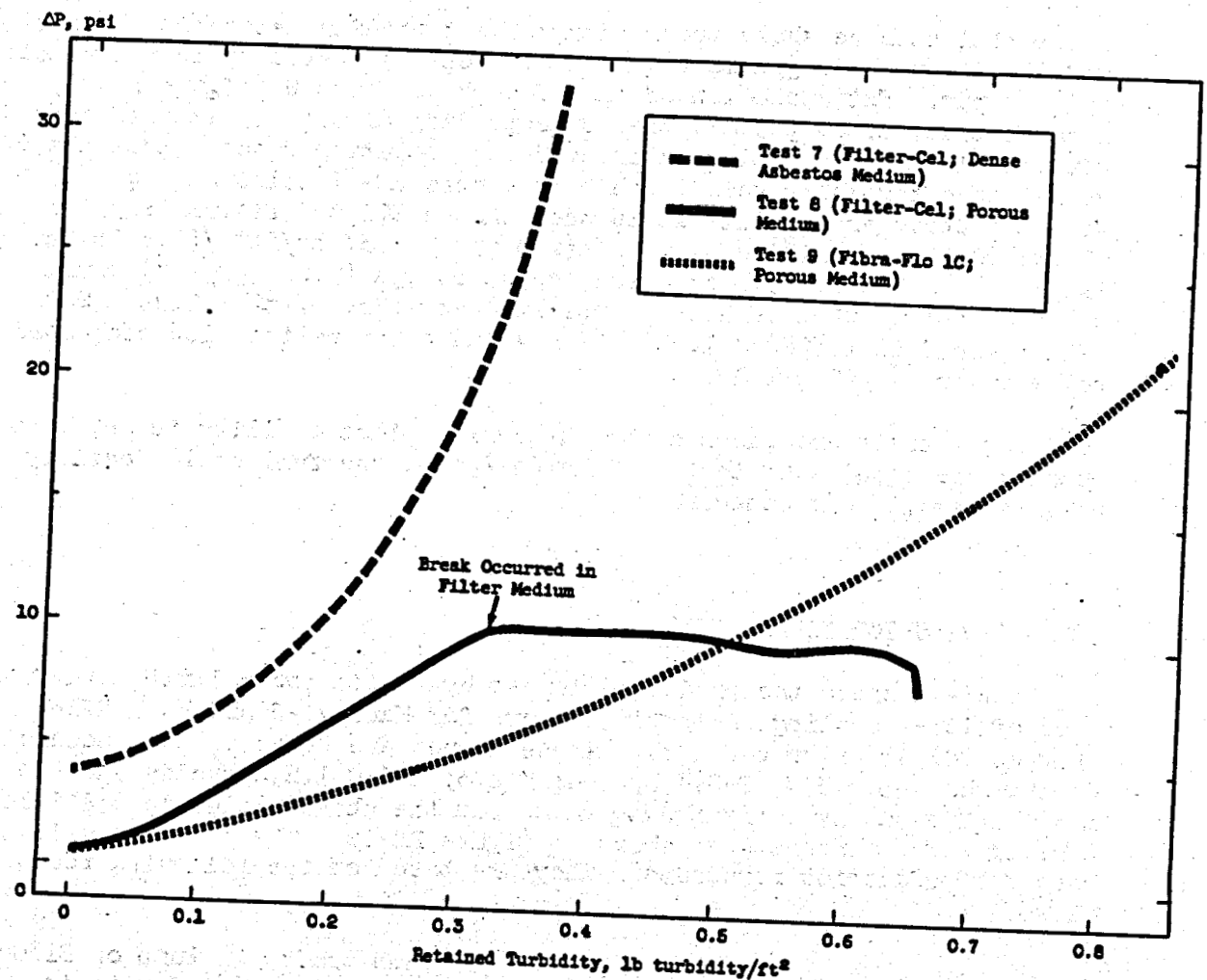


Figure RE-17. Comparative ΔP Buildup, Tests 7, 8, and 9 (Flow = 0.183 gpm/ft^2)

2. It is desired to maintain the turbidity level in the bulk moderator at 5 ppm or less.

3. The filter is operated up to 60 psi ΔP.

4. Extrapolation of test 9 results from 21 psi to 60 psi ΔP is valid, ie, the cake specific resistance is predictable at high ΔP.

A material balance shows that under these turbidity generation conditions a filter flow of 50 gpm is required to keep the turbidity in the reactor below 5 ppm. Extrapolation of the ΔP curve of test 9 (figure RE-17) indicates that, at 60 psi ΔP, the corresponding amount of retained turbidity would be 1.57 lb/ft^2. The calculated extrapolation assumes that the ΔP vs retained turbidity relationship in figure RE-17 holds at high ΔP. With the above assumptions it is seen that a 273 ft^2 filter operating at 50 gpm (0.183 gpm/ft^2) would last (273 ft^2 × 1.57 lb/ft^2)/(3.4 lb/day) = 126 days. A 200 ft^2 filter operating at 50 gpm (0.25 gpm/ft^2) would reach 60 psi ΔP when retained turbidity was 1.23 lb/ft^2. Thus a 200 ft^2 filter would last (200 × 1.23)/3.4 = 72 days before the flow decreased below 50 gpm at 60 psi ΔP.

Additional tests are planned for the L-Area pilot facility in order to confirm the results of test 9. The design of the full scale facility will be carried out concurrently.

POWER LEVELS FOR FORECASTING

Figure RE-18 shows the predicted lowest hydraulic power limit as a function of inlet cooling water temperature for Mark VI-J and Mark VII-A charges for all five reactors. These curves are based on data accumulated during the R-1,2,3,4; P-3,4,5; L-4; K-2,3; and C-1,2,3 cycles and are made continuous by extrapolating data from the above cycles to all inlet cooling water temperatures shown on figure RE-18. These curves will be used for production forecasts. They are based on the following reactor flows:

Area	Fuel	CW Flow, %	PW Flow, %	Pump Shaft Break Reduction Factor, R	Pump or Elbow Cavitation Limit, °C
RPLK	VII-A	~966	~1007	0.81	91.0
C	VII-A	~966	~1027	.80	108.0
RPLK	VI-J	~966	~ 927	.83	94.4
C	VI-J	~966	~ 980	0.81	109.0

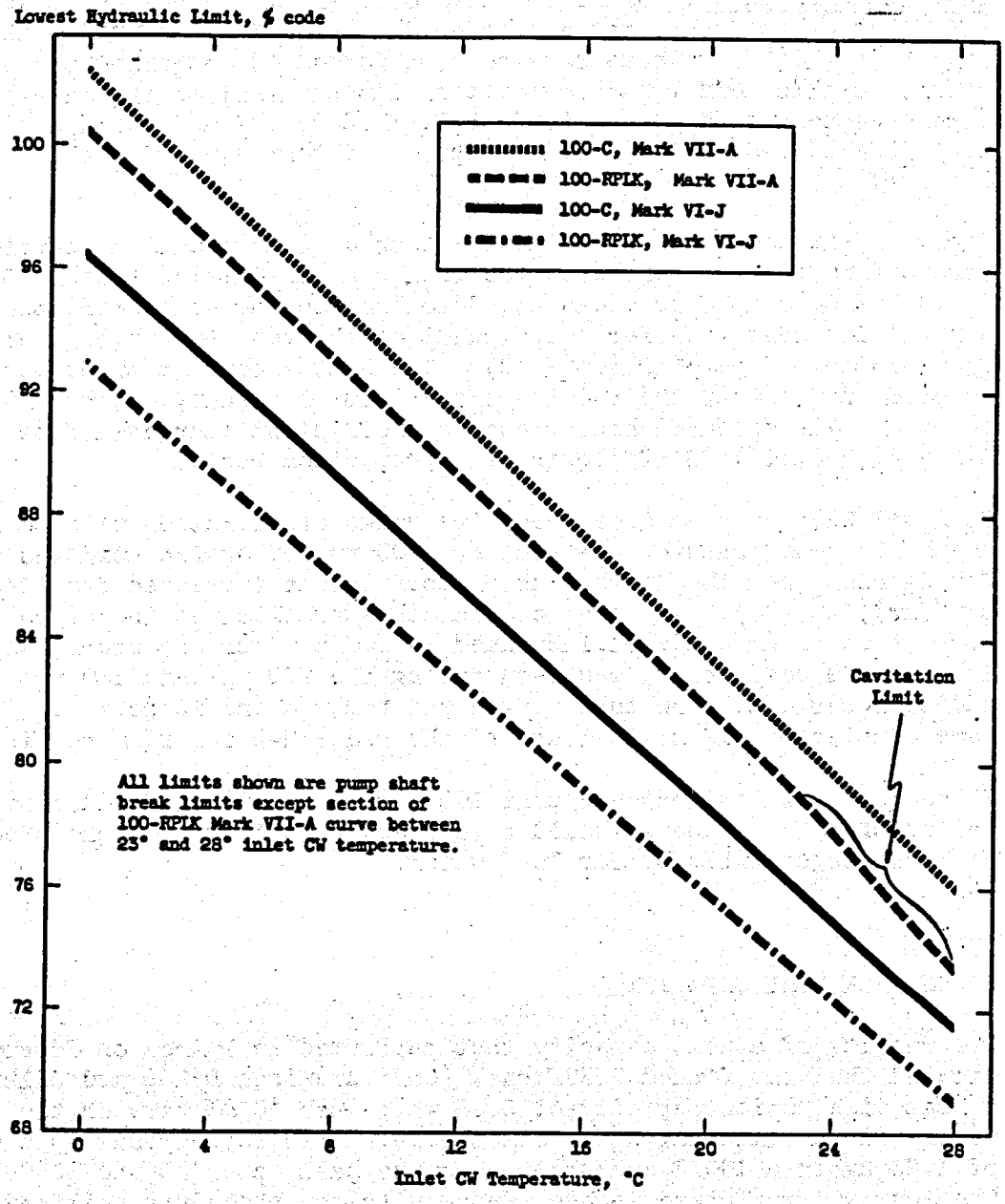


Figure RE-18. Lowest Hydraulic Limit versus Inlet Cooling Water Temperature

PAR POND

Test Heat Exchanger. After 130 days of operation, the heat transfer coefficient of the test heat exchanger decreased from about 490 to 416 pcu/(hr)(ft^2)(°C). Tests have been performed to determine the effectiveness of oxalic acid for removing the film built up on the tube bundle. (See December monthly report, DPSP 58-1-12, p 107 and March monthly report DPSP 59-1-3, p 117 for description of test equipment and film buildup).

Figure RE-19 shows a plot of heat exchanger heat transfer coefficient versus time for the period of the oxalic acid addition tests. On April 8, 70 ppm of oxalic acid was added to the cooling water for about 3½ hours. The heat transfer coefficient of the test exchanger increased from 416 to 427 pcu/(hr)(ft^2)(°C). The heat exchanger tubes were inspected and part of the reddish-brown film previously observed had been removed. The measured heat transfer coefficient increased from 430 to 446 pcu/(hr)(ft^2)(°C) following the inspection period.

On April 22, after an additional two weeks of operation with no further change in heat transfer coefficient, 100 ppm of oxalic acid was added for 3½ hours and the heat transfer coefficient increased from 446 to 472 pcu/(hr)(ft^2)(°C). The oxalic acid concentration was increased to 200 ppm for one hour and then increased to 300 ppm for one hour. No further increase in coefficient was seen during the 200 ppm and 300 ppm acid-addition periods. The tube bundle was removed and inspected. The tubes were completely clean; no trace of the reddish-brown film remained.

These test results indicate that 100 ppm of oxalic acid added over a period of 3½ to 7 hours should remove any film formed on the process heat exchanger tubes by Par Pond water.

SCRUP CASK HEAT CAPACITY

The results of a heat capacity test performed in P Area on February 6 indicated that the finned SCRUP cask (ORNL drawings D8289 and D21292) has a heat dissipation capacity of 35.8 watts per °C ΔT between the cask water and the external still atmosphere. This corresponds to a capacity of 1.86 kw in a 100°F atmosphere and the cask water at the 90°C limit. The test was performed in accordance with the procedure outlined in RTM-1720. The SCRUP cask will not be used for SNAP shipments as originally planned because the LMF cask is dimensionally adequate and has a higher heat dissipation capacity (4.2 kw).

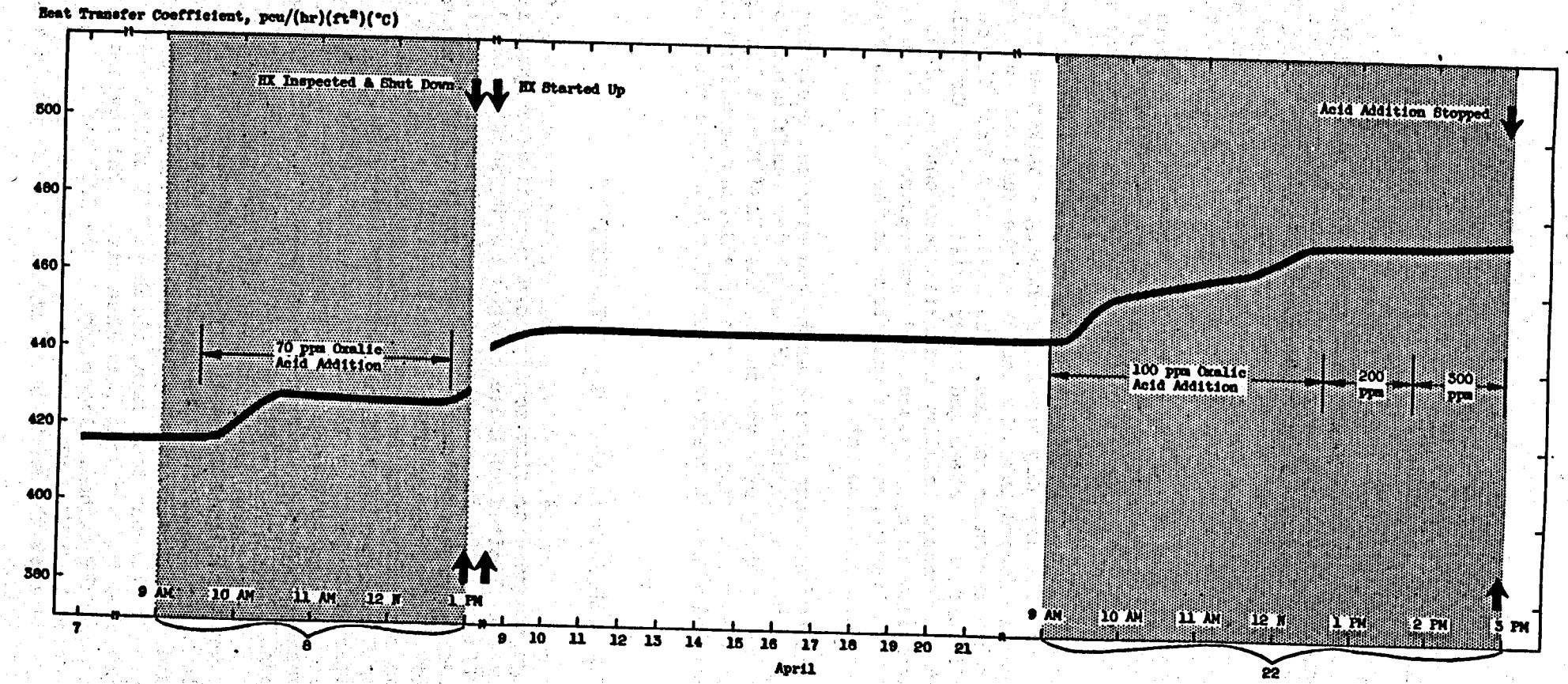


Figure RE-19. Par Pond Test Heat Exchanger Heat Transfer Coefficient versus Time

OXIDE FILM RESISTANCE OF MARK VI-J FUEL

Equipment has been installed in the K-Area disassembly basin to determine whether the heat transfer coefficient of irradiated Mark VI-J fuel is significantly different from that of fresh fuel. There is a possibility that aluminum oxide on the surface of irradiated fuel causes the heat transfer coefficient to be considerably less than that of fresh clean fuel. The equipment, which was described in last month's report, p 326, has been operated successfully with aluminum dummy fuel elements.

Difficulty has been experienced in inserting the steam and water seal plugs into irradiated fuel elements, and the equipment has been dismantled temporarily to allow additional machining of the plug surfaces.

The equipment utilizes Mark VI-J fuel tubes in outer housings (the stem and orifice assembly and inner housing - basket assembly are removed). Steam is forced into the inside of the fuel tube while water is forced through the outer annulus. The steam flow rate is controlled so that an interface of steam and water exists inside the fuel tube several inches from the bottom to allow complete steam condensation. The steam and water flow rate, pressure, and temperature are measured, and the amount of heat transferred from the steam to the water is calculated from the steam condensation rate.

Two dummy elements have been tested with the equipment. One of the dummy fuel tubes was smooth and the other was sandblasted to approximate the surface pitting observed on the Mark VI-J fuel tubes from flow Zone I of the K-2 cycle. The observed over-all heat transfer coefficients for both dummy elements at various water flow rates are shown in figure RE-20.

INCREASED FLOW IN MARK VI-J FUEL ELEMENTS

Present blanketed Mark VI-J charges are restricted by fuel element design and reactor flow zoning to a process water pump flow of about 937%. Power is limited in these charges by the shaft break limit and by fuel effluent temperature. These power limits could be increased 3.6% to 5.2% power by widening the coolant annuli, thereby increasing the coolant flow (two cases for such a redesigned Mark VI element were described in the report for November 1958, DPSP 58-1-11). It was shown in the report for December 1958, DPSP 58-1-12, that when total flow is increased, the power limit imposed by surface heat flux (BOSF) is reduced considerably below the present BOSF power limit. This pronounced decrease in BOSF is caused by the adverse effects of geometry, velocity, and pressure on the burnout heat flux correlation presently used at SRP.

It was shown in last month's report that the revised Bernath burnout heat flux correlation (DPW 59-133) differed only slightly from that presently used at SRP; in addition, SRL has proposed a new correlation in DP-355, based on burnout test data, which yields significantly higher values of Mark VI-J burnout heat flux than either the revised Bernath method or the present SRP method. Accordingly, the BOSF's for the two cases of increased Mark VI-J flow have been recalculated for the condition of winter operation at the shaft break limit. The results are shown in the following table.

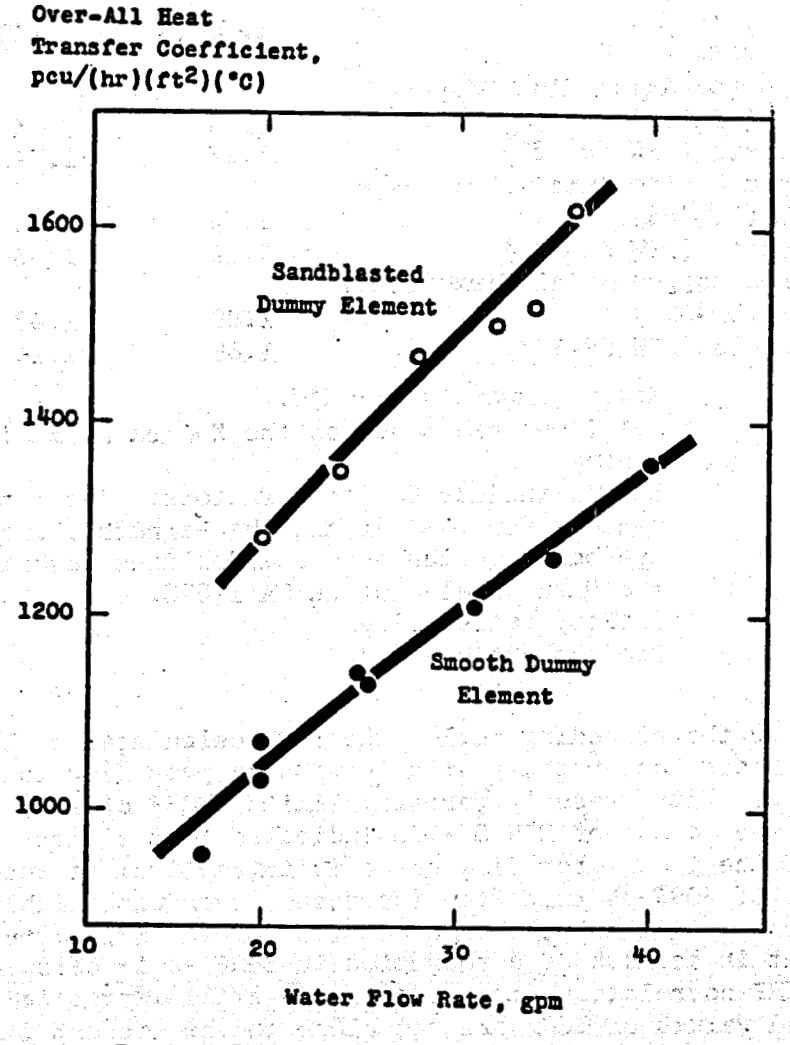


Figure RE-20. Heat Transfer Coefficient versus Water Flow Rate with and without Surface Pitting (Dummy Elements)

	Case 1	Case 2	Case 3
Reactor Power (Shaft Break Limit),* %	85.6	89.2	90.8
Total Pump Flow,** %	937	1014	1059
Fuel Element Flow (Zone 1),** %	1.756	1.940	2.051
Minimum BOSF			
Rooftopped Axial Flux Shape††			
SRL (DP-355)	2.56	2.03	1.81
Bernath (DPW 59-133)	2.24	1.54	1.26
Chopped Cosine Axial Flux Shape			
SRL (DP-355)	2.32	1.74	1.50
Bernath (DPW 59-133)	2.12	1.45	1.14
Antirooftopped Axial Flux Shape‡			
SRL (DP-355)	1.86	1.48	1.32
Bernath (DPW 59-133)	1.66	1.14	0.93

* River temperature = 9°C.

** All flows are based on the K-Area hydraulic system.

† BOSF's include hot spot factors. The hot spot factors for the SRL and the Bernath correlations are based on the same nonidealites described for the Mark VI element in TA 1-575.

†† Rooftop ratio = 1.4.

‡ Rooftop ratio = 0.7.

As shown in the preceding table, the BOSF calculated by the SRL correlation decreases less rapidly with increasing pump flow than the method of DPW 59-133. With present operating limits (BOSF of 1.4) and calculation methods, the method of DPW 59-133 indicates that element redesign for either of the two higher flow cases is impractical because of the severe reduction of BOSF as pump flow increases, even with highly rooftopped flux shapes. Similarly, though adoption of the SRL correlation would not result in as much of a reduction in BOSF as is calculated with the present SRP correlation, the redesign is still impractical because the calculated values of BOSF are too close to the present limit of 1.4 for high flows and adverse axial flux shapes. Present knowledge of Mark VI-J operation does not permit relaxation of the 1.4 limit on BOSF.

Physics Studies

REACTOR PHYSICS

C-5 CHARGE DESIGN

The C-5 charge will be a Mark VI charge designed to make use of the maximum amount of Mark VI fuel and target material now in the 300-Area inventory. The nominal fuel concentration for the charge will be 16.70 wt % oralloy rather than 16.90 wt % as is specified for the standard Mark VI charge. The Technical Standard limits on oralloy concentration of individual tubes will be relaxed so that fuel tubes can be accepted in the range of 16.70 ± 1.20 wt %. In other respects the charge will be standard. The fuel and target will be matched for the standard bucklings of 963 microbucks in the control zone and 833 microbucks in the buckled zone. The charge will have standard control system and shadow rods and will have twelve poison tubes in the buckled zone initially that will be replaced with spike elements after a reactor exposure of 10 to 20%. The margin of control for the charge will be about 51 microbucks and the attainable cycle exposure will be about 54.4%.

SPIKE ELEMENTS FOR R-6 AND P-7 CHARGES

To make use of the maximum number of Mark VI fuel tubes which are now in the 300-Area inventory, it will be necessary to use some fuel tubes which have a fuel concentration greater than 18.00 wt % oralloy. These tubes can be used as spike elements in an unreflected Mark VII-A charge without limiting reactor power. However, restrictions must be placed on the way in which these tubes are handled in the Assembly and Disassembly Areas. The rules for handling Mark VI elements are based on handling fuel tubes which have a fuel concentration no greater than 18.00 wt % oralloy. The handling rules for Mark VI-J fuel elements, based on a maximum fuel concentration of 27.0 wt % oralloy, will be made applicable for handling the spike elements for a charge if the fuel concentration of any element is greater than 18.00 wt % oralloy.

The R-6 and P-7 charges will use Mark VI-S spike elements with a fuel concentration greater than 18.00 wt % oralloy. The average concentration of the group of 24 new Mark VI-S elements in each charge will be about 18.00 wt % oralloy. The fuel concentration of individual elements will be within the range of ±0.20 wt % from this average concentration.

MARK VI-J SCRAM TEST

A scram test was performed at the end of the K-2 cycle to determine the kinetic behavior of a Mark VI-J charge. Neutron and gamma flux transients, fuel effluent and reactor effluent temperature transients, and an inference of moderator flow patterns and velocities were reported in the February monthly report (DPSP 59-1-2). The following analysis is concerned with bulk moderator temperature changes after a scram.

Moderator temperatures were measured at 31 thermocouple positions during the scram test. The primary purpose of the test was to determine the delay (transport lag) and response times of the bulk moderator to permit accurate analysis of the kinetic reactivity behavior of the reactor.

Representative temperature transients are presented in figure RP-1 for three spatial locations near the radial center of the reactor. Analysis of these and other temperature transients showed that the flow at any given spatial location was displacement-type (very little mixing) flow and could not be described by a simple exponential relationship with time and power.

To determine a more suitable mathematical description of the behavior of moderator temperatures during the scram, the reactor was divided into 31 regions with one monitored thermocouple near the center of each region. Each region was statistically weighted, and the statistically weighted average temperature of the moderator was determined as a function of time (see figure RP-2). The statistical weights used were as follows:

Radial Weights		Axial Weights	
Gang I	0.28	+6' level	0.10
Gang II	0.26	+3' level	0.25
Gang III	0.31	0' level	0.30
BZ	0.15	-3' level	0.25
		-6' level	0.10

The statistical weights used to calculate the bulk moderator temperature affect the response time of the bulk moderator. For example, if the radial statistical weights were altered to 0.25 for Gangs I and II, 0.30 for Gang III, and 0.20 for the buckled zone, the response time would be approximately 10% longer than for the above case. This result is caused by the fact that the fast-acting central portion of the tank is weighted less heavily than the region corresponding to Gang III.

Calculated moderator temperatures for an 8-second and a 10-second response time and a 1.1-second delay time also are presented in figure RP-2. The best fit to calculated curves indicated a moderator temperature response time of 9.0 ± 1.0 seconds with a delay of 1.1 ± 0.3 seconds after power begins to decrease (2.0 ± 0.3 seconds after the scram time).

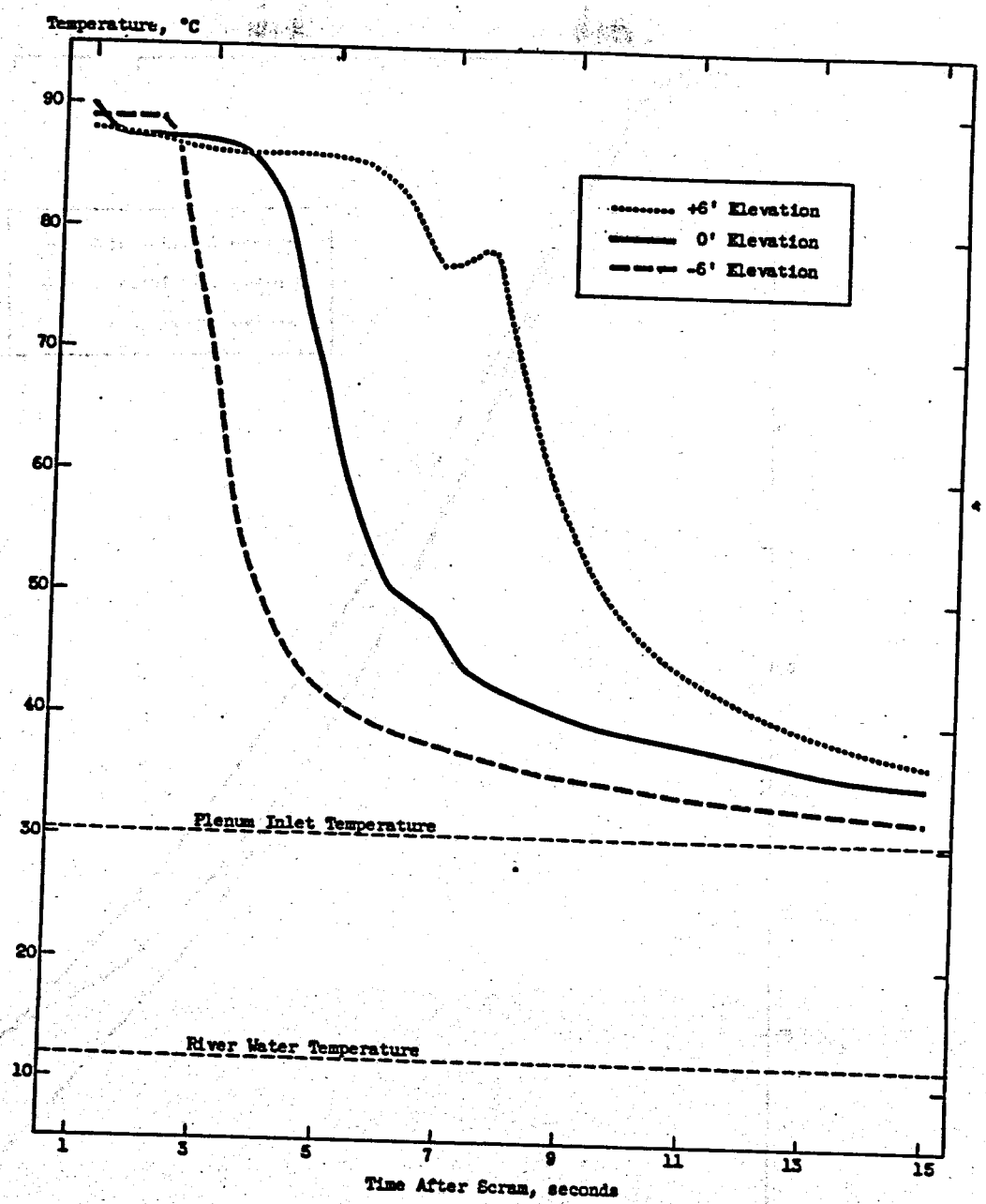


Figure RP-1. Bulk Moderator Temperatures (Coordinate X24,Y56) Following Scram

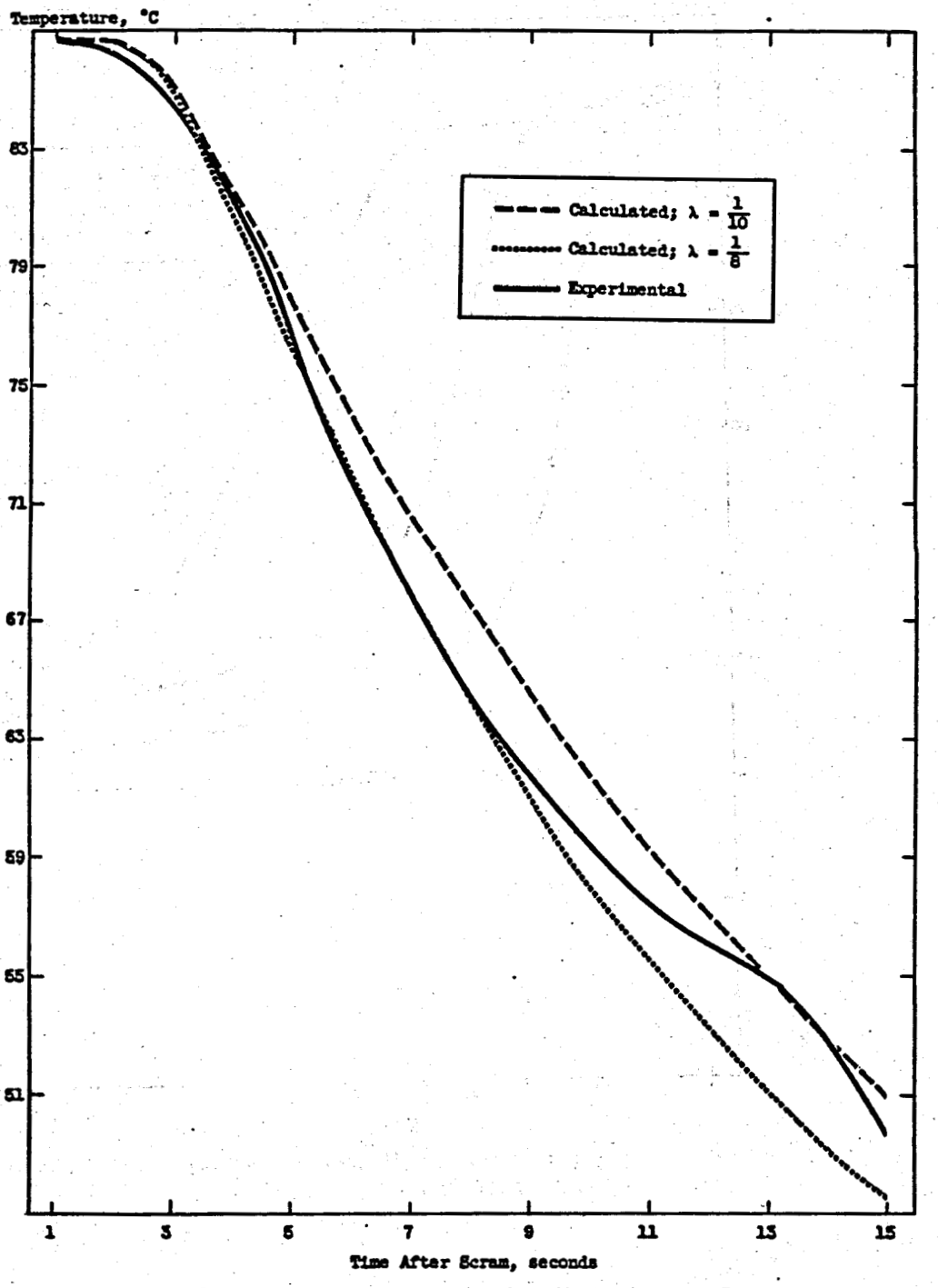


Figure RP-2. Statistically Weighted Bulk Moderator Temperatures

The calculated curves were derived from the expression:

$$\frac{T_m/T_{m_O}}{dt} = \lambda \left(\frac{T_F - PI}{T_{F_O} - PI} - \frac{T_m - PI}{T_{m_O} - PI} \right)$$

where:

T_m = Moderator temperature at time t

T_{m_O} = Initial moderator temperature

PI = Plenum inlet temperature

T_F = Fuel effluent temperature, determined empirically from a flat zone tube as $[59(0.8e^{-1.1t} + 0.2e^{-0.08t}) + 30]$

T_{F_O} = Initial flat zone effluent temperature

λ = Response constant of the moderator

It is seen from the above equation that the effluent temperature of the flat zone fuel elements was used as the driving force for bulk moderator temperature changes. The initial 0.5°C decrease in measured bulk moderator temperatures during the first second after the power decrease (1.0 to 2.0 seconds after the scram) is attributed to the loss of neutron and gamma heating in the moderator.

A plot of moderator delay time as a function of spatial location in the tank is presented in figure RP-3. The delay times are measured in seconds from the instant power began to decrease rapidly until the moderator temperature at a given point began to decrease rapidly. This plot may be used to study flow patterns and velocities in the bulk moderator.

MARK VI-J POWER COEFFICIENT OF REACTIVITY

The power coefficient of reactivity of a Mark VI-J charge (with blanket) was measured at several power levels during the K-2 cycle. The coefficient was determined both by the rod oscillation and the xenon transient techniques, and the test results are presented in figure RP-4.

Because the extrapolated power coefficient at zero power is very close to the same value for both the rod oscillation and the xenon transient techniques, and because the results of the moderator coefficient test indicate a temperature coefficient that is independent of temperature, it is felt that the relatively constant power coefficient as measured by the xenon transient technique is accurate. The average power coefficient as measured by xenon transients is -0.00066% $\Delta k/k/MW$ for Mark VI-J.

(Text continued on page 122.)

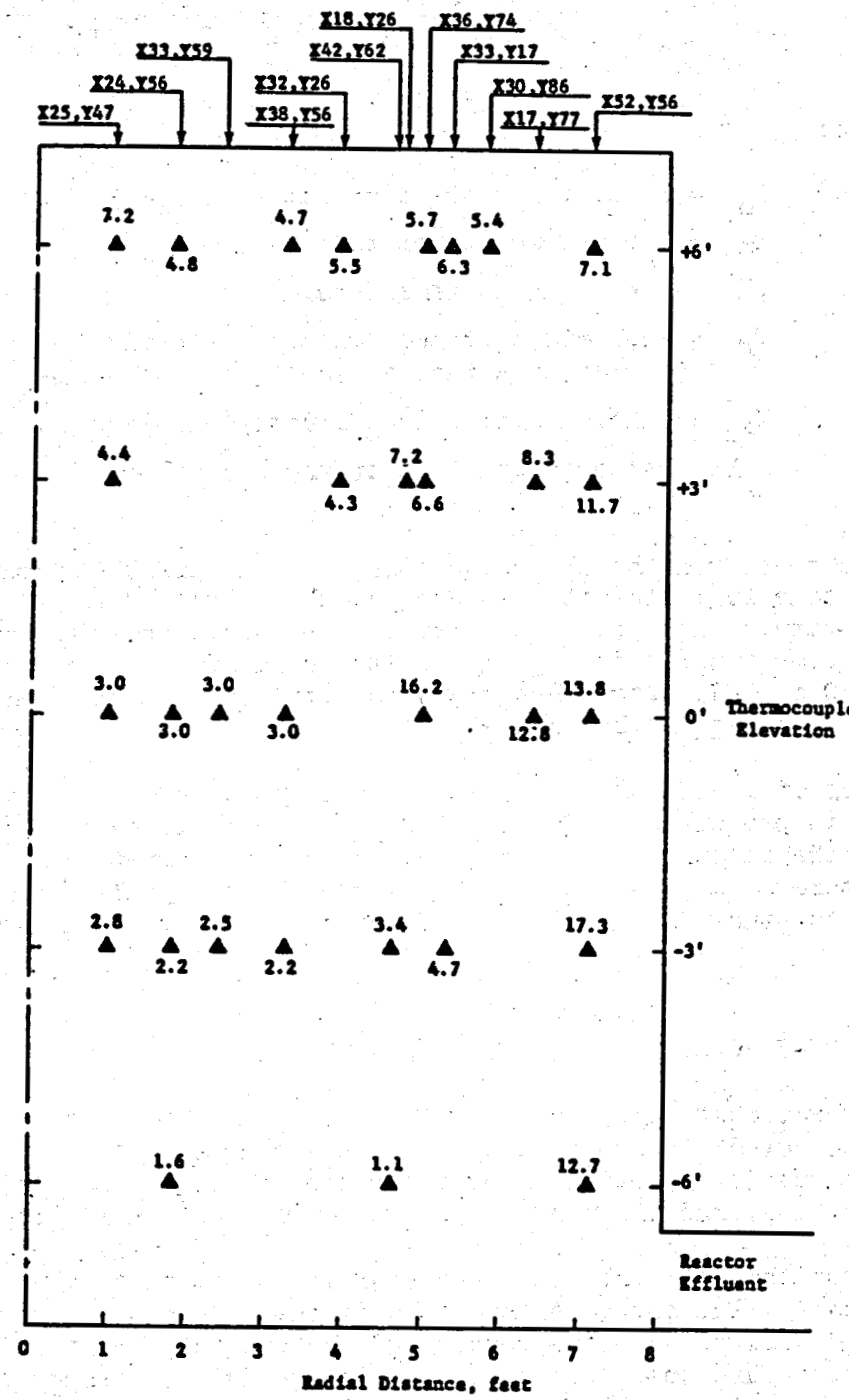


Figure RP-3. Bulk Moderator Delay Times (Seconds)

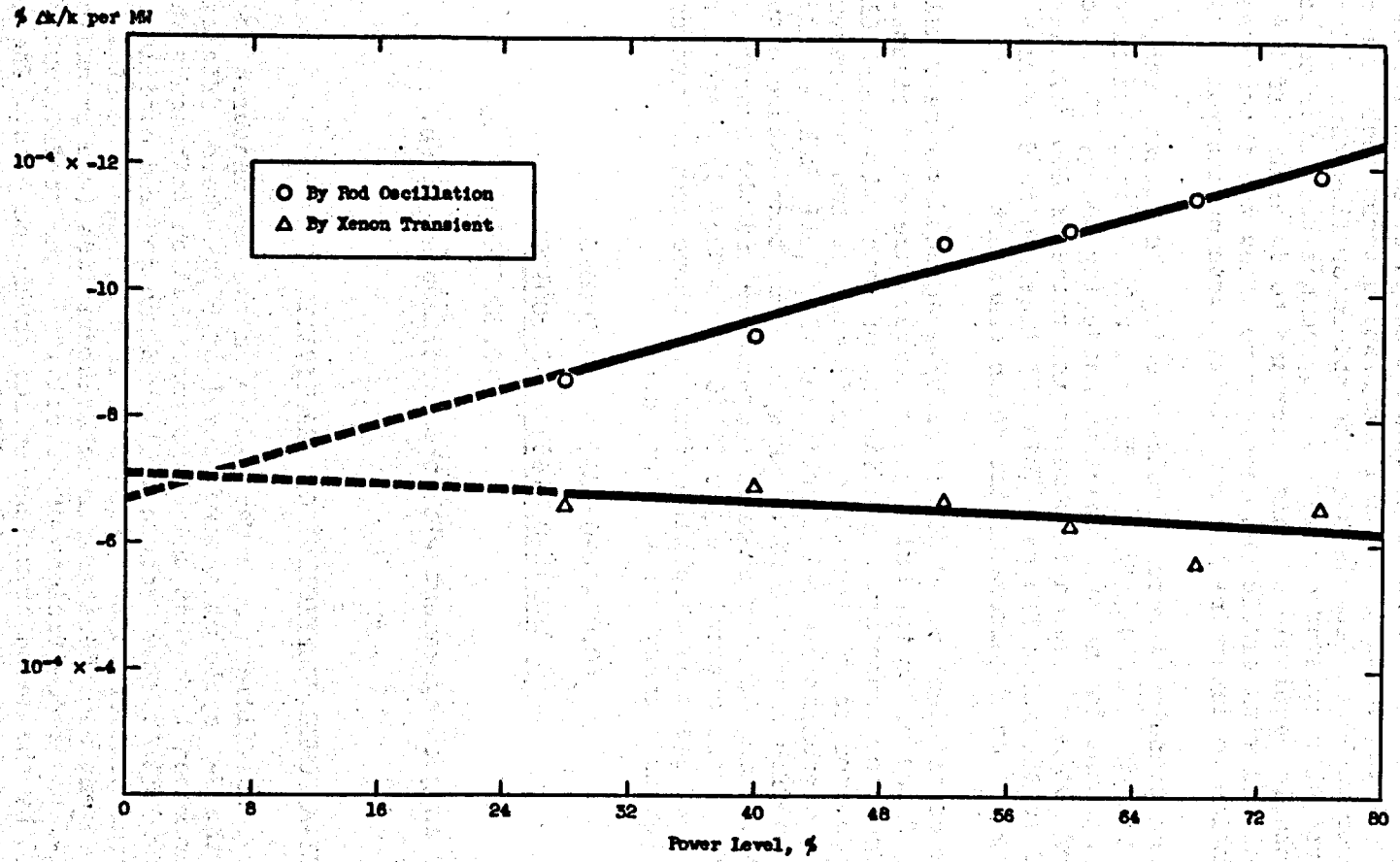


Figure RP-4. Mark VI-J Power Coefficient of Reactivity

The discrepancy between the rod oscillation results and the results of the xenon transient calculations is attributed to the effect on the reactivity of the temperature oscillations being out of phase with the reactivity of the rod oscillation by more than 180°. This effect of the temperature oscillations could cause a large error in the determination of control rod worth by the rod oscillation technique. The magnitude of the error would be dependent on the moderator flow characteristics of the particular type of charge. A program has been started to evaluate the probable magnitude of this error and to determine if the error can be predicted with sufficient accuracy to allow the future use of rod oscillation to determine the reactivity worth of control rods.

The xenon transients were calculated using the latest method of calculating the value of θ (MW/kg). The refinements included increasing θ by 10% to account for neutron spectrum hardening and by 3% to account for a value of 194 Mev/fission instead of the value of 200 Mev/fission that is assumed in the Xenon Tables. This 13% increase in θ results in a 13% increase in the magnitude of the xenon transient.

A power coefficient of -0.00066% $\Delta k/k/MW$ agrees extremely well with known values of the moderator and coolant temperature coefficients. The measured moderator temperature coefficient of reactivity was -0.0195% $\Delta k/k/°C$. This moderator coefficient is a combination of two temperature coefficients: a coolant channel coefficient of -0.005% $\Delta k/k/°C$, and a bulk moderator coefficient of -0.0145% $\Delta k/k/°C$. The temperature changes per MW for the average coolant channel temperature and the average bulk moderator temperature are approximately 0.024°C and 0.0392°C, respectively. The power coefficient as calculated from the above temperature coefficients and temperature changes per MW would be:

$$\begin{aligned} &(0.024)(-0.005) = -0.000568 \\ +\ &\underline{(0.0392)(-0.0145) = -0.000120} \\ &-0.000688\%\ \Delta k/k/MW \end{aligned}$$

This is in good agreement with the experimental value.

MARK VII-A POWER COEFFICIENT OF REACTIVITY

The power coefficient of a Mark VII-A charge was measured in R reactor during the R-3 cycle at a power level of 85%. The coefficient as determined by xenon transient and by rod oscillation techniques was -0.00061 and -0.00065% $\Delta k/k/MW$, respectively. Since the rod oscillation technique for measurement of control rod worth is suspected to contain errors that are dependent on moderator flow characteristics (see preceding section on Mark VI-J power coefficient of reactivity), the coefficient that was determined by the xenon transient method should be used. The xenon transient was corrected by a value of 1.13 to account for neutron spectrum hardening and the change in the value of Mev/fission.

ROD REPLACEMENT IN MARK VII-A CHARGES

Control Rods. Replacement of control rods in Mark VII-A charges having an exposure greater than 30% was discussed in the January monthly report (DPSP 59-1-1). The analysis reported there was too limited in scope, considering only the reactivity of Gang I with SRP fuel, to provide an adequate or economic replacement schedule. An extension of the analysis has been made to include reactivities of all three gangs with both SRP and SC fuel.

The number of half rods of a gang which must be replaced during the charge-discharge prior to the start of the cycle is determined by the curves of figure RP-5. These curves show the maximum of the average exposure of the rods in a gang with which a cycle may be started as a function of the anticipated length of the cycle. Rods must be changed until the average rod exposure in the gang is less than the number read from the appropriate curve. This method of rod replacement will insure that at the end of the cycle, the half rods will be capable of holding sufficient buckling to permit good flux shaping (ie, q_{max}/q_{avg} less than 1.30).

The curve relating to half rod productivity and costs which was published in the January report shows that the worth of production minus rod costs varies little between rod exposures of 24% and 48%. Thus during a shutdown between cycles, if charge-discharge time is not limiting or if a septifoil must be visited anyway, all half rods with exposures greater than 24% should be replaced. In addition, half rods should not be exposed beyond 48%.

Full rods should be replaced after alternate cycles unless they receive an exposure of 12% at insertions greater than 500 veeder units. If this condition occurs, the rods should be changed at the end of the cycle in which the exposure is obtained.

Shadow Rods. The function of the shadow rods is to prevent flux peaking in fuel adjacent to the sparjets. To provide adequate protection against flux peaking, the exposure of 4.9 wt % Li-Al shadow rods will be limited to 60%. A curve of flux peaking versus exposure is presented in figure RP-6. The curve was obtained by combining curves of: (1) peaking versus reactivity worth of shadow rods, and (2) reactivity worth of 4.9 wt % rods versus exposure. There have been 4.9 wt % shadow rods exposed to 60%, at which time the peaking was 0.8%, corresponding to about 0.5°C in effluent temperature. No problem was encountered with this amount of peaking, but the peaking should not be allowed to be greater than the 0.5°C.

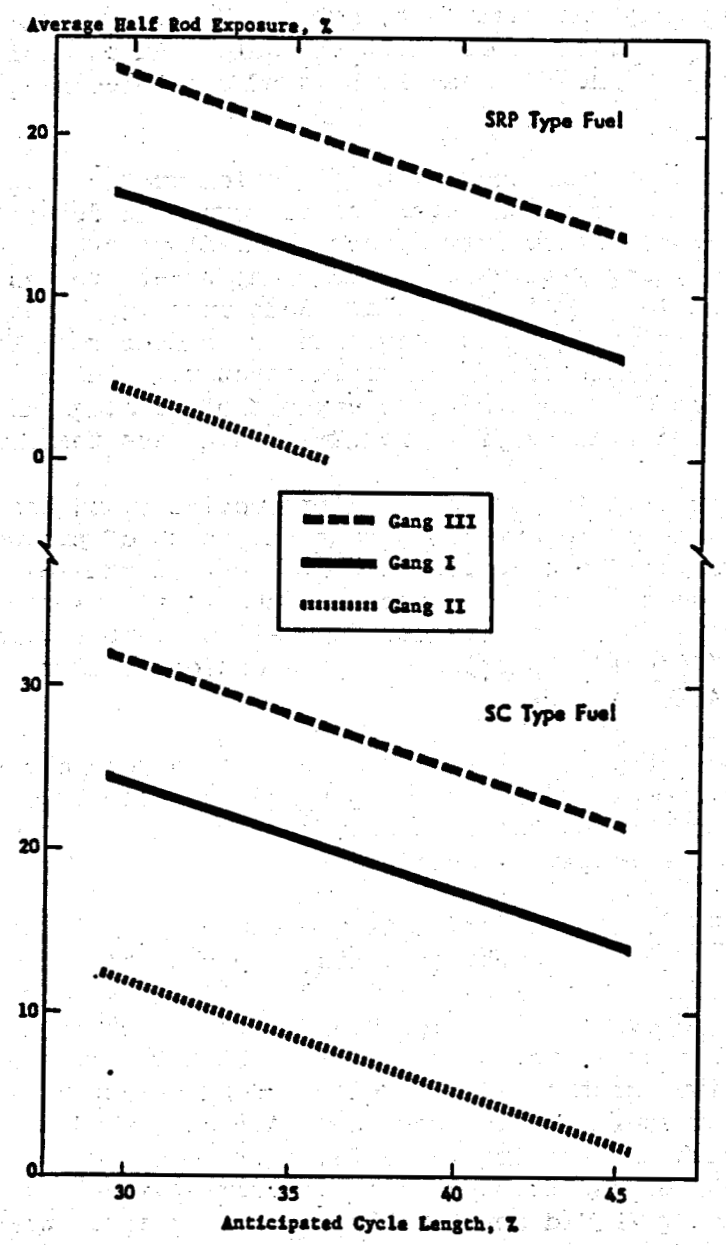


Figure RP-5. Limits on Average Half Rod Exposure

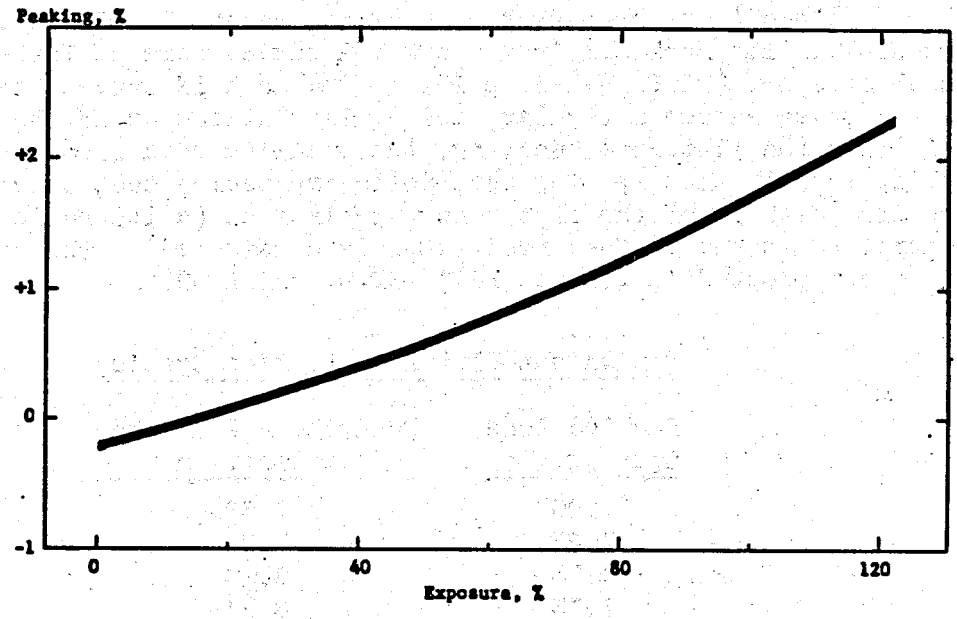


Figure RP-6. Percent Peaking versus Exposure of 4.9 wt % Shadow Rods

OPERABILITY OF MARK VII-A, MARK V-A, AND MARK IX LATTICES

A general survey has been made to determine the relative flux shaping characteristics of the Mark VII-A, Mark V-A, and Mark IX as fuel assemblies for PTPR lattices. The buckling available with lattices of these elements was calculated and compared with the buckling necessary for various degrees of axial flux flattening. This comparison is given in figures RP-7, RP-8, and RP-9, in which the buckling values are plotted against M^2 as a parameter. Each figure corresponds to a particular reactor height.

Buckling Available. The buckling available is the material buckling of a flat-zone, zero-rod configuration of the fuel assemblies. The bucklings were obtained from the curves of buckling versus lattice spacing in figure RP-10. The Mark VII-A and Mark V-A curves were published in last month's report (DPSP 59-1-3, p 124). The Mark IX curves have been added. These curves assume a regular, triangular lattice as in the buckled zone. To derive the flat zone buckling, the buckling read from the curves at the appropriate lattice spacing (see following table) must be corrected for the irregularity of the flat zone type lattice (a 1:6 ratio of number of control positions to fuel positions was considered). The corrected bucklings are plotted in figures RP-7, RP-8, and RP-9.

Equivalent Flat Zone Lattice Spacing

Buckled Zone Spacing, in.	Equivalent Flat Zone Spacing, in.
7.00	7.56
7.25	7.84
7.50	8.10
7.75	8.37
8.00	8.64

In the determination of the migration area ($M^2 = L^2 + \tau$), the value of L^2 for a flat zone lattice was taken to be 1.25 times the value for the corresponding buckled zone lattice. (Buckled zone values were shown in last month's report.) The 1.25 factor is correct for a 7-inch buckled zone lattice spacing and is thought to be a good approximation to spacings at least as large as 8 inches. The values of τ for flat zone lattices were determined from the equation given in last month's report.

The five points on each "buckling available" curve correspond to the five lattice spacings given in the preceding table. (The smallest lattice spacing corresponds to the lowest M^2, etc.)

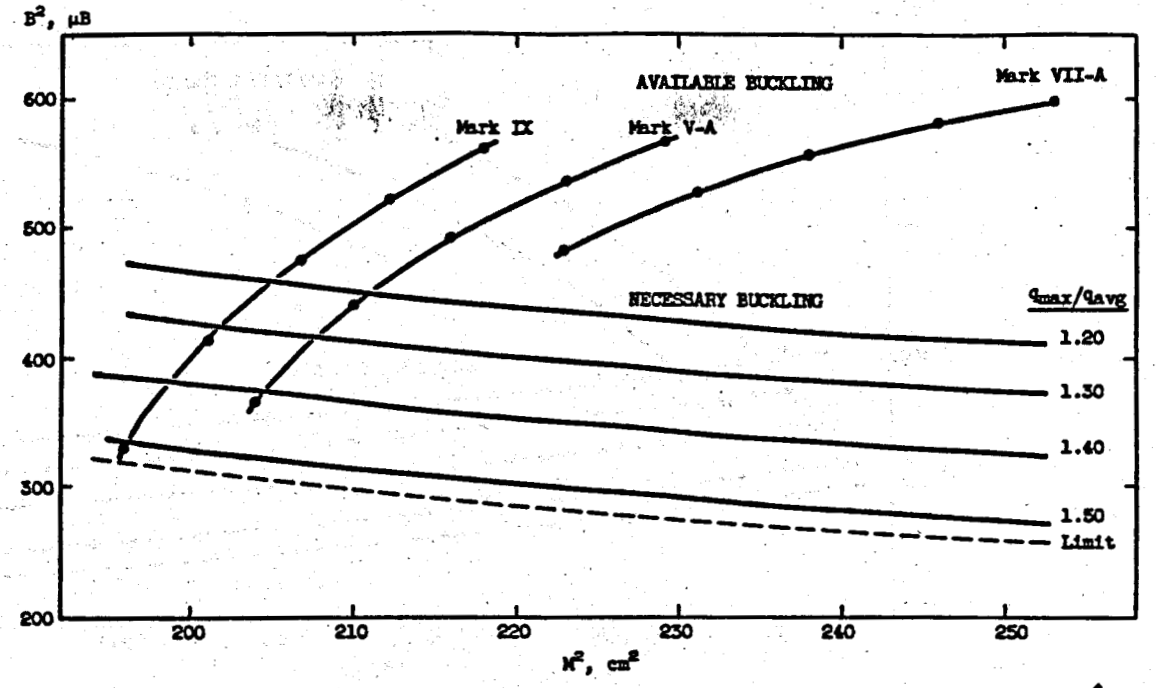


Figure RP-7. Available and Necessary Bucklings with a Reactor Height of 14 Feet

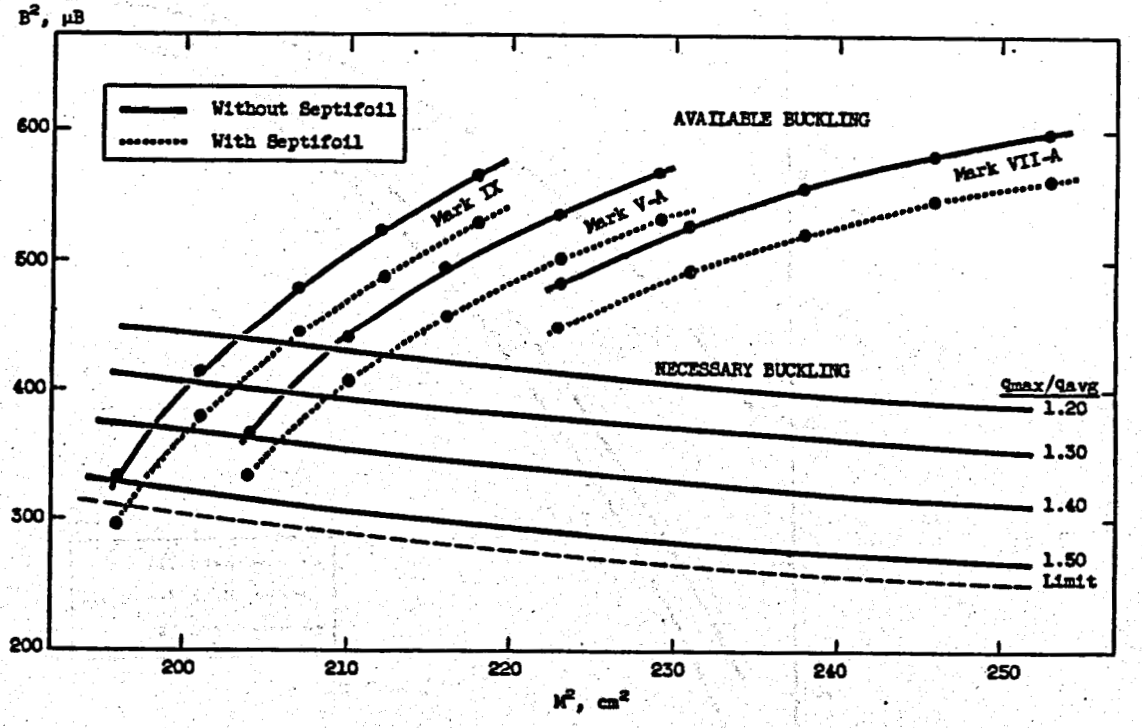


Figure RP-8. Available and Necessary Bucklings with a Reactor Height of 15 Feet 2 Inches (Current Mark VII-A)

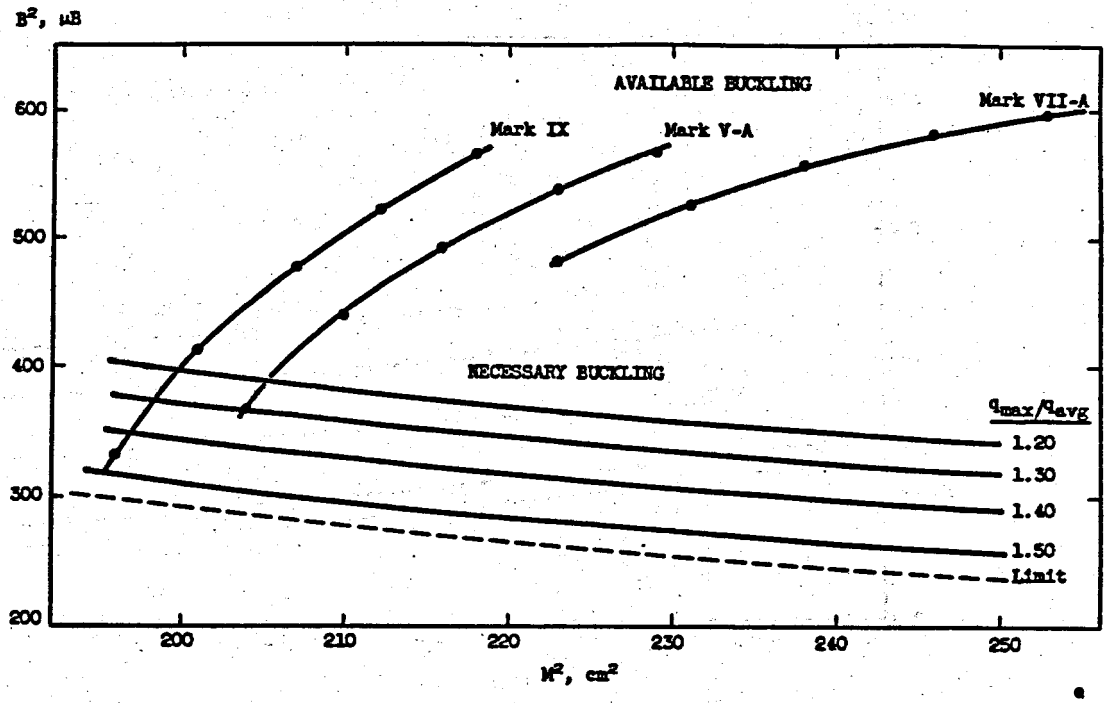


Figure RP-9. Available and Necessary Bucklings with a Reactor Height of 18 Feet

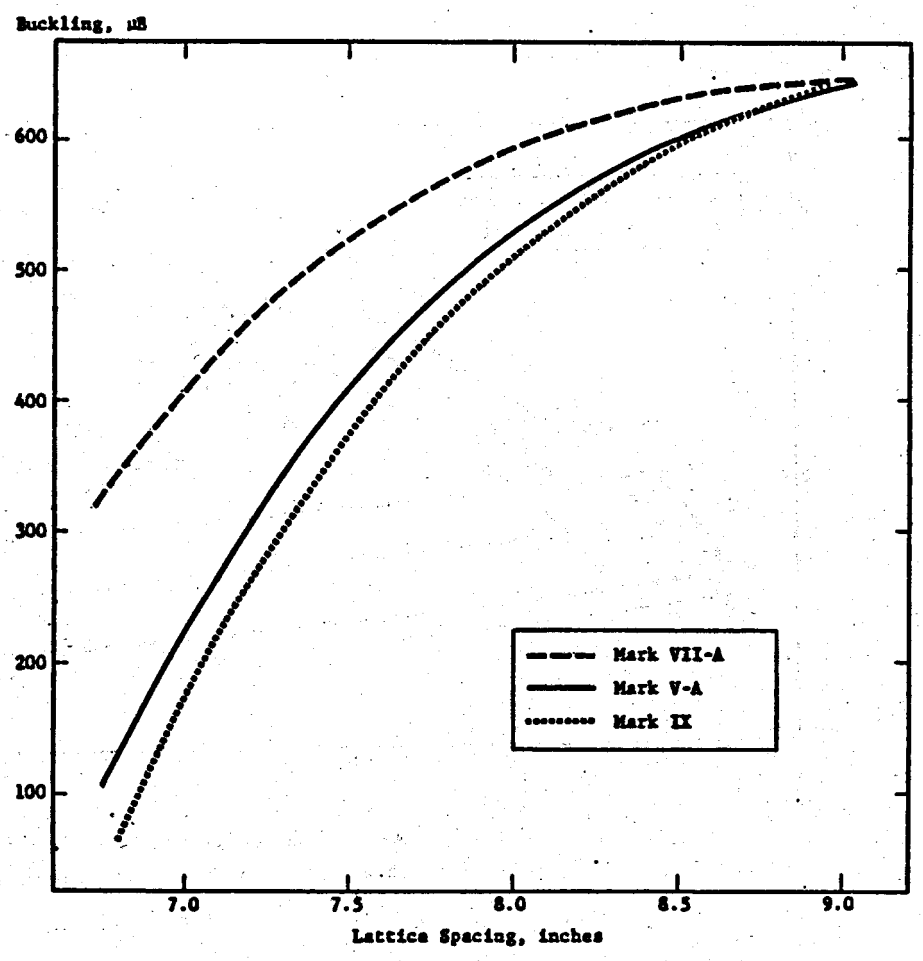


Figure RP-10. Buckling versus Lattice Spacing (Regular Triangular Lattice)

Buckling Necessary. Losses in reactivity ($\Delta k_{eff}/k_{eff}$) expected in the operation of the PTPR are listed in the following table. The basis for the value of the loss is listed along with the loss itself. These losses of reactivity were converted to buckling losses by means of the following equation, in which the factor 1.06 is the reciprocal of the nonleakage probability:

$$B^2 = \frac{k_{eff}}{k_{eff}} \times \frac{1.06}{M^2}$$

PTPR Reactivity Losses

Loss	Value, % $\frac{k_{eff}}{k_{eff}}$	Basis
Xenon	2.76	Current VII-A
Samarium	0.53	Current VII-A
Moderator temp coefficient	0.94	VII-A to 65°C
Metal temp coefficient	0.28	Twice current VII-A
Miscellaneous	0.35	Unexplained current VII-A loss
Total	4.86	

In addition to the losses, the axial buckling and the buckling held by control rods in flux shaping must be supplied. (The radial buckling is zero in the flat zone.) The buckling necessary to support the axial flux shape depends upon the desired flatness of the flux shape, which is measured by the ratio of the maximum to the average axial flux (q_{max}/q_{avg}). Axial flux shapes with half rods positioned for minimum q_{max}/q_{avg} were calculated for several strengths of half rods and for several reactor heights. In all cases the full rods were positioned to 200 veeder units. The ratios of full rod length, half rod length, and the length of 1000 veeder units to the height of the reactor were taken to be the same as with current Mark VII-A charges. The sum of the bucklings held in rods (B^2_{CR}) and in axial leakage (B^2_Z) is given in figure RP-11 as functions of q_{max}/q_{avg}.

To determine a "buckling necessary" curve for a particular q_{max}/q_{avg} and reactor height, the value of $B^2_{CR} + B^2_Z$ was taken from the curves of figure RP-11. This value was added to the value of ΔB^2 which was determined as a function of M^2 from the equation.

The "buckling necessary" curves are more or less average curves. The worse condition of axial flux shaping expected during a cycle would occur with the full rods at about 350 veeder units. This case will be investigated.

Since this analysis was primarily concerned with the PTPR, no consideration was given to the sparjets and their associated shadow rods.

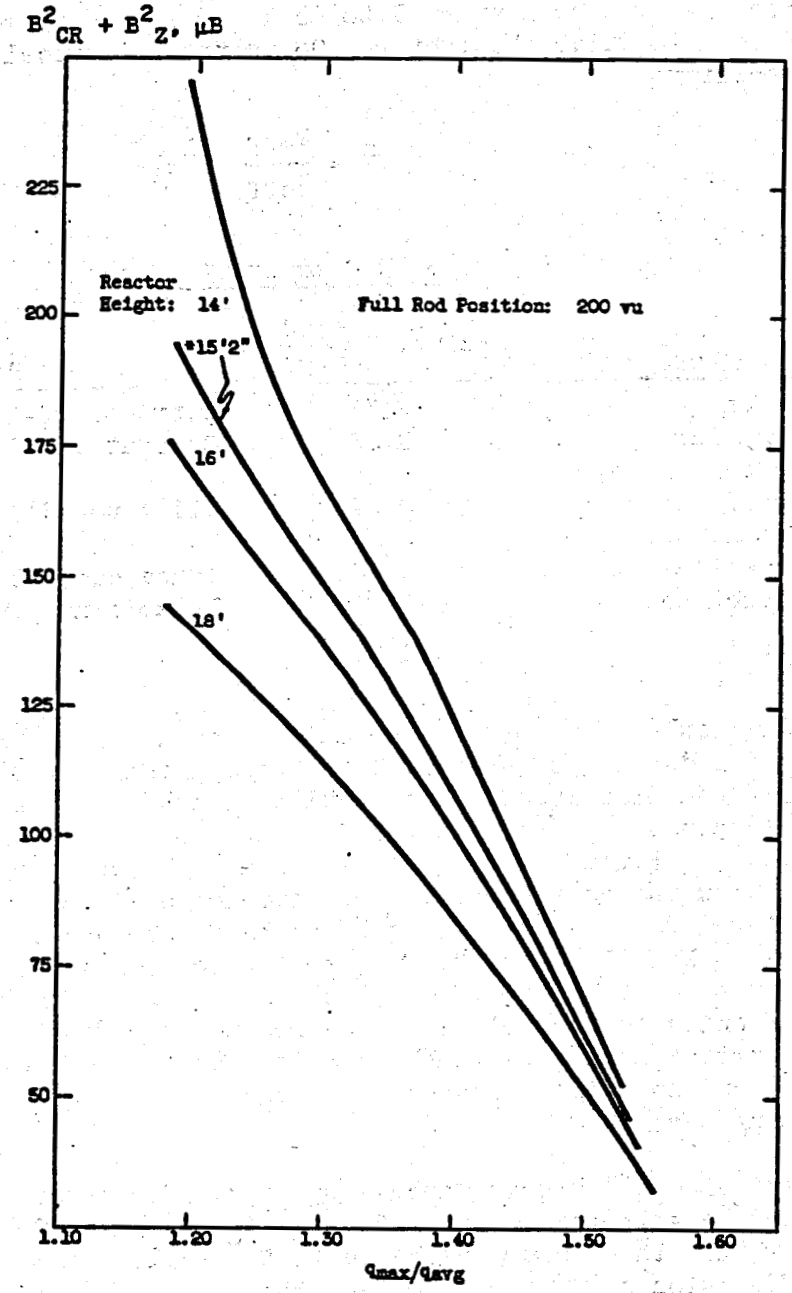


Figure RP-11. Axial Buckling and Buckling Held in Control Rods Necessary to Produce a Particular Value of q_{max}/q_{avg}

* Current Mark VII-A.

REFLECTOR CALCULATIONS

Additional radial flux calculations were made for the Pressurized Tube Production Reactor using the Lil Abner IBM 650 routine. The neutron leakage from the reactor core and the tritium production in producer tubes located at the outer boundary of the reflector were determined as functions of reflector thickness. The machine routine was described in last month's report (DPSP 59-1-3, pp 125-128).

The total neutron leakage was obtained by the following expression:

$$\text{Total Leakage} = \underbrace{-\frac{D_F\int\nabla_R\Phi_F dS}{\Sigma_F\int\Phi_F dV + \Sigma_S\int\Phi_S dV}}_{\text{Fast Leakage}} \underbrace{-\frac{D_S\int\nabla_R\Phi_S dS}{\Sigma_F\int\Phi_F dV + \Sigma_S\int\Phi_S dV}}_{\text{Slow Leakage}}$$

where:

F, S are subscripts referring to the fast and slow neutron groups, respectively

dS is a surface increment

dV is a volume increment

D is the diffusion coefficient

Φ is the neutron flux

∇_R is the radial component of the gradient operator

Σ is the absorption cross section

This expression differs slightly from the one given in last month's report and is taken as the more appropriate one for this analysis. The gradient of the fast and slow fluxes and the volume-weighted fluxes were determined directly from the calculated flux shapes.

Figure RP-12 is a plot of leakage as a function of reflector thickness. The difference between the total leakage from the core (buckled zone) and the total leakage from the reflector is equal to the absorption in the reflector. This difference increases as the reflector thickness is increased. The bottom curve represents the slow leakage from the reflector. It shows that for increasing reflector thicknesses, the fast leakage from the reflector approaches zero.

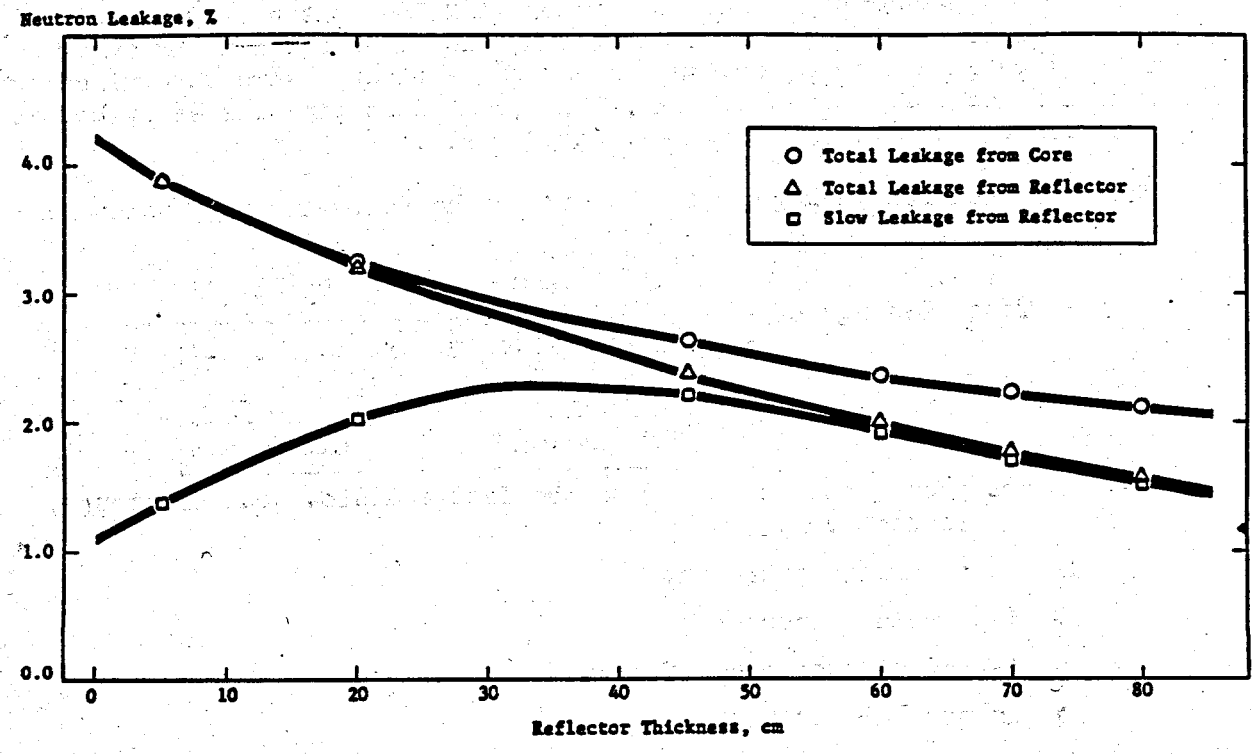


Figure RP-12. Neutron Leakage versus Reflector Thickness

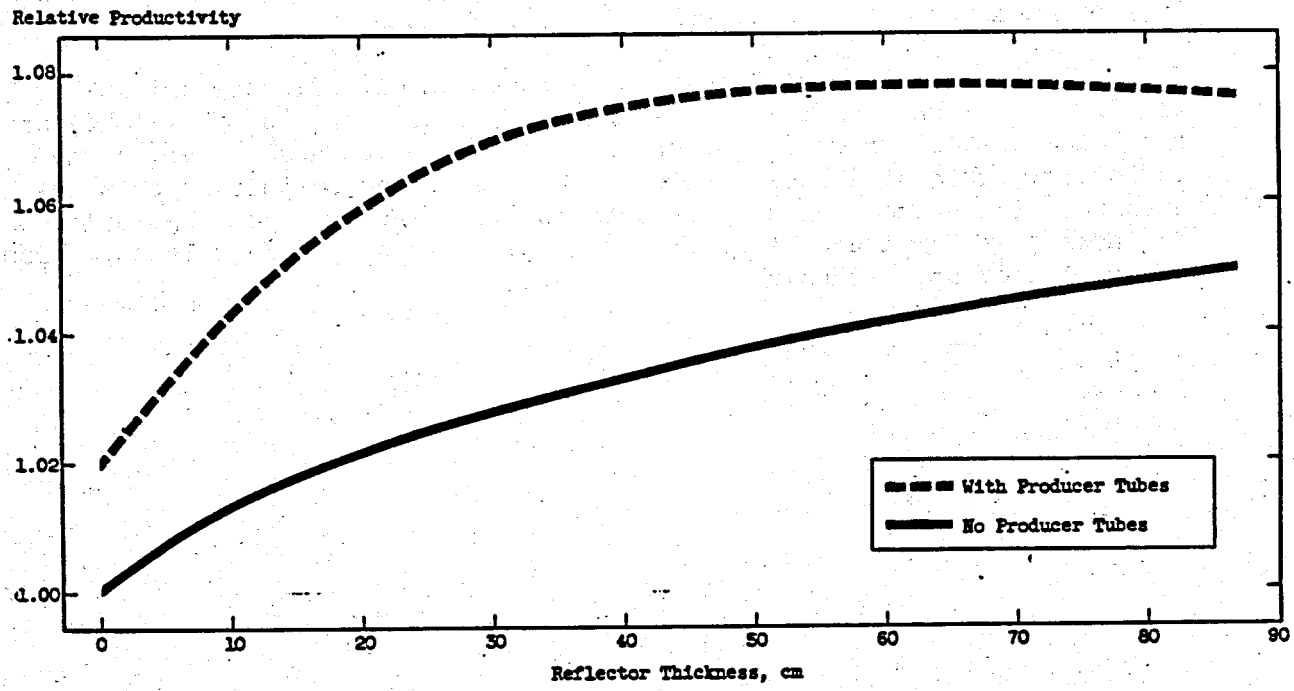


Figure RP-13. Relative Productivity versus Reflector Thickness

The slow neutrons leaking from the reflector can be utilized for tritium production, if lithium elements ("producer tubes") are arranged around the outer boundary of the reflector. In calculating the production to be gained through the use of producer tubes it was assumed that a ring of producer tubes would absorb 80% of the slow neutrons leaking from the reflector.

Relative productivity values were determined by the method given previously for both the producer tube and no producer tube cases. Relative productivity, or fractional increase in production, is plotted in figure RP-13 as a function of reflector thickness. Without producer tubes, an increase in tritium production in control rods occurs as the reflector thickness is increased. Tritium production in producer tubes only, which is proportional to the "slow leakage from reflector" curve shown in figure RP-12, reaches a maximum at a reflector thickness of approximately 35 centimeters. The result is a flattening out of the "relative productivity with producer tubes" curve of figure RP-13, which has a maximum at approximately 60 centimeters of reflector. With producer tubes the gain in production from an increase in reflector thickness from 40 to 60 centimeters is relatively small.

NUCLEAR PHYSICS

TRENDS IN AREA RADIATION LEVELS

A knowledge of trends in area radiation levels and system activities during shutdown, in relationship to operating parameters, is of interest in connection with current and future maintenance of reactor equipment. Information concerning radiation levels in the 105 Buildings during charge-discharge shutdowns is available from the "Area Activity Levels - Monthly Reports," which were issued by Reactor Technology Section from April 25, 1956 to October 14, 1958, and from DPSOL 105-1226, "Status of Reactor System Radioactivity," which replaced the monthly reports. A document entitled "Trends in Radiation Levels - 100 Areas," April 21, 1959, (DPSP 59-886) has been issued. It summarizes this radiation level information for each of the 100 Areas and enables graphic analysis of trends in radiation levels during shutdown as associated with the following operating parameters:

- Reactor Power - One Day Prior to Shutdown
- Cycle Exposure at Shutdown
- Process Water Activity - One Day Prior to Shutdown
- Turbidity Prior to Shutdown
- Type of Fuel

Figures RP-14 through RP-18 enumerate the activity data for five specific locations and a single process water system (System 3). Then data are considered to be representative of system activities in general. Further details are given in DPSP 59-886.

General conclusions which result from an inspection of the data are as follows:

- Radiation levels during shutdown have increased in all areas since April 1956 and are associated primarily with increases in process water activity.
- Increased process water activity is associated with increased cycle exposure and turbidity.
- General trends in radiation levels following shutdown correlate fairly well with process water activity prior to shutdown; fluctuations in radiation levels and hot spots correlate with turbidity and fuel element failures.
- No extensive buildup of long-lived activities in the process water piping is indicated.

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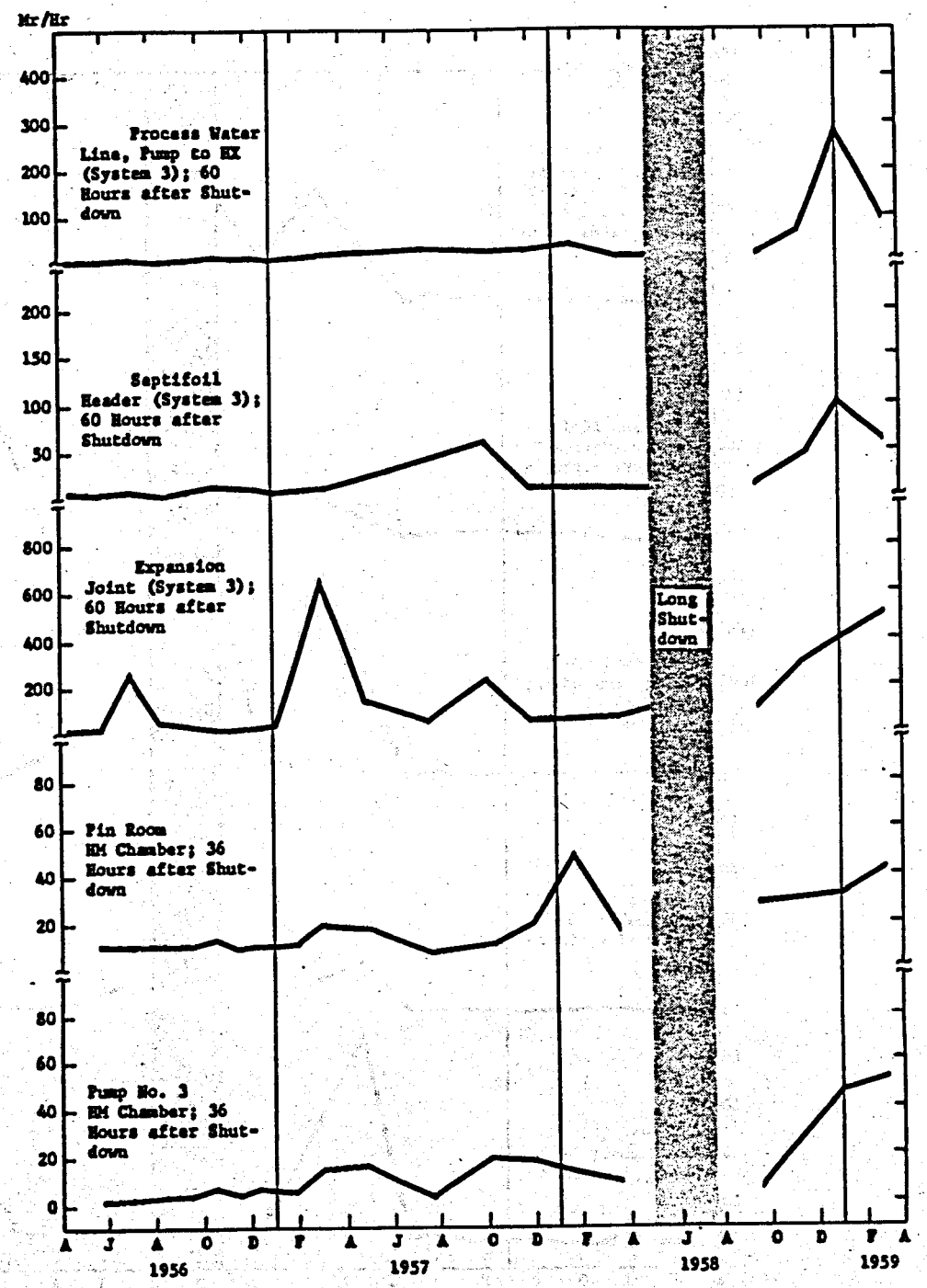

Figure RP-14. Activity Information, R Area

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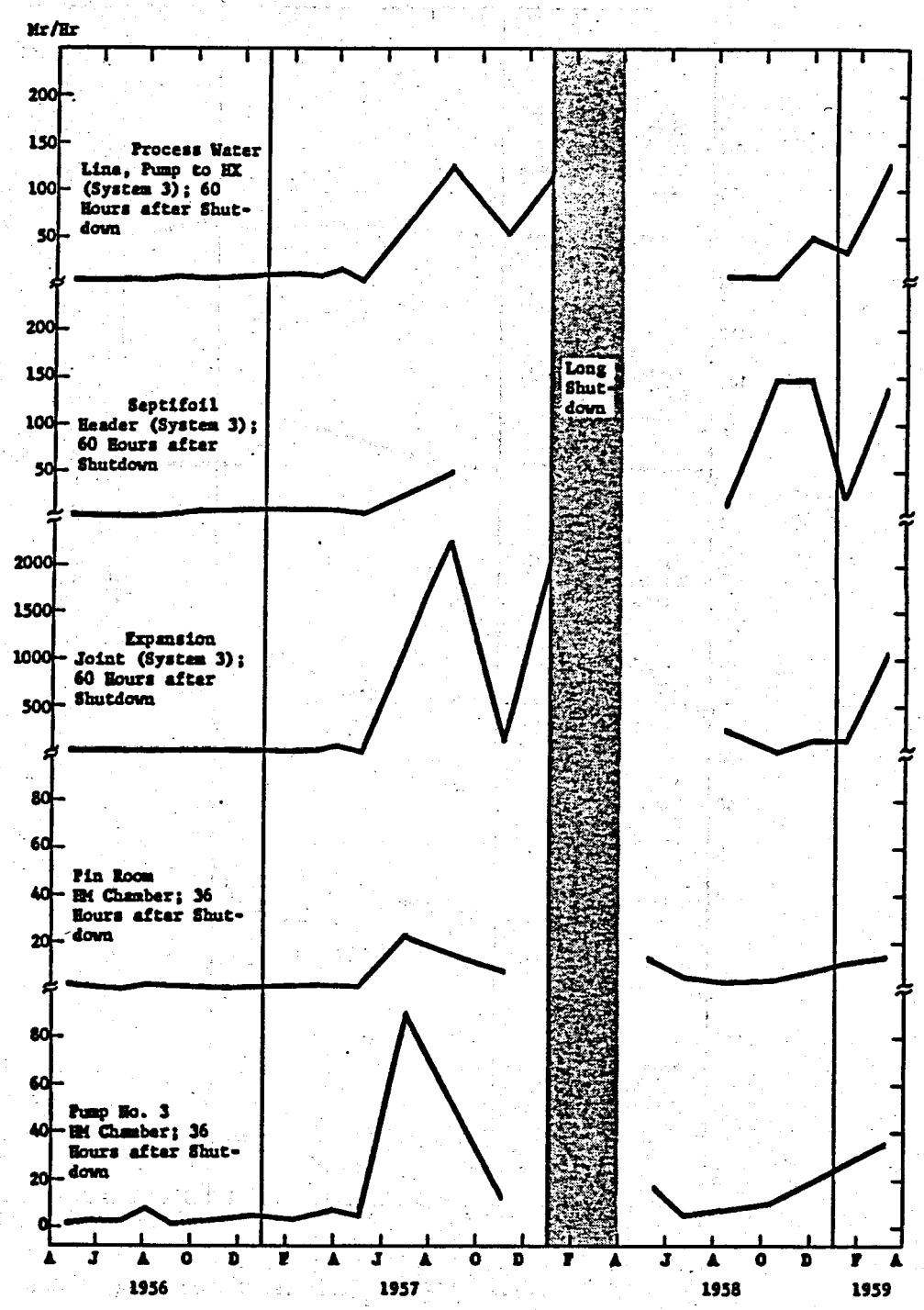

Figure RP-15. Activity Information, P Area

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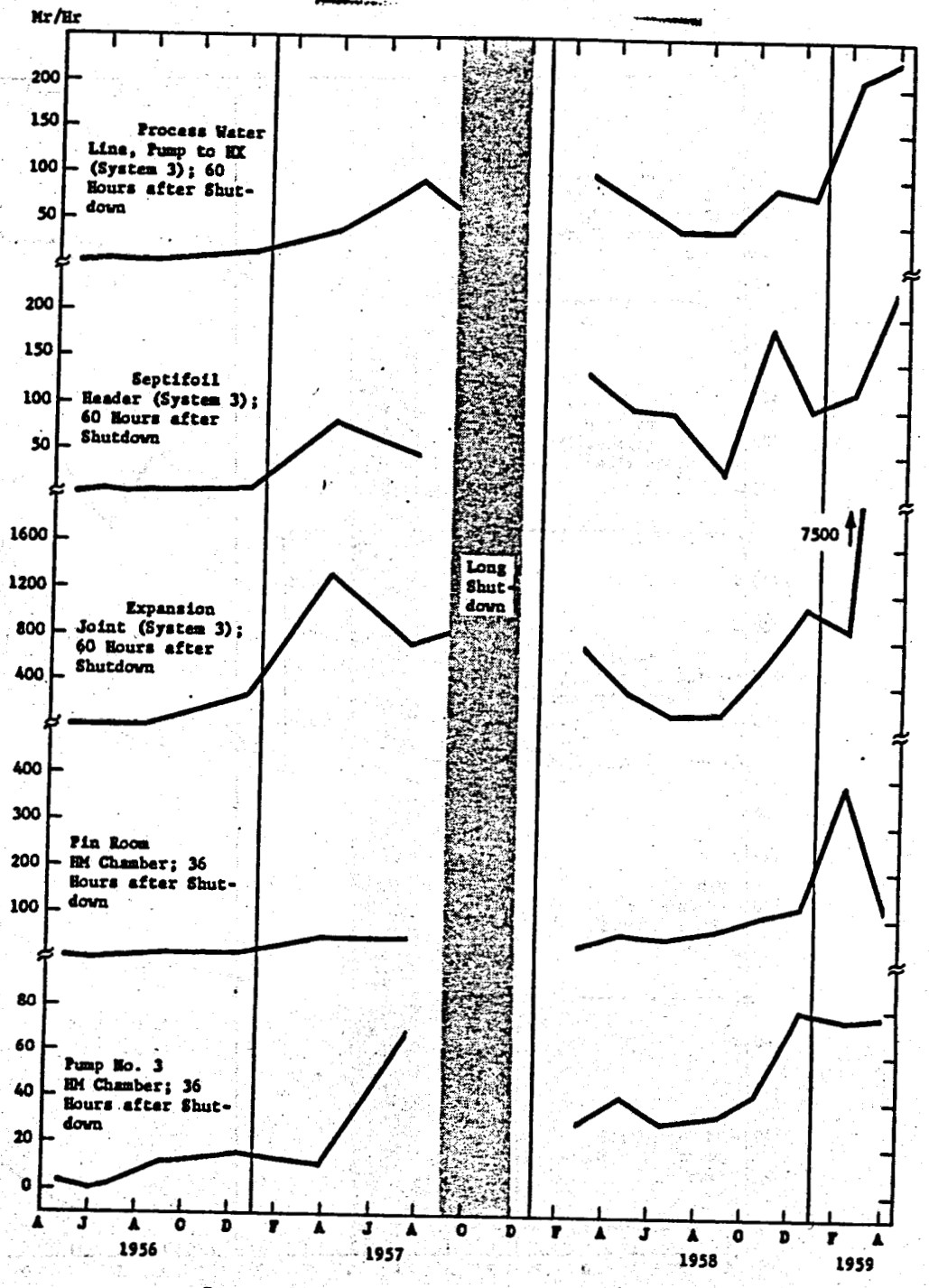

Figure RP-16. Activity Information, L Area

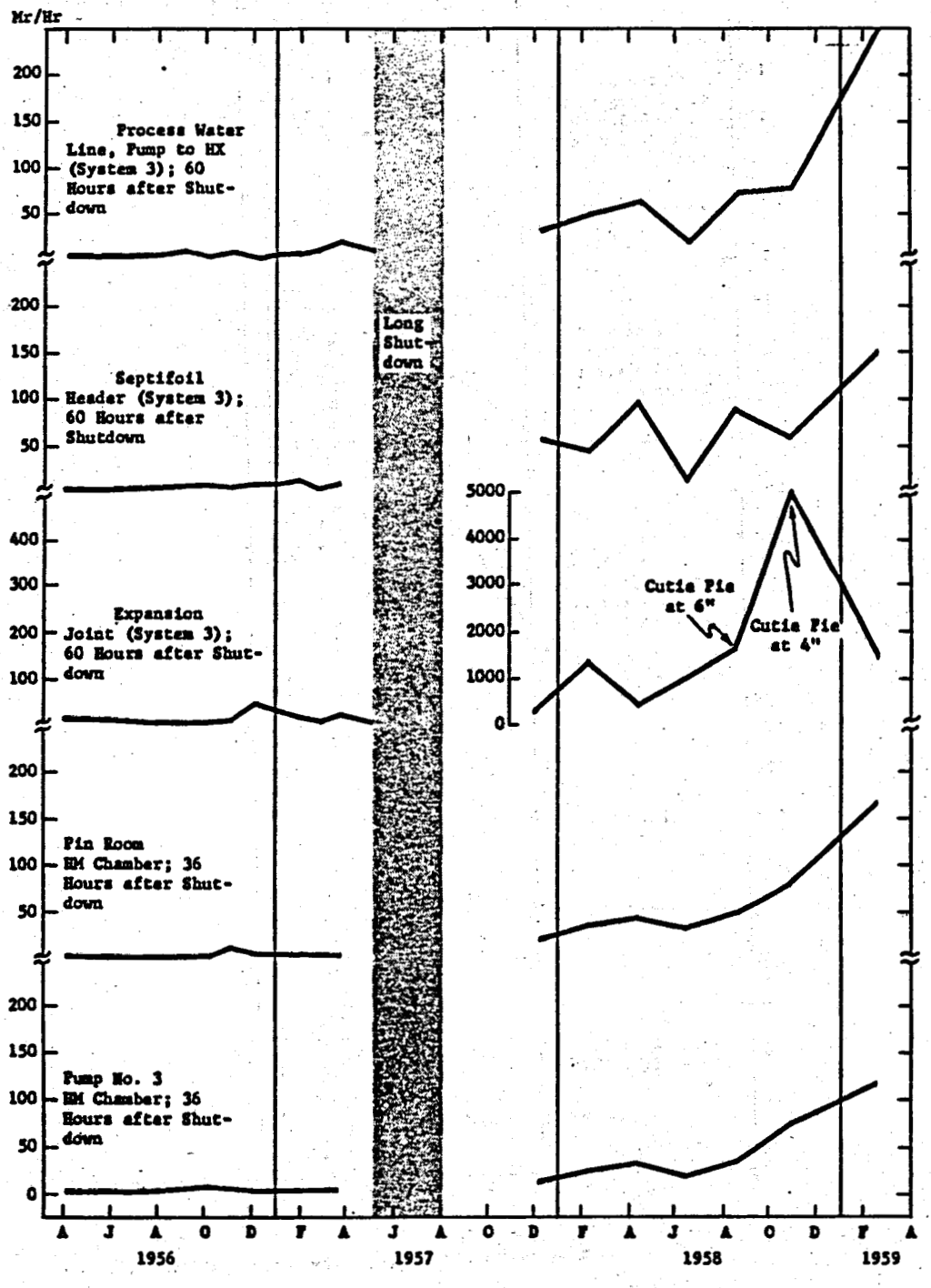

Figure RP-17. Activity Information, K Area

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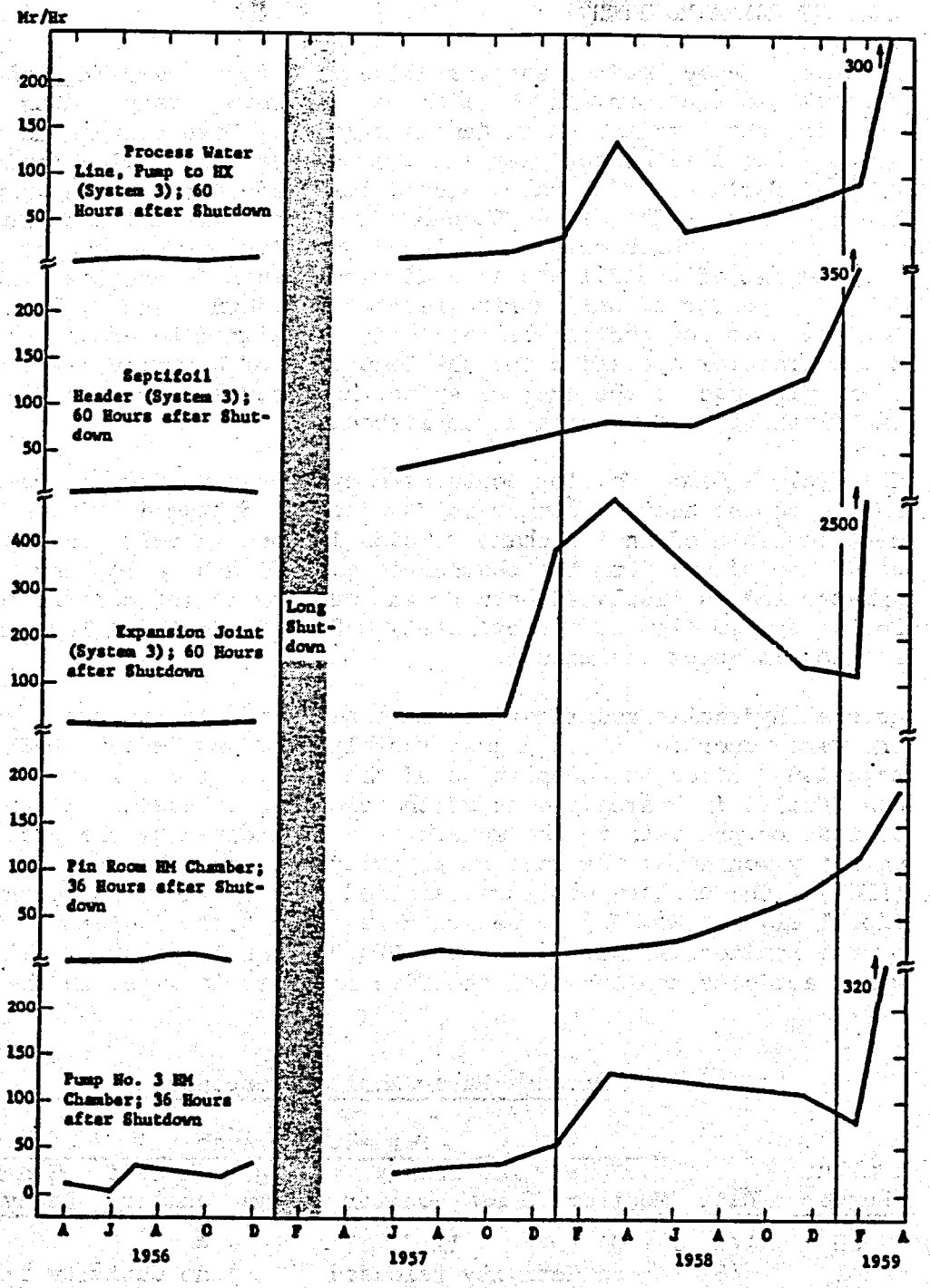

Figure RP-18. Activity Information, C Area

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CYCLONE GAMMA MONITOR

A Cyclone Gamma Monitor was installed in P Area under TA 1-538 during the P:13-14 shutdown (March, 1958) and is being evaluated as a replacement for the Single Channel Gamma Monitors. Both systems monitor the gross process water activity and are intended to provide rapid indication of fuel failures which have relased large amounts of fission products to the moderator. The Single Channel Gamma Monitors utilize porous stainless steel filters which must be replaced periodically because of pluggage. Replacement of the filters is a difficult maintenance task (SWP) and is normally performed only during reactor shutdown. The Cyclone Gamma Monitor does not contain filters and is designed to enable backflushing during reactor operation for the reduction of background activity. However, pluggage of the cyclone system has been a problem and effective backflushing has not been accomplished.

The Cyclone Gamma Monitor contains a cyclone settling pot which is connected to the heat exchanger inlet line for System 4. The sensing element consists of an ion chamber which is located near the settling chamber. The signal from the chamber is amplified by a Beckman micromicroammeter and is displayed on a Brown recorder in the central control room. Normal system flow is approximately 0.25 gpm and the delay time from the reactor is about one minute.

Several hydraulic modifications have been made to the prototype system and were reported in the August Monthly Progress Report, DPSP 58-1-8, page 145. Since the completion of the modifications, five fuel elements have failed in P Area. None of the elements released a significant amount of fission products to the moderator as indicated by the process water activity monitors; the maximum activity increase observed was 11 percent (LEGM). The cyclone monitor responded to only the most recent failure (April 21, SRP No. 53). The Brown recorder trace indicating the response of the monitor is shown in figure RP-19. The response of all the process water activity monitors for the five failures is given in the following table.

Process Water Monitor Response

Failure Number	Date	Percent Increase			
		Cyclone Monitor	Low Energy Gamma Monitor	Single Channel Gamma Monitor	Ten-Minute Delay Monitor
35	1/1	0	11	4	0
41	3/6	no activity released		no activity released	
49	4/3	0	0	0	3
51	4/16	0	0	0	3.9
53	4/4	9.7	11	0	10.8

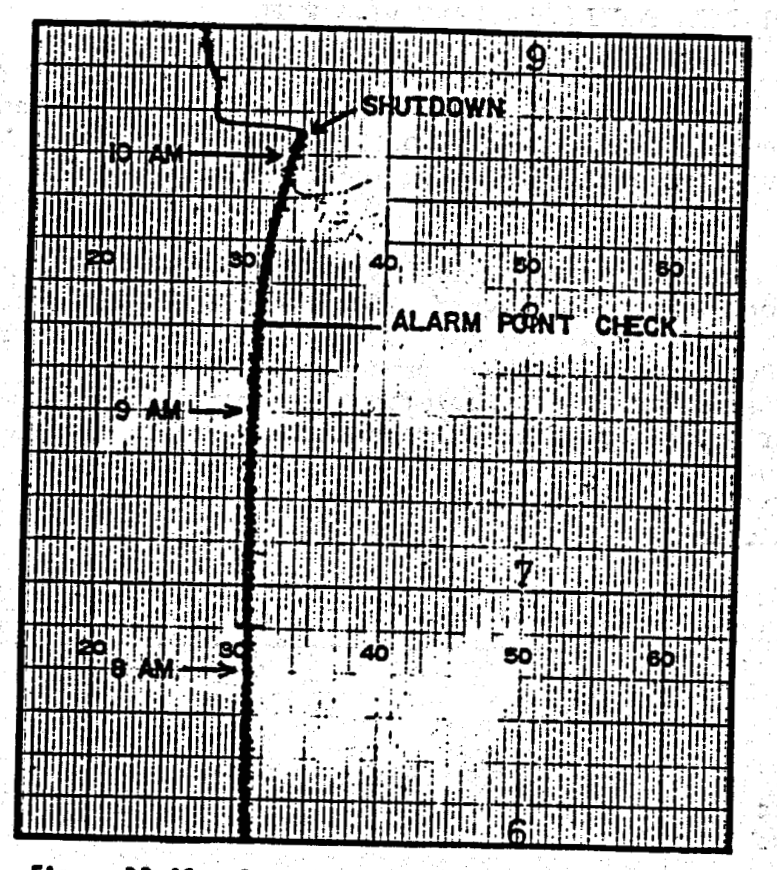

Figure RP-19. Response of Cyclone Gamma Monitor to Fuel Failure No. 53

The backflushing difficulties currently experienced with the cyclone monitor will be investigated during the forthcoming P:6-7 shutdown. The inability to backflush can result from (1) pluggage of the cyclone settling pot, (2) pluggage of the flush drain line, or (3) malfunction of one or more valves. If the investigation indicates that turbidity has caused the pluggage of the system, a larger cyclone settling pot and increased diameter hydraulic tubing will be tested.

FOOD IRRADIATION FACILITY - MARK VI-J FUEL

The food irradiation facility is designed to use Mark VI fuel elements as a gamma source. These elements are arranged in two concentric rings of nine elements each. Because of scheduled operations, Mark VI elements will no longer be available. Criticality considerations preclude the use of 18 Mark VI-J elements without modifications to the food machine.

A possible means of adapting the machine to the use of Mark VI-J elements is to use only a single ring of nine elements. Calculations were made to determine the gamma dose rates for 9 Mark VI-J elements after an exposure of 80%. These data are presented in figure RP-20 as a function of time after shutdown. Calculated and experimental values for 18 Mark VI elements after an exposure of approximately 40% are shown for comparison. The ratio of the dose rate from 9 Mark VI-J elements, to the dose rate from 18 Mark VI-J elements, determined from figure RP-20 are summarized in the following table.

Time After Shutdown	Gamma Dose Rate Ratio: 9 Mark VI-J Elements (one ring) / 18 Mark VI (two rings)
1 day	0.81
1 week	0.90
2 weeks	0.94
4 weeks	1.00
2 months	1.09
3 months	1.14

SHIELDING CALCULATIONS FOR PTPR

Calculations were made to determine the concrete shield heat load from capture gammas in the thermal shield and from pile gammas for several shield thicknesses and compositions. Results of the calculations are shown in figure RP-21 and in the following table.

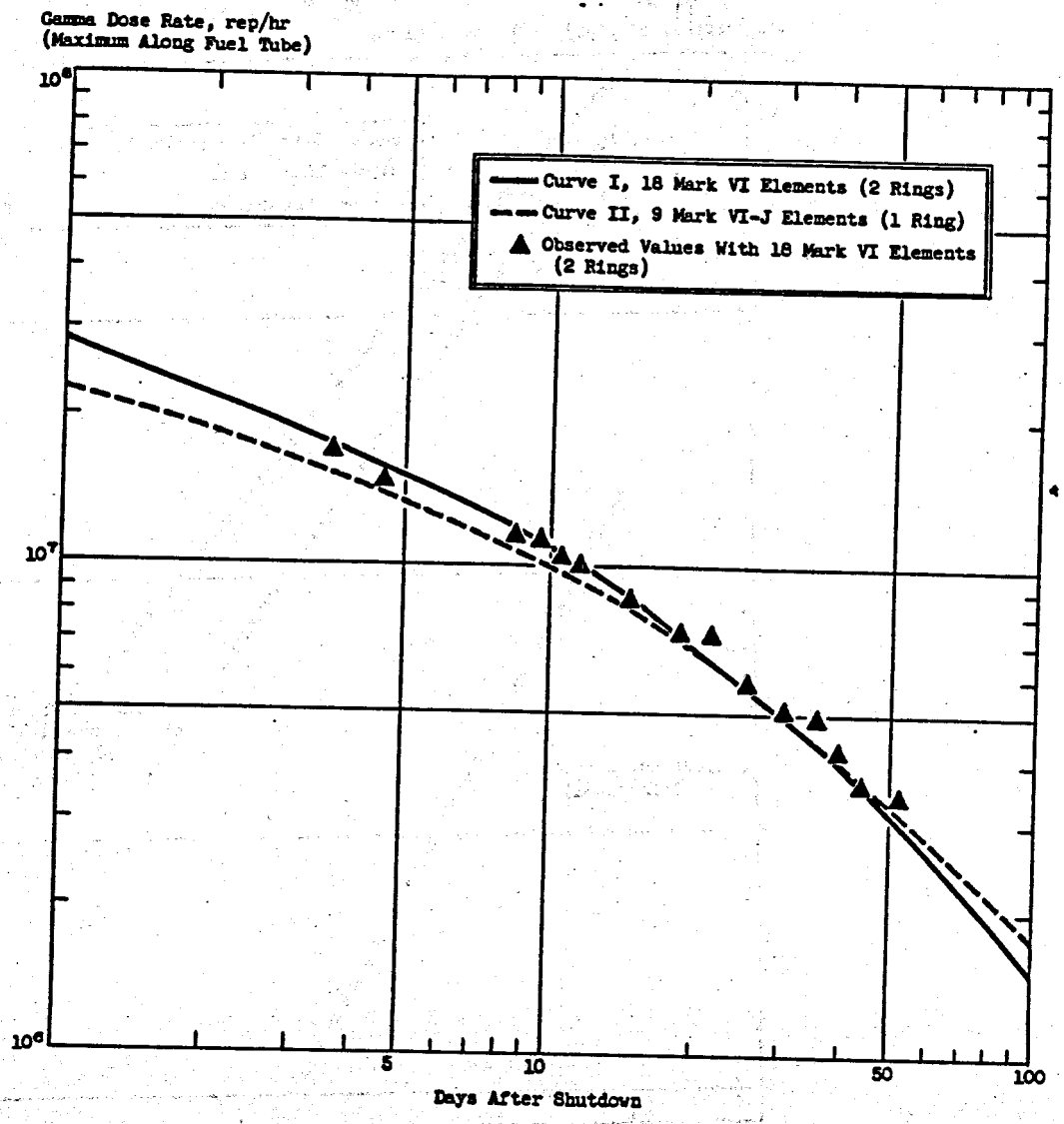

Figure RP-20. Predicted and Observed Gamma Dose Rates in Food Irradiation Machine

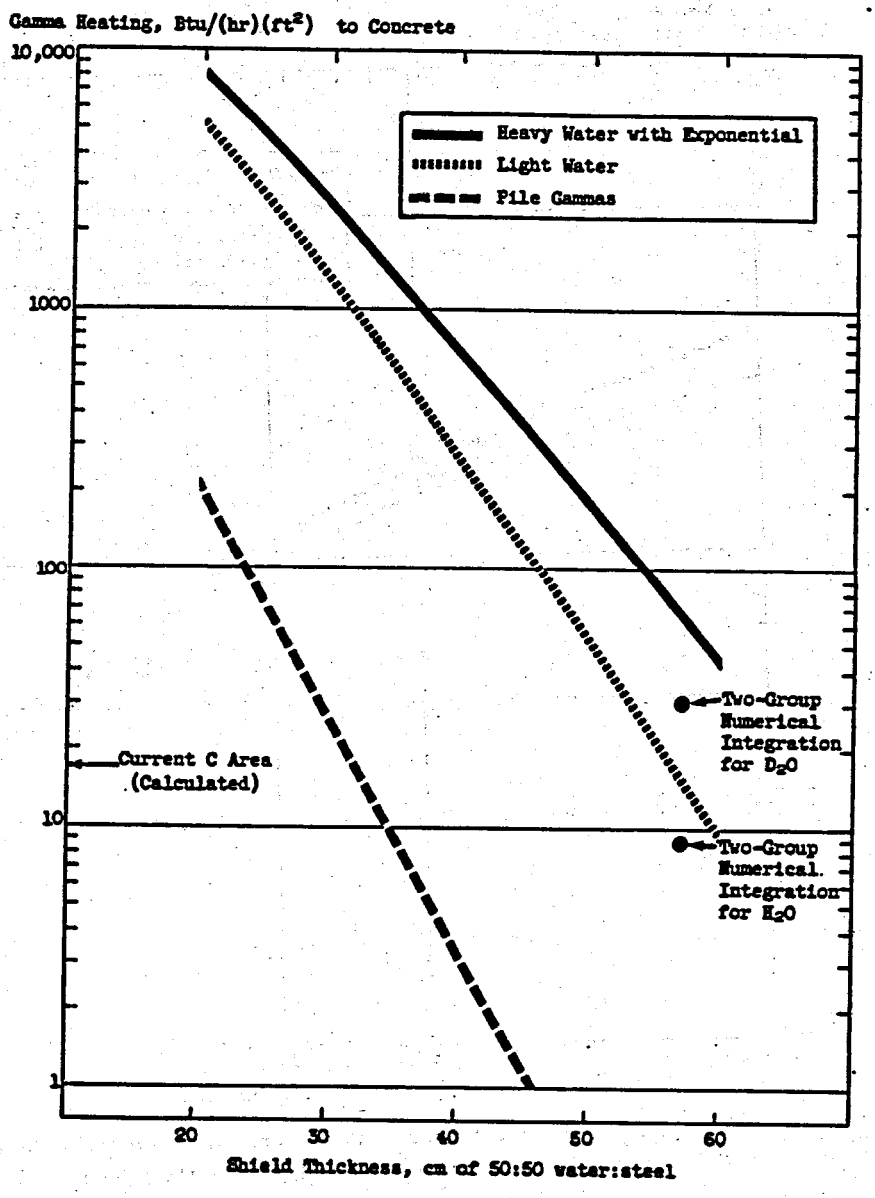

Figure RP-21. Gamma Heating of Concrete as a Function of Thermal Shield Thickness

Shield Composition	Heat to Concrete, (Btu/(hr)(ft^2)
• A 57 cm thick homogeneous mixture of 50:50 stainless steel:H_2O with a 43 cm reflector.	9
• A 57 cm thick homogeneous mixture of 50:50 stainless steel:D_2O with a 43 cm reflector.	31
• An interpolation of the 57 cm D_2O case above to 50 cm thick.	63
• A 50 cm thick homogeneous mixture of 50:50 stainless steel:D_2O with 40 cm reflector and a 3-inch-thick producer tube region just inside the inner shield wall consisting of a single ring of 1-inch-diameter 4.5% Li-Al rods with a 3-inch center-to-center spacing. ($\Sigma_{a_{eff}}$ of producer tube region was .0312 cm^{-1}.)	29
• A 50 cm thick 50:50 stainless steel:D_2O mixture consisting of two solid 4.5-inch-thick stainless steel plates separated by 9 inches of D_2O, and a 3/4-inch stainless steel tank wall separated from the outer 4.5-inch plate by 1 inch of D_2O. Reflector thickness was 40 cm.	24
• Slab case above with 1% boron added to the second 4.5-inch steel plate.	18
• Slab case above with 1% boron added to the tank wall.	16

The results shown above are approximate because of the assumptions which were made to simplify the calculations. More accurate calculations will be made on the IBM 650 at SRL when suitable codes for gamma shielding problems are programed. One set of codes which will be investigated was devised by Alco Products, Inc. and is described in APAE-Memo-142.

A description of the method of each calculation follows.

Exponential Neutron Attenuation in Shield. 50:50 stainless steel:water mixtures of various thickness were assumed for D_2O and H_2O. Neutron attenuation used was based on PDP experimental data. Each neutron capture was assumed to produce 8 Mev of 7 Mev single-point spectrum gamma rays. Heat to the concrete is shown in figure RP-21 as a function of shield thickness. A reactor power of 200% was assumed.

Pile Gammas. 50:50 stainless steel:water mixtures of various thickness were assumed. Each fission was assumed to produce 21.75 Mev of 2.5 Mev single-point spectrum gamma rays. Heat to the concrete is shown in figure RP-21 as a function of shield thickness. Reactor power is 200%.

Two-Group Theory. Numerical integration using thermal flux values was calculated with the Lil Abner code on the IBM 650 as the source term. Flux plots used for the 57 cm D_2O and H_2O cases may be found in the Works Technical Monthly Progress Report for March, DPSP 59-1-3. Each neutron capture was assumed to produce 8 Mev of 7 Mev single-point spectrum gamma rays. Heat to concrete is shown in the preceding table for a reactor power of 200%. Heat produced by neutron capture in the concrete is not included although it becomes significant at neutron flux levels of greater than about 10^{10} $n/cm^2/sec$ in the concrete. Shield problems calculated by this method and results are shown in the preceding table.

MODERATOR CHEMISTRY

CHARACTERIZATION OF TURBID MODERATOR

Twenty-five additional moderator samples have been analyzed by the technique of filtration through Millipore paper described in last month's report (DPSP 59-1-3, p 141). In general these data confirmed the results reported last month, and have yielded additional results.

Particle Size Distribution. Twenty-five new samples showed the same median values (with same standard deviations) of the "average," "minimum," and "maximum" turbidity particle size as were reported last month. The particle size distribution of moderator turbidity below 4 microns is thus constant.

Data taken in P Area during a scram showed the same abrupt rise in the "large particle index" within a half hour after the scram that was reported last month for C Area. This indicates the appearance in the moderator of particles of size greater than 10 microns after the scram.

Amounts of Elements. Nineteen samples have been analyzed for content of elements by the filtration technique (including those reported last month). The amount of solids in these samples ranged from 14 ppm to 448 ppm. A plot of ppm of manganese in the solids fraction against ppm of solids gives points scattering about a straight line through the origin, along which the ratio (ppm of Mn)/(ppm of solids) is constant. Furthermore the spread of values of this ratio is substantially less than the spread of values of (ppm of Mn).[1]

For other elements the scatter is large (over 1 to 2 decades) so that existence of a straight line correlation is not at once evident. In some cases the scatter is significantly reduced by dividing by ppm of solids (indicating a partial correlation) while in others it is increased (as would be expected if there were no correlation).

The results of this type of treatment of the data are as follows:

Element →	Mn	Fe	Cr	Na	Cu	Mg
Grams/Kg of Solids	0.012	1.3	~0.048	≤0.076	0.1	low
Degree of Correlation	high	low	medium	low	none	none
% of Total Element in Solids Fraction	76	>93	>86	40	57	0

[1] If the spread is expressed by the ratio of highest to lowest value (ignoring the tails of the distribution) the spread of (ppm Mn)/(ppm solids) is less by a factor of 2.6 than the spread of (ppm Mn).

The elements B, Li, Ni, U are not ordinarily detected in turbid moderator. The Al content of the solids is too high to be measured by the spectrographic method in current use; substantially all (99.9% of it is found in the solid fraction.

The amounts of elements in the filtrate fraction are more or less independent of the amount of solids present, the values being as follows:

Element →	Al	Mg	Mn	Na	Cu	Fe	B	Co	Cr	Li	Ni	U
ppb	20	2	0.1	3	<2*	<1*	<10**	<2**	<5**	<2**	<2**	<50**

* Normally below the listed limit of detection in the filtrate.
** Always below the listed limit of detection in the filtrate.

Radioisotopes. Five samples were analyzed radiochemically using the filtration technique (including the two reported last month). These were barely sufficient to show the correlation between amount of activity carried by the solids and the amount of solids. The numerical values summarized below must be considered approximate.

Isotope →	Na-24	Mn-56	Fe-59	Co-60	Cr-51	Zr-95	0.5 Mev*
Cpm/Mg of Solids	-	600	5	12	2	10	~2.5
Degree of Correlation	-	med	med	high	low	none	none
% of Total on Solids	5	94	95	97	88	97	96
Cpm/Ml in Filtrate	high	30	30	30	30	20	30

* Calculated as Ru-103

The range of values for the filtrate (except Na-24) is rather low, and the values are independent of amount of solids. This may be the effect of corrosion in the radiochemical analysis but this is not likely.

Efficiency of Filtration Technique. All of the work reported above was done using a 1.2 micron-pore diameter Millipore filter. Some of the data suggested that this filter was not removing all of the solids. To check this possibility, aliquots of the same moderator sample (C Area, taken April 4; turbidity = 16 ppm) were filtered, one through 1.2 micron and one through 0.3 micron-pore diameter Millipore paper. The decay curve of a β-γ planchet of each of these two aliquots was determined. Analysis of these decay curves showed that

- The 0.3-Micron paper removes about half of the long-lived activity passed by the 1.2-micron paper
- The 0.3-micron paper passed about the same amount of Mn-56 and of Na-24 (or 15-hour half-life isotopes) as was passed by the 1.2-micron paper.

Deionizer Effluent. A pair of samples consisting of reactor effluent and deionizer effluent drawn nearly simultaneously from R Area and another pair from P Area were analyzed by the filtration technique.

The analysis for elements showed Cr, Cu, Mn and Fe to be contained in the solids fraction, which was not removed by the deionizers. The concentrations of remaining elements were too near the limits of detection to give any reliable indication of the effect of the deionizers on their concentration.

Radiochemical analysis of these samples gave the following decontamination factors, calculated as ratio of activity in the filtrate fraction of the deionizer effluent to that in the filtrate fraction of reactor effluent:

Isotope →	Na-24	Mn-56	Fe-59	Co-60	Cr-51	Zr-95	0.5 Mev
Decanlow Factor							
Present Data	424	47	>6	>>1	>4	~12	~25
Nonturbid Samples*	4000	43 & 2	5	12	-	-	-

* Data previously reported in RTR 85 (DPSP 58-814), based on two pairs of nonturbid samples (filtration technique not used).

Of the present data, only the value for Na-24 can be considered to be established with any precision. The remaining values serve only to indicate that there is substantial removal of radioisotopes by the deionizers.

CHLORIDES IN MODERATOR

The high initial corrosion rate of aluminum components as measured by deuterium evolution could conceivably result from presence of chlorides in the moderator, which cause pitting corrosion. The concentration of dissolved chloride has been determined during C&D shutdown and the first few days after startup, twice in L Area and once each in R, P and C. Samples were taken on at least 4 days for each shutdown.

For the L:4-5 shutdown, 14 samples were taken -- one per day starting at the beginning of the shutdown (April 2) and continuing until April 14, five days after full power was reached. The concentration of chloride remained less than the 0.1 ppm limit of detection throughout this period. The deuterium evolution rate showed the usual peak at startup (65 scf/day maximum on April 8 as compared to 36 scf/day on April 11).

Of the 21 samples obtained from the other four shutdowns, two each gave 0.3 and 0.2 ppm and one gave 0.1 ppm of chlorides. One gave 0.6 ppm, but a resample on the same day gave <0.1 ppm.

It therefore seems unlikely that chlorides dissolved in the moderator are contributing in any major way to the general occurence of high initial aluminum corrosion. Data so far accumulated do not rule out the possibility that it may contribute to increased corrosion in individual reactor cycles.

INLINE pH SYSTEM

The prototype inline pH system which is designed for continuous pH measurement of process water, was installed in Building 105-P (TA 1-678). A block diagram of the equipment as arranged for operation is shown in figure RP-22. The system consists of three parts: (1) flow assembly and electrodes, (2) pH meter, and (3) recorder. The flow assembly contains a glass pH electrode, a reservoir type reference electrode, and a thermo-compensator. The pH meter measures the potential between the glass electrode and the reference electrode. This potential varies with the pH of the process water circulating through the flow assembly. The thermo-compensator is a resistance thermometer which adjusts the pH meter reading to compensate for changes in the electrodes caused by temperature. The strip chart recorder provides a continuous record of pH. A diagram of the piping and equipment arrangement is shown in figure RP-23.

The pH meter was placed in operation on April 15. Measurements of process water pH since that time have been recorded during a reactor shutdown, a reactor startup and during normal reactor operation. The pH has varied from 6.8 during shutdown to 7.8 during normal reactor operation. Because the inline pH system has been in operation only 10 days, insufficient results have been collected to determine upper and lower limits of process water pH and the variation with reactor parameters. These factors will be determined as more experience is gained.

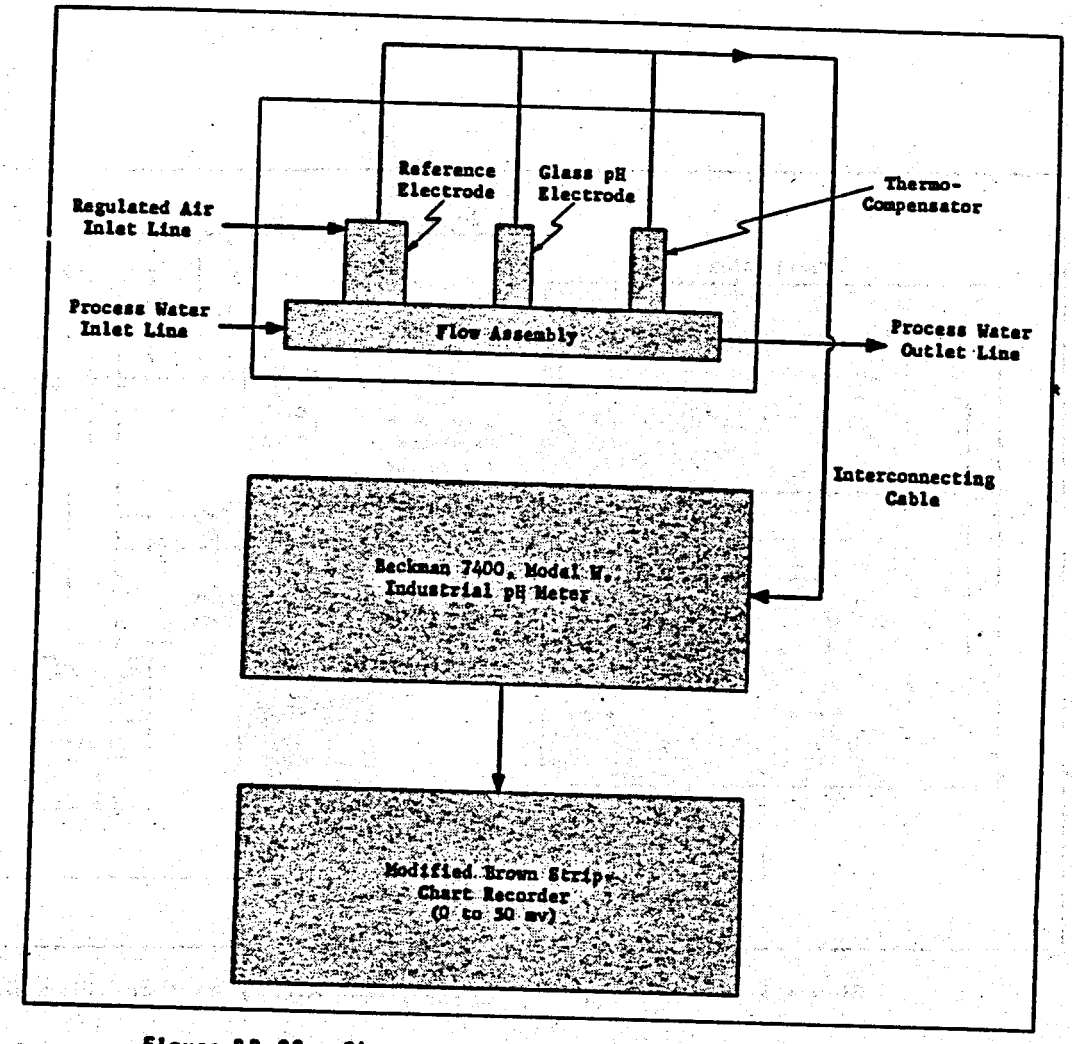

Figure RP-22. Elementary System Block Diagram of P-Area In-Line pH Meter

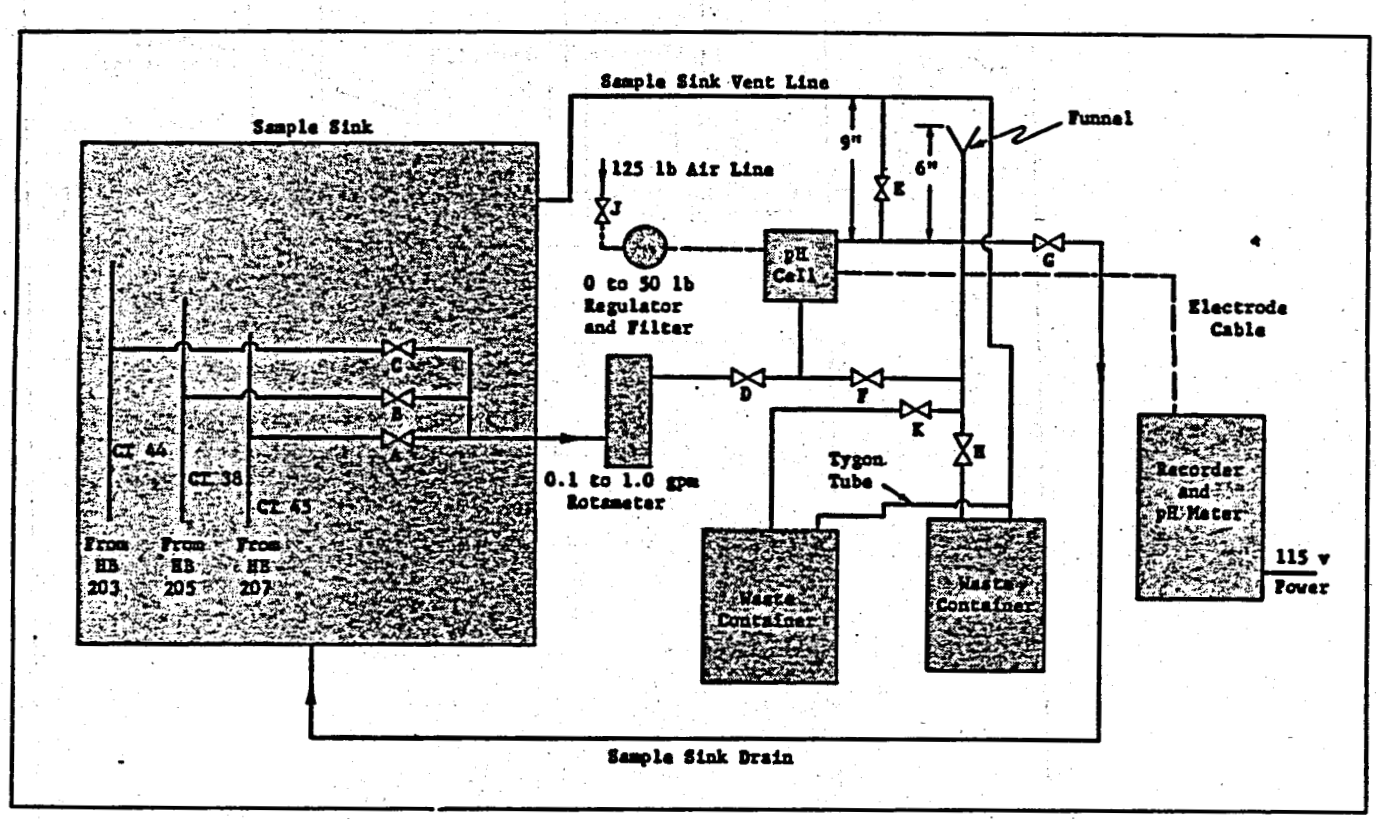

Figure RP-23. pH System -- Equipment Arrangement and Flow Diagram

PRODUCTION PHYSICS

TRITIUM PRODUCTION

Mark VI-J Target Production Rate. About 34% of the targets from the K-2 discharge have been processed through the 200 Area. A tritium production rate of was derived from the He-4 yield of this material. [The predicted value, given in DPSP 58-1393 (RTM 1684), was] The measured tritium yield was less than (The Material Balance Ratio was) The Separations Technology Section has postulated that the missing tritium is trapped in the form of water, and can be recovered.

Comparison of Mark VI and Mark VI-J. The measured Mark VI-J target production rate may be combined with other recent observations to provide a realistic estimate of the advantage of Mark VI-J operation over Mark VI operation. Blanketed, unspiked charges operating at 80% power, to exposure levels of 42% (Mark VI) and 72% (Mark VI-J) are assumed for the comparison. The tritium production rates are listed in the following table.

Source	Rate, mg/MWD	
	Mark VI	Mark VI-J
Targets		
Control Rods		
Blanket		
Total		

Both target production rates are measured values. The control rod production rate for Mark VI was derived from the curves given in DPSP 58-1138 (RTM 1621), which agree with 200-Area measurements. The control rod production rate for Mark VI-J was derived from the average reactivity held in rods during K-2. The two blanket production rates are consistent theoretical estimates, supplied by SRL. (Neither has been measured.)

The K-2 cycle exposure was 72%. If this is assumed to be the standard Mark VI-J cycle exposure, then Mark VI-J operation would generate times as many MWD per year as Mark VI operation at 42% per cycle. The net advantage of Mark VI-J over Mark VI may therefore be summarized as follows:

Item	Percent Gain
gm/MWD	
MWD/year	
gm/year	

PLUTONIUM PRODUCTION

Total Production in Natural Uranium Charges. Natural uranium charges produce tritium (in control rods and spikes) as well as plutonium. Both products must be included to provide a useful index of neutron economy. One such index is the total production rate, which may be expressed in units of grams of plutonium per reactor MWD (by equating an atom of tritium to an atom of plutonium). Measured values of this total production rate have been derived from Accountability records to indicate the changes in neutron economy which have taken place during the history of SRP reactors, and to compare the measured rates with what should be expected theoretically. Past production was divided into three categories: low ngs operations, high g/T operations (except for Mark VII-A charges), and those Mark VII-A charges which had been processed in the 200 Area through February 1959. The measured total production rates are listed below.

Category	Fuel	Rate, g/MWD
Low ngs	Mark I, VII	
High g/T	Mark I, VII	
Current	Mark VII-A	

The measured rate for low ngs operation is about higher than the current theory would predict. The high g/T (I, VII) rate agrees with theory. The Mark VII-A rate is about than the theoretical rate. In each case, the comparison with theory is based entirely on plutonium production, because empirical rates are used to predict tritium production. The comparison is considered valid because plutonium represents about of the total production from natural uranium charges. The decrease in total production rate, from the early low ngs operations to the present, is attributable primarily to the higher specific exposure in current charges. Details are given in DPSP 59-1005.

Revisions of Plutonium Production Rates.

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PURSUANT TO SECTION 148, ATOMIC ENERGY ACT OF 1954, AS AMENDED

AND

DEPARTMENT OF ENERGY REGULATION 10 C.F.R. 1017

PAGE DELETED

PURSUANT TO SECTION 148, ATOMIC ENERGY ACT OF 1954, AS AMENDED

AND

DEPARTMENT OF ENERGY REGULATION 10 C.F.R. 1017

Glossary of Terms

FUEL ASSEMBLIES

Mark I	Quatrefoil loaded with solid slugs of natural uranium.
Mark V	Two concentric tubular fuel elements of natural uranium. Series flow of coolant (down the outer annulus, up the intermediate annulus, then down the center of the assembly).
Mark V-A	Two concentric columns of individually clad $8\frac{1}{2}$-inch-long natural uranium fuel tubes. [Tubes clad at SRP (triple-dip bonding) are designated Mark V-A-SRP, and tubes clad at Sylcor (hot-press bonding) are designated Mark V-A-SC.]
Mark VI	Tubular fuel element of U^{235}-Al alloy with an internal column of solid Li-Al target slugs.
Mark VI-J	Similar to Mark VI except that: (1) the fuel alloy has a higher uranium content, (2) hollow-core, larger diameter target slugs are used, and (3) the basket tube and dead annulus are eliminated.
Mark VI-S	Similar to Mark VI except that the target alloy has a lower lithium content. (This element is used to spike low-reactivity fuel charges.)
Mark VII	Quatrefoil loaded with hollow slugs of natural uranium. [Slugs canned at SRP (triple-dip bonding) are designated Mark VII-SRP, and slugs canned at Sylcor (hot-press bonding) are designated Mark VII-SC.]
Mark VII-A	Similar to Mark VII except that the hollow slugs and quatrefoil are larger to permit operation at the higher coolant flows provided by the Bingham pumps. [Slugs are designated Mark VII-A-SRP or Mark VII-A-SC, on the same basis as Mark VII slugs.]
SPR	Power reactor fuel element.
TFEN	Tubular fuel element, natural uranium (used for SRL test elements).

MISCELLANEOUS

CMX	Savannah River Laboratory development facility for 100 Areas.
HWCTR	Heavy Water Components Test Reactor.
PTPR	Pressurized Tube Production Reactor.
Strength	Ratio of the Li-6 content of a Li-Al alloy to the Li-6 content of an equal volume of Li-Al alloy containing 1.0 wt % natural lithium.
Sylcor	Sylvania-Corning Nuclear Corporation.

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Natural Uranium Processing

PROCESS STATUS

Solvent extraction of irradiated uranium in the jumbo mixer-settlers in F Area started on April 6 and continued intermittently, with numerous shutdowns in addition to the scheduled weekend shutdowns. Performance was generally irregular for several hours following startups until steady state operation was attained; losses and decontamination were satisfactory during all periods of steady state operation. First cycle rate varied between five and ten batches per day with better control of flow rates and of bank stability at higher rates. By the end of the month, techniques were developed which to some extent minimized the startup problems.

Plutonium refluxed in the 1B bank on one occasion until it was estimated that had accumulated, and the bank was stripped. The buildup had persisted for several hours in spite of large increases in the amount of ferrous sulfamate fed to the bank. Procedures were modified to provide additional protection from overadjustment of 1AF with nitrite which could have caused the reflux. A loss of of plutonium to the 2AW occurred, presumably because insufficient sodium nitrite was available for 2AF adjustment. Both occurrences were recorded on charts showing neutron readings.

Tests of the bismuth phosphate process for recovery of neptunium from high activity waste were successfully completed in H Area, and solvent extraction of neptunium from solutions of bismuth phosphate product cakes was demonstrated in the second plutonium cycle. Recycle of neptunium through the second cycle is in progress, to obtain the required decontamination before transfer to B-Line.

FEED MATERIAL

Characteristics of Feed

Reactor Charge →	R-15	R-2	R-2	P-3
Composition, wt %				
Mark I	-	100	-	-
Mark VII	100	-	-	-
Mark VII-A	-	-	100	100
Plutonium Content % of Estimated Content				
Processing Period	3/24 - 4/3	4/3 - 4/10	4/8 - **	4/10 - **
Cooling Time,* days	330	144	149	130
Raw Metal Solution,* γ c/m/ml				
Zr	0.18×10^8	1.2×10^9	2.3×10^9	2.8×10^9
Nb	0.36×10^8	2.2×10^9	4.1×10^9	4.7×10^9
Ru	0.10×10^8	2.0×10^8	4.4×10^8	6.5×10^8
I	nil	7.3×10^2	2.7×10^4	1.7×10^5
U-237	nil	3	12	70
Gross	0.55×10^8	3.6×10^8	6.8×10^8	8.3×10^8

* At start of processing period.
** Still in process.

DISSOLVING

Hydrostatic tests early in the report period showed that the column gasket was intact on each dissolver. The problem of low dissolver vacuum was corrected by replacing the gaskets in the off-gas jumpers. Dissolver 6.1D currently is processing P-3 material while 6.4D is processing R-2 dingots under Test Authorization 2-228.

Material balance calculations based on composite raw metal solution acidities indicated possible acid deficiency on three dissolver cuts and low acid (less than 0.8%) on two others during the month. Since the precision of the acid analysis in raw metal solution is only about 80% and an individual cut may represent only 15% to 20% of the raw metal solution composite, material balance calculations are subject to considerable error. The following precautions were taken to prevent the loss of a dangerous amount of plutonium to the head end centrifuge cake:

- Transfers to head end were limited to no more than 1200 grams of plutonium from the questionable cuts.
- Analyses indicated that the plutonium in the raw metal solution had not hydrolyzed to any great extent.
- In one instance, comparison showed the ratio of plutonium to uranium in the centrifuge run tank to be essentially identical to that in the raw metal solution hold tank. This indicated that no significant amount of plutonium had been lost to the centrifuge cake.

Dissolving end points were revised as a precaution against potentially acid-deficient cuts. The final specific gravities for the first and second cuts are now 1.70 and 1.71, respectively; the former values were 1.715 and 1.730. The Laboratories Section developed a routine control analysis to determine the amount of hydrolyzed plutonium in the raw metal solution by comparing the amount of TTA-extractable plutonium with the gross alpha count corrected for uranium and americium. Experiments with hydrolyzed plutonium [determined spectrophotometrically to be free from Pu (IV)] show that only 10% to 20% of the hydrolyzed material is extracted by TTA. Although the analysis is crude and may indicate hydrolysis where none exists, it should reliably detect dangerous amounts of hydrolysis (ie, greater than 33%).

The uncertainties in the material balance calculations for the acidity and uranium content of an individual cut also affect the regulation of dissolver heel and cut size. The amount of acid used per dissolver cut was increased 5% to obtain larger cuts and to limit excessive heels.

Analyses of coating waste solutions were very erratic during the month with uranium values ranging between 0.19 and 6.9 g/ℓ and plutonium between With time as a parameter, there is no apparent trend in the analyses, and for individual batches no correlation between uranium and plutonium concentrations. Average losses to coating waste for uranium and plutonium for the report period were 0.24 and respectively.

Dissolving Summary, F Area

Dissolver →	6.1D	6.4D
First Cut		
Uranium Dissolved, batches/cut	3.91	4.14
Avg Cut Time, hours	11.65	12.81
Avg Cut Acidity, % HNO_3	0.76	1.16
Acid Economy, mols HNO_3/mol U	4.68	4.27
Second Cut		
Uranium Dissolved, batches/cut	3.70	3.80
Avg Cut Time, hours	12.5	14.08
Avg Cut Acidity, % HNO_3	3.52	2.06
Acid Economy, mols HNO_3/mol U	4.61	4.61
Raw Metal Solution		
Avg Acidity, % HNO_3	1.77	
Acidity Range, % HNO_3	0.91 - 4.40	

HEAD END

A new jumper was installed on the precipitation tank to allow permanganate to be added at the tip of the agitator blade, as was successfully done in H Area. Visual inspection of the centrifuge effluent during one cold uranium run with the new jumper and a feed rate of about 100 lb/min showed no signs of cake carryover.

High range ΔP transmitters were installed in the 11-1C skimmer control system to permit use of the normal skimmer controls at the higher pressures required to move the skimmers. The higher pressures are related to the use of stronger return springs in the actuator mechanism. Operation thus far appears satisfactory.

SOLVENT EXTRACTION

The first cycle inventory of cold uranium was depleted April 3, and irradiated uranium was processed in the first cycle starting April 6. The gamma activity of the 1AF increased gradually from a startup value of 6.4×10^7 c/m/ml to the current value of 1.2×10^8 c/m/ml as shorter-cooled material was processed. First cycle end stream losses and decontamination factors were satisfactory during all periods of steady state operation despite initial poor control of saturation in the 1A bank. Decontamination factors across the first cycle decreased from the unusually high initial values, as fission product activities built up throughout the process, to values more indicative of normal operation.

Considerable difficulty was experienced throughout the report period with the migration of uranium toward the aqueous-out ends of the 1A, 1B, and 1D banks during weekend shutdowns; this migration appears responsible for the "aqueous block" of the 1A bank reported last month as well as blockage of the 1B bank during the early part of this month. Following hot startup on April 6, an excessive amount of uranium was contained in the 1BP, and 100 pounds of uranium was lost to the 1DW; as a result of the uranium in the 1BP, about 3000 pounds of 2BP was recycled to head end; the ratio of uranium to plutonium was 4.6:1 in the recycled 2BP compared to a Technical Standard limit of 1:1 and a normal value of less than 0.02:1. It has been postulated that the uranium bearing aqueous phase migrates through the interstage weep holes during shutdown periods as a result of hydraulic pressure gradients within the banks.

Prior to the shutdown on April 10, the uranium and plutonium were stripped from the 1A and 1B banks to prevent startup losses to the 1AW and recurrence of uranium contamination in the 1BP; the 1BX acidity was increased to 7.5% to compensate for the acid lost because of the absence of 1AP during the flush. No uranium or plutonium losses were detected during shutdown and immediately following startup on April 13 but the gamma activity of the 1AP and 1BP increased about 400-fold, presumably due to low startup saturation in the 1A bank and the presence of fission products in the scrub stages following the flush. A similar shutdown procedure was used on April 17 except that the 1A bank was only partially stripped. Fission product contamination of the 1BP relative to the 1AF after this startup was about the same as that from the previous, more extensive stripping of the 1A bank.

Uranium migration data obtained during 1D bank shutdowns (summarized in the following table) indicate that uranium is partially stripped from the migrating aqueous when operating the impellers after shutdown, resulting in lower losses at startup. Further testing is planned to leave the impellers running following high saturation conditions at shutdown since it is not yet established how much the migration of uranium is affected by saturation and how much by running the impellers. Use of the technique on the 1A bank is planned.

Migration of Uranium in the 1D Bank

Shutdown Date	Stage 11 Sp Gr*	Stage 18 (1DW), g/ℓ				Total Hours Shutdown	Hrs Impellers Operated After Shutdown**
		Immediately After Shutdown	6 Hours After Shutdown	12 Hours After Shutdown	Prior to Following Startup		
4/3	1.0	-	-	-	13.5	54	0
4/8	1.25	1.8×10^{-3}	49.4	-	49.4	6	0
4/9†	1.27	1.9×10^{-3}	121	137	165	17	0
4/10	1.05	1.8×10^{-3}	-	2.7×10^{-3}	0.8	52	12
4/14	1.10	1.4×10^{-3}	80×10^{-3}	0.2	2.7	28	16
4/17	1.05	0.8×10^{-3}	-	-	2.3	54	12

* Specific gravity is indicative of the relative saturation at shutdown; the normal desired operating condition corresponds to a specific gravity of 1.25 in stage 11.

** Impellers at 1/2 normal speed.

† Aqueous outlet stage interface raised prior to shutdown.

On April 14, approximately 98% of the plutonium fed to the 1B bank refluxed for a 37-hour period, ending with the shutdown of the first cycle for a 1B bank flush. Approximately of plutonium collected in the bank before shutdown. The reflux appeared caused by destruction of the ferrous ion in the bank presumably as a result of nitrous acid extraction in the 1A bank. The exact source of the nitrous acid is unknown; however, the 1AF adjustment procedure was revised to prevent accidental addition of nitrite to the 1AF after the usual simmer period.

Approximately of plutonium were lost to the 2AW while feeding a batch of 2AF that may not have been properly adjusted with sodium nitrite. This batch of feed was adjusted in a tank not used previously and all or part of the adjustment material (~100 lb) may have filled the piping from the head tank to the canyon.

Initial checkout of the 1B bank neutron monitor was still in progress at the time the 1B reflux started so that reflux conditions were not immediately recognized. However, subsequent close review of the monitor charts showed the progress of the inventory buildup in the 1B bank. Although recent performance of the dual-channel 1B monitor has been satisfactory, minor modifications still are needed to decrease the degree of fluctuation in the recorded count rate.

The original F-Area 2A bank monitor was used for startup because the new dualchannel monitor was not ready in time. The old monitor showed good response to the large 2AW loss reported above by a sharp drop in indicated neutron count as the bank inventory was depleted. This monitor later failed and was replaced with the old H-Area 2A bank monitor since the new dual-channel monitor was not operable upon installation. The "tube-train," as designed, apparently resulted in excessive noise level at low neutron count rates. Redesign of the tube train is in progress.

Process Performance

Period →	4/6 - 4/8	4/9 - 4/15	4/16 - 4/22
Decontamination Factors			
RM/1EU ($\times 10^6$)			
Gross γ	-	7.4	5.1
Zr-Nb	-	9.5	-
Ru	-	1.7	-
RM/1AF			
Gross γ	4.7	3.7	4.0
Zr-Nb	-	5.2	5.6
Ru	-	0.7	0.8
1AF/1DF ($\times 10^3$)			
Gross γ	100	46.4	0.52
Zr-Nb	-	35.0	0.39
Ru	-	97.0	1.36
1DF/1EU ($\times 10^2$)			
Gross γ	0.02	0.22	15
Zr-Nb	-	0.25	-
Ru	-	0.15	-
RM/2BP ($\times 10^6$)			
Gross γ*	-	38	16
Zr-Nb	-	46.2	18.5
Ru	-	10.2	5.4
1AF/1BP ($\times 10^3$)			
Gross γ	54.5	61.9**	1.93†
Zr-Nb	-	-	1.43
Ru	-	-	5.96
1BP/2BP ($\times 10^3$)			
Gross γ*	0.02	7.7	8.8
Zr-Nb	-	-	10.9
Ru	-	-	0.44
End Stream Quality			
1EU, c/m/ml			
Gross γ	80	150	220
Zr-Nb	-	100	-
Ru	-	50	-
2BP, c/m/mg			
Gross γ*	752	592	962
Zr-Nb	592	438	768
Ru	60	154	194
Pu to Raffinate, %			
1AW			
1DW			
2AW			

* Corrected for Pu gamma activity
** Omits material produced immediately after startup and during 1B bank flush.
† Omits material produced during first 24 hours after startup.

Flowsheet and Operating Conditions, F Area
(Status on April 23)

Bank	Impeller Speed, rpm	Stream	Component	Nominal Wt %	Flow Ratio by Wt	Rate,* Batches/Day	Gamma Activity, c/m/ml
1A	250*	1AF	UN	38.9		10*	
			HNO_3	5.0	1.0		
		1AX	TBP	30.0	2.08*		
		1AS	HNO_3	17.2	0.553		
1B	250*	1BX	HNO_3	0.63	0.354*		
			Fe Sulfamate	4.2*			
		1BS	TBP	30.0	0.496*		
		1BP	-	-	-		
1C	225*	1CX	HNO_3	0.063	3.83		
1D	200	1DF	UN	39.8		10*	
			HNO_3	0.44			
			Fe Sulfamate	0.065	1.0		
		1DX	TBP	30.0	1.890*		
		1DS	H_2O	100	0.090*		
		1DS'	HNO_3	17.2	0.193*		
1E	225	1EX	HNO_3	0.06	3.600		
		1EU	-	-	-		
2A	350	2AF	HNO_3	23.2	1.00	15*	
		2AX	TBP	33.0	0.333		
		2AS	HNO_3	3.7	0.315		
2B	233	2BX	Hydroxylamine Sulfate	0.41	0.250		
		2BP	-	-	-		

* Varied during month.

Glossary of Terms - Separations Areas

RM - Dissolver Solution
1AF - First Cycle Feed
1AX - First Cycle Organic Extractant
1AS - First Cycle Aqueous Scrub
1AW - First Cycle Aqueous Waste
1AP - First Cycle Extraction Bank Product
1BX - Partitioning Bank Aqueous Extractant
1BS - Partitioning Bank Organic Scrub
1BP - First Cycle Aqueous Plutonium Product
1BU - Partitioning Bank Organic Uranium Stream
1CX - First Cycle Aqueous Strip
1CU - First Cycle Aqueous Uranium Product
1CW - First Cycle Spent Solvent
2AF - Second Plutonium Cycle Feed
2AX - Second Plutonium Cycle Organic Extractant
2AS - Second Plutonium Cycle Aqueous Scrub
2AW - Second Plutonium Cycle Aqueous Waste
2BX - Second Plutonium Cycle Aqueous Strip
2BP - Second Plutonium Cycle Aqueous Product
2BW - Second Plutonium Cycle Spent Solvent
1DF - Second Uranium Cycle Feed
1DX - Second Uranium Cycle Organic Extractant
1DS - Second Uranium Cycle Aqueous Scrub (Stage One)
1DS' - Second Uranium Cycle Acid Scrub (Stage Four)
1DW - Second Uranium Cycle Aqueous Waste
1EX - Second Uranium Cycle Aqueous Strip
1EU - Second Uranium Cycle Aqueous Product
1EW - Second Uranium Cycle Spent Solvent

LOSSES

Average Waste Losses (To Building 241 Tanks)

	Pu, %	U, %
From Head End		0.03
From Low Activity Waste		0.77
From High Activity Waste		0.29
From Rerun		0

SOLVENT RECOVERY

The F-Area second plutonium and second uranium cycle solvent inventories were increased with additions of 24,000 pounds and 17,000 pounds of H-Area second uranium cycle solvent on April 10 and April 13, respectively. The solvent was added to the acid washer in each system in the Bldg 211 area. Laboratory examination indicated that the H-Area solvent was equivalent to present F-Area solvent in terms of plutonium and uranium retention.

Activity in Process Solvent

Period →	4/2 - 4/8	4/9 - 4/15	4/16 - 4/22
1st Cycle, Unwashed*			
Gross γ, c/m/ml	-	-	-
Zr-Nb, %	-	-	-
Ru, %	-	-	-
1st Cycle, Washed			
Gross γ, c/m/ml	210	440	1700
Zr-Nb γ,%	34	-	15
Ru γ,%	66	-	85
TBP, vol %	31.6	31.0	30.9
Z Number	<5	-	-
2nd U, Washed			
Gross γ, c/m/ml	20	130	270
TBP, vol %	30.8	30.6	31.3
Z Number	<5	-	-
2nd Pu, Washed			
γ, c/m/ml	25	470	850
TBP, vol %	33.8	32.5	32.9
Z Number	<5	-	-

* 1CW sampler has not yet been installed.

WASTE DISPOSAL

Over-all operation of the continuous evaporators in both the high activity and low activity waste systems continues to be good. The high waste volumes reported for high activity waste are attributed primarily to the intentional low volume reduction factor (a ratio of 16 volumes of feed to one volume bottoms) used to check the decontamination performance of this system. At the end of the report period, fission product equilibrium was attained with condensate gamma activities in the range of 500 to 1500 c/m/ml; the system gross gamma decontamination factor (feed to condensate) is estimated to be 5×10^5, which is lower than anticipated and essentially the same as that obtained with the old batch type evaporators.

The contents of the low activity waste evaporator were transferred to re-run for product recovery during the period and this system has not yet regained its equilibrium concentrations.

Eight grams of plutonium were discarded to waste through the laboratory waste system this month.

SHIPMENT OF HAW TO ORNL

In connection with the development of a new order-of-magnitude estimate for a minimum facility to ship acidic high activity waste to ORNL, the following requirements were included in the scope:

- Isolation of an existing 5500-gallon tank in the Building 211-F laboratory waste pit from the rest of the laboratory waste tankage and piping.
- Installation of a new 5500-gallon storage tank in the spare cell adjoining the existing tank (one of the two 5500-gallon tanks would serve for collection and storage while shipments were being made from the other).
- Installation of a 100-gallon transfer tank equipped to receive waste by jet transfer from one of the 5500-gallon storage tanks.
- Equipment for transferring the waste from the 100-gallon tank through shielded lines to ORNL 500-gallon shielded flasks on a truck-trailer.
- Installation of a covered loading shed and a concrete pad, with a drain and sump.
- Installation of 3-inch storage tank fill and return lines, approximately 400 feet total, from Section 18, Building 221-F hot canyon (lines to be provided with standard waste line encasement).
- Installation of a new waste pit sump with a jet to return only to one of the storage tanks.
- Installation of additional shielding, equivalent to about two feet of concrete, above the cells housing the storage and measuring tanks.

- Installation of a ruthenium trap and a waste tank type filter for the tank ventilation air, to reduce potential airborne activity discharge into the recycle vent system from the tanks.

An important consideration is that the operation of this facility for storing and shipping wastes must not interfere with or compromise the normal operation of the laboratory waste facility, or compromise any other production operation, (eg, by spreading contamination in the area).

It is estimated that the installation would permit shipment of waste with the desired 180 to 210 days' total decay time at an average rate of about 3500 gallons per month, assuming a total cooling period of 135 days by the time the waste is moved to the new facility. It is estimated that approximately six months would be required to complete the facilities after project authorization.

ACID RECOVERY

The gamma activity to the acid recovery unit, although low, has been increasing slowly during the period. Overhead condensate activities have increased from 0 to 890 β c/m/ml, while the recovered acid activities have reached 1000 γ c/m/ml. Fission product analyses indicate that 90% of the recovered acid activity is Zr-Nb, whereas the condensate activity is primarily I-131.

UNDERGROUND WASTE STORAGE

Status of Waste in Underground Tanks

Tank	Fill, inches	Maximum Temp, °C Sludge	Supernatant	Remarks
1-F	269.7	32	32	Cooling water through 1/8" orifice, all 36 coils.
2-F	267.2	28	26	Full flow of cooling water through one set of 17 vertical coils and two horizontal coils. All others have 1/8" orifices.
3-F	204.0	32	28	Full flow of cooling water through all 36 coils.
5-F	7.0	34	30	Full flow of cooling water through all 36 coils.
7-F	270.4	-	27	Low level waste. Cooling water through 1/8" orifice, all 36 coils.
8-F	209.5	-	29	Low level waste. Cooling water through 1/8" orifice, all 36 coils.

Building 241-F Cooling Water

	Maximum	Minimum	Specification
Na_2CrO_4, ppm	760	710	500 to 1000
pH	10.3	5.9	>7.2

WASTE TRANSFERS

Waste Volumes, F Area
(3/20 through 4/23)

Streams to Bldg 241	Transfers, gal/batch Actual	Flowsheet
High Level		
HAW Evaporator	189*	132
LAW Evaporator, Continuous	22**	51
LAW Evaporator, Batch	27.5	-
Head End	24.8	26
Rerun	14.4	-
Buildup in Bldg 241	216	-
Low Level		
Coating Waste	233†	187
GP Concentrate	9.5	-
Lab Waste Concentrate	9.0	-
Silica Gel	11††	~3
Buildup in Bldg 241	299	-

* A 43% increase above flowsheet attributed to concentration changes.

** Low volume resulted from transferring evaporator contents to rerun for product recovery.

† Contains 20 gal/B of B-Line recovery waste.

†† High volume due to citrate regeneration of silica gel column.

RELEASE OF RADIOACTIVITY TO THE ENVIRONMENT

Seepage Basins, F Area. The following table shows the activity discharged to the F-Area seepage basins.

Activity Discharged to Basins, 3/19 - 4/15	
Nonvolatile β, curies	2.18
I-131, curies	0.12
pH of Water Discarded During Month	1.8
Avg Seepage and Evaporation Rate, gal/day	44,000
Level in Basin No. 3 on 4/23, inches	74

Activity in Basins, 4/15	Basin No. 1	Basin No. 2	Basin No. 3
I-131, curies/cc	37.2×10^{-12}	4.0×10^{-12}	2.0×10^{-12}
Nonvolatile β, curies/cc	5.14×10^{-10}	4.61×10^{-10}	3.34×10^{-10}
Seepage Basin pH, 4/15	1.5	1.9	3.5

Stack Releases, F Area. Release of I-131 from the F-Area stack totaled 0.72 curie for the period April 3 through 15.

RERUN

At the end of the report period, 2700 pounds of uranium and of plutonium were in rerun. Rerun receipts for the month are shown in the following table.

Rerun Receipts and Volume of Waste to Building 241-F

Source	Reason	No. of Transfers	% of Total Wt Received (78,333 lb)	% of Rerun Waste to Bldg 241-F
Sumps	Process leaks, spills, flushes	5	23.4	100*
Solvent Recovery	Metal loss to 1CW	1	31.8	0
LAW Evaporator	Pu loss to 2AW	3		0
2nd Pu Cycle (10.5)	Aqueous waste from 2BW tank	2	1.1	0
2BP Run Tank	Organic accumulation	1	1.7	0
B-Line Waste	Off-standard 2BP solution	1	15.5	0

* Waste from preceding month.

Approximately 700 pounds of unirradiated uranium was returned to head end with no known adverse effects on March 26, following an inspection which showed the concentrated solution to be clear with no indication of suspended solids. This uranium had been received in rerun in an acidified carbonate wash solution from the first stage hot solvent washer and was concentrated by evaporation.

COLD FEED PREPARATION

The initial shipment of 50% ferrous sulfamate, after several months' storage, assayed only 23% ferrous sulfamate and was discolored and contained large amounts of yellow sludge. During this storage period, infrequent elevated temperatures and agitation were used to dissolve crystals; these conditions apparently contributed to oxidation of the ferrous sulfamate, which now appears unsuitable for use. A fresh shipment of 50% solution was diluted to 30% ferrous sulfamate and acidified to 0.5% HNO_3. No degradation of the diluted and acidified ferrous sulfamate has been observed since it was received on March 26. As further precautions, steam has been blanked from the storage and make-up tanks, agitation of the hold tank has been minimized, and instrument air has been shut off to the storage tank.

A 4' × 3' acid measuring tank has been installed over the 3.7% HNO_3 make-up tank to increase the accuracy of measuring the amount of 50% HNO_3 added to the 9' × 36' horizontal make-up tank. Addition of the relatively small amount of acid directly to the large make-up tank, was not considered sufficiently accurate to maintain the close acid tolerances required on the critical 2AS stream.

A-LINE

The last batch of H-Area uranium has been processed, and the E-1-1 storage tank has been flushed. One off-standard lot of uranium trioxide resulted when total metallic impurities exceeded the maximum limit. The increase in metal coincided with processing of low-uranium-concentration (35 g/ℓ) flush material from H Area. The off-standard material will be blended with future production.

Backpressuring of silica gel column S-8-1 with water failed to relieve the plugged condition noted last month (DPSP 59-1-3, p 240), and this column was shut down after processing 431 column volumes (1,161,000 pounds) of EU concentrate; approximately 10 curies of Zr-Nb were adsorbed by this column since the last regeneration. Sodium citrate was used under TA 2-271 as a substitute for oxalic acid (citrate will not precipitate uranium at 75 g/ℓ) in regenerating a silica gel column to evaluate citrate for use in the refining of enriched uranium.

Column S-8-1 was regenerated with 15 column volumes of 0.1M sodium citrate followed by the usual six column volumes of 5% oxalic acid. The detailed conditions and results of this test are given in the following table. Note that 50% of the total Zr-Nb removed from the column was taken off in the first 4.3 column volumes of citrate; the remaining volumes (10.7) removed very little Zr-Nb, even though the column was held with citrate on the gel for a period of 24 hours (during which the second batch of citrate was made up and analyzed). The remaining Zr-Nb was removed during the oxalic acid flush. Radiation profiles of the column before and after the citrate test and after the oxalic acid flush showed maximum readings of 4500, 1200, and 450 mr/hr, respectively. The first 6.9 column volumes of citrate and 2.2 volumes of oxalic acid were neutralized and sent to the Building 241-F waste tank; the remaining solutions were sent directly to the seepage basin.

Results of Silica Gel Column Regeneration

Test Procedure

Column	S-8-1
Period Column Used	2/4/59 - 3/24/59
Curies Absorbed	10.1
Total Curies Eluted	8.4
Sodium Citrate	0.08 to 0.109M; pH 6.3 to 6.6; fed at 7 gpm (30-minute residence time) and 85°C
Oxalic Acid	5.0%; fed at 1.5 gpm and 85°C

Test Results

Material		Column Volumes Fed	Curies Eluted
1.5% HNO_3 Flush		3.31	-
Sodium Citrate		2.16	3.88
		2.16	0.19
		2.54	0
		2.70	0.09
		2.70	0.02
		2.70	0
	Subtotal →	14.96	4.18
Water Flush		~9	-
Oxalic Acid		2.16	3.40
		3.78	0.84
	Subtotal →	5.94	4.24
Water Flush		3.33	-

Finished Product Analyses. The most significant of the analyses of the finished oxide shipped during the period are summarized in the following table.

A-Line Finished Product Analyses

	Specification	Range	Average
Parts Pu/bpU	<10	1	1
Parts Fe/mpU	-	70 - 225	123
Parts Total Metal Impurities/mpU	<250	142 - 327	211
Activity Level, % Aged Natural U			
Beta	<100	1.3 - 4.7	3.3
Gamma	<100	7 - 45	24
Reactivity Ratio	-	0.92 - 0.95	0.93
UO_3, %	>97	97.9 - 98.6	98.26

PLANT TESTS OF NEPTUNIUM RECOVERY

$BiPO_4$ Precipitation Plus Solvent Extraction. Three batches of HAW concentrate have been treated in Building 221-H head end equipment by the $BiPO_4$ precipitation process (described in the past four monthly reports). The dissolved product cake from the first two of these batches was sent through one pass of decontamination in the second plutonium cycle.

The precipitation tests are summarized in the following two tables and figure S-1; for completeness, the data for run 1 (reported last month) are included.

$BiPO_4$ Performance Data

Run No.	Strike No.	Type Strike	Np Loss, % (Based on Feed to Run)	DF = Curies in Feed / Curies in Product		
				Zr-Nb	Ru	Gross
1	1	Byproduct	2.4	3.7	1.3	2.4
	2	Product	26.7*	3.6	66.4	8.2
	3	Product	0.9	0.8	1.2	0.7
		All Strikes →	30.0	10.6	96.9	14.3
2	1	Byproduct	6.5	0.9	1.4	1.0
	2	Product	3.5	3.0	19.7	4.4
	3	Byproduct	4.9	11.3	1.4	8.1
	4	Product	1.9	5.8	71.0	8.8
		All Strikes →	16.8	170.	2661.	300.
3	1	Extraction	8.8	5.4	144.	8.3
	2	Byproduct	2.8	8.1	1.3	6.9
	3	Product	2.8	2.1	18.8	2.6
		All Strikes →	14.4	92.	3570.	149.
		All Strikes plus precentrifugation** →	-	131.	4000.	215.

* High loss caused by operating error.

** This run had a centrifugal clarification before the first strike (due to an operating error - see text) which increased the over-all DF slightly, as shown.

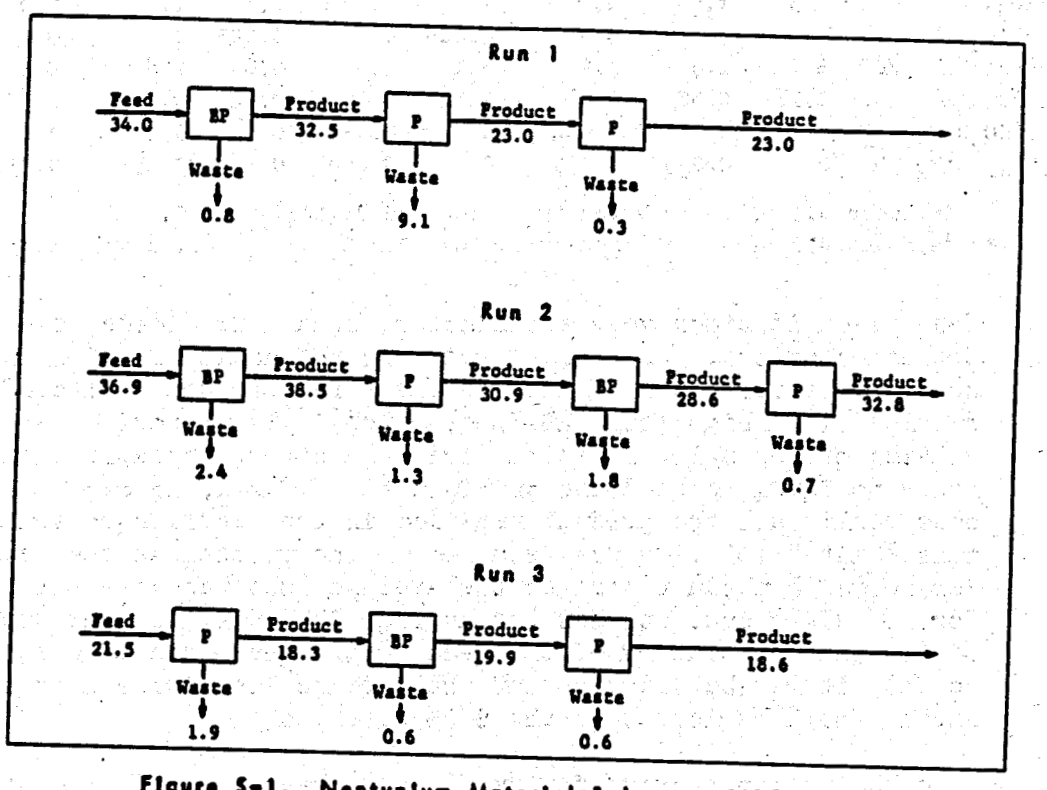

Figure S-1. Neptunium Material Balances, $BiPO_4$ Tests
(All values shown are grams of neptunium)

P Product Strike

BP Byproduct Strike

Concentration of Reagents at Time of Strike

Constituent	Run No. 1 Byprod Strike	Run No. 1 Product Strike	Run No. 1 Product Strike	Run No. 2 Byprod Strike	Run No. 2 Product Strike	Run No. 2 Byprod Strike	Run No. 2 Product Strike	Run No. 3 Extraction Strike	Run No. 3 Byprod Strike	Run No. 3 Product Strike
HNO_3, M	1.3	1.7	1.0	1.4	0.8	1.5	1.6	0.8	1.9	1.1
$NaBiO_3$, M	0.006	0	0	0.006	0	0.005	0	0	0.005	0
Total Bi, g/ℓ*	5.0	1.3	1.9	5.1	2.0	4.8	1.8	2.3	5.5	2.0
H_3PO_4, M	0.16	0.57	0.58	0.12	0.57	0.12	0.55	0.55	0.12	0.53
$Fe(SO_3NH_2)_2$, M	0	0.014**	0.032	0	0.032	0	0.11	0.039	0	0.069
$(NH_4)_2SiF_6$, M	0	0.035	0.053	0	0.054	0	0.052	0.038	0	0.049

* Includes all Bi contributed by $NaBiO_3$ and $Bi(NO_3)_3 \cdot 5\ H_2O$.

** Ferrous sulfamate was erroneously low; 0.03M ferrous sulfamate should have been added.

Test precipitations were satisfactory regarding yields, except for two operating errors. The first error was the addition of an insufficient quantity of ferrous sulfamate in the first product strike of run 1; this resulted in a high loss (26.7% of batch). The second error was the misvalving of the H_3PO_4 addition during an attempted extraction strike; because no H_3PO_4 reached the precipitation vessel, no cake was formed and essentially all the product remained in the centrifuged solution (most of this "waste" solution was returned to the process as the feed for run 3). For normal byproduct strikes the average loss was 4.2% per strike, and for normal product strikes the average loss was 2.3% per strike; in view of the unusual history of the feed to the one extraction strike made (see above), it may be assumed that the average loss for a normal extraction strike would be less than the 8.8% observed.

Decontamination factors for run 1 were adversely affected by contamination from the head end equipment. Evidence for this is the fact that the product cake solution from the last strike of run 1 contained brown-black solids (probably MnO_2) and that the streams leaving this strike contained 2.8 times as much fission product activity as did the feed to the strike. On later runs, when equipment contamination was reduced, process DF's improved and became less variable. The activity of the solution sent to the Warm Canyon for solvent extraction is shown in the following table.

$BiPO_4$ Test - Activity of Feed and Product Solution

Stream	Weight, lb	Sp Gr	Gamma Activity, c/m/ml Zr-Nb	Ru	Gross
Run 1 HAWC Feed	8689	1.29	1.01×10^{10}	4.51×10^{9}	1.56×10^{10}
Run 1 Product (after 3 strikes)	9424	1.26	8.56×10^{8}	4.19×10^{7}	9.80×10^{8}
Run 2 HAWC Feed	7960	1.31	8.80×10^{9}	5.00×10^{9}	1.44×10^{10}
Run 2 Product (after 4 strikes)	8344	1.25	4.71×10^{7}	1.71×10^{6}	4.37×10^{7}
Run 3 HAWC Feed	6190	1.28	8.93×10^{9}	3.13×10^{9}	1.44×10^{10}
Run 3 Product (after 3 strikes)	8824	1.26	4.74×10^{7}	5.38×10^{5}	4.65×10^{7}

The waste volume generated by the $BiPO_4$ precipitation runs was 30,791 gallons (to Building 241-H). This volume is not indicative of the waste volume to be expected from a routine operation, because many extra flushes and decontaminations were made and because only alkaline evaporation was used. It is estimated that a routine three-cake $BiPO_4$ treatment of HAW concentrate in F Area would raise Purex waste output by 150 ± 50 gallons of high level waste per canyon batch, and that this figure might be reduced if, for example, the acid from tank decontamination could be returned to the HAW evaporator instead of neutralized and sent to waste.

Three batches of neptunium-bearing cake solutions from the head-end bismuth phosphate treatment were received in the 2AF feed vessels for solvent extraction decontamination. Recovery during processing of the first cake solution from head end under the reducing high acid flowsheet (see following section) was poor with at least 50% of the neptunium appearing in the 2AW waste stream. It is believed that this loss was due to the presence of residual MnO_2 cake (contamination from head end equipment, see preceding paragraphs) in the feed solution, as evidenced by subsequent inspection of samples of this material, which destroyed the reductant (ferrous sulfamate) added to the feed by its oxidizing ability. The 2AW containing the loss has been concentrated in the LAW system and will be reprocessed in the second product cycle. Laboratory tests on the concentrate indicate that the neptunium is extractable in a reducing solution.

Before the second $BiPO_4$ cake solution was processed, the solution was divided into two equal batches so that a solid bismuth phosphate precipitate (visible in samples of this material) could be dissolved by adding more 50% HNO_3 and by heating to 70° to 75°C. Prior to being fed to the cycle, the solution was diluted to 5.8M acid and the reducing strength was established to assure that an adequate ferrous sulfamate adjustment had been made. Recovery from the first batch of this solution was good based on waste stream analysis, 2AW analysis indicating a loss of only 0.5% and 2BW analysis indicating a loss of about 2%. Results from the second batch are pending. The 2AS ferrous sulfamate concentration was increased to 0.1M for these two batches to prevent loss to 2AW. No other flowsheet modifications were made.

The combined 2BP from the first two $BiPO_4$ cake solutions and the recovered 2AW concentrates will be reprocessed for further decontamination. Gross gamma DF for the two cake solutions processed was about 224 (on a volume basis, 2AF:2BP) from the one pass through the extraction cycle. The third head end cake solution will be processed separately as a final demonstration run.

Solvent Extraction and Cation Exchange Recovery. The neptunium recovered in the recent tests of a stirred-bed anion exchanger in Building 221-H (see the February monthly report, DPSP 59-1-2) was decontaminated by solvent extraction in the second plutonium cycle, and the neptunium-rich 2BP was transferred to HB-Line coupling for concentration and further decontamination. The purpose of these operations was to provide process information regarding the behavior of Np in solvent extraction and cation exchange flowsheets. A total of 13.7 grams of decontaminated Np was obtained from these tests and shipped to SRL. Two solvent extraction flowsheets were tested; a reducing, high acid flowsheet using ferrous sulfamate, and an oxidizing, low acid flowsheet using ceric sulfate, both shown in detail in the following table.

Second Product Cycle Flowsheet for Decontamination of Np

	Oxidizing		Reducing (Revised)*	
	Relative Flow, vol	Composition	Relative Flow, vol	Composition
2AF	100	2.0M HNO_3; 0.015M $Ce(SO_4)_2$	105	5.0M HNO_3; 0.015M FS
2AX	80	33% TBP	59	33% TBP
2AS	20	0.5M HNO_3; 0.015M $Ce(SO_4)_2$	14	1.0M HNO_3; 0.015M FS
2BX	40	0.01M HNO_3	20	0.10M HNO_3

* Added FS to the 2AS.

Neptunium solution from the anion exchange tests (stored in a 2BP hold tank) was processed in two separate batches. The first batch consisted of 62% of the material and was processed by three reducing runs followed by an oxidizing run. The remaining material was processed by two oxidizing runs followed by two reducing runs. Decontamination and recovery performance for each run are shown in the following table.

Decontamination and Recovery of Np in Second Product Cycle

	Pass	Flowsheet	DF			% Recovery*	Final 2BP Gross γ, c/m/ml**
			Gross γ	Zr-Nb	Ru		
Batch 1	1	Reduction†	260	110	2580	45.5	4020
	2	Reduction†	1.9††	1.6	2.4	50.8	
	3	Reduction†	3.4	3.2	3.3	39.1	
	4	Oxidation	5.7	6.3	2.3	74.4	
Batch 2	1	Oxidation	130	>10,000	80	103.6	8040
	2	Oxidation	6.3	<1	6.1	73.7	
	3	Reduction	2.5	1.2	2.6	83.8	
	4	Reduction	1.9	<1	2.3	73.6	

* For each batch, based on feed to first pass for that batch.
** Gross gamma in feed to batch 1 and batch 2 = 2.18×10^7 c/m/ml.
† No FS in the 2AS on these runs; refluxing was high.
†† Low apparent DF attributed to contamination of 2BP when recycled from pipe hold-up of preceding feed material (estimated at 7% by wt).

Over-all neptunium recovery was 74% from both batches, based on final analysis of the 2BP sent to B-Line. The apparent poor recovery during the first three reducing runs is attributed to reflux in the scrub section of the 2A bank due to the lack of ferrous sulfamate in the 2AS scrub stream. Addition of reductant to the scrub stream during the two reducing runs of the second batch eliminated the reflux condition.

Over-all gross gamma DF for four passes was 9500 for the first batch and 4000 for the second batch, resulting in final gross gamma activity in the 2BP of 4020 c/m/ml (88% Zr-Nb) and 8040 c/m/ml (14% Zr-Nb), respectively. As expected, the low acid flowsheet favored zirconium-niobium decontamination and the high acid flowsheet favored ruthenium decontamination. Gross DF's for the first pass of each batch were 260 and 130, respectively, while the succeeding passes netted DF's in the range of 2 to 6.

The 2BP resulting from the solvent extraction runs made with anion exchange eluate was processed in the HB-Line cation exchange coupling facility. The volume reduction across the coupling step was 250 and the gross DF was 9.5 (97.0 for Zr-Nb and 5.5 for Ru). Pertinent data are summarized in the following table.

Quantity	Coupling Feed	Coupling Product Eluate			
		Cut 1	Cut 2	Cut 3	Over-all
Volume, ℓ	4580	9.44	4.30	4.54	18.28
Np Concentration, g/ℓ	0.00411	1.26	0.24	0.16	0.75
Np, g	18.8	11.9	1.1	0.7	13.7
Gamma; Zr-Nb, c/m/ml	2.70×10^3	4.69×10^3	1.64×10^4	1.88×10^3	6.73×10^3
Gamma; Ru, c/m/ml	3.25×10^3	1.33×10^5	2.69×10^5	6.42×10^4	1.48×10^5
Gamma; Gross, c/m/ml	5.72×10^3	1.28×10^5	2.94×10^5	6.77×10^4	1.52×10^5

The column feed was adjusted to 0.0025 to 0.0050M $Fe(SO_3NH_2)_2$ before absorption. Absorption was performed at 2 ℓ/min; absorption waste losses totaled 0.5 gram. Thus, since 18.8 grams were fed and 13.7 grams were eluted, there are still 18.8 - (13.7 + 0.5) = 4.6 grams unaccounted for. Some of this material may have been left on the column, in which case it may appear next month when the 2BP from solvent extraction of the bismuth phosphate product is processed.

Plutonium Isolation and Recovery (221-F)

PROCESS STATUS

The JB-Line was put into operation at the end of the month although construction still was in progress at startup. Coupling performance generally was satisfactory although some concentrates containing excessive uranium were obtained (the acceptable limit is about 0.3% of the plutonium). The first plutonium trifluoride precipitations went very well with low waste losses and comparatively short filtration times. A small amount of hydrolysis was observed in the first roasted cakes, which indicated that the drying and roasting cycles should be modified. Three test reductions were made with cerous trifluoride before plutonium production started. The first reduction yields were somewhat low, so in addition to drying changes, the amount of iodine booster was increased for subsequent runs. The final quality of these buttons still is to be determined. The anion exchange test facility in recovery continued to operate well. The cation exchange system was contaminated by high-salt waste solutions and is being cleaned.

COUPLING

The startup of the JB-Line coupling facility was made without incident. At the initial operating conditions, the average performance for the column load of was: product fraction of recycle fraction of and loss of These original feed solutions contained low plutonium concentrations (down to) and relatively high uranium concentrations (up to of the Pu). The coupling concentrates that went into the first JB-Line buttons contained residual uranium contamination to the extent of 0.5 to 1.0% of the plutonium. It became apparent that better uranium decontamination in coupling would be needed because uranium is not decontaminated further by the fluoride precipitation; the observed contamination is sufficient to bring the buttons below the acceptable assay of 99.5% plutonium. At the end of the month, coupling was being operated for maximum uranium decontamination (which means a lower throughput rate).

Work on the various coupling instruments continued. The liquid level indicators (capacitance probes) still gave variable readings although the situation was improving. The neutron counters on the column segments and various vessels also were giving erratic readings. The in-line gamma absorptometer for measuring plutonium concentration gave good precision although the original laboratory calibration with uranium solutions had a bias of about 3.7 g/ℓ compared with plutonium analyses from the old absorptometer and from laboratory alpha counting; when recalibration is completed, the instrument should be giving rapid and accurate analyses.

WET CHEMISTRY

Precipitation of plutonium trifluoride was started at the end of the month and chemically, at least, operation appeared to be quite satisfactory. Results from the first runs with respect to filtration times and losses are shown in the following table.

Filtration of Plutonium Trifluoride

Run No.	Batch, grams	Time, minutes	Losses, grams
1		59	
1A*		18	
2		63	
3		78	
4		70	

* The first run was made in two parts because the concentrate feed rotameter plugged at 80% through the original run.

The filtration times were slightly less than the estimated design time, and the losses were less than the nominal design value of 0.1%. The filtrates listed above were neutralized and sent to waste; the only JB-Line solutions sent to recovery were the aluminum nitrate-nitric acid flushes of the precipitators and boats. A noticeable feature of these runs was the difficulty in removing the plutonium trifluoride from the precipitator after a run. A coating was left on the walls of the first stage vessel and on the overflow line to the second stage; in addition, approximately settled and remained on the bottom of the second stage after normal filtration, so that excessive volumes of washes were required. The jets which were designed to assist removal of the settled slurry were ineffective as installed, and modifications of the washing technique and jet design are being studied. To prevent high losses from oxidation of residual plutonium trifluoride between runs, ascorbic acid (0.06M) was added to the diluent solution which is left in the vessels. The procedure appeared to be satisfactory and losses did not increase materially on successive runs. The standard flush of aluminum nitrate and nitric acid was used in order to clean the precipitator before the weekend shutdown, and the plutonium trifluoride dissolved rapidly and completely.

The major difficulties so far have been with equipment. Although the construction schedule had been accelerated tremendously, there was insufficient time for many cold runs before actual operation started and many items were still incomplete or not functioning properly. The system was not flushed enough to remove all particulate matter and the rotameters on the precipitator feed lines clogged sufficiently often to interfere with the first runs. As mentioned above, the flushing jets also were plugged.

Many minor modifications were made to the precipitation system during the testing period before the facility was put into use. The two major modifications were made to the plastic second stage precipitator vessels. The vent system was changed to avoid collapsing the vessel under vacuum as was reported last month. In addition, the supporting steel frames were strengthened because the vessels expanded when they were filled with solution so that the allowable limit for nuclear safety was exceeded (3.5 inches maximum internal width). The original steel frames bulged as well as the plastic sides in the areas between the support grid. Additional lattice members were added midway between the original vertical members in order to strengthen the frame, and these also supported small bearing plates which pushed against the sides in the middle of the original exposed areas. After modification, the vessels expanded from 3-1/4 inches to a maximum of 3-7/16 inches, which is within the limit. The revised design was considered in detail for its effect on criticality and does not require any changes in the present precipitator batch limits or operation.

The cake drying stations functioned properly in tests and during the first regular runs. Air flow and heat control were satisfactory, and the design drying cycle of 6 hours at 70° to 80°C was used. The first cakes were somewhat discolored after roasting and indicated that further drying was needed to prevent hydrolysis, so the drying period was lengthened for the subsequent runs.

ROASTING AND REDUCTION

Operations started in the new mechanical line following a short period of testing. In the various cold tests, temperature response on the roasting furnaces was excellent, the cake transfer stations operated satisfactorily, and dummy charges charges in the reduction furnaces heated at the designed rate, which would give firing times of 40 to 50 minutes (this can be varied further by changes in the power level).

Three reductions of cerous trifluoride were made and the equipment operated satisfactorily, although the slag and crucibles containing iodine were very difficult to remove from the pressure chamber. The amount of iodine added as a booster was varied to simulate different total heats of reactions in plutonium reductions.

Reduction of Cerous Fluoride

Cold Run No.	Total Heat of Reaction	Results
C-1	Approximated the reduction of as PuF_3 using I_2/mol Pu and excess Ca (flowsheet).	a) reduction yield. b) Properly formed button. c) Slag and crucible very difficult to remove from pressure chamber.
C-2	Approximated 1/3 the reduction of of Pu as PuF_3 with no I_2 added.	a) Incomplete reduction - no button. b) Slag burned when exposed to the cabinet atmosphere. c) Charge dumped easily.
C-3	Approximated the reduction of as PuF_3, with I_2/mol Pu and excess Ca.	a) reduction yield. b) Properly formed button. c) Slag and crucible very difficult to remove from pressure chamber.

Buttons were obtained from the first three plutonium trifluoride reductions. Reduction yields were The first button was very rough and poorly formed, but the others were properly formed. The quality analyses have not been completed on any of these buttons. The roasting cycle on these first runs was 1.5 hours at 600°C, and the reduction charge contained - mol I_2/mol Pu. Contrary to the experience with cerium, the reduced charges all were relatively easy to dump and the iodine booster ratio was raised to mol I_2/mol Pu for subsequent runs. (The line was still under construction during the cerium tests, and it is possible that the combination of high iodine and high humidity were responsible for the original difficulties in dumping.) The purple plutonium trifluoride cakes hydrolyzed to some extent in the roasting operation, as indicated by the formation of the yellow-green oxide. It is known that an excess of the oxide lowers the reduction yield so efforts are being made to reduce the hydrolysis, and modifications of the drying and the roasting cycles are being considered.

RECOVERY

Process performance of the anion exchange test facility was generally satisfactory during the first 12 runs. The majority of the difficulties encountered were mechanical in nature and are attributed to poor performance of the equipment. The average product concentration was while the losses averaged The average recycle was low, at A complete heel elution made on one of the columns indicated that a heel of approximately remained after a normal product elution.

Filtration of silica from the slag and crucible solution has been adequate and there is no indication that pressure drop through the anion columns is increasing. One of the anion columns periodically seems to give improper operation and also may be the source of a small amount of resin found in the product stream. High absorption losses (which were recycled) and low product concentrations were experienced on this unit and correspond to the behavior of the cation columns when the follower plate is not moving and the bed fluidizes. If performance does not improve, the unit will be replaced.

The present concentration of aluminum in the anion product prevents direct processing of these solutions in the JB-Line precipitation process. Previous work indicates that an aluminum content less than 100 ppm of plutonium must be attained to avoid harmful effects on plutonium density and ngs; and the aluminum in the anion product is currently averaging 595 ppm. Cross-contamination between the feed material and the product stream may be occurring in the piping and column headers. The bed wash procedure is being varied to insure maximum decontamination from aluminum.

Precipitation of plutonium hydroxide, high absorption losses, and high aluminum contamination in the cation exchange system were caused by solution from the waste neutralization tank. This material had boiled out of the tank and the drain on the vessel vent header was plugged so that the solution ran back into other vessels. The cation system experienced high aluminum contamination in the concentrate (ranging from 800 to 100,000 ppm of plutonium) and high absorption losses (as much as 10%) for several weeks, and efforts to determine the source of the difficulties had failed.

The cause finally was found when one cation feed batch lost its usual light-blue color during absorption and white solids were observed in the feed rotameter. Subsequent analysis showed that the feed was basic. A thorough flushing program was undertaken and revealed that approximately 700 grams of plutonium had precipitated in the cation feed tanks and collected on the column filters. This material was dissolved by the high-acid flush solution and recycled. A material balance has accounted for 96.2% of the product involved in the incident. The discrepancy, approximately 50 grams, may be due to errors in analytical results and tank volumes. Although this type of incident is particularly hazardous with respect to criticality, there was no indication that any of the recovery tanks approached their respective nuclear safety limits, and the regular batch control system appeared adequate. To prevent a recurrence of the incident, a blank was installed in a line from a currently unused tank from which some waste might have been transferred. Also, the addition rate of caustic during neutralization will be controlled to prevent the neutralization tank temperature from exceeding 70°C. Revisions to the vent system are being considered.

Enriched Uranium Processing

PROCESS STATUS

Equipment changes required to switch from Purex to recovery of enriched uranium were largely completed, and cold runs to check out the new process were in progress at the end of the month. In rerun, plutonium and uranium that came from flushes of the canyon sumps at the conclusion of Purex operations are being held for extraction in second product cycle after the neptunium extraction runs are completed.

DISSOLVING

One of the two dissolver columns for processing of Mark VI elements has been installed in the canyon, and cold dissolving tests were started at the end of the month. The second column will be received about the end of May.

Before installation, the first column was pickled on the water side of the condenser coils with 30% HNO_3 at 135° to 140°F for 16 hours. Following the pickling the coils were hydrostatically tested at 150 psig; no leakage was observed. The column was installed on dissolver 6.1D.

Dissolver Pot Residue. Inspection of the dissolver pots immediately after the columns were removed showed scattered metal pieces, mainly quatrefoil components, partially submerged in what appeared to be a black sludge. After the pots had been open to the atmosphere for several weeks, the sludge had dried, leaving a fine powder on the dissolver bottom. During an unsuccessful attempt to sample the powder, the layer was estimated to be no more than 1/8 inch deep. It was decided to leave the metal pieces and powder in the dissolvers; the amount of metal, estimated as less than 100 lb, does not reduce the dissolver volume significantly or interfere with installation of the column insert.

Plans for Cold Runs. An extensive series of cold dissolving runs is considered necessary because of the great difference between the Mark VI elements and the natural uranium elements dissolved for Purex operations. It is planned to develop the following information during these runs:

1. The relation between element submergence (the length of element exposed to acid), catalyst concentration and dissolving rate.

2. The relation between dissolving rate and foam height.

3. A method for determining the end point at which dissolving should be stopped. In dissolving natural uranium, specific gravity is used to determine the end point, but in dissolving of Mark VI elements there will be little change in specific gravity.

4. Retention of catalyst on the element fragments. Two dissolvings will be made for each charge of elements; the first will dissolve 90% of the charge; the second, a heel cleanout, will dissolve the remainder. The heel cleanout cut will be transferred to a hold tank and then returned to the dissolver for the next large cut. Catalyst may plate out on the element fragments near the end of the large cut and redissolve in the next heel cleanout cut. Catalyst retained in the heel solution could cause a rapid reaction when returned to the dissolver.

Cold runs in dissolver 6.4D will be made with bundles of aluminum tubing, as follows.

- Run 1. Four Bundles (28 elements). Procedure will be based on SRL experience with dissolving at TNX.
- Run 2. Four Bundles. Catalyst concentration and addition rate will be altered, depending on experience in first run.
- Run 3. Four Bundles. Final check of proposed startup procedure, based on runs 1 and 2.
- Run 4. Two Bundles. Simulate dissolving procedure for Mark VI-J elements.

This will leave two bundles for checkout of dissolver 6.1D when the second column is available.

HEAD END

The changes made in head end process piping for neptunium recovery runs were removed and standard piping was reinstalled. Piping revisions necessary for the change over to the enriched uranium process were completed and the system was leak-checked with water. The 10.3C centrifuge is out of service awaiting completion of repairs to the high pressure spray pump.

The first cold run, consisting of a solution of $Al(NO_3)_3$, HNO_3 and sufficient "Ludox" (SiO_2 suspension) to form a gelatin cake, was completed on April 24. No uranium was added to this run. No cake was observed in the centrifuge feed sample but it is questionable that cake could be detected visually because of the nature of the cake and the small volume of cake and sample involved. Further runs will be made with solutions containing U-238 to obtain process and cake wash data.

SOLVENT EXTRACTION

Following removal of Purex solvent from the 1A, 1B, and 1C banks, an extensive flush of the 1A heat exchanger-Hackman hat piping and 1A bank was made on March 25 to March 30. This work completed the first cycle cleanout program for conversion to the processing of enriched uranium. The flush procedure, which was similar to the flush made in March 1958, consisted of the following individual flushes.

- The heat exchanger-Hackman hat piping was flushed with 3000 lb of 5% HNO_3 - 5% $NaNO_2$ followed by 150 lb of 30% $NaNO_2$ and hot process water to remove any deposits of MnO_2 and uranyl hexahydrate.

- Both the heat exchanger-Hackman hat and the scrub inlet were flushed with 14,000 lb of 10% NaOH at 70° to 75°C followed by process water to remove any deposits of siliceous material.

- A final cleanup flush was made with 5000 lb of 50% HNO_3 at 65° to 70°C followed by process water.

Material removed by the various flushes is indicated in the following table.

Flush	Pu, g	U, lb	Gamma, gross curies
5% HNO_3 - 5% $NaNO_2$	0.1	-	2.4×10^4 (84% Zr-Nb)
10% NaOH	0.3	0.3	2.9×10^4 (95% Zr-Nb)
50% HNO_3	2.2	0.5	3.2×10^5 (84% Zr-Nb)

The aluminum cold run for the enriched uranium recovery process was started in first cycle on April 24. Operation to date has shown need for only a few minor adjustments in the flow control and recording instrumentation. The colorimeters and neutron monitor demonstrations will be made during the cold natural uranium and first hot runs, respectively.

SOLVENT SYSTEMS

Purex solvent was removed from the first cycle and second uranium cycle during the period. About 208,000 pounds of solvent that came from first cycle and from skimmed solvent tank 511 were sent to the burial ground for storage and eventual burning or reuse. About 70,000 pounds of solvent from the second uranium cycle were sent to F Area.

The solvent washers were charged with normal wash solution, ie, 2.5% carbonate in the canyon washers and 0.63% nitric acid in the Building 211-H acid washers. Make-up of solvent for the recovery of enriched uranium began after the Purex solvent had been removed. Ultrasene (110,000 pounds) was added to the first cycle system, flushed through the batch washer and decanter, and recirculated through the 1A and 1C banks to collect any remaining Purex solvent before the TBP concentration was adjusted. After this flushing and circulation, the TBP concentration was 1.2% and the gamma activity was in the range of 500 to 800 c/m/ml. Sufficient TBP has been added to this solvent to bring the TBP concentration up to about 2.3%. The allowable TBP concentration of the first cycle solvent is 2.5 ± 0.25%.

Sixty thousand pounds of Ultrasene were added to the second cycle and circulated similarly, resulting in an unadjusted TBP concentration of about 2%. The initial TBP adjustment brought the TBP concentration up to about 5%. The allowable TBP concentration of second cycle solvent is 7.5 ± 0.75%. This solvent attained a gamma activity of 100 to 200 c/m/ml during this period of circulation.

Solvent from the Purex second product cycle has been intermittently used in the neptunium recovery program. Since the feed activity approximated that of Purex 1AF, there was an increase in the solvent gamma activity up to a maximum of 3.3×10^4 c/m/ml, compared to 200 to 400 c/m/ml during Purex operation. This solvent will not be used for recovery of enriched uranium or reused for normal Purex plutonium recovery.

OUTSIDE FACILITIES

Hydrostatic Testing. The GP evaporator heater, the EU evaporator reboiler, and the acid recovery unit reboiler were hydrostatically tested at 45, 38, and 42 psi, respectively, according to the ASME code for unfired pressure vessels. All three units are in satisfactory condition.

Solvent Removal. Solvent found in miscellaneous tanks throughout Building 211 Area has been collected in tank 717 where it is being skimmed before it is sent to the burial grounds. Following removal of the solvent that had collected in the recycle sump, the sump was visually inspected and found free of solvent.

Solvent entrained in 1EU was skimmed from the EU concentrate tank and the tank was flushed with Ultrasene to remove the remaining TBP; this Ultrasene was also skimmed. Prior to skimming the tank, water flushes were added to remove the uranium remaining in the aqueous heel. These flushes stripped the uranium in the solvent from 47 g/ℓ to 2.4 g/ℓ.

ACID RECOVERY UNIT

The reboiler of the acid recovery unit was inspected and repaired; details are shown in the following table. Following repairs the reboiler was hydrostatically tested and found satisfactory.

Acid Recovery Unit Reboiler Inspection and Repair

Item Needing Repair	Type of Attack	Method of Repair
Vertical seam weld on SS Shell (Process Side).		
Top	Knife line and pitting	*
Bottom	Honeycombed.	*
Inside seam weld on nozzles to column.		
Top & Bottom	Cracked ~75% of circumference.	*
Inside seal weld around tube sheet.		
Top & Bottom (Process Side)	Corrosion down center of weld ~100% of circumference.	*
Weld on face of top flange to column.	Severely honeycombed.	*
Inside seal weld on nozzle in bottom reboiler head.	Corrosion down center of weld.	*
24 Tubes leaking at top tube sheet.	Leaking between tube and sheet.	Welded in 19 chill blocks (Total 38 chill blocks & plugs) and welded balance of the 153 tubes to top tube sheet.
Inside carbon steel shell (steam side) from bottom tube sheet to 2" above sheet	Pitting corrosion where condensate stands when unit is not operating.	6"-wide SS band was welded around outside of carbon steel shell above the tube sheet.
Head flanges (42") on reboiler were gasketed with Blue African Asbestos gasket.	Leaking air into unit during operation, and acid out when unit down.	Installed an Asbestos-"Teflon" envelope gasket in place of Blue African Asbestos.**

* Ground out and welded.

** To obtain sufficient pressure to seal the new gaskets it was necessary to replace the original bolts (ASTM A193 B.8) with bolts of higher tensile strength (ASTM A193 B.8 cold-drawn to a Brinnell hardness of 320). Both types of bolts are type 304 stainless steel.

SILICA GEL COLUMNS, H AREA

Dummy runs are proceeding in the silica gel columns. A dummy regeneration run using water was made at flows up to 5 gpm. The steam to the regenerant heater had to be controlled manually because the steam control valve was oversized. This difficulty is being corrected. A maximum flow rate of 10 gpm was obtained during the dummy dilute nitric acid wash. Dummy feed runs are in progress.

WASTE TRANSFERS

Waste Volumes, H Area
(3/19 through 4/21)

Streams to Bldg 241	Actual Transfers, gallons
High Level	
HAW Evaporator	3,579*
LAW Evaporator	1,069
Head End	-
Rerun** (Tank 10H)	14,210
(Tank 14H)	18,729
Buildup in Bldg 241†	-
Low Level	
Coating Waste	-
GP Concentrate	2,368
Buildup in Bldg 241†	-

* NaOH and water flushes of 1st cycle plus discard of 2nd stage evaporator bottoms.
** See rerun section.
† Change in level in Bldg 241 tank is small and incremental increases cannot be determined accurately.

UNDERGROUND WASTE STORAGE

Status of Waste in Underground Tanks*

Tank	Fill, inches	Maximum Temperature, °C Sludge	Supernatant	Remarks
9-H	268.0	36	37	Cooling water through 1/8" orifices, all 36 coils.
10-H	268.25**	80	38	Water through 28 vertical coils; 1 vertical valved out, 5 vertical and 2 horizontal coils have failed.
11-H	274.5	35.5	35.5	Low level waste; cooling water through 1/8" orifices, all 36 coils.
12-H	264.75	89	37	Full cooling water flow through all 36 coils.
13-H	202.25	37.5	42	Low level waste; cooling water through 2 horizontal coils.
14-H	142.0	117	45	Full cooling water through all 41 coils.

* Pluggage of the level dip tubes in Building 241 pump pit tank 4 has occurred frequently since the processing of bismuth phosphate waste.
** On April 6 about 3 inches of waste were jetted from tank 10 to tank 12 without incident.

The "necklace alarm" on waste tank 14-H energized indicating the presence of liquid shorting the probes. A visual check indicated liquid in the south riser but none in the north riser. A swab was dipped into the liquid and found to be contaminated to only ~270 c/m of activity, indicating the liquid had not come from the tank. Upon standing the observed liquid evaporated from the annulus; however, the alarm did not deactivate. The cause of the current alarm is unknown. The water table level in the area of the tank is 277'8" above sea level which is 16 inches below the top of the steel annulus pan. The water table movement in this area is being examined to determine if it could account for the liquid in the tank annulus.

Building 241 Cooling Water, H Area

	Maximum	Minimum	Specification
Na_2CrO_4, ppm	900	824	>500
pH	10.7	10.45	>7.0

RERUN

Rerun Receipts and Waste Volume

Source	Reason	No. of Transfers	% of Total Wt Received in Rerun (581,000 lb)	Rerun Waste Volume, gal
1AF Adjustment Tank	Special product	42	85.0	30,791*
Raw Metal Tank	Flush	1	0.6	-
Vent Scrubber	Routine change	1	1.6	-
Sumps	Process leaks & flushes**	11	8.4	-
Head End Strike Tank	Flushes to remove special product waste	5	3.7	-
1AW Decanter	Solvent removal	2	0.1	-
1CU Decanter	Solvent removal	2	0.6	-
Rerun	Flushes	-	-	491
Rerun	Solvent washing and miscellaneous	-	-	1,657

* This volume is somewhat larger than optimum since additional waste was generated arbitrarily as an operating convenience, or because of gross operating errors (eg, misvalving of H_3PO_4 discussed on page 220).

** Approximately 280 grams of plutonium were received in rerun from sumps during the period. About 120 grams of this have been concentrated and are being stored in condensate tank 11.4 HE; this tank will not be used in enriched uranium processing. The balance is being held in rerun until flushes of the canyon cells are complete.

The rerun solvent was prepared for discard to the burial ground by washing it with Na_2CO_3, HNO_3, and oxalic acid. A final alkaline wash was required as the activity of the solvent was greater after the acid wash; this was probably due to removal of activity from the vessel walls. See the following table. After the solvent is removed from rerun, the canyon floors will be washed to rerun. The rerun station will then be flushed with Ultrasene to remove TBP. Following the Ultrasene flush the rerun station will be flushed in turn with NaOH, HNO_3, and water.

Rerun Solvent Washing

	Gross γ, c/m/ml	Zr-Nb γ, c/m/ml	Ru γ, c/m/ml
1st Half of Solvent:			
Before Wash	3.5×10^7	1.9×10^7	2.0×10^7
	3.75×10^7	1.9×10^7	2.2×10^7
Following 4 washes, 2½% Na_2CO_3	1.30×10^7	4.3×10^6	8.8×10^6
	1.26×10^7	4.6×10^6	4.6×10^6
Following 1 wash, 5% HNO_3	2.4×10^7	9.3×10^6	8.3×10^6
Following 1 wash, 5% oxalic acid	8.5×10^6	2.7×10^6	7.1×10^6
	8.5×10^6	2.0×10^6	6.9×10^6
Following 3 alkaline washes, 2½% Na_2CO_3, plus 30 days decay	1.99×10^6	2.48×10^5	1.84×10^6
2nd Half of Solvent:			
Before Wash	1.2×10^7	9.0×10^6	3.5×10^6
	1.1×10^7	8.1×10^6	3.6×10^6
Following 4 washes, 2½% Na_2CO_3	1.1×10^7	9.3×10^6	2.8×10^6
Following 1 wash, 5% HNO_3	5.0×10^7	4.9×10^7	3.5×10^7
Following 1 wash, 5% oxalic acid	6.9×10^6	3.6×10^6	3.3×10^6
Following 3 alkaline washes, 2½% Na_2CO_3 plus 30 days decay	3.72×10^6	1.36×10^6	1.17×10^6
	3.04×10^6	1.15×10^6	1.37×10^6

RELEASE OF RADIOACTIVITY TO ENVIRONMENT

Activity Discharged to Seepage Basins, H Area

Activity Discharged to Basins 3/19 to 4/22, curies	
Nonvolatile β	1.91
I-131	0.32
Activity in Basins on 4/15, curies/cc	
I-131 Activity	
Basin No. 1	6.6×10^{-12}
Basin No. 2	4.8×10^{-12}
Basin No. 3	2.7×10^{-12}
Nonvolatile β	
Basin No. 1	4.4×10^{-10}
Basin No. 2	5.5×10^{-10}
Basin No. 3	1.2×10^{-10}
Seepage Basin pH	
Basin No. 1	2.2
Basin No. 2	2.6
Basin No. 3	9.8*
pH of Water Discarded during Month	2.0
Average Seepage and Evaporation Rate, gal/day	74,000
Level in Basin No. 3 on 4/22, in.	134.0

* Approximately 1800 lb of 60% nitric acid were added to Basin No. 3 in an attempt to reduce the pH. Water was circulated from one part of the basin to another at 1000 to 1500 gpm during addition of the acid. Prior to the acid additions the average pH in the basin was about 9.5. During the additions the pH increased to about 10.5 then decreased to between 9.0 and 9.5. It is assumed that sections of the basin which had not been sampled were strongly alkaline thus preventing a reduction in pH.

Tritium Process Direct Assistance

MARK VI-J PROCESSING

No slug breakage and very little chipping was observed during the decanning of twenty-four charges of Mark VI-J slugs. Product loss from decanning averaged approximately per slug. The unmachined interior of all slugs was coated with a white powder, presumably oxide from reaction with air sealed in the hollow space.

All charges were satisfactorily melted, including ten with a banded partial second layer containing 1/4 of the total charge.

HIGH PURITY PRODUCT PROGRAM

Plans are completed for the production of approximately of tritium with an absolute purity (neglecting He-3) of 99.9% or better, to be used in calorimeter determinations. The column feed, column, and product collection systems are to be evacuated, outgassed, and leak checked prior to processing this material. The columns will be operated with a raffinate loss of from of the product fed to "sweep" all deuterium and nearly all protium into the raffinate stream. The raffinate will then be collected and reprocessed at the conclusion of the program. Product rates are expected to be comparable to normal operation producing product averaging tritium with less than loss to the raffinate.

FURNACE WASHING

In preparation for repair, the activity level in the north furnace was reduced about 95% by a two-hour treatment with 1% nitric acid, followed by a complete water flush and a short water rinse. A weighted crucible inserted in the furnace reduced the volume of acid required to 45 gallons. The gamma activity radiated from the drum of collected washings was 4.2 r/hr at three inches, probably due to Zn-65. The drum contained about one curie of Zn-65 and about 100 curies of tritium. According to the stack monitor, less than 1 cc of product was lost to the furnace hood during treatment.

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PURSUANT TO SECTION 148, ATOMIC ENERGY ACT OF 1954, AS AMENDED

AND

DEPARTMENT OF ENERGY REGULATION 10 C.F.R. 1017

Wet and Dry-Stored CR Slugs

Byproduct Mass		
18	<0.01	0.02
19	.01	.02
21	.05	.02
22	.01	.02
24	.05	.03
28	.02	.04
40	<0.01	0.01

PRIMARY SEPARATION

After a steady decrease in peak first stage backpressures early in the month, the diffuser was given an air treatment. No lasting improvement was achieved because of a number of pump failures immediately following the activation. The extraction Sprengel pump drive motor failed during run 8, and during the next two runs (9 and 10), diffuser Sprengel pump 233B was inoperable because of a sheared pin in the drive assembly. The high first stage backpressures resulted from the reduced pumping capacity for stage 1 of the diffuser. During run 13, the drive belt slipped off the motor pulley of extraction pump 223A and was not replaced for two runs. Since the start of Mark VI-J processing, diffuser operation has been quite uniform, except during pump failure, showing no evidence of decreasing permeability.

BYPRODUCT LOADING

Loading of one trailer was completed. Analyses of cylinder banks are shown in the following table.

Z BED RECOVERY

Data from two recovery runs completed this month are summarized in the following table. In the second run, the tritium recovered was equivalent to per extraction run compared to the largest prior recovery of per run. This bed had been used to recover tritium from twelve Mark VI-J runs, and part of this increase is believed due to an increase in tritium oxide from these Mark VI-J extractions.

In the last recovery run, the circulating gas stream bypassed the Z bed during cooling to reduce the quantity of methanes readsorbed on the bed. To further reduce the quantity of hydrocarbons recycled through the stack stripper - Z bed recovery cycle, recovery byproduct will be stacked directly, bypassing the stack stripper. Laboratory data indicate that organic materials may poison the CA vessel in the stack stripper system. The reduction in hydrocarbons is also expected to reduce diffuser element poisoning, resulting in more uniform diffuser performance and fewer air activation treatments.

STACK LOSSES

Stack losses are summarized in the following table. Total losses decreased from last month to this month, because of the fewer runs processed. Losses averaged about run both months. No Z bed recovery byproduct was stacked during this period. Charge-discharge losses increased to an average of per run. Losses attributable to the decanning of Mark VI-J slugs averaged per run during the period in which these elements were processed.

Since reinstallation of the stack line air purge system, raffinate losses as calculated from mass spectrometer analyses have agreed within with the stack monitor indications. Prior to the air purge, these measurements of raffinate losses differed by as much as a factor of ten.

Reassembly of the booster ejector pump in the stack stripper system has been completed following modifications to the first stage mercury return line. The pump has been out of service for some time because of problems encountered in modifying the first stage and in replacement of English-thread bolts inadvertently lost during handling of the pump parts. An O-ring of Viton, an elastomer with exceptional temperature tolerance, was installed in the inlet flange on a test basis. After an adjustment of the power input to the pump as determined by measurement of the mercury boiler pressure, the pump will be returned to service.

The zeolite bed in the stack stripper system was replaced after processing waste from 52 runs.

Stack Loss Summary

Operation
Discharge-charge
First Evacuation
Byproduct
Raffinate
Miscellaneous
Total per Run
Total Combined Loss

DELETED VERSION

Tritium Process Improvement

DELETED VERSION

DELETED VERSION

EXTRACTION IMPROVEMENT PROGRAM

Assembly of the experimental extraction system was completed. Following calibration and checkout of individual components, the operability of the system was demonstrated by a dummy run with an unirradiated slug and subsequently by extraction and separation of gases from a single, low content Mark VI slug.

Initial extraction studies will be aimed at determining whether losses, incurred in extracting uncleaned, clad elements, eg, Mark VI-A or undecanned slugs, can be reduced or eliminated by processing at low furnace pressures. It is believed that the low pressure extraction process will reduce the extent of tritium exchange with hydrated oxides on clad elements and improve the yields which would otherwise be obtained with or without prior cleaning.

RECOVERY OF PRODUCT FROM NONSTANDARD ZEOLITE BEDS

Oxides of hydrogen isotopes were transferred from a Line 1, 6" zeolite bed (232-H-I-ST-F) to a standard 4" bed, and the receiver bed was delivered to the Separations Department for subsequent recovery in the Line 2 recovery system. The transfer run was terminated after the 6" bed had been heated to a final temperature of 700°C, and no further desorption was detectable.

STACK LOSS PROGRAM

Results of recent laboratory stripping studies (see last month's report) showed that trace quantities of hydrocarbons in the feed gas to the stack stripper system contribute to a reduction in the efficiency of the stripper operation. Preliminary laboratory runs were made with feed obtained from the Line 2 recovery system, which was unusually rich in hydrocarbons, principally methanes. However, since it was felt that the (relatively) high concentration of methane in recovery byproduct might lead to misleading results in analyzing stripping problems, normal byproduct material was used for all runs this month. However, the concentration levels of hydrocarbons were too low to permit an evaluation of catalyst performance on the basis of mass spectrometer data, and ion chamber data were generally inconclusive because of shifting backgrounds.

The stripping procedures used in the normal byproduct runs were identical to those in earlier studies. Feed was circulated, at a constant pressure of 700 mm, first through Hopcalite (at 500°C) for one hour, and then through zeolite (at 25°C) for an additional hour. Data from the first three runs are summarized in the following table and figure S-4. Data indicate that concentrations of all trace constituents were of the same order of magnitude as the normal analytical error of the mass spectrometer (±0.1%). Conversion efficiencies calculated from the changes in total tritium activity, as measured by ion chamber, were 100%, 46%, and 87% for runs 6, 7, and 8. However, the accuracy of these measured

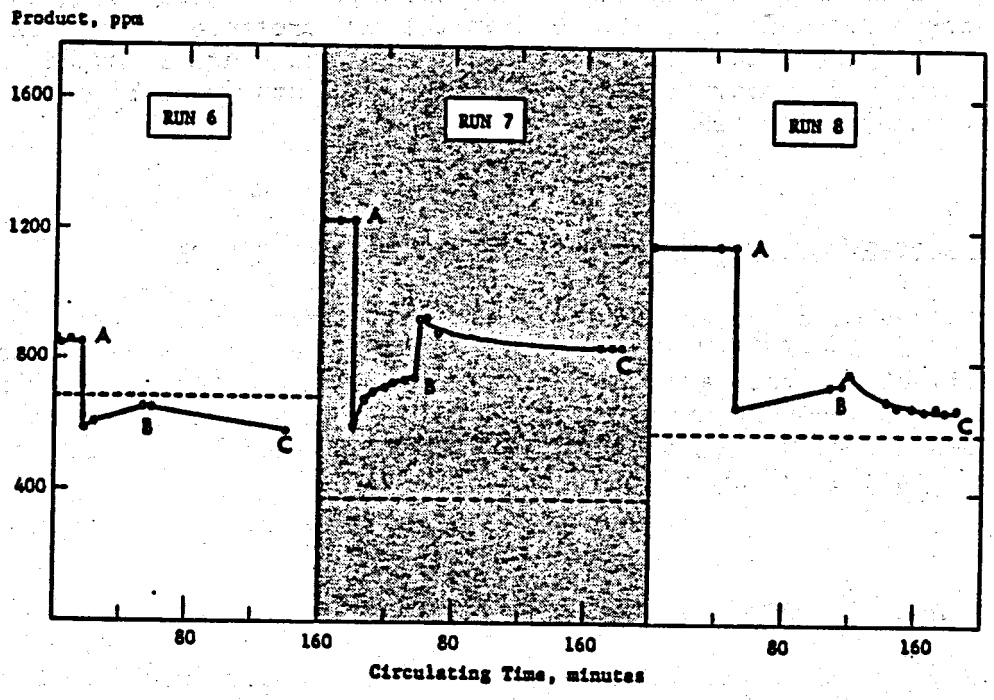


Figure S-4. Circulating Stripper System (685 mm)

----- Ion Chamber Background
A Hopcalite Cut In; Zeolite Bypassed
B Zeolite Cut In; Hopcalite Bypassed
C End of Run

efficiencies is questionable in view of shifting background levels. As can be seen from the curves in figure S-4 the shift in ion chamber background for run 7 was of the same order of magnitude as the apparent decrease in conversion efficiency. In general, the shape of the concentration-time curves in figure S-4 indicates an approach to equilibrium on both the Hopcalite and zeolite beds, paralleling the behavior obtained with recovery byproduct.

Before the start of the recent series of runs, blank samples were obtained from the system with helium circulation through the hot Hopcalite and zeolite beds. Results showed an apparent desorption of Mass 44, presumably carbon dioxide, from the Hopcalite, which subsequently disappeared on the zeolite when contacted with this absorbent. After a fourth run has been completed with normal byproduct feed, the conversion efficiency of the Hopcalite will be checked independently with elemental protium and/or tritium. The Hopcalite will then be contacted with air in an effort to recover all material adsorbed on the bed.

Mass Spectrometer Data
(Concentration in %)

Run No.→	6	7	8
Hydrogen			
Initial Feed	0.07	0.01	0.02
After Hopcalite Circulation	.05	.06	.01
After Zeolite Adsorption	.06	.13	.01
Tritium (Mass 6)			
Initial Feed	.04	.14	.05
After Hopcalite Circulations	.05	.05	.01
After Zeolite Adsorption	.03	.03	.02
Tritiated Methane (Mass 18)			
Initial Feed	.05	.11	.02
After Hopcalite Circulation	.12	.05	0.01
After Zeolite Adsorption	.07	.10	ND
Total Methanes (Mass 16 - 19)			
Initial Feed	.07	.11	0.05
After Hopcalite Circulation	.14	.05	.02
After Zeolite Adsorption	.07	.10	.05
Carbon Dioxide (Mass 44)			
Initial Feed	.11	.21	.12
After Hopcalite Circulation	.15	0.10	.15
After Zeolite Adsorption	0.04	ND	0.07

A new catalyst vessel was loaded with the Baker hydrogenation catalyst (0.5% palladium on activated alumina) which was scouted in March 1958, as a potential oxidation catalyst. The catalyst will be installed in the laboratory system to determine its performance as a high temperature oxide conversion catalyst in comparison with Hopcalite. Equipment is also being assembled to obtain information on the effect of coadsorbed methane and/or carbon dioxide on the adsorption capacity of zeolite for water.

GETTERING STUDIES

The laboratory program to develop a substitute gettering material was completed with a series of runs in which the effect of adsorbed water on the capacity of zeolite for protium adsorption was investigated. In all five runs completed, type 5A zeolite was loaded with 10.5 g of H_2O/100 g of adsorbent. The adsorptive capacity for protium at -195°C was reduced by 50% to 70% of the capacity of dehydrated zeolite at equilibrium vapor pressures between 50 and 300 microns. However, the trend of the curves illustrated in figure S-5 suggests that the reduction in adsorptive capacity decreases rapidly to a negligible amount as the equilibrium vapor pressure increases above 400 microns. These results, although significant, are not sufficiently critical to prejudice the use of zeolite as a getter. The capacity of water-loaded zeolite at 260 microns is comparable to that presently attainable with charcoal at 100 microns. An increase in equilibrium pressure of this magnitude would not interfere with the process as presently operated.

On the basis of work completed to date, type 5A zeolite appears superior to charcoal as a getter in nearly all respects. The specific advantages of zeolite are as follows:

- The potential explosion hazard inherent in low temperature adsorption of oxygen by charcoal (as the result of air inleakage) is not present with zeolite.
- The capacity of dehydrated zeolite is greater than that of charcoal throughout the entire range of pressures studied, and the capacity of water-loaded zeolite is adequate for the proposed application.
- The rate of adsorption by zeolite is greater than that of charcoal. Equilibrium is attained over zeolite in approximately one-half the time required with charcoal (20 to 22 minutes versus 45 minutes).
- The higher bulk density of zeolite as compared with charcoal permits the adsorption bed to be charged with 80% more adsorbent, increasing the effective adsorption capacity by the same percentage, and compensating for any potential reduction in capacity as the result of hydration.
- Zeolite has a slightly higher sorptive capacity for helium, a potential impurity in the Building 234-H process, than does charcoal (see last month's report).
- Zeolite can be desorbed satisfactorily at temperatures as low as 150°C, comparable to the required desorption temperature of charcoal.

Additional work will be aimed at determining the sorptive capacity at -78°C. Refrigeration of the bed with a dry ice-triclene mixture would effect a considerable cost savings through elimination of the large quantities of liquid nitrogen consumed during each adsorption.

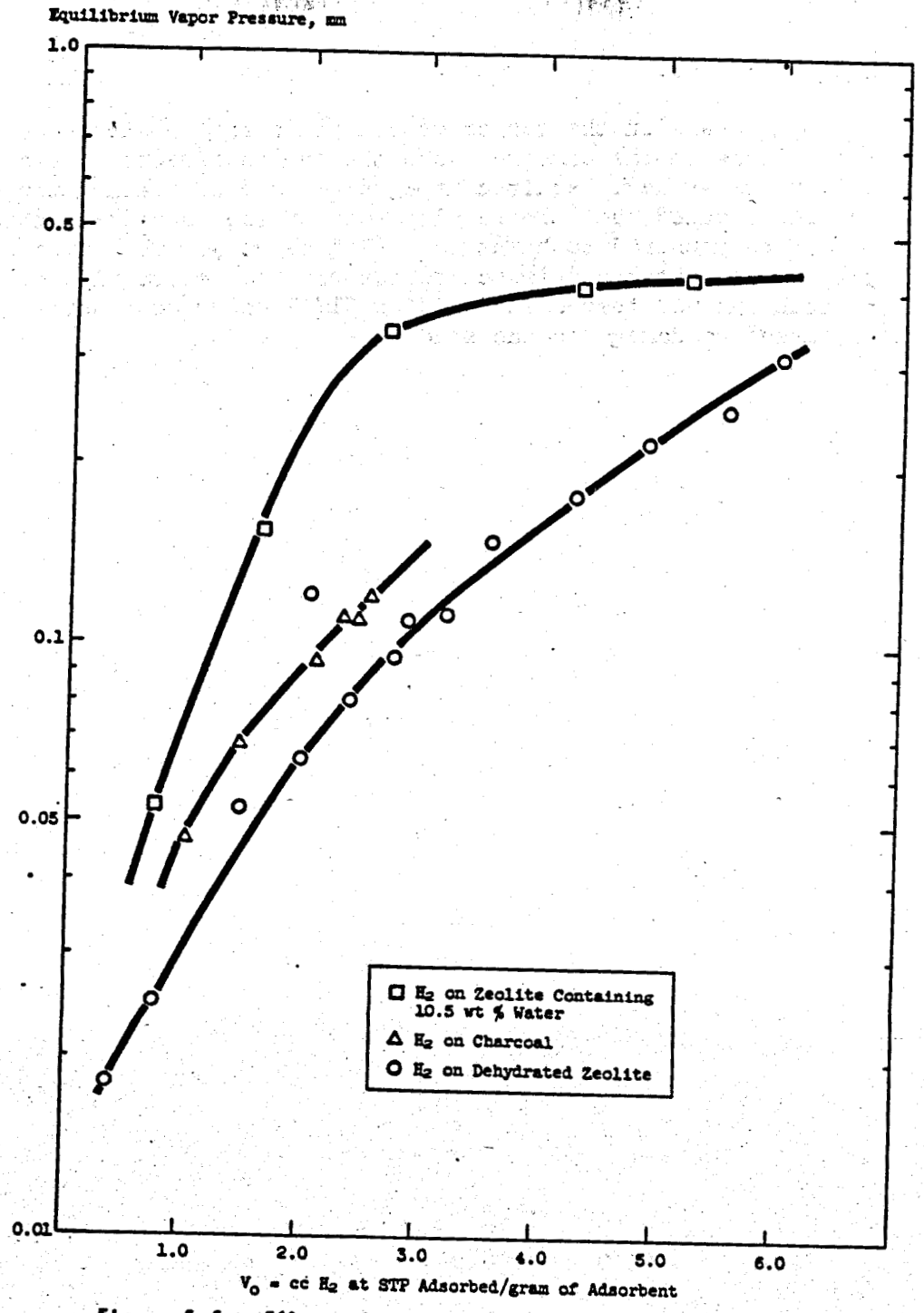


Figure 5-5. Effect of Water Vapor on Hydrogen Adsorption

Mechanical Assistance

CANYON ACME NUTS

The top threads on the canyon vessel studs are sometimes smashed by the beveled ends on the acme nut when the nut is impacted. Two vessels in Building 221-F have required repairs because of smashed threads. It has been recommended that the beveled ends of the threads at the bottom of the nut be machined to blunt ends (Higbee cut). The blunt end of the threads will withstand impact forces without deformation of the threads. One such nut was tested in Building 717-F under most severe conditions and caused no damage to the stud.

Heavy Water Separation

Extraction Area Operating Summary

	Building 412	
	March	April*
Performance Index	980	1020
**Bldg Gas Quality, % H_2S	99.9	99.8
First Stage		
Operating Time, %	97.0	95.7
Gas Flow, 1000 lb/hr	651	649
Liquid Flow, gpm to CT-1	343	341
HT-1 18th Tray Temp, °C	130	134.5
CT-1 Gas Inlet Temp, °C	33.9	33.4
**CT-1 Base Conc, mol % D_2O	0.054	0.051
**PC-1C Gas Inlet Press., psig	291	288
**Flow Limitations:		
CT-1 Carryover	0	0
FRCp-1 Wide Open	8	8
CT-1 Level Control	0	0
Second Stage		
Operating Time, %	96.8	95.4
Gas Flow, 1000 lb/hr	196	194
**CT-2B Base Conc, mol % D_2O	7.6	7.7
**Flow Limitations:		
CT-2B Loading	0	0
FRCp-2 Wide Open	5	6
CT-2A Level Control	0	0
CTD-2B Level Control	1	0
FRCp-4 Wide Open	2	2

* Downtime for overhaul 6 unit-days.
** Represents month-end figure, all others are monthly average.

PROCESS PERFORMANCE

The net transfer of heavy water from the Extraction Area to the DW Plant during April was 30,889 pounds of D_2O, an average of 1030 pounds per day. Hot tower temperature was increased from 130°C to 135°C on April 1 to increase productivity. As a result, the average Performance Index was about 4% greater than last month despite greater equipment outage. A summary of Extraction Area operation is given in the above table.

Unit 23 was shut down during the period April 11 to April 17 for repairs to the damaged 11th tray of HT-1, replacement of the leaking SC-1A tube bundle, and general overhaul and hydrostatic test.

All four units in the east wing of Building 412 were unintentionally shut down on April 10 by a failure in the electrical switchgear. Normal operation was restored in all four units within 12 hours.

Unit 27 was shut down on April 13 to replace a floating head gasket in the SC-1B heat exchanger. A large tube-to-shell leak of H_2S was detected prior to the wing shutdown on April 10. Hydrostatic test of the tube bundle revealed no leaking tubes. The floating head gasket was replaced and the unit returned to normal operation on April 14.

A siege of severe carryover was experienced on April 4 which persisted until April 10. Process flow rates were decreased by 2 to 9% in both stages to combat carryover from the first stage cold towers and the attendant loading in second stage cold towers. The worst condition was passed by April 7 and flow rates were gradually returned to maximum by April 10. Average production rate during the 6-day siege was about 4% below normal.

The Rework Unit was shut down during the period April 8 to April 17 for lack of suitable feed material. Operation was resumed when the Rework Unit evaporator had purified enough feed material to assure continued operation. The evaporator was started on April 13 and purification of about 62,000 pounds of high conductivity feed material was completed on April 30.

SHUTDOWNS AND FAILURES

During the month of April there were 7 partial or complete unit shutdowns compared with 9 in March. Following is a list of causes of shutdowns.

	Number of Shutdowns		Hours Downtime	
	Entire Unit	2nd Stage Only	Entire Unit	2nd Stage Only
Hydrostatic Test and Overhaul	1	-	154	3
Electrical Failure	4	-	40	5
Gasket Failure (27 SC-1B)	1	-	40	4
Electrical Failure*	1	-	15	3
		Total Downtime →	249	15

* An additional 13 hours of this shutdown occurred in March.

The east wing shutdown on April 10 occurred when the main 440-volt circuit breaker developed a short-circuit to ground. An inspection of the section of the damaged bus bar indicated poor contact existed between the "C" phase bus and the corresponding main breaker contact. It is believed that arcing and overheating at the point of poor contact reached the proportions necessary to ionize the surrounding air, which afforded a low-resistance path to ground.

REVISIONS IN ELECTRICAL SWITCHGEAR

Revisions are being made in the emergency electrical tie system that permits the transfer of power between wings in Building 412. In the past, malfunction of the tie system has caused shutdown of one wing when electrical difficulties occurred in the other wing. The revisions can be made without additional equipment shutdown. A summary of the principal changes to be made is as follows:

- The crash circuit for the gas blowers will be revised to prevent shutdown of blowers during failure of the 440-volt system. The blower crash circuit will be wired directly to the crash switch. The blower crash relays have been blocked in the closed position until a unit shutdown provides the opportunity to complete the necessary wiring. During the interim, the blowers will have to be shut down manually when a unit is crashed.

- Tie breakers and valve breakers will be converted to manual operation. Key interlocks will be provided to prevent closing both valve and tie breakers at the same time. This will eliminate the present, although remote, possibility of paralleling separate power sources through the 440-volt switchgear.

- The automatic features of both valve and tie breakers, which are inactivated by the changes described above, will be removed at the first wing shutdown.

REPLACEMENT OF TUBE BUNDLES

It is planned to replace extensively corroded tube bundles from raw-water-cooled heat exchangers in Building 412 with tube bundles from Building 411. Use of the bundles from Building 411 will delay the expense of a large scale retubing program for about two years. Otherwise, complete retubing of about four SC-1 tube bundles from Building 412 could be expected during 1959 because of extensive corrosion. One bundle has already been retubed this year with type 316 SS tubes at a cost of about $20,000. A summary is shown below of the condition of the twenty SC-1 tube bundles now in service in Building 412 or being held as spares.

- 2 bundles are extensively corroded and probably will require complete retubing.

- 12 bundles have a known history of severe corrosion or have had leaks in tubes that have since been plugged or replaced.

- 4 bundles have been in service for less than one year.

- 2 bundles have not been inspected.

Four SC-1 tube bundles that show little or no corrosion will be maintained as spares at all times for Building 412. All bundles removed from Building 411 will be inspected visually and with the Probolog.

TUBE SHEET VENTS FOR RAW WATER EXCHANGERS

Tube sheet vents on raw water exchangers are being activated to prevent accumulation of chlorides, corrosion sludge, and stagnant water at the fixed tube sheet. Such accumulation may have caused the stress-cracking of tubes in 21SC-1A in early January. Nozzles to permit activation of the tube sheet vents in Unit 23 were installed during the recent routine overhaul. Piping will tie the existing vent port on the fixed tube sheet of each raw water exchanger to the new nozzle; each nozzle is located on the raw water outlet line downstream from the flow control valve. Flow indication will be given by a bulls-eye flow indicator. Calculations indicate that a vent flow rate of 20 gpm maximum per exchanger can be obtained with this piping. The total cost for activating the vent ports on all SC-1's, SC-2's, and AC-2's in Building 412 was estimated at $1500.

A less expensive installation, that tied the vent port into an existing nozzle upstream of the control valve, was tried in 26SC-1B. This installation did not permit sufficient flow.

SULFUR PLUGGAGE

Operating difficulties from sulfur pluggage have decreased during the past year. Improved deaeration of feed water to the Extraction Area has reduced substantially the formation of sulfur in the units. Installation of a trap has caused the sulfur formed to accumulate in a less troublesome location at the base of CT-2B.

Improved Deaeration. Improvement in vacuum deaerator performance in the Water Treatment Plant is attributed to two factors. First, both extensive maintenance and improved performance checks of the deaerator system have reduced the amount and duration of upsets in the system. Second, the decreased load on the deaerators from the shutdown of Building 413 has apparently improved deaerator efficiency.

Fifteen months ago, the oxygen concentration of feedwater was seldom lower than 0.1 ppm. This concentration of oxygen will react with hydrogen sulfide to form 250 pounds of sulfur in one unit during a year. The oxygen concentration during the past year, after improvements in deaeration, has averaged less than 0.05 ppm, which is equivalent to an annual deposition rate of less than 125 pounds of sulfur per unit. Previously, 300 to 400 pounds of sulfur have been removed from the CT-1 during overhaul. In Units 22 and 23, only about 25 pounds were removed from this location.

Sulfur Trap. During the round of unit overhauls last year a standpipe, designed to trap sulfur in the base of CT-2B, was installed in the liquid outlet nozzle of CT-2B in four units. Prior to the installation of the traps, approximately 200 pounds of sulfur were removed from the CTD-2B, which receives liquid leaving the base of CT-2B. Since this amount of sulfur filled the drum, it was frequently necessary to "hot flush" the CTP-2B pump and the associated piping in order to dislodge the sulfur which overflowed from the drum.

The recent overhaul of Unit 23 afforded the first opportunity to inspect a trap; this trap was installed 9 months ago. Approximately 300 pounds of sulfur were removed from the trap in the base of CT-2B and only 5 pounds were found in CTD-2B. In contrast the major portion of the sulfur removed from Unit 22, which had no trap, was from CTD-2B. Unit 22 was inspected just 6 weeks ahead of Unit 23 and had operated for a period of 7 months since the last overhaul.

Removal of sulfur in excess of the estimated 125 pounds per year formed is attributed to accumulation of sulfur in the unit during the past six years and incomplete removal during previous overhauls.

UNIT 23 INSPECTION

Routine overhaul, hydrostatic test, and inspection of Unit 23 was carried out during April 11 to 16. The unit was last inspected 9 months ago. The significant results of the inspection are summarized below.

First Stage Hot Tower. The wall of the liquid drawoff box at the 11th tray was corroded and perforated over the entire surface. This wall was made of carbon steel instead of the specified type 304 SS. Perforation and disintegration of approximately 15% of the wall had allowed an excess of liquor to flow from the hot tower into the humidifier section. This damage was the same as that observed in Unit 22 last month.

First Stage Cold Tower. The first separation of lining from the weld metal at a tray support ring was observed in CT-1, manufactured by Wyatt Metal and Boiler Works. About 4 feet of weld separation was present at the ring supporting the bottom tray.

About 98% of the horizontal welds and 25% of the vertical welds were corroded. No separation of lining was observed at vertical welds.

Two samples of lining welds were removed with the Weld Prober from the base of CT-1 where separation of the horizontal welds had occurred. Visual inspection and examination by dye penetration of the samples and sample cavities revealed no cracking of the carbon steel base metal. The samples were sent to the Engineering Assistance Section for metallurgical examination.

Waste Stripper. The bottom downcomer (across the center of the tower) and the associated seal pan and weirs were damaged in a manner indicative of rapid condensation of steam in the bottom of the tower. This damage was also seen in Unit 22 but has not yet been traced to any unusual operating condition.

Stripper Exchangers. No pluggage was found between the turbulence promoters and the tube walls in the SX-1's. Severe pluggage at this location had caused shutdown of Unit 22 about one week after overhaul. Two promoters from SX-1C in Unit 23 were examined for erosion and found to have a maximum of 5 mils wear (original thickness, 40 mils) since they were installed 9 months ago. The turbulence promoters were left in service and no indication of pluggage has been noted during the first two weeks of operation after overhaul.

Secondary Condensers, SC-1A and SC-1B. Hydrostatic test showed 22 leaking tubes in SC-1A. Severe pitting and pluggage had occurred along the entire length of tubes located on the periphery of the bundle. No visual damage or pluggage was found in SC-1B. A Probolog examination also revealed no corrosion of SC-1B.

FRCp-3 Reducer. Corrosion and erosion 3/8" deep had occurred on the 3-inch flange of the FRCp-3 reducer since it was last inspected 9 months ago. The reducer was replaced.

BUILDING 412, PURGE TOWER SYSTEM INSPECTION

The purge tower system in Building 412 was shut down on April 8 for routine hydrostatic test and inspection. No increase in the pitting of the type 316 SS cladding in PT-1B had occurred since the last inspection 6 months ago. At that time, pinpoint pitting of the cladding on the top head and walls measured a maximum depth of 3/16".

The weld cap on the discharge tee of the purge tower pump failed approximately one week before the pump was shut down for inspection. This cap was eroded in an area 4" in diameter and 3/8" deep. The leak was through a hairline crack at the thinned section.

BUILDING 413, PURGE TOWER SYSTEM INSPECTION

The Building 413 purge tower system was hydrostatically tested and inspected on April 21. Inspection of these facilities followed six months of operation in stripping carbon dioxide from inert gas. No corrosion or pitting was observed on any of the type 316 SS cladding in PT-1A or PT-1B.

BUILDING 412, RICH LIQUOR AND WEAK LIQUOR STORAGE TANKS (RLT AND WLT) INSPECTION

The RLT and WLT Tanks in Building 412 were hydrostatically tested and inspected on April 27. The tanks were last inspected approximately one year ago.

There was no corrosion or pitting of the type 316 SS cladding on the walls of either the RLT or WLT. The bottom dished heads on both tanks were generally roughened and corroded 1/64" to 1/32" deep. No other attack was observed.

COST OF REWORKING DEGRADED HEAVY WATER

The basis for a schedule of charges for reconcentrating heavy water from off-Plant sources has been calculated using the Rework Unit operating experience during the last six months of 1958 for both costs and production rate. The production rate was adjusted to reflect an anticipated annual outage of 7%. A breakdown of the reworking costs per pound of D_2O received is presented in the following table as a function of the concentration of material to be reworked.

The total unit cost is accumulated from three different expenses: the cost for separative work, the scheduled discards, and the handling losses. These three items are explained in more detail below.

- The units of separative work required to separate the degraded moderator into 99.75 wt % D_2O product and 2.5 wt % D_2O waste is shown as a function of the initial concentration in column 3 of the following table. The cost for separative work in dollars per pound of D_2O received is shown in column 4. This cost includes all of the direct charges for operating the Rework Unit and allocated overhead and depreciation charges.

- The scheduled discard of D_2O waste is shown in column 2 as percent of the D_2O received. The cost for replacing this loss is based on replacement from the stockpile where the production cost was assumed to be $24.35 per pound ($28.00 less the 15% service charge added by the AEC).

- The handling loss, shown in column 6, was estimated from the experience in 1958 and is for nonscheduled losses such as leaks, spills, etc. This cost was also based on replacement of the D_2O from the stockpile at $24.35 per pound.

The total cost per pound of D_2O received is shown in column 7. This is a production cost to which a service charge will be added by the AEC to form a charge schedule.

Moderator Rework Cost

Conc Received, wt % D_2O	Scheduled Discard,* %	Separative Work Units**	Cost, $/lb degraded D_2O received			
			Separative Work	Scheduled Discard	Handling Loss	Total
10	23.1	19.138	7.06	5.62	0.47	13.15
20	10.3	15.474	5.71	2.51	.38	8.60
30	6.0	12.808	4.72	1.46	.31	6.49
40	3.8	10.888	4.02	0.93	.27	5.22
50	2.6	9.382	3.46	0.63	.23	4.32
60	1.7	8.108	2.99	0.41	.20	3.60
70	1.1	6.946	2.56	0.27	.17	3.00
80	0.6	5.780	2.13	0.15	.14	2.42
90	0.3	4.392	1.62	0.07	.11	1.80
95	0.1	3.356	1.24	0.02	.08	1.34
99	nil	1.452	0.54	nil	0.04	0.58

* D_2O discarded as % of D_2O in material received.

** Units of work required to separate each pound of D_2O in material received into fractions at 99.75 wt % D_2O and 2.5 wt % D_2O. Calculations were based on Cohen's Value Function. See "Theory of Isotope Separation," Karl Cohen, 1951, National Nuclear Energy Series.

REWORK EVAPORATOR OPERATION

Satisfactory performance has been obtained thus far with the evaporator recently installed for purification of Rework Unit feed and product. Emphasis has been on a production run to obtain a quantity of feed for the Rework Unit and consequently a number of characteristics of the equipment have not yet been defined.

During the period April 13 to April 30, about 62,000 pounds of degraded heavy water were processed. The conductivity of this material was reduced from 625 micromhos to 2 micromhos. Thus far, the throughput has been limited to a maximum of 240 pph by a steam flow restriction at the pressure regulating valve ahead of the calandria.

The buildup of solids in the calandria was controlled by a small purge to maintain the solids content at less than 10% to control foaming. About 2 drums of calandria purge liquor were accumulated during the production run. Eventually, this purge liquor will be processed in the evaporator and the solids content increased further. The D_2O will then be displaced from the calandria with H_2O and the residue discarded.

The purification factor, defined by the ratio of solids in the calandria to the solids in the condensate, has been as high as 130,000 but the accuracy of this figure is limited by ability to analyze the condensate.

A normal heat transfer coefficient in the calandria of 400 to 500 $Btu/(hr)(ft^2)(°F)$ was obtained during the production run with a calandria temperature difference of 14° to 16°F.

Glossary of Terms - GS Area

FRCp-1	Flow recorder-controller, first stage process gas
FRCp-2	Flow recorder-controller, second stage process gas
FRCp-3	Flow recorder-controller, first stage liquid
FRCp-4	Flow recorder-controller, second stage liquid
CT-1	First stage cold tower
CT-2A	Second stage cold tower (column A)
CT-2B	Second stage cold tower (column B)
CTD-2B	Second stage cold tower drum
HT-1	First stage hot tower
HT-2A	Second stage hot tower (column A)
HT-2B	Second stage hot tower (column B)
PT-1A	Purge tower - lower section
PT-1B	Purge tower - upper section
S-1	Waste stripper
SX	Stripper heat exchanger
LH	Liquid heater
PC	Primary condenser
SC	Secondary condenser
AC-2	Blower aftercooler - second stage

ENGINEERING ASSISTANCE SECTION

Raw Materials Technology

Mark V-A Fuel 302

Mark VI-J and Mark VI Fuel 302

Mark VII-A Slugs 307

Li-Al Alloy 316

Engineering Assistance

Instrument 321

Mechanical 328

Metallurgical 332

Miscellaneous 342

Reports and Technical Papers 601

Raw Materials Technology

MARK V-A FUEL

MARK V-A HOT-PRESS BONDING

A program has been initiated to continue the Mark V-A hot-press bonding development carried out by the Technical Division. It is desirable to gain additional cladding and irradiation experience and to continue to investigate the economics of the Mark V-A element. A Test Authorization is now being circulated for approval, which will authorize the cladding and irradiation of as many as 12 assemblies, or reactor positions, in each natural uranium charge in R Area over a one-year period.

Initially, cores will be fabricated at NLO by extruding oversize tubes and machining the cores to final size. Cores provided previously by this process contained large inclusions, or voids, which interfered with nickel plating at Sylcor. Revised casting procedures and less-impure charge material are being evaluated as methods of improving core quality.

Long uranium tubes, extruded to size by the American Brass Company, will be used if satisfactory cores are not immediately available from NLO.

The cores will be beta-treated using air cooling through the beta-to-alpha transition temperature until an improved heat treatment is available.

Cladding and welding will be done at Sylcor.

MARK VI-J AND MARK VI FUEL

Mark VI-J Reject Summary, 3/26 - 4/22

Type Reject	Number Inspected	Number Rejected	Yield, % This Month	Yield, % Last Month
Casting	48	0	100	100
Core Machining	48	0	100	100
Tube Fabrication	105	11	89.5	78.5
Tube Finishing	79	1	98.7	99.2
NTG	119	2	98.3	94.6
		Over-all Yield →	86.8	73.7

Mark VI Reject Summary, 4/2 - 4/22

Type Reject	Number Inspected	Number Rejected	Yield, % This Month	Yield, % 11/25 to 12/24/58
Casting	192	4	97.9	100
Core Machining	188	0	100	99.3
Tube Fabrication	87	5	94.3	85.4
Tube Finishing	50	0	100	99.5
NTG	23	0	100	99.3
		Over-all Yield →	92.3	83.8

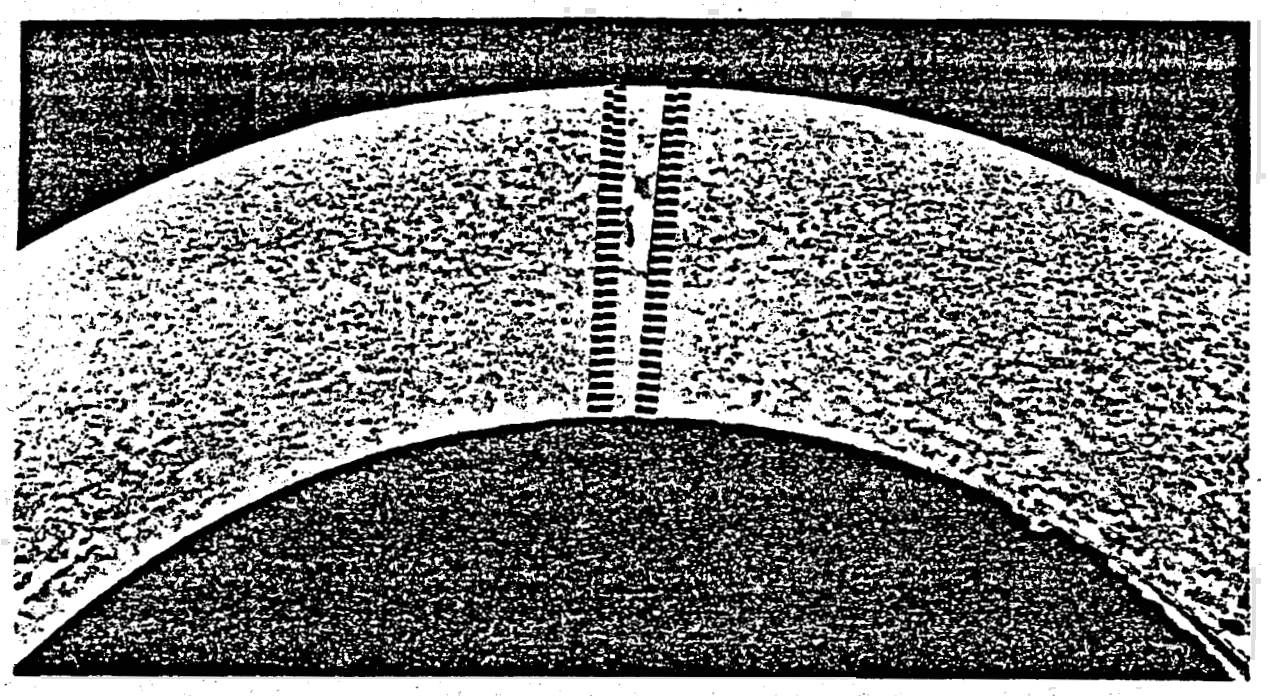
Increased fabrication yield for both Mark VI-J and Mark VI fuel was due to fewer rejections for tube blisters.

COMPONENTS

Tubing of type 1100 aluminum alloy, for Mark VI-J inner sheath fabrication, was test extruded to finished diameters with a porthole die. Adequate diametral control, within ±0.005", was obtained. Recrystallization across weld lines appeared complete (see figure EA-1) on tubing from the initial billet through the die. However, subsequent extrusions had poor surfaces and an excessive (36") length of coring. Coring and laminations in the porthole-extruded inner sheath tubing are shown in figure EA-2. Because of these defects associated with the porthole-extrusion of sheath material, inner sheath requirements will be met using mandrel-extruded tubing that requires machining to finished diameters.

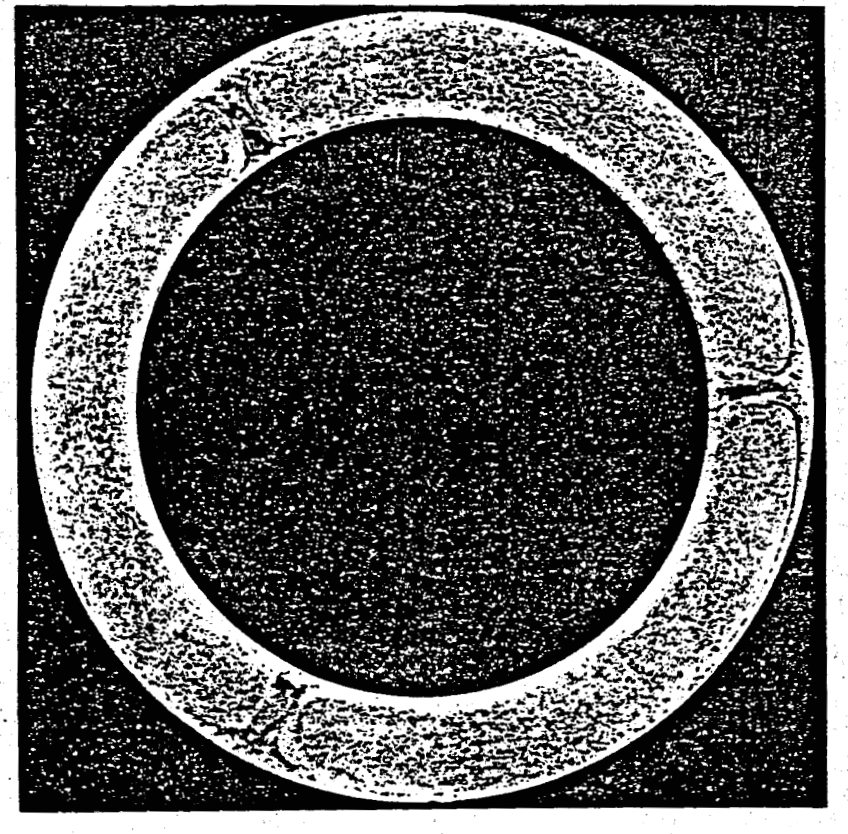
CORE THICKENING

A new end plug alloy of type 5050 aluminum (1.0 - 1.8% magnesium, remainder aluminum) was recommended to reduce the core thickening tendency in Mark VI-J tubes. Two test billets containing type 5050 aluminum plugs were extruded. A 16° plug-core interface was selected because of its good control of end-defect configuration on Mark VI tubes. Preliminary destructive results indicate that the front defect, shown in figure EA-3, is satisfactorily centered and exhibits no thickening. However, the rear defect (plug thickened), shown in figure EA-4, appears different from any previously observed. Thickening would occur when the hardness of the hot plug exceeded that of the core. This could result from the progressive increase in billet temperature normally experienced during extrusion. When the front interface passes through the extrusion die there is a match of plug and core hardnesses, whereas the higher temperature of the rear interface results in a hardness mismatch. Subsequent tests will be run with lower extrusion temperatures and different rear core-plug interface angles.



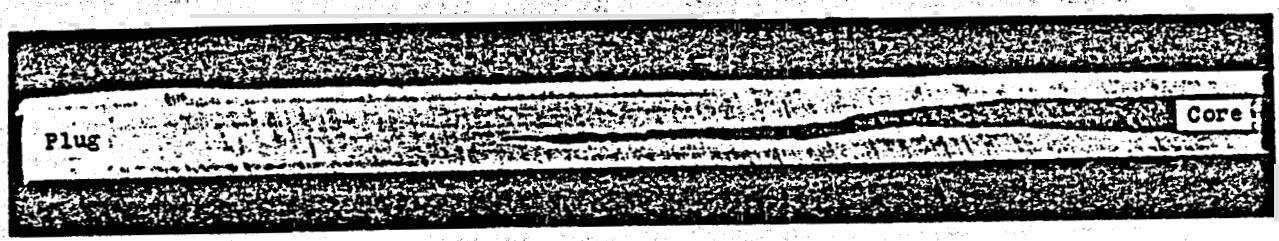
Approx 3X Neg EA-8918-M

Figure EA-1. Recrystallized Weld in Porthole-Extruded Inner Sheath Tube



Approx 1X Neg EA-9542-M

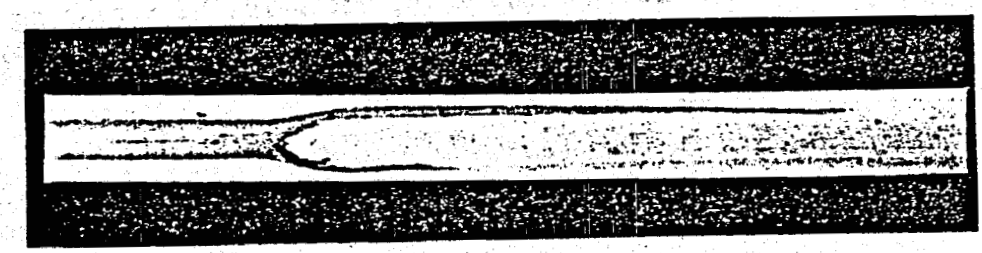
Figure EA-2. Transverse Cross Section of Inner Sheath Tubing 11 inches Behind Billet-to-Billet Interface. Note coring and laminations.



Approx 3X

Neg EA-10239-M

Figure EA-3. Front End Defect of Mark VI-J Fuel Tube Showing a Normal Tapered, Centered Core Section



Approx 3X

Neg EA-10241-M

Figure EA-4. Rear End Defect of Mark VI-J Fuel Tube Showing Thickening of Type 5050 Aluminum End Plug

ALUMINUM OXIDE FILMS

As reported in last month's report, films produced by the boiling ammonia solution treatment were difficult to identify. SRL has now identified most of the samples, formerly marked as unidentified, as the desired alpha aluminum oxide monohydrate film. Identification was possible because of a new chemical film stripping technique that supplied better samples for X-ray diffraction.

In view of these results, the ammonia treatment appears to produce the desired film in acceptable thicknesses. However, the corrosion protection afforded by this film remains undetermined. Electron microscope examinations have shown the film is built up of fibers extending outward from the film surface, as compared to a large crystalline structure for the monohydrate film produced by steam autoclaving. The fibrous structure may not be as corrosion resistant as the crystalline structure. Loop tests of the various oxide films are being initiated by SRL at the Babcock and Wilcox Research Station, Alliance, Ohio, to determine corrosion merits of the various films.

FLUOROSCOPE MODIFICATION

An ionization chamber, fabricated by the Instrument Assistance Section, was installed in the Building 321-M fluoroscope. This will be used in the determination of the density of "hot spots" found in Mark VI and Mark VI-J fuel elements. "Hot spots" were first reported in the monthly report for December 1957, DPSP 57-1-12, and have since been found infrequently. The ionization chamber, with a Beckman amplifier and a recorder, measures the difference in current produced by the X-ray beam passing through a high density area as compared with an adjacent normal density area. The current difference will be calibrated so that specifications authorized in TA 3-399, "Nonuniform Tube Concentrations," can be followed.

MARK VII-A SLUGS

MARK VII-A-SRP YIELDS

About 115,000 Mark VII-A-SRP slugs were canned during April. The overall yield was 63.6% compared to 68.5% for March. The major causes for rejects were poor bonds, X-ray and pits, as shown in the following table.

Reject	% Defective April	% Defective March
Nonseat	3.4	4.9
Spire Damage	1.5	2.6
Poor Bond	5.0	3.1
Nonbond	1.8	3.4
X-ray	10.2	8.5
Pits	7.1	0.5

X-Ray Rejects. Beginning April 14, the X-ray reject rate declined from a level of about 13% to a level of 8%. This decrease is attributed to the following:

1. Close attention to aluminum component cleaning, particularly degreasing.

2. An operation evaluation test designed to determine the effect of operator variability on reject rates and yields.

In the operator evaluation test, the operators hold their assignments during the canning of an entire 100-slug lot rather than rotating assignments. This same test was run twice before. X-ray rejects were decreased and yields were increased during both tests.

Further work will be done to develop an optimum operator assignment schedule.

Nonbonding. It was reported last month that a correlation was observed between nonbonding and the tin content of the AlSi dip bath. The same correlation was found to exist between nonbonding and the tin content of AlSi canning bath during April. When the tin content of the canning bath increased to (maximum allowed nonbonds increased from 1% to 25%. The nonbond reject rate dropped to 1% when this AlSi was discarded and replaced with virgin AlSi.

Spire Pitting. The daily pitting reject rate, as determined by final inspection, increased from less than 0.5% to a maximum of 12.0% in April. These pits show intergranular corrosion and have been found only in the spire. The pits first show up after the slug has been canned and most of them are found within one inch of the top of the slug. Figure EA-5 shows these pits as they appear on a spire.

Destructive examination of selected samples show the pits to be as much as 0.015 inch in diameter and 0.004 inch deep. Figure EA-6 shows a section through a typical pit.

To determine whether the stainless steel spring inserts affected the pitting reject rate, 600 cores were canned without springs in the spires. The spires with no springs had no visible pits after canning, while spires containing springs had numerous pits, as shown below.

	No. of Slugs	Percent Defective				
		Pitting*	Nonseating	Spire Damage	X-ray	Nonbonding
No springs	600	0.0	8.9	6.2	10.1	1.9
Springs (control)	800	10.0	2.5	2.9	9.9	0.3

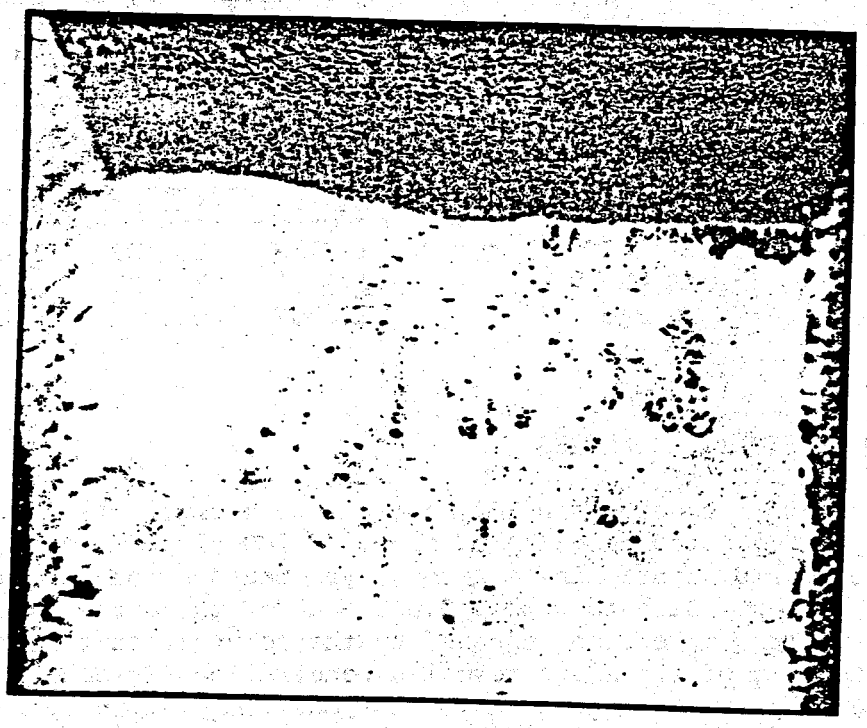
* Estimated since slugs have not been autoclaved and final inspected. Pits are more readily seen on an autoclaved surface.

Spire damage was higher when no springs were used, confirming earlier tests. The higher nonseating reject rate was caused by floating of the unweighted spires.

It has been reported that water, in contact with stainless steel and aluminum at high temperatures, will cause rapid galvanic corrosion. To check this, 200 slugs were shallow-quenched keeping the spire chimney above the quench water level so that no water splashed into the spire. Another 200 slugs were deep-quenched by submerging the slug and allowing water to enter the spire. The shallow-quenched slugs appeared to be free from pits while the deep-quenched slugs appeared to have numerous pits in the spires. Final results will be determined after the test and control slugs have been autoclaved. All slugs are now being shallow-quenched to prevent water from entering the spire.

CANNING SLEEVES

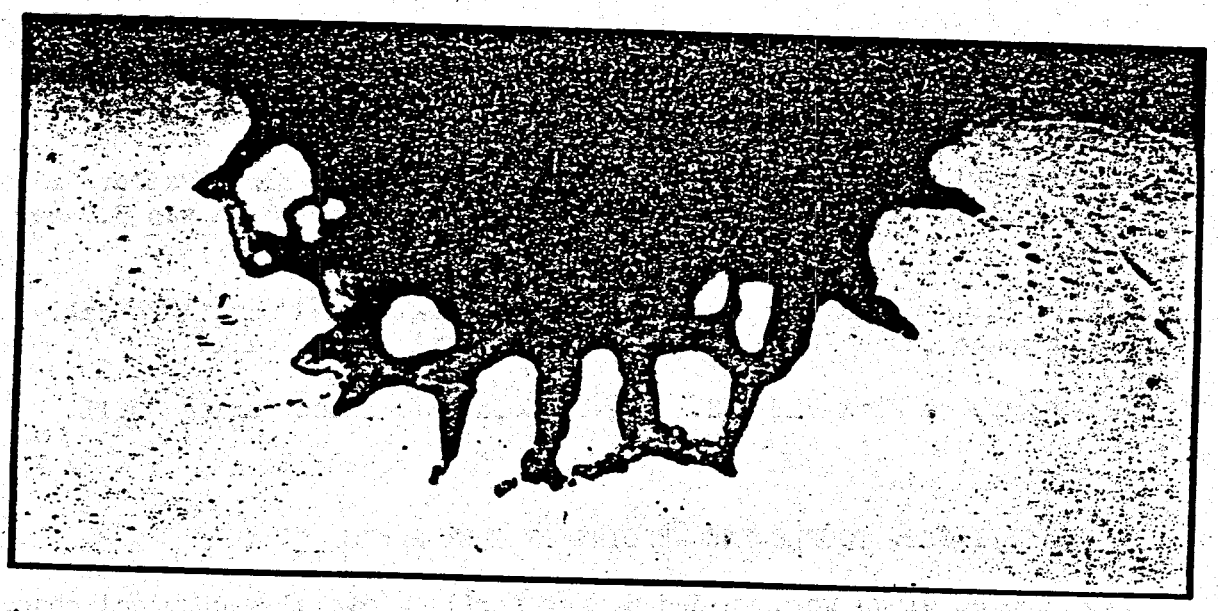
Lots of 100 Mark VII-A canning sleeves from each of the three current sleeve fabricators have been studied for dimensional changes during their first fifteen canning cycles. Inside diameter measurements were taken after each cycle and compared to the original average sleeve size. The results, shown in the following table for the initial and final measurement, show the Gougler sleeve to be dimensionally more stable than either the Martin or Slocomb sleeve. The Gougler sleeve is fabricated from SAE 1025 steel; the Martin and Slocomb from SAE 1015. Since the differe in steel composition is believed to be an important factor affecting sl life, the sleeve steel specification has been changed from SAE 1015 to SAE 1025 for current orders.



Approx 10X

Neg EA-10352-M

Figure EA-5. Typical Cluster of Pits in a Mark VII-A-SRP Canned Slug Spire, After Autoclaving



Approx 500X

Neg EA-10410-M

Figure EA-6. Photomicrograph of a Typical Pit. (The pit is 0.004 inch deep and 0.005 inch in diameter.)

An order of sleeves currently being fabricated by Slocomb from SAE 1025 steel will be checked for an increase in cycle life. The previous "half-life" for Slocomb SAE 1015 sleeves was 20 to 35 compared to about 80 for the Gougler SAE 1025 sleeves.

Sleeve Fabricator →	Inches		
	Martin	Slocomb	Gougler
Initial ID	1.2138	1.2138	1.2133
Final ~~OD~~ ID	1.2131	1.2124	1.2134
ID Change	-0.0007	-0.0014	+0.0001

BOND-PENETRATION TESTING

A laboratory model of an eddy current penetration tester has been used to inspect the outer cladding of about 10,000 slugs. The tester was set to reject a slug containing a groove-type penetration having a 0.060-inch-wide base and extending to within 0.010 inch of the outer cladding surface. Twelve slugs have been rejected by the penetration tester. Destructive examination of the slugs revealed penetrations in some of the rejected slugs as shown in figure EA-7. The maximum penetrations found in these slugs ranged from 0.004 inch to 0.029 inch, with minimum cladding thickness of 0.007 inch to 0.030 inch.

The tester will be modified to inspect the outside and inside cladding as soon as the necessary equipment can be fabricated and installed. It is expected that this modification will be completed by May 15. Design and fabrication of the Mark VII-A production bond-penetration tester is proceeding.

AUTOCLAVE AND REACTOR FAILURES

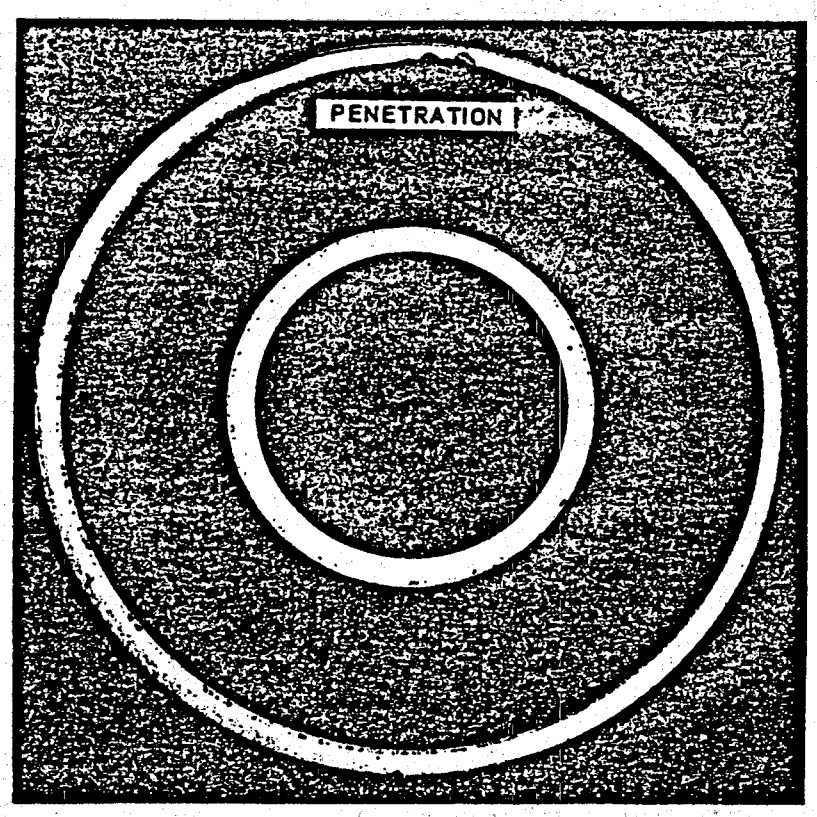
Six Mark VII-A-SRP slugs failed the autoclave test for a failure rate of 0.9 per 10,000 slugs tested. The failure rate for March was 3.0 per 10,000.

All failures during April, which have been examined metallurgically, were found to be caused by AlSi penetration of the spire wall.

There were no Mark VII-A-SC slug autoclave failures during April.

TA 3-395/1-662, DOUBLE-END WELDING OF FUEL SLUGS

Two hundred slugs were welded during April. Several mechanical changes were found necessary and the machine was shut down for modification (see Engineering Assistance portion of this report.) Necessary equipment changes have been made and welding should resume the first week in May.



Approx 3X Neg EA-10209-M

Figure EA-7. AlSi Penetration Into Cladding of Mark VII-A-SRP Slug

TA 3-418, BLOOMING AND CONTINUOUS-MILL PRACTICE

About 5,000 Mark VII-A-SRP and 3,000 Mark VII-A-SC fuel cores fabricated from rod rolled by a new schedule of blooming-mill and continuous-mill passes have been canned and finished. The new schedule comprises 19 rather than 21 blooming-mill passes and gives a uniform reduction per pass through the continuous mill.

Detailed canning and finishing data for the Mark VII-A-SRP test cores were reported in the March monthly report, DPSP 59-1-3, p 310. Results showed test cores were comparable to standard cores in canning quality.

Data are reported below for the Mark VII-A-SC test cores. Process yields for standard production from the same period are reported for comparison.

	Test		Standard
	No. Cores	%	%
NLO			
Cores Produced	3024	-	-
Process Yield	1869	61.8	76.0
Surface Quality Rejects	1078	35.0	16.0
Dimensional Rejects	22	0.7	6.0
Machining Rejects	77	2.5	2.0
Sylcor			
Cores Canned	1530	-	-
Canning Yield	1463	95.6	96.6
SRP			
Cores Finished	1463	-	-
Finishing Yield	1306	89.3	93.2

The data show that the NLO process yields were significantly lower for the test cores than for the control cores. Sylcor canning and SRP finishing yields for both test and control groups are comparable, within normal variations.

Metallurgical examination of as-received core blanks and finished slugs showed that both metal quality and canned slug quality were comparable to regular production material.

A further fabrication evaluation may be made pending the completion of irradiation tests in the R-6 reactor cycle.

MARK VII-A-SC YIELDS

	No. of Slugs	Yield, %
Canned at Sylcor	9,805	96.4 (96.6)*
Finished at SRP	18,091	91.8 (93.2)*

* Last month's yield.

TA 3-384 SPECIALLY HEAT-TREATED MARK VII-A-SC SLUGS

About 40,000 core blanks were heat-treated during the month. This increase over the normal 17,000 to 20,000 per month was due to an increase in forecasted Sylcor production and anticipated requirements of the forthcoming alpha canning tests at SRP. One core blank per NLO shipping box (310 cores) was destructively examined for transformation and grain size. No untransformed cores were found, and grain size was between 0.2 to 0.3 mm. The success in controlling grain size was due to an improved method of operating the bronze furnace. The current procedure requires a startup period of about two hours to establish operating temperatures before beginning core heat treatment. The rate of heat treatment now used is three cores per 55 seconds, but either 2 or 4 cores per 55 seconds may be run by adjusting the furnace heat input. However, a continuous rate of 4 cores per 55 seconds can significantly shorten furnace life and cause unscheduled shutdowns.

TA 3-442/1-693, SINGLE WELD SYLCOR SLUGS—IRRADIATION TEST

Approximately 2400 single-weld Sylcor slugs have been finished for irradiation in the P-7 reactor cycle. Finishing yields and reject rates for these slugs were equivalent to those of standard double-weld slugs as shown below.

		% Defective		In-Process
	No.	Poor Bonds	Rework	Yield, %
Single-weld	2338	4.1	4.2	90.2
Double-weld (control)	2000	3.6	4.0	90.9

Metallographic examination of ten acceptable single-weld slugs showed their quality to be slightly better than that of double-weld slugs. Data are summarized below.

		Minimum Thickness, inch					
	No.	Jacket (off core)		Jacket (off cap)		Weld Throat	
	Examined	Avg	Min	Avg	Min	Avg	Min
Single weld	20	0.038	0.036	0.031	0.024	0.030	0.007
Double weld	21	0.038	0.031	0.029	0.012	0.013	0.000

A group of 860 single-weld slugs were welded without preliminary machining of the interface. The finishing yield of the unmachined slugs was about 12% higher than for the machined slugs. Reject rates for both poor bonds and rewelds were lower as shown below.

	No. of Slugs	% Defective Poor Bonds	% Defective Rewelds	In-Process Yield, %
Not machined	860	3.3	1.3	93.8
Machined	700	7.7	8.0	82.3

Metallographic examination of welds from five slugs each of both unmachined and machined single weld slugs showed an average weld throat thickness of 0.035 inch for the unmachined slugs compared to 0.026 inch for the machined slugs. Unmachined and machined single weld slugs are compared in figure EA-8.

The unmachined slug welds were found to contain only one interface as compared to two for the machined slugs. The better yield and increased weld throat thickness of the unmachined slugs apparently result from more metal being available to cover the interface during welding. Based on these results, machining of single-weld slugs prior to welding is being discontinued.

Production of further single-weld slugs has been stopped pending post-irradiation examination of 300 single-weld slugs irradiated in the P-6 reactor cycle. If the results of this examination are acceptable, twenty thousand single weld slugs will be manufactured for exposure in the P-9 reactor cycle.

Approx 10X Neg EA-9522-M

MACHINED PRIOR TO WELDING

Approx 10X Neg EA-9285-M

NOT MACHINED PRIOR TO WELDING

Figure EA-8. Comparison of Machined and Unmachined Single-Weld Slugs

Li-Al ALLOY

TA 3-408, RAMMED CRUCIBLES FOR MELTING Li-Al

Rammed crucibles are being evaluated in an attempt to find one which will have a longer useful life than present preformed clay-graphite crucibles. Rammed crucibles are not preformed but are fabricated in place by ramming refractory cement between the induction coil and an iron core. The core is subsequently removed and the crucible is sintered with a gas torch.

Two types of rammed crucibles have been evaluated. The first, rammed mullite, could not be hardened satisfactorily by available sintering techniques (see February monthly report, DPSP 59-1-2, p 317). Evaluation of a second rammed crucible made of Ajax 108 refractory cement (40% alumina, 60% silica) was begun during April. Operating performance during 8 heats run in this crucible to date was satisfactory, and no significant deterioration of the crucible occurred. However, iron contamination was higher than usual (1000 vs 500 ppm) in metal from the first two heats, apparently due to contamination by the crucible. Analyses are being made on the remaining heats to determine whether contamination is continuing or only occurred on the first few heats.

If iron contamination decreases, the crucible will be evaluated throughout its useful life. If results are satisfactory, a crucible for the 350-pound-capacity vacuum furnace will be constructed and tested.

GRAPHITE MOLD DETERIORATION

The rate of deterioration of graphite molds increased significantly when outgassing of molds in the new Stokes furnaces was begun in October 1957. Deterioration is apparently the result of heating by induction rather than resistance heaters as were used in the old outgassing furnaces. The induced currents create hot spots in the molds causing localized oxidation or loss of binder. Graphite from the damaged areas is released as inclusions into the ingots cast in these molds.

To prevent inclusion pickup, molds of either zirconia-coated or uncoated steel are being evaluated. Two Li-Al pigs cast to date in a steel mold which had been flame-sprayed with zirconia, picked up some zirconia contamination on the surface. Removal of this contamination by chemical milling is being evaluated.

Ingots from uncoated steel molds were found unsatisfactory previously because iron contamination and extreme cold shutting were present on the ingot surfaces. Though contamination was only present on ingot surfaces, it penetrated the ingots deeply at folds formed by the cold shuts. This iron contamination was removed by scalping the surface deeply, lowering metal yields. Chemical milling of billets, however, may permit removal of the iron from the surface without removing appreciable quantities of Li-Al.

An uncoated steel mold has been designed and ordered. Evaluation will begin in about one month.

TA 3-385/1-622, ONE-PIECE BOTTOM END FITTINGS FOR CONTROL RODS

Control-rod one-piece bottom end fittings with double-taper cones (described in the March 1958 monthly report, DPSP 58-1-3, p 326) were adopted for full production use. Since their adoption, approximately 200 rods with these fittings have been fabricated. The only production difficulty encountered was the occasional stripping off of the nose of the fitting during rod swaging. When this happened, the rods had to be manually pushed rather than mechanically pulled through the swager. This difficulty may have occurred because the swager collet did not grip the nose of the fitting tightly. Stripping of the fitting nose does not affect rod quality since this nose is machined off subsequent to swaging. To eliminate this problem, the following changes are being evaluated:

- Redesign swager collet.
- Fabricate fittings from a stronger alloy such as 1100 aluminum.
- Alter swager to permit pushing rods through the swager mechanically.

TA 3-420/2-258/1-719, CHEMICAL MILLING OF Li-Al BARS

Approximately 1000 chemically-milled Mark VI-J slugs have been fabricated for the L-6 reactor charge. As shown below, over-all yield for these slugs was the same as that for normal slugs fabricated during November and December 1958.

	% Yield (Pound Basis)		
Operation	Normal Machined Slugs, 11/13 to 12/17/58	Chemically-Milled Slugs to 4/17/59	Improvement
Casting	95.8	97.2	+1.4
Billet Preparation	95.3	97.5	+2.2
Extrusion and Composition	93.0	93.0	0
Straightening	100.0	89.7	-10.3
Core Machining or Milling	79.2	85.0	+5.8
Canning, Single and Double*	100.0	100.0	0
Final Inspection*	100.0	100.0	0
Over-all Yield →	67.2	67.2	0

* Corrected for rejected slugs salvaged by recanning.

Potential over-all yield for chemically milled slugs is about 78 to 85%. The lower yield obtained to date resulted primarily from severe cracking of bar stock at roll straightening. Details of the fabrication are given below.

Extrusion. A cobalt-steel ID mandrel insert for the porthole extrusion die was used in place of the normal tool-steel insert in an attempt to reduce ID surface galling. However, the new mandrel did not improve core ID surface, and about 8% of all chemically milled cores were rejected for this defect. The use of slower extrusion speeds will be evaluated.

Straightening. Extensive cracking occurred during straightening of the extruded tubing using the SRL roll-straightener. This resulted in rejection of about 10% of the as-straightened tubing. In addition, about 5% of the milled cores were rejected for excessive warp when roll pressure was reduced to minimize cracking.

Metallographic examination indicated that the principal cause of cracking was low extrusion temperature. This resulted in poor recrystallization across the weld interface, and large-grained, brittle alloy. An additional cause was found to be inclusions in occasional billets which effectively reduced the weld area of the extruded tubing. A section of tubing which cracked along the weld line during straightening is shown in figure EA-9. Poor bonding at the interface is apparent, as are excessive inclusions. A section of nonfailed tubing, broken at the weld line for examination, is shown in figure EA-10. The excellent bonding and absence of inclusions are apparent. A comparison of the grain size of failed and nonfailed tubing is shown in figures EA-11 and EA-12.

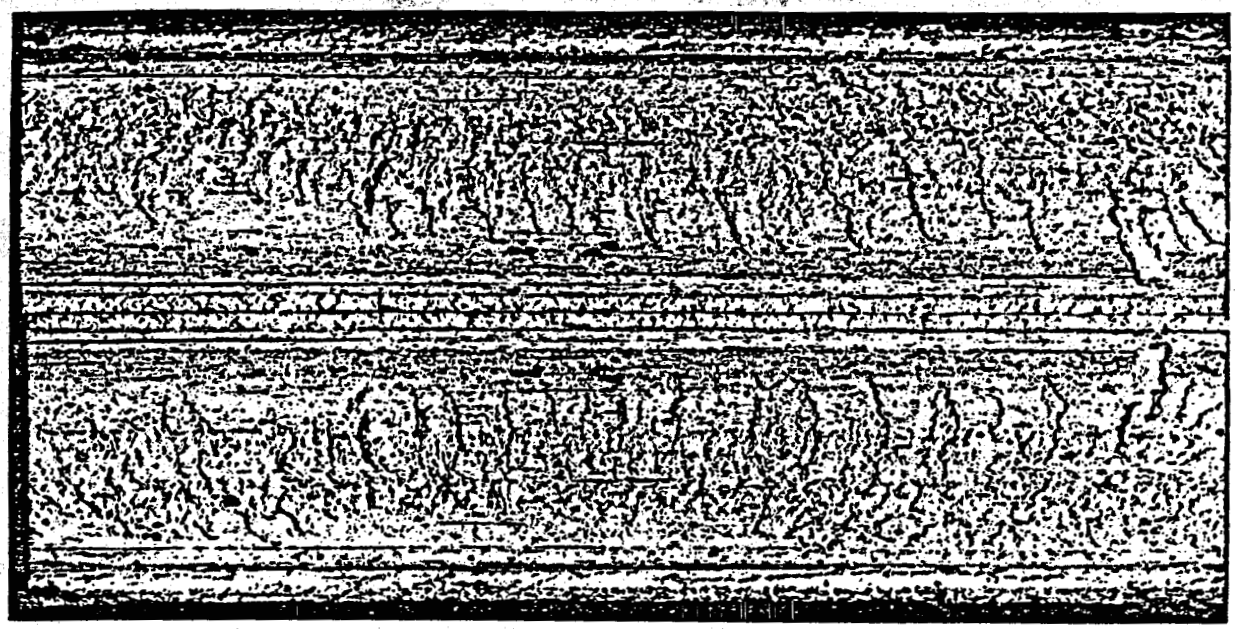
Cracking was largely eliminated by increasing the billet temperature to 850°F for extrusion. Sources of inclusions in billets are being investigated.

Chemical Milling. About 0.002 inch was removed from the OD and ID of each core by milling for 3 minutes in Oakite 160. Cores were then rinsed, desmutted in nitric acid, thoroughly rinsed and hot-air dried. The cores were degreased prior to milling to remove identification ink and oil from the roll-straightening operation. No cores were rejected for defects attributable to the milling operation.

TA 3-421, CHEMICAL MILLING OF CAST Li-Al INGOTS

Evaluation of chemical milling of ingots in Building 320-M in preparation for extrusion was continued (for initial results, see February monthly report, DPSP 59-1-2, p 318). Chemical milling can increase yields, reduce costs, and permit removal of inclusions from areas which normally are not machined, such as the primary pipe.

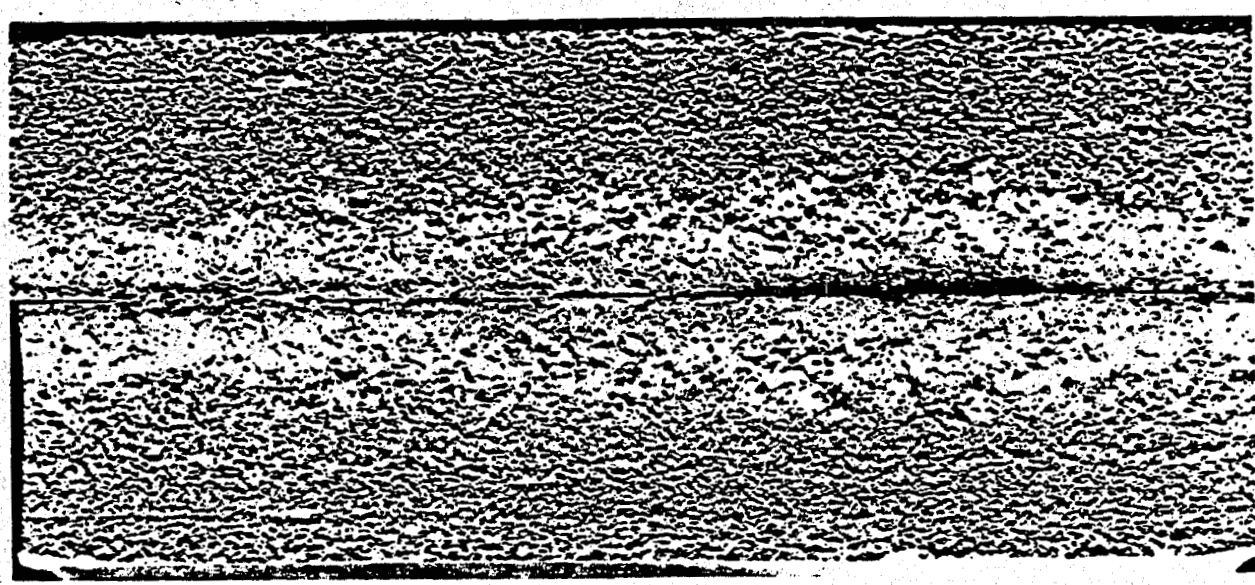
To date, approximately 60 ingots have been milled in this manner. Analyses of bar extruded from 15 of these ingots showed no contamination resulting from chemical milling. Bar from the chemically milled billets will be examined for metallographic appearance. If the bar is satisfactory, slugs from chemically milled billets will be evaluated for reactor and separations performance.



Approx 5X

Neg EA-10283-M

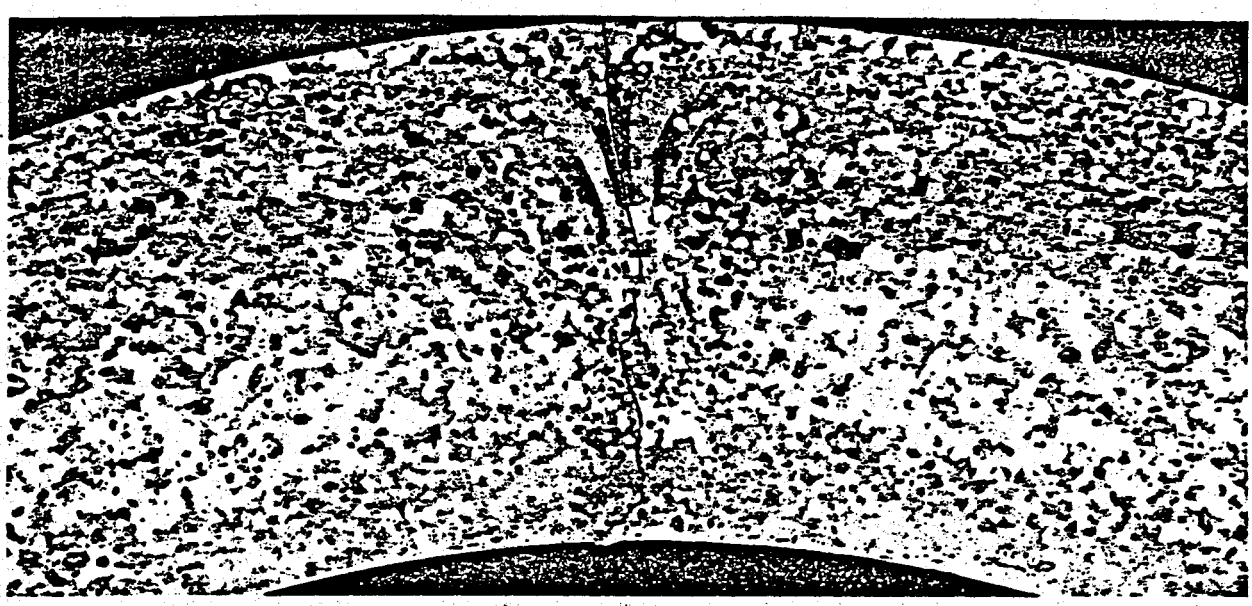
Figure EA-9. Mark VI-J Tubing Which Failed at Weld Interface During Roll-Straightening. Note lack of bonding at interface, and numerous small inclusions.



Approx 5X

Neg EA-10285-M

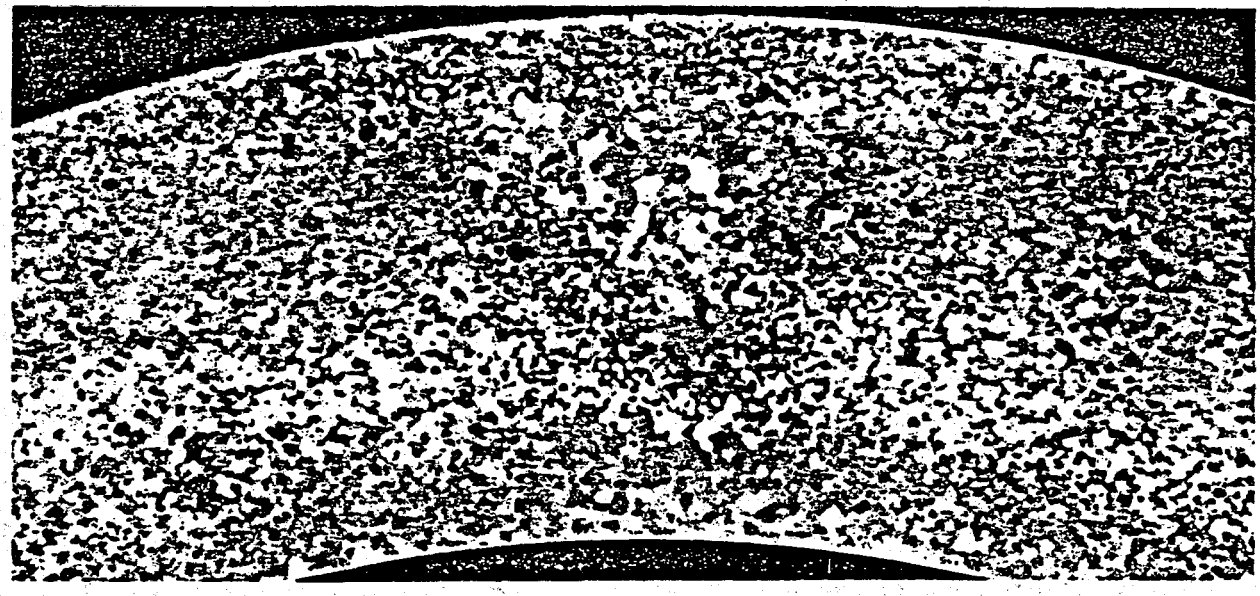
Figure EA-10. Weld Interface of Mark VI-J Tubing Which Did Not Fail During Roll-Straightening. Note excellent bonding at interface, and lack of inclusions.



Approx 10X

Neg EA-10268-M

Figure EA-11. Cross Section of Mark VI-J Tubing Which Cracked During Roll-Straightening. Note lack of recrystallization across weld interface, and relatively large grain size.



Approx 10X

Neg EA-10269-M

Figure EA-12. Cross Section Showing Weld Interface of Mark VI-J Tubing Which Did Not Fail During Roll-Straightening. Note complete recrystallization across weld interface, and relatively small grain size.

Engineering Assistance - Instrument

SEPARATIONS AREAS

BLACKNESS TESTER FOR DISSOLVER COLUMN INSERT

An instrument was built to measure the neutron absorption efficiency of the boron steel plate in the bottom of the insert for the D Column in Building 221-H. During tests in Building 717-F, before the insert was installed in the column, the boron steel plate decreased the count rate of the instrument by a factor of about 4, indicating thermal neutron absorption equivalent to the sheet of cadmium which was used during calibration tests. The cadmium was 1/32 inch thick, so there was a negligible transmission of thermal neutrons through it.

The blackness tester consists of a dual probe assembly containing a Po-Be neutron source with a paraffin moderator in one probe, and a fission counter with a preamplifier in the other. The preamplifier signal is fed to a linear amplifier and a scaler. The probe assembly is designed for remote handling by the hot canyon crane so that future rechecks of the neutron absorption efficiency of the insert may be made in the canyon. Figure EA-13 shows the blackness tester arrangement.

During the test the probe assembly is lowered into the column insert until the boron steel plate is between the neutron source and the fission counter. The neutron count rate is then compared with similar readings taken with only the stainless steel upper portion of the insert between the source and the detector.

HEALTH PHYSICS

AUTOMATIC FILM BADGE READER

The Film Reader portion of the handling system for film badges has been checked out and operated successfully in the Building 735-A laboratory. Except for the densitometer output circuit, all parts of the reader, including the IBM card punch, have been thoroughly tested.

The reader determines the radiation dose and the badge number of the wearer and punches the data on IBM cards. Once the unit has been loaded with film trays, all operations are automatic. The equipment, shown in figure EA-14, consists of an IBM type 026 card punch and a rack-console containing the reading head and the electronic circuits. The films are introduced into the machine at tabletop level in trays containing 50 pieces each. The reading head is located directly above the tray. The control and computer functions are located in the rack sections above the reading head, while the power supplies are below.

(Text continued on page 325.)

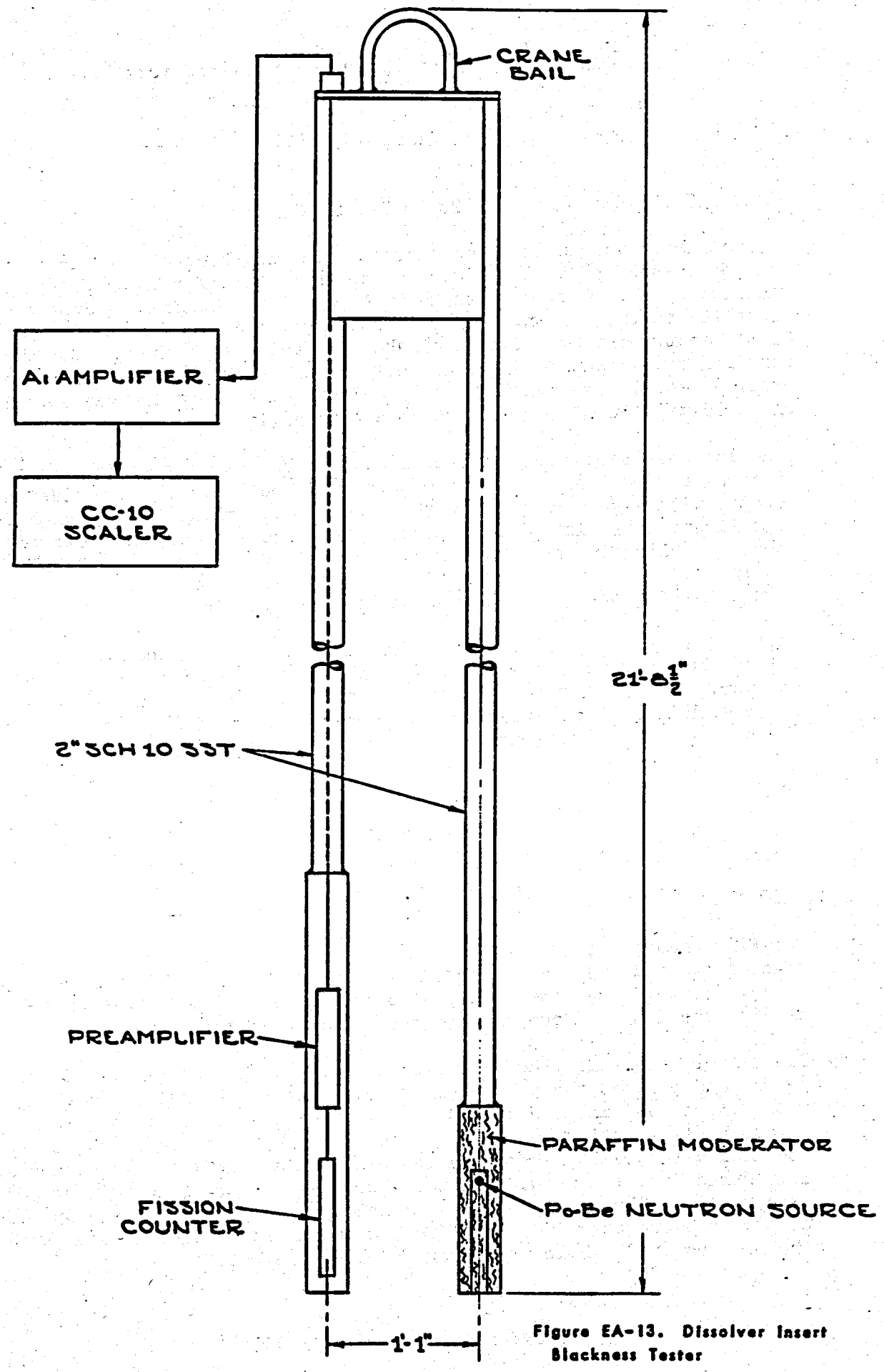
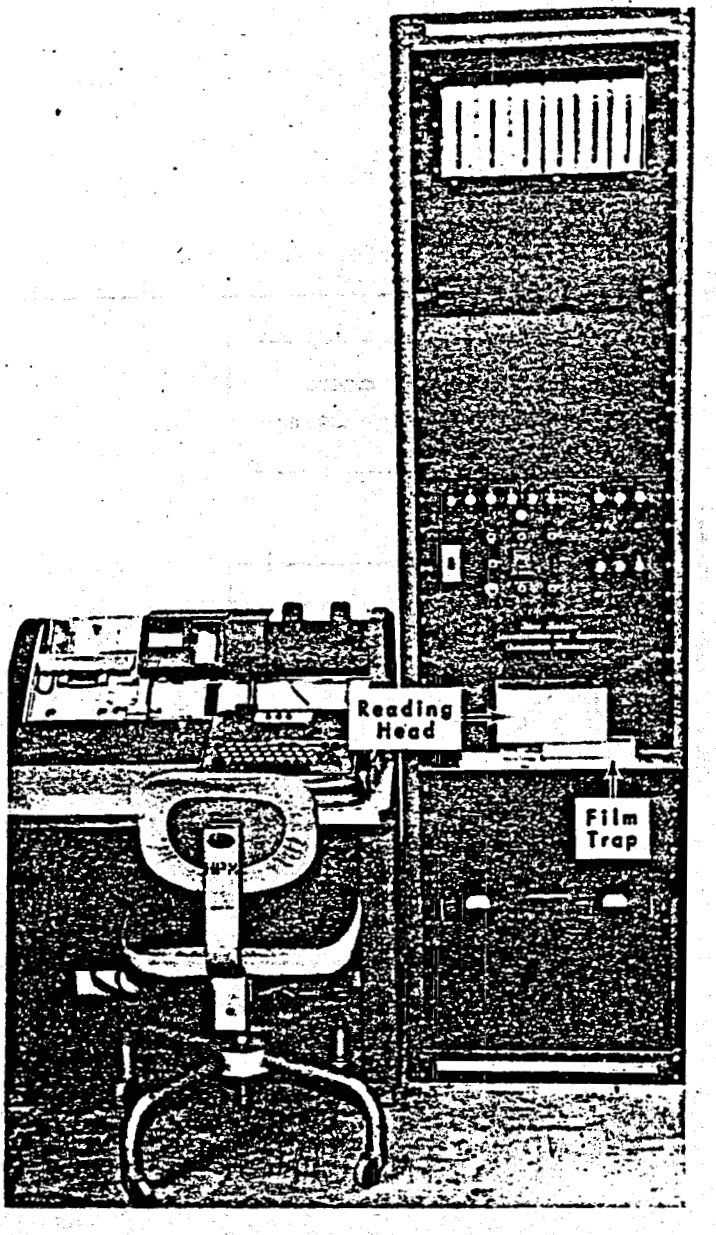


Figure EA-13. Dissolver Insert Blackness Tester



DPSPF 5856-1

Figure EA-14. Automatic Film Badge Reader

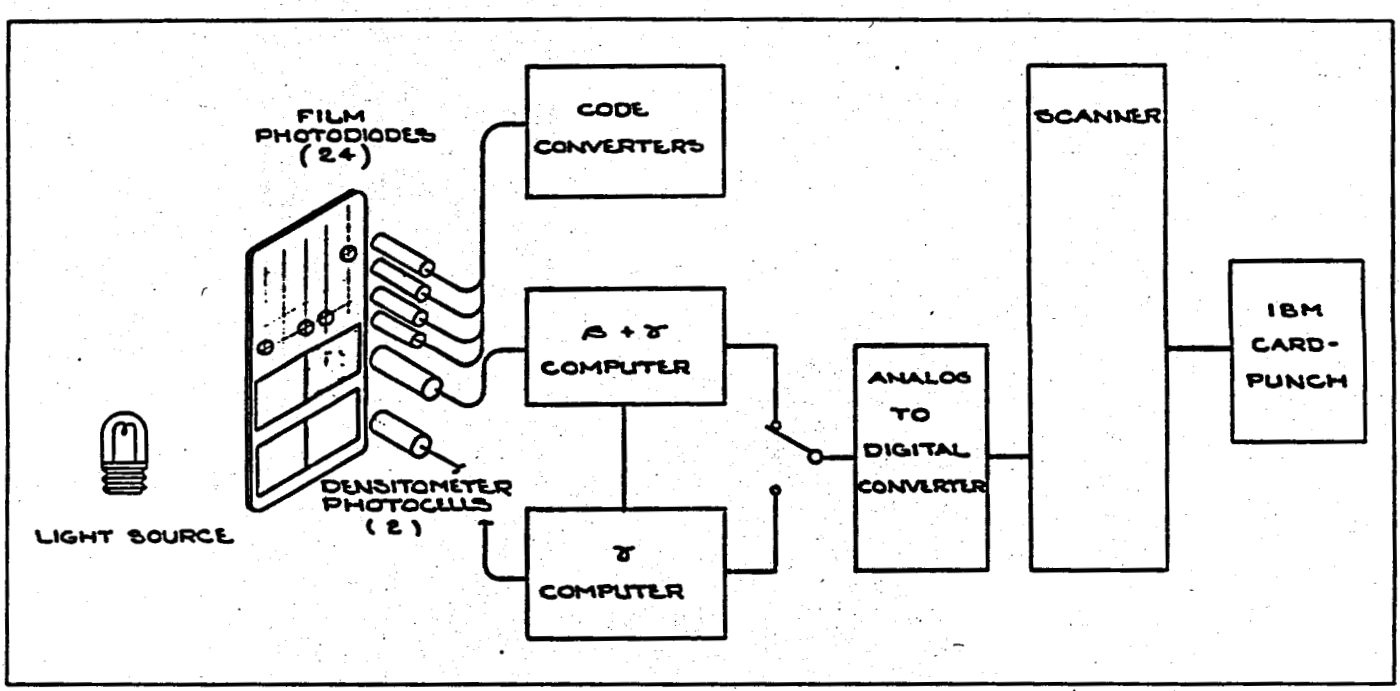


Figure EA-15. Block Diagram of Automatic Film Badge Reader

Figure EA-15 is a block diagram of the reader. Identification is made by means of binary-coded dots on the upper portion of the film. The location and number of dots viewed by the photodiodes and interpreted by binary-to-decimal code converters determine the badge number to be printed out on the cards. The lower portion of the badge is darkened by the radiation exposure. This darkening is measured at two locations: in an open window where both beta and gamma radiation darken the film, and between silver shields where only the gamma rays penetrate. Photocells measure the amount of light transmitted through the two regions on the film and feed the information to two computers. The computers determine the dosage and make corrections for the gamma exposure on the beta-plus-gamma portion of the film. The outputs of the computers are in the form of voltages which are digitized by an analog-to-digital converter to allow punching in IBM cards.

A stepping switch scans the code converters and the analog-to-digital converters in sequence to permit serial reproduction of the numbers on the punched cards.

PULSE READER FOR POCKET DOSIMETERS

A modified version of an instrument designed at Hanford is being evaluated as a reader for pocket-type radiation dosimeters. It measures the pulse required to recharge the dosimeter to a precisely controlled voltage. Exposure of standard Landsverk pocket-dosimeters can be read over a range of 1 to 170 milliroentgens. Increments of 2 mr can be distinguished easily on the meter indicator. The instrument also provides an analog output voltage suitable for IBM automatic data recording equipment.

The pulse reader will replace the General Electric "string electrometer," which is no longer commercially available. The new instrument has the following advantages over the GE electrometer:

- Small amounts of discharge can be more accurately measured by determining the amount of charge required to bring the dosimeter to a standard voltage, rather than by measuring the remaining charge and calculating a small difference in two large numbers.
- The pulse reader is all electronic and should operate more reliably.
- Reading and charging are performed at the same time.
- The output signal is suitable for automatic data-processing equipment.

Figure EA-16 is a block diagram of the pulse reader. To measure the radiation exposure received by a dosimeter, the dosimeter is inserted into the reader well, where it is charged to a precisely controlled reference voltage. Since the dosimeter had been charged to exactly this voltage before being exposed to radiation, the potential difference between the exposed dosimeter and the reference voltage is a measure of the exposure. This voltage difference appears as a pulse at the input of the amplifier during the charging operation. The pulse is amplified, inverted, and transmitted to a pulse stretcher circuit where a large capacitor is charged to the maximum height of the pulse. This capacitor potential is measured by a vacuum tube voltmeter and read out on a calibrated meter. The meter reading is maintained without noticeable change for several minutes and is not affected by removal of the dosimeter from the reader well. A reset switch is provided to discharge the capacitor in preparation for the next reading.

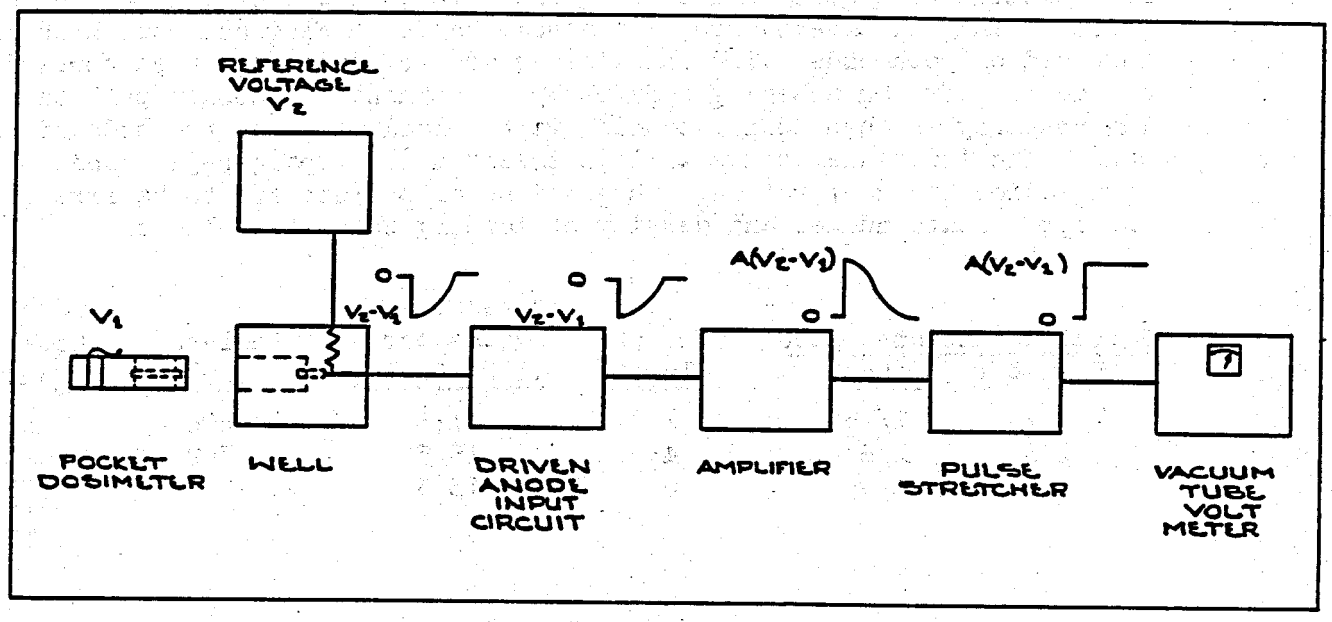


Figure EA-16. Block Diagram of Dosimeter Pulse Reader

Engineering Assistance - Mechanical

REACTOR AREAS

COBALT IRRADIATION

Slugs, to contain cobalt pellets during irradiation in a sparger-jet position in the production reactors, have been designed to avoid loss of the small individual pellets in case of accidental physical damage to the slug. The pellets are contained in drilled holes in aluminum sleeves, as indicated in figure EA-17. The holes are peened-over to contain the pellets, and the sleeves are then assembled on an aluminum rod which is threaded on both ends. The end sleeves are threaded, and when screwed on the rod form an assembly completely containing the cobalt pellets. The assembly is then double-canned, with welded plugs on the ends of both cans. The individual slugs will be inserted in a sparger-jet tube for irradiation in the reactors. Three sizes of pellets are to be irradiated, the approximate number and density of loading being as follows:

Pellet Dimensions, inch		No. of	Density, grams per	Pellets	Total Cobalt,
Diameter	Length	Slugs	inch of slug	per Slug	grams
1/16	1/16	1	11.8	4800	118
1/8	1/8	4	15.5	702	620
1/8	1/4	5	15.5	351	775
					1513

SPRAY NOZZLE TEST

Spray nozzle tests are being run to determine the best type nozzles to use for cooling a fuel element which may fall on the reactor tank top or process room floor during discharge of the fuel.

Of the six nozzle designs tested, a Spraying Systems Company No. 1½H230SQ, full-jet square-spray nozzle appears to give the best coverage. The test was run with the nozzle at a height of 26 feet from the ground, at nozzle angles of 0°, 30°, and 45° from the vertical. The nozzle has a flow capacity of 23, 27, and 36 gpm of water at pressures of 10, 15, and 25 psig, respectively. The data indicate that the square nozzles should be operated at 25 psig and located approximately 15 feet apart to have the required coverage of 0.4 pound of water per hour per square inch of floor area.

Further tests are under consideration to determine if the ventilating air flow in the process room will disturb the spray pattern.

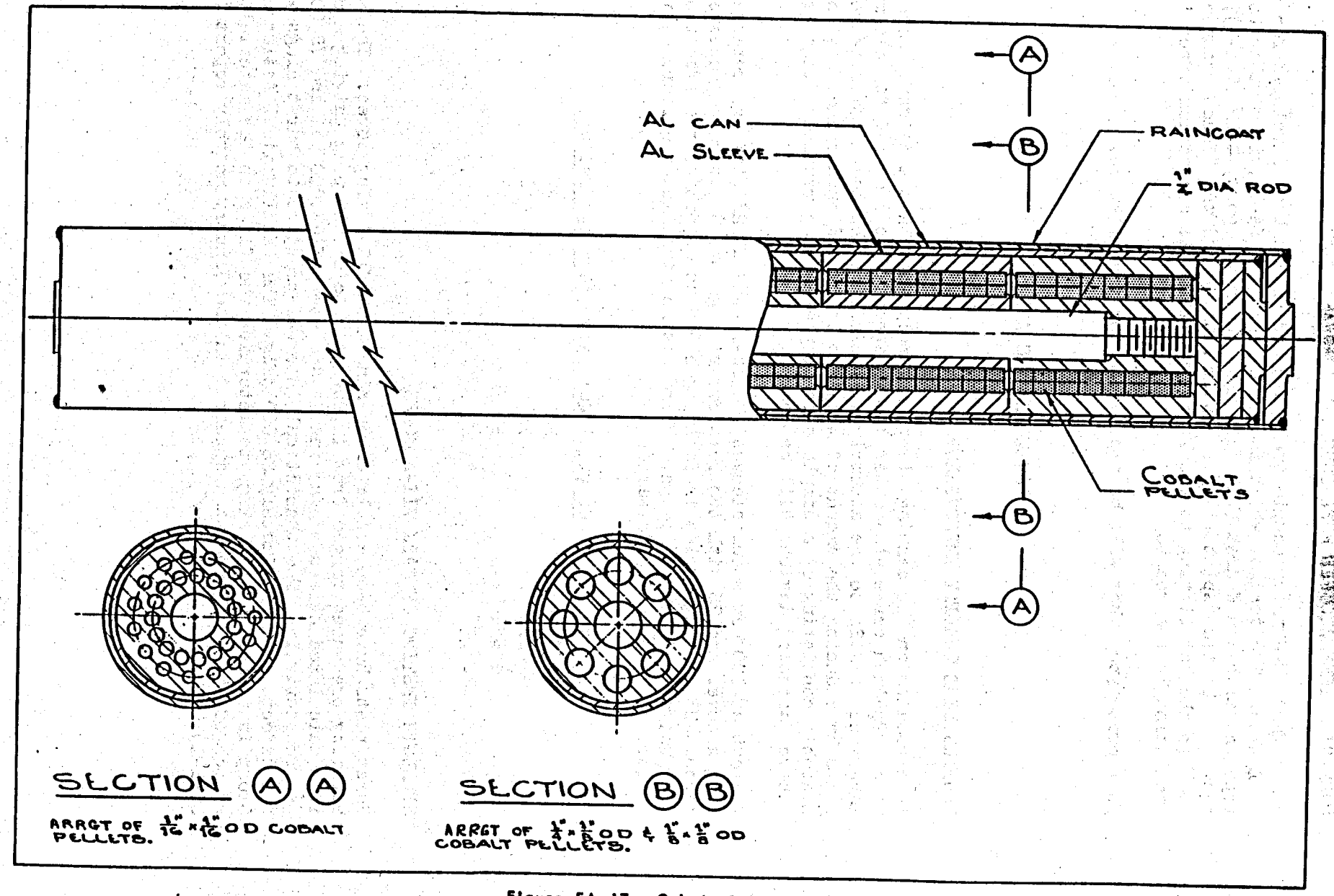


Figure EA-17. Cobalt Pellet Slug

MARK VI-J THERMOCOUPLE FUEL TUBE

A new Mark VI-J thermocouple fuel tube design has been developed for use in L Area. The design, which is shown in figure EA-18, is similar to the type installed in K Area (see July 1958 monthly report, DPSP 58-1-7), except for the flow splitter type bottom fitting and the "Jones" connectors used in the top fitting to facilitate thermocouple connection after installation in the reactor.

Four prototype tubes will be fabricated for installation.

RAW MATERIALS AREA

CANNING LINE QUALITY IMPROVEMENT

Test results on the canning sleeve assembly show that a 3-inch, rather than a $2\frac{1}{2}$-inch center-to-center sleeve spacing, is required for minimum adherence of AlSi to the assembly. To minimize the heat load requirements by reducing the amount of metal to be heated each cycle, a sleeve assembly consisting of a single support plate using retainer rings as sleeve fastening devices will be tested.

A group of 100 new steel sleeves (50 as-received, 25 with one retainer-ring groove, and 25 with two retainer-ring grooves) has been processed through 25 canning cycles in Building 313-M to determine the effect of the grooves on sleeve warpage. There was no significant difference in failure of sleeves due to warping.

Sleeve Type	Original Quantity	Warpage Failures
Plain	50	21
1-groove	25	9
2-groove	25	12

STRENGTH TESTING OF MARK VII-A CORES

Burst tests have been completed on nine Mark VII-A slug cores. The bursting pressures ranged from 42,000 psig to 51,000 psig, giving a maximum calculated stress on the inside diameter of 65,000 psi and 79,000 psi, respectively. The theoretical internal pressure to burst slugs without defects is approximately 85,000 psig. The Metallurgical Assistance Group is inspecting the slugs to determine the flaws in the material. A sketch of the test apparatus appeared in the January monthly report (DPSP 59-1-1).

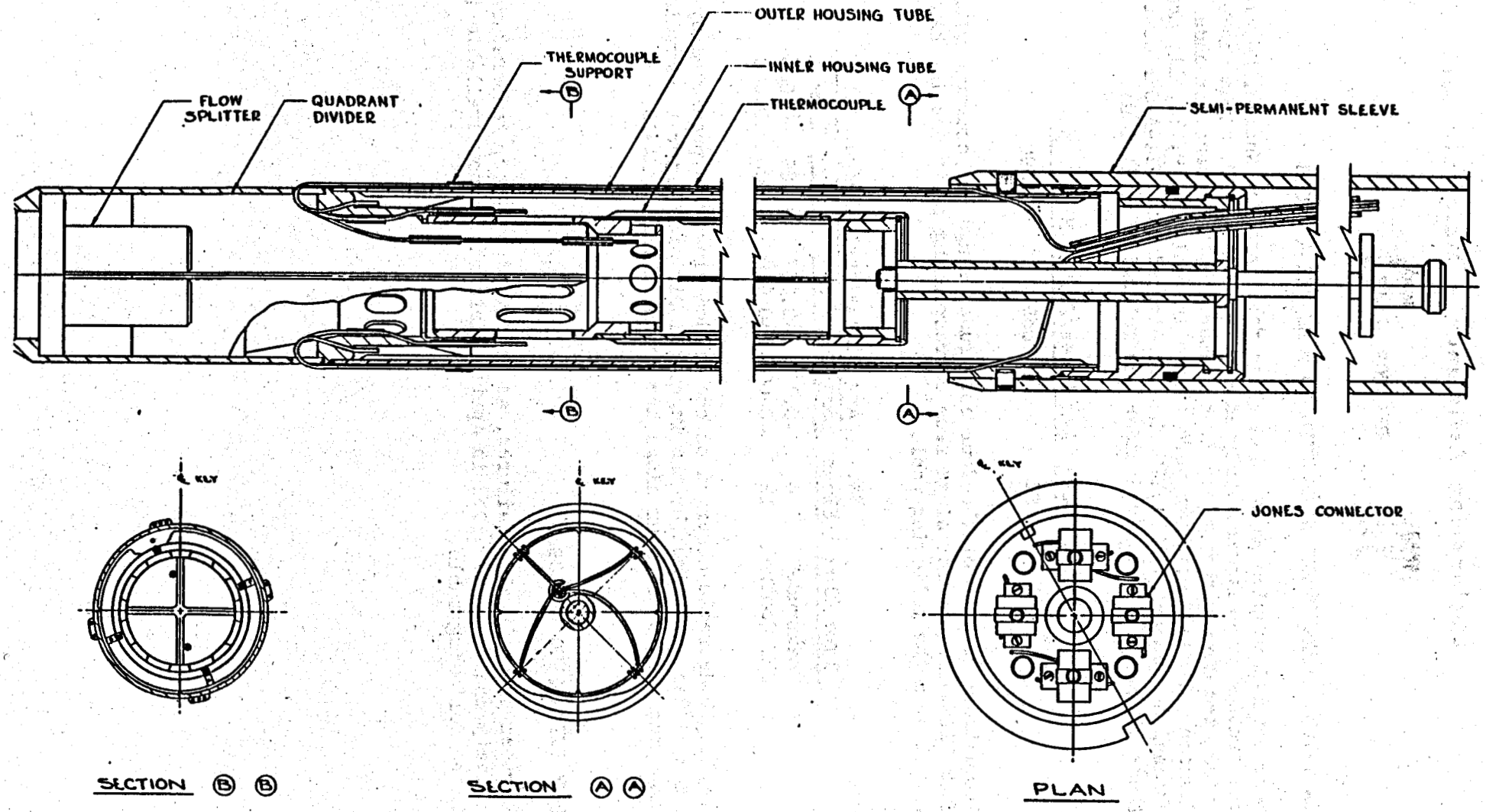


Figure EA-18. Mark VI-J Thermocouple Fuel Tube, RPL Areas

Engineering Assistance - Metallurgical

REACTOR AREAS

TA 1-705, IRRADIATION OF COEXTRUDED MARK VII-A SLUGS

Four natural uranium Mark VII-A slugs fabricated at Nuclear Metals by a coextrusion process were stripped of cladding at SRP, alpha canned, and irradiated during the R-4 cycle to determine their behavior under irradiation. These four slugs were placed in positions 9, 10, 11, and 12 of channel A, quatrefoil X31,Y75 which contained Mark VII-A-SRP control slugs. These slugs were irradiated under the following conditions.

Cycle	Days	Avg Power/Ft, %	MWD/T, %
R-4	54	0.0039	0.590

Dimensional changes of the four coextruded slugs are compared with the 12 Mark VII-A-SRP control slugs from corresponding positions in channels B, C, and D in the following table.

Type Slug	No. Slugs	Avg Dimensional Change, inch			
		Length	Warp	OD	ID
Coextruded	4	-0.163	+0.037	+0.009	-0.005
SRP Control	12	+0.005	+0.010	+0.003	0.000

The length decrease observed on coextruded slugs was predicted by growth index values determined by the Technical Division prior to irradiation. Photographs of these four slugs are shown in figure EA-19.

MARK VII-A SLUG FAILURES

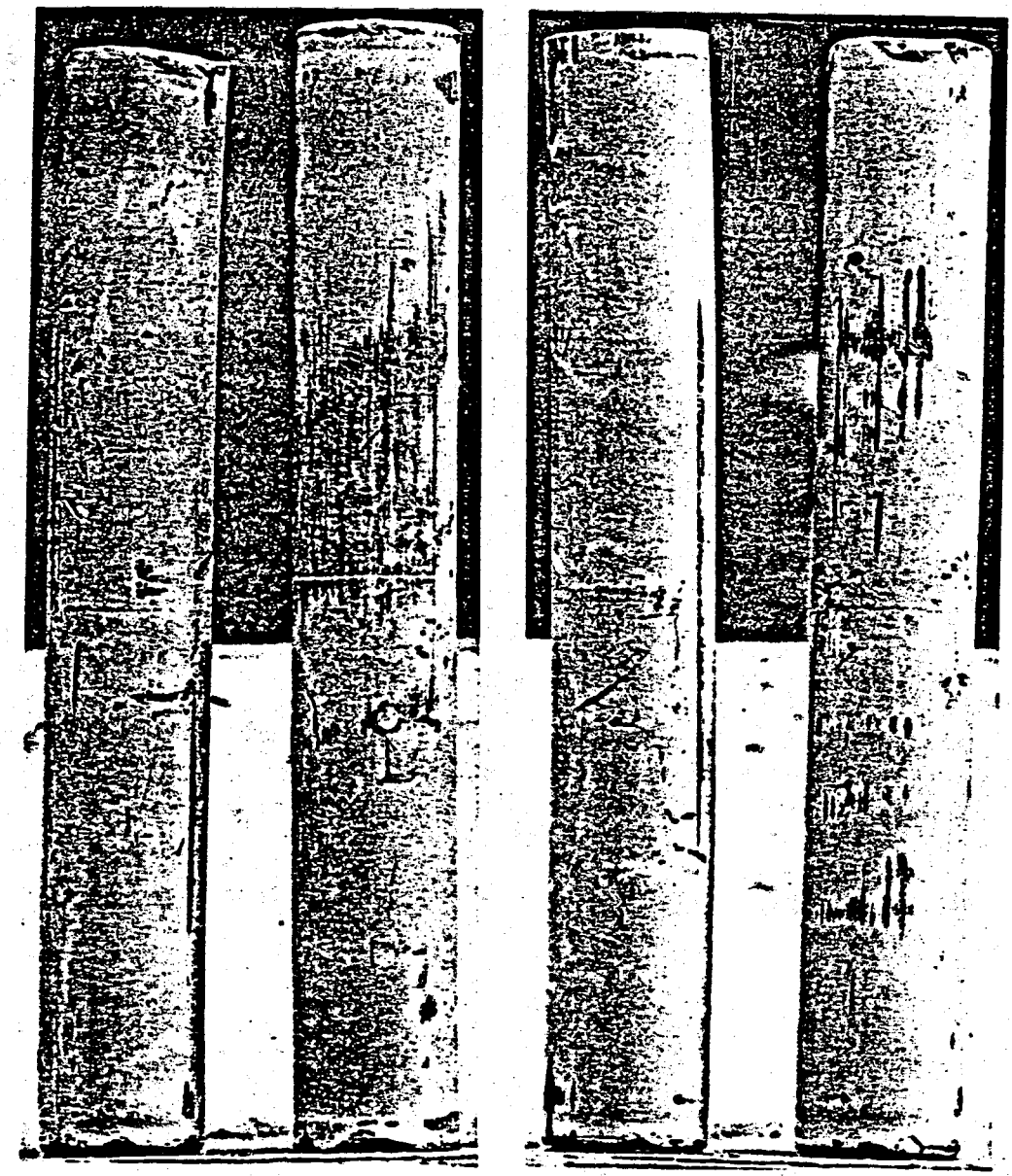
Thirteen probable slug failures, as indicated by flow, temperature, and activity monitors, occurred in April. Information now available is shown in the following table.

Failure No.	Cycle	Date	SRP Type	Coordinates X	Coordinates Y	Channel	Days Exposed	Exposure, %
47	L-4	4/1	Dingot	14	66	B	47	33.6
48	K-3	4/1	Dingot	08	54	B	39	28.4
49	P-6	4/3	Ingot	34	54	D	25	19.9
50	K-3	4/4	Ingot	15	21	A	39	28.5
51	P-6	4/16	Ingot	35	51	D	38	28.7
52	R-5	4/18	Ingot	39	21	A	44	34.6
53	P-6	4/21	Ingot	30	42	C	42	31.6
54	C-4	4/21	Ingot	44	78	A	28	22.0
55	R-5	4/21	Ingot	19	21	D	47	35.5
56	C-4	4/23	Ingot	11	45	A	30	22.2
57	L-5	4/26	Ingot	10	48	D	22	13.7
58	C-4	4/29	Ingot	22	12	B	37	25.9
59	P-6	4/30	Ingot	15	45	D	52	37.7

METALLURGICAL EXAMINATION OF IRRADIATED SLUGS

Failure 37. Mark VII-A-SRP medium hydrogen (3.1 to 5 ppm) dingot slug, serial number 3N32907, which failed on February 9 during the R-3 cycle, was destructively examined by the Technical Division in the high level caves. Small cracks were observed in the core. No extensive general corrosion of the aluminum cladding was evident. A definite cause of the failure has not been determined.

Photographs of the failure are shown in figures EA-20 through EA-23.

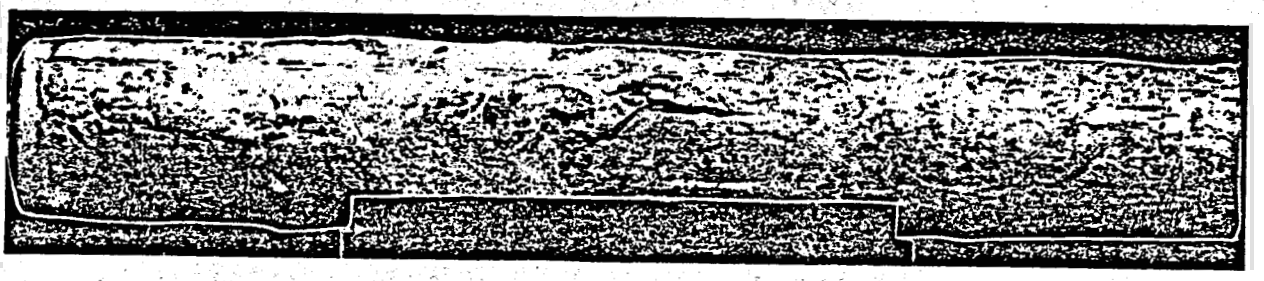


Approx 2/3x DPSPF 5863-1 Approx 2/3x DPSPF 5863-2

Figure EA-19. Irradiated Coextruded Mark VII-A Slugs. Note severe warp.

Slug identification from left to right as follows

Cycle	Serial No.	X	Y	Channel	Position	Warp, inch
R-4	7	31	75	A	10	+0.032
R-4	45	31	75	A	9	+0.032
R-4	23	31	75	A	11	+0.048
R-4	8	31	75	A	12	+0.035



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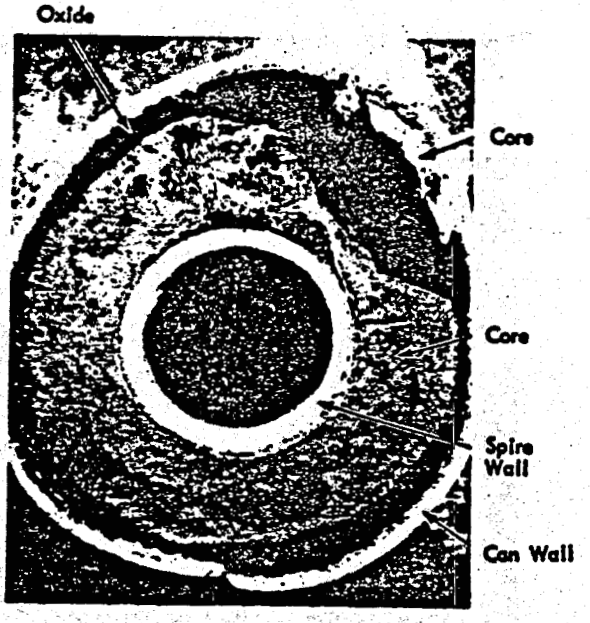
Neg SRL-19707

Figure EA-20. Failure No. 37 - Mark VII-A-SRP Dinget Slug



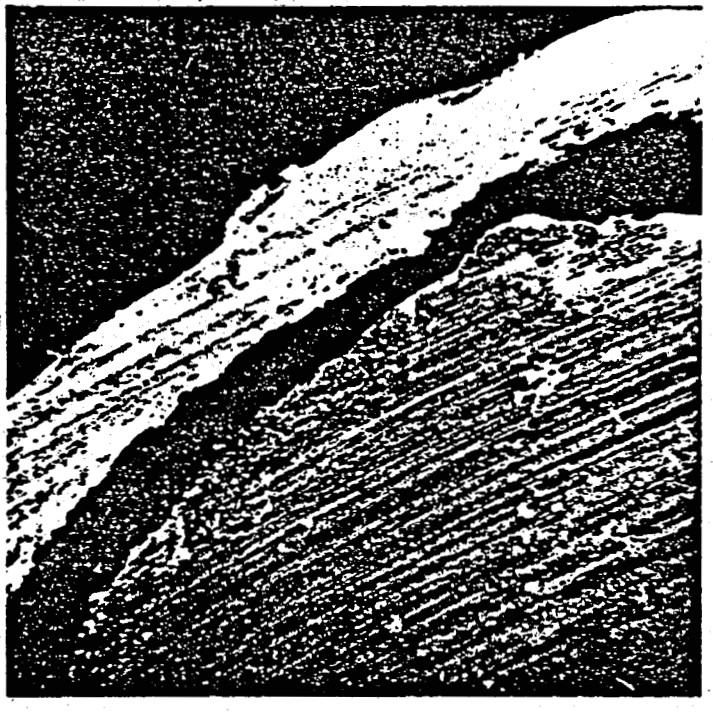
Approx 2.7x

Neg SRL-19973

Figure EA-21. Failure No. 37 (Section Through Failed Area)

Approx 15X Neg SRL-19976

Figure EA-22. Enlarged Section Showing Small Radial Cracks in Core of Failure No. 37



Approx 15X Neg SRL-19760

Figure EA-23. Enlarged Section Showing Circumferential Cracks in Core of Failure No. 37

SEPARATIONS AREAS

FULFLO FILTER CASES, EP 421.91-1, BUILDING 221-F

A Fulflo filter case 12-1/4" OD by 40-3/8" high and fabricated from 0.120" thick AISI type 304L stainless steel was examined after failure caused by corrosion and penetration of the wall. The filter uses glass wool filter cartridges to filter 50% HNO_3 at room temperature. This filter case operated for approximately 2-1/2 years and then lay idle for approximately 1-3/4 years. During the latter period the filter cartridges were not removed from the filter case and the equipment was not cleaned or drained.

Results of the examinations are listed below:

1. Corrosion and failure were caused by chloride pitting. The attack probably occurred during the 1-3/4-year period in which the filter was not in operation and was coated internally with a sludge which contained up to 14,000 ppm chlorides.

2. New glass wool filter cartridges contained 1.8 ppm total chlorides.

3. Analysis of the case wall is typical for AISI type 304L stainless steel.

It was recommended that filter cases having similar service history be inspected; that operating procedures provide for frequent filter changes and sludge removal; and that filter cases be thoroughly cleaned before being placed in standby.

INSPECTION OF NITRIC ACID REBOILER, EP 341.22, BUILDING 211-H

On March 30, leakage occurred through the carbon steel shell of the reboiler just above the bottom tube sheet. A temporary patch was placed over the 1¼"-diameter hole and operation was continued until April 14. The reboiler had operated for approximately 3-3/4 years in hot 30 to 50% HNO_3. Parts of the reboiler exposed to acid are AISI type 309SCb stainless steel while the remainder including the shell, is carbon steel.

Visual inspection of the component parts of the reboiler showed the following:

1. Type 309SCb welds on the acid side at the top and bottom of the reboiler were corroded and cracked. Welds exhibited honeycomb attack, knife line attack, preferential attack down the center of welds, and pitting attack at weld craters.

2. Type 309SCb base metal (shell and tube sheet) was slightly etched.

3. A corroded band approximately 1½ inches wide was found in the carbon steel shell between the bottom of the condensate drain line and the top of the lower tube sheet. Corrosion is attributed to attack from HNO_3 which leaked from the tube side to the shell side.

4. A hydrostatic test showed 26 leaks at the rolled tube joints in the tube sheet at the top of the reboiler. Crevice corrosion from HNO_3 is the primary cause of these leaks.

The reboiler is presently being repaired.

SEPARATOR PLATE TESTS

Specimens of cadmium sheet are sandwiched between plates of stainless steel in a test to determine if galvanic corrosion of the cadmium occurs in presence of water. Previous results were described in the February monthly report, DPSP 59-1-2.

Examination after 90 days exposure at room temperature in uncontaminated basin water revealed corrosion averaging less than 0.0001 inch/mo along one edge of the cadmium specimen. The stainless steel suffered no detectable corrosion.

Because corrosion appeared only in an area where contact between the stainless steel and cadmium was good, additional assemblies are now being tested in which the cadmium sheet contains raised spots to insure intimate contact.

RAW MATERIALS AREA

DOUBLE-END SLUG WELDING

Approximately 100 good Mark VII-A slugs were double-end welded early in April. The appearance of these welds was generally good except for a few extremely small pinholes which disappeared after etching and for the arc pulloff area which was rather rough due to high current. Metallurgical quality was good.

Certain mechanical and electrical changes seemed desirable for more reliable operation and still better quality, however, and the following modifications were made.

1. Circuits were changed to permit a lower preheat current and higher first-step weld current. This both reduced pitting of slug at arc start, and maintained short preheat time.

2. A pneumatically-actuated rheostat was added to the weld current control circuit to permit a gradual reduction of weld current during the arc pulloff. A smoother, more uniform weld is produced.

3. An air cylinder was added to automatically retract one torch to permit slug loading. A foot operation was thus eliminated.

4. Increased power to rotate the slug was provided by driving both slug supporting rolls through an added idler roll. Rotation stoppage was eliminated.

5. Larger air cylinders with hydraulic controls were provided on torch pulloff mechanism. Jerky and unreliable action was eliminated.

6. Slug guide roll shaft was redesigned to provide a sturdier and more accurate mechanism.

The above changes permitted better quality and greater reliability during test welds on several hundred slugs. Additional slugs are being welded, and sample welds are being examined in the Metallurgical Laboratory.

ISOTHERMAL HEATING OF LITHIUM-ALUMINUM CONTROL ROD SLUGS

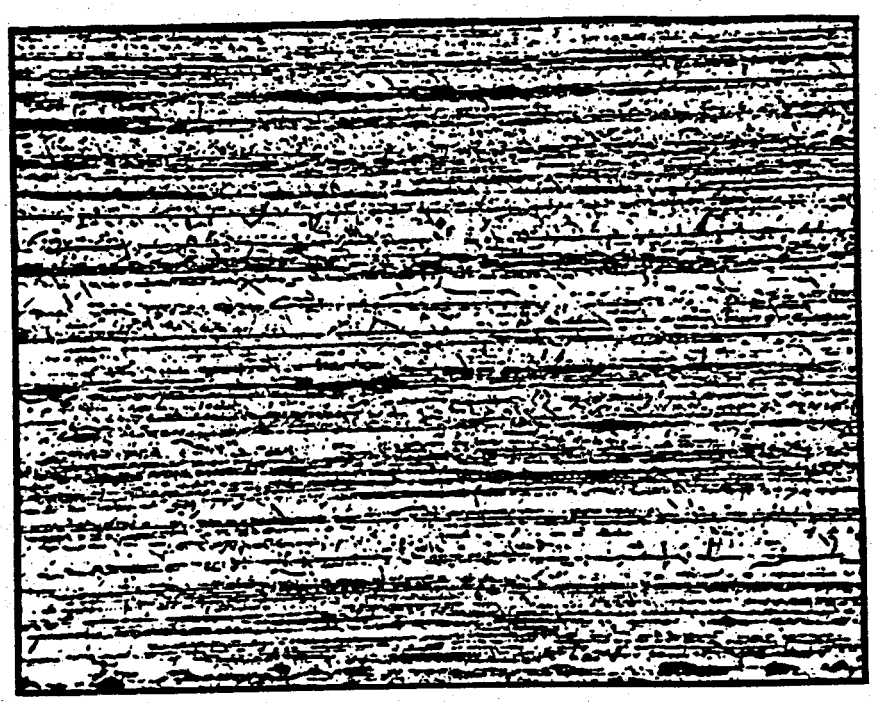
In the normal as-extruded condition, lithium-aluminum control rod cores may contain some lithium in supersaturated aluminum solid solution. During reactor service, precipitation of additional beta phase (Li-Al) from this supersaturated solution could alter the properties of the alloy and affect its behavior in the reactor. Sample control rod slugs of 3.5 normal wt % lithium are being heated in various ways to determine what structural changes occur.

In one slug heated isothermally at 300°C for 240 hours, additional beta phase was precipitated along grain boundaries of the aluminum solid solution, as illustrated in figure EA-24. The structure of an as-extruded core with no heat treatment is shown for comparison.

This work is continuing.

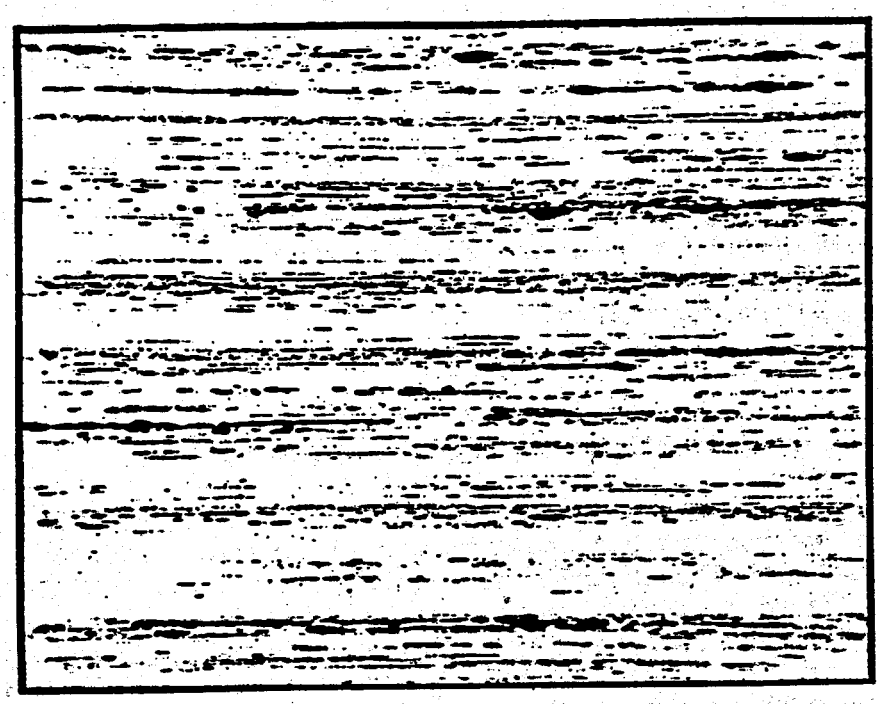
FAILURE OF MARK VII-A-SRP SLUGS IN AUTOCLAVE TEST

Metallographic examination of Mark VII-A-SRP slugs which failed during autoclave testing has shown that weld pinholes and AlSi penetrations of the aluminum sheath are major causes of failure. The frequency of failures due to AlSi penetrations through the spire near the top end of the slug has increased sharply since January 1959. For example, of 9 failures examined since January and found to be caused by AlSi penetrations, all were through the spire, and 8 of the 9 occurred near the top end of the slugs.



Approx 100X No Etch Neg EA-10342-M

Longitudinal Center Section, Pin No. 680 FHA, Heated Isothermally at 300°C for 240 Hours



Approx 100X No Etch Neg EA-10340-M

Longitudinal Center Section, Pin No. 680 FGC, As-Extruded -- No Heat Treatment

Figure EA-24. Microstructural Change Produced in 3.5 wt % Lithium Control Rod Core by Isothermal Heat Treatment. Note beta phase (blackened by exposure to air) precipitated in a network along grain boundaries of the aluminum solid solution (white).

X-RAY DIFFRACTION LABORATORY

GROWTH INDEX DETERMINATIONS

Data necessary for the calculation of growth index (a measure of the preferred orientation in uranium and probable stability during irradiation) have been determined for 160 samples from 16 Mark VII-A fuel cores. These cores included as-received SRP cores and specially heat-treated Sylcor cores.

Calculations completed to date indicate that the typical as-received, alpha-rolled, uncanned, SRP uranium fuel core has a growth element of 5.31, a shrinkage element of 5.47, and a growth index of -0.16 in the direction of the fuel element axis. In the circumferential direction, the growth element is 3.67, the shrinkage element is 2.80, and the growth index is +0.87.

Growth index values will next be determined for SRP fuel cores after canning by the triple-dip process. The cores tested will have known fabrication histories and will be from the same NLO rods as the as-received cores reported above.

The data obtained will provide complete characterization of the preferred orientation pattern of the standard Mark VII-A-SRP fuel slug. We may then be able to evaluate, prior to actual reactor exposure, the effect of changes in NLO or SRP procedures on slug stability during irradiation.

EFFECTS OF SPECIMEN PREPARATION ON DIFFRACTION MEASUREMENTS

Present procedures for preparing specimens for diffraction require abrasive cutting, automatic metallographic polishing, and electrolytic etching.

Diffraction tests show that approximately 0.006 inch must be removed by polishing to eliminate any effect on orientation of abrasive cutting. Approximately 0.0001 inch (3 minutes in mixed acid-glycerine etch at 0.5 amp/cm^2) must be removed to eliminate any effects of polishing.

SAMPLING SIZE DETERMINATION

For as-received, alpha rolled, natural uranium, ingot fuel cores, one fuel element from each rod appears to be an adequate sample. Eight samples from each fuel core, rather than the 16 now taken, seem to be adequate.

VARIATION IN ORIENTATION WITHIN A SINGLE CORE

There seems to be no correlation between the growth element, shrinkage element, and growth index and the position from which a sample is taken from an alpha-rolled, natural uranium fuel core. Figure EA-25 shows the variation of growth and shrinkage elements and growth index of transverse sections taken at 27 different places along a fuel core.

MISCELLANEOUS

BOILER FAILURE, BUILDING 284-F

Boiler No. 2 in Building 284-F was shut down when a leak occurred in a carbon steel water header.

Examination of the header revealed many deeply gouged areas in the $\frac{1}{2}$-inch wall. These gouges were made during maintenance work in January when carbon steel fins were cut off using oxyacetylene cutting torches and were replaced with stainless steel fins.

The leak and the gouged areas were repaired by welding. In addition, several cracked fillet welds joining the new stainless fins to the header were ground out and repaired.

ELLIOTT FAN TURBINE, EP 284F-808

An Elliott fan turbine drives the forced and induced draft fans for boiler No. 2 in Building 284-F. Steam leakage was observed at the casing joint, and an examination showed the lower casing was cracked at 4 blind bolt holes. This is shown in figure EA-26.

The bolts which were used to join the upper and lower casings were measured and found to be 1/16 to 1/8 inch longer than the combined flange thickness on the upper casing and tapped bolt hole depth in the lower casing. Stresses caused from tightening the bolts during installation and from expansion and contraction during turbine operation probably caused the cracking at the bolt holes.

The casing will not be repaired but will be assembled with shorter bolts and returned to service. It was recommended that a spare turbine casing be purchased.

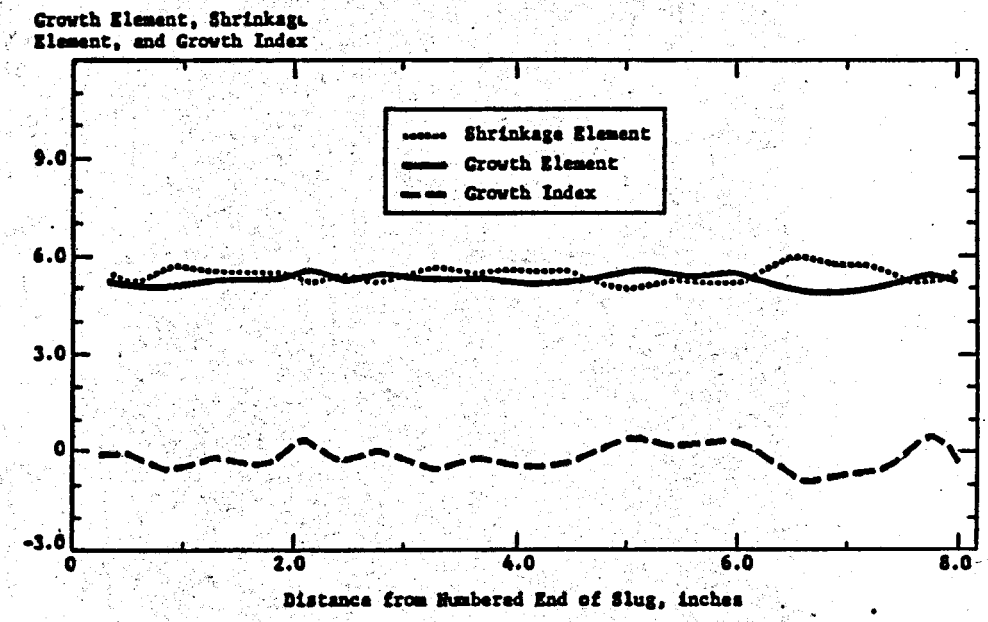
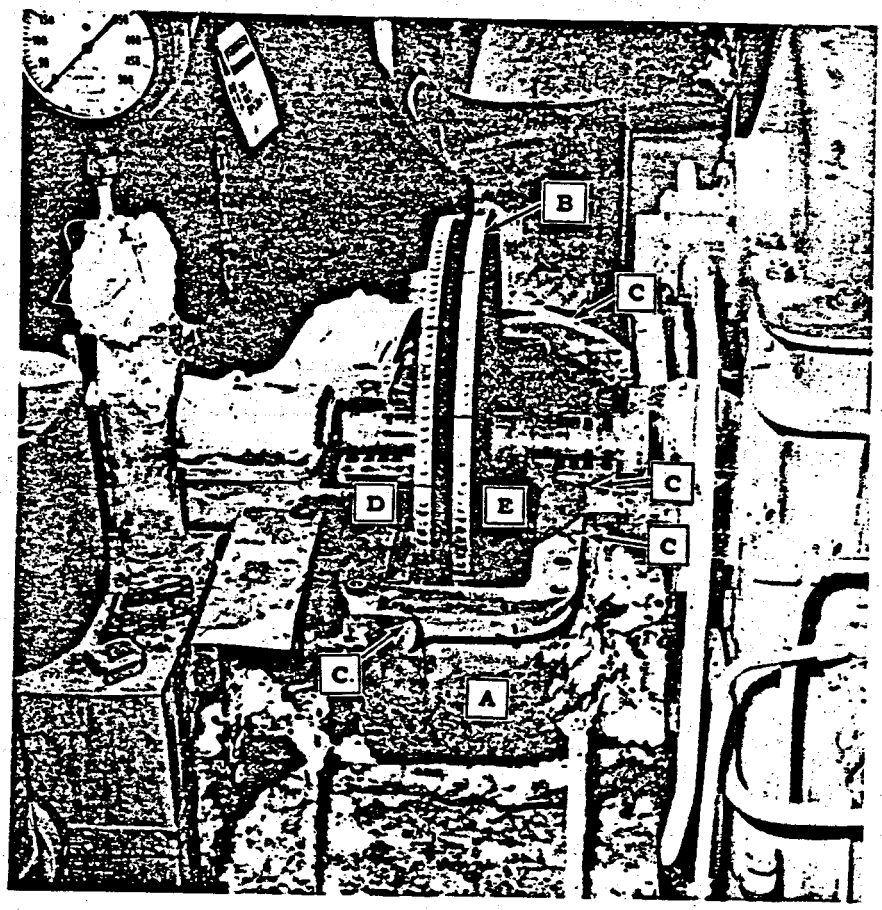


Figure EA-25. In-Slug Variations in Orientation, Alpha Rolled Core, Slug No. F51690-1-2 (27 samples were diffracted with 0.296° between samplings)



Approx 0.2×

DPSFF 5858-1

Figure EA-26. Elliott Fan Turbine with Upper Casing Removed

A Lower Casing (Cast Iron)
B Wheel
C Location of Cracks
D High Pressure Side of Turbine (310 psig at 425°F)
E Low Pressure Side of Turbine (20 to 30 psig)

INSPECTION OF DEEP WELL PUMP PARTS, WELL 905-37F

Trouble in deep well pump 905-37F was indicated by a loss in head and water flow. Tests prior to removal showed that there was a hole through the carbon steel casing. The pump had been in service for approximately 2 years, during which time pump components were exposed to water of pH 4.2 containing 42 ppm carbon dioxide. Inspection of this pump approximately 1 year ago had revealed no noticeable corrosion of the pump parts; however, the impellers, the wear rings, and the one length of shaft securing the impellers had been replaced due to damage from excessive vibration.

The pump was removed, and the shafts, shaft couplings and sleeves, column casing and couplings, impellers and impeller bowls were examined. These parts were covered with a thick deposit having an extremely high chloride content (30 ppm on shafts and 6000 ppm on impellers and impeller bowls). Results of the visual examination are summarized in the following table. Corrosion of the shaft next to the couplings and sleeves is attributed to galvanic action.

This investigation is being continued.

Inspection of Deep Well Pump 905-37F

Pump Part	Visual Inspection Results	Remarks
8" Schedule 80 Carbon Steel Column Casing	Top end of the fourth section from the top of the well was severely eroded through the wall of the casing. This was due to improper seating of the casing in the cast iron coupling. Other sections of pipe contained heavy layers of rust on both the inner and outer surfaces. Slight surface roughening was found beneath the rust.	Fourth section of column casing was replaced.
AISI type 416 SS Shafts, Monel Shaft Sleeves, and Everdur Couplings	These components were covered with a brownish-red deposit which was adherent to small localized areas on only the shafts. Pitting a maximum of 1/16" deep was found in the shaft material next to the Everdur couplings, next to the Monel sleeves, at pipe wrench indentations, and beneath adherent surface deposits. Everdur couplings were uncorroded; however, the top end of 3 sleeves was pitted.	When the present shafting for this pump can no longer be used, it will be replaced with AISI type 410 SS shafting containing sleeves and couplings of the same material. A strap wrench will replace the pipe wrench for turning the shaft and couplings.
SAE 40 Impellers and SAE 62 Impeller Bowls	Both were covered with a thick adherent deposit. A reddish-brown deposit was found on the bowls, and a bluish-gray deposit was found on the impellers. The surface beneath the deposit showed no pitting or evidence of erosion.	SAE 40 impellers were replaced with SAE 62 bronze impellers.
Cast Iron Column Couplings	Showed evidence of graphitic corrosion.	Cast iron couplings were replaced with SAE 62 bronze couplings.

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STATISTICS

	R	P	L	K	C	D	F	H	M/A	Total April	Total Year to Date
SWP's Processed	727	540	594	648	452	45	1481	1524	47	6,058	21,377
Extended SWP's in Effect	54	52	56	46	51	7	73	95	23	457	-
Surveys	1583	964	1226	959	913	83	3240	4231	508	13,807	50,105
Air Samples	261	202	587	307	196	96	5533	1408	125	8,715	30,373
Personnel Contamination Cases	6	2	2	2	3	0	7	5	0	27	77
Vessel Entry Permits	4	0	0	4	0	0	9	4	0	21	105
Industrial Hygiene Surveys	7	4	4	6	1	0	20	11	12	65	287
H-3 Bio-Assay Analyses	0	0	0	0	0	0	0	2259	0	2,259	10,038
Pencils Processed	1400	874	1300	1360	1400	0	3302	5752	16	15,404	54,959
Training Sessions	7	3	7	4	6	2	19	8	3	59	311

REACTOR AREAS

GENERAL

Reactor operations continued to be plagued by fuel element failures. There were thirteen failures, with three failures occurring in P Area and four in C Area. Better containment of radioactivity was experienced since the majority of the failures were discharged into "harps." However, "harps" were not available in all cases. The following table summarizes the failed fuel element experience for this period.

FAILED FUEL ELEMENT EXPERIENCE

Area	Date of Failure	Vented in "Harp"	Weir Water Maximum Activity, μc/ml	Factor Increase in Weir Water Activity	Emerg Basin Maximum Activity, μc/ml	Estd Curies to Seepage Basin	Estd Curies Nonvolatile Beta to Effluent*	Failed Fuel Element Disassembled
R	4/18	Yes	4.0×10^{-4}	40	-	**	24.0	In Progress
R	4/21	Yes	2.5×10^{-3}	200 (C&D)	-	**	24.0	†
P	4/3	No	1.0×10^{-5}	Negligible	1.0×10^{-3}	**	1.5	Completed
P	4/16	Yes	5.0×10^{-5}	10	3.0×10^{-5}	**	1.5	†
P	4/21	No	5.0×10^{-5}	10	1.0×10^{-1}	11	1.5	Completed
L	4/1	Yes	1.0×10^{-5}	12	No Basin	**	6.5	†
L	4/26	Yes	1.5×10^{-5}	50	No Basin	**	0	†
K	4/1	No	2.0×10^{-5}	10	1.0×10^{-2}	2.0	33.0	In Progress
K	4/4	No	6.5×10^{-5}	20	7.5×10^{-2}	2.0	33.0	†
C	4/21	Yes	††	Negligible	3.0×10^{-2}	55†	29.5	†
C	4/21	No	††	Negligible	3.0×10^{-2}	55†	29.5	Complete
C	4/23	No	††	Negligible	3.0×10^{-2}	55†	29.5	†
C	4/29	Yes	2.0×10^{-4}	3	3.0×10^{-2}	55†	29.5	†

* Total releases to effluent during the report period.
** Not pumped.
† Not started.
†† No significant increase in weir activity.
† Activity from previously reported failures.

Frequent reactor outages and subsequent scheduled work in radiation zones resulted in a total of 720 authorizations to exceed the daily control limit for whole body radiation exposure. This was a factor of two increase in the number of authorizations as compared to the previous six months average.

On separate occasions in R, P and C reactor buildings gaseous airborne activity was detectable in clean areas. These observations were concurrent with fuel element failures. Low level skin, clothing and equipment contamination resulted. The contaminants were short-lived and appeared to be daughters of even shorter-lived gaseous precursors. The most probable cause for this radioactivity in clean areas was a temporarily unbalanced air flow.

R AREA

Routine operation of the R-Area isolation and decontamination facility continued to be hampered with severe contamination problems. Concentration of nonvolatile beta activity in the isolation tank ranged from 4.5×10^{-2} μc/ml to 2.5×10^{-4} μc/ml. Smearable contamination in the work area was as high as 5 rad/hr at 3", necessitating the continued use of air supplied plastic suits. Containment of radioactive contamination in the vicinity of the work area was a major problem. Plans for facility equipment improvement include:

- Complete enclosure of the isolation tank from the floor to the ceiling with necessary access doors.
- Adequate ventilation to maintain the enclosure under negative pressure with respect to disassembly - air being exhausted through adequate particulate filters.
- Breathing air manifold.
- Separate clean water supply.
- Additional electric hoists with positive protection to prevent raising irradiated fuel components from the basin.
- Shielding for exposed piping.
- Better underwater tools.

"Harps" were removed from the disassembly and isolation basins on a number of occasions for shipment to other areas for containment of failed elements. Since "harps" cannot be completely decontaminated prior to removal from these basins, it is necessary to wrap them to prevent spread of radioactive contamination during handling and transporting. Maximum contamination encountered was 40,000 c/m as measured by Thyac on a paper towel smear. The maximum radiation level was 4 rad/hr at 3".

The reactor remained down for charge-discharge operations following the second fuel element failure of the R:5 reactor cycle. Scheduled work with high radiation exposure rates included the replacement of thermocouples in the -40' level pin room. Personnel exposure rates averaged 700 mrad/hr to the head, with hand exposure rates of 1.2 rad/hr. Maximum tritium air concentrations were as high as 3.5×10^{-3} μc/cc, necessitating the use of plastic suits.

Routine processing of an Instrument mechanic's film badge indicated an exposure of 1.8 r for a two-week film badge period. The exposure pattern on the film was unusual and could only be duplicated under carefully controlled conditions with the radiation flux incident on the badge in the same plane as the film. The film image was duplicated using a Co-60 field calibration source. Special Hazards Committee investigation concluded that the employee's badge had been tampered with during the badge period.

P AREA

The reactor was shut down on three separate occasions to discharge suspected fuel element failures. Only one of these elements was placed in a "harp" since "harps" were not available on two occassions. The two failures which were not contained in "harps" were completely disassembled in the emergency basin. Less than one curie of nonvolatile beta activity was released to the effluent stream as a result of these failures. The failure which was discharged on April 21 was apparently severe and resulted

in water contamination in the emergency basin as high as 1.3×10^{-1} μc/ml.

Following disassembly of this failure, a "rigging choker" which was contaminated to 15 rad/hr at 3" was removed from the emergency basin without proper controls. This resulted in extensive contamination of a Reactor Department operator as well as widespread clean area contamination.

Water from the SCRUP railroad cask containing spent U-238 fuel elements from Chalk River was pumped to the seepage basin prior to unloading the cask. Water samples taken prior to pumping showed nonvolatile beta concentrations of 2.6×10^{-7} c/ml. The cask was flushed for four hours, using clean makeup water, reducing the concentration of activity in the cask by a factor of 30. Additional flushing will be done prior to removing the elements for shipment to the Separations Area for processing.

Two commercial trailers (Rowe Transfer Company) which were used in the shipment of irradiated Mark VII-A material to Oak Ridge were returned to the vendor after extensive decontamination including replacement of the wooden floor.

L AREA

The reactor remained down for charge-discharge operations, L:4-5 outage, following a fuel element failure. Radiation exposure rates were comparable to those experienced during the last reactor outage for the purpose of charge-discharge. Scheduled work required respiratory protection on a number of occasions due to extensive contamination, particularly at 0' level following C&D operations and at -20' level on the repair of the C-1 fission chamber.

The reactor was shut down due to leaking heat exchanger 4-A. The radiation exposure rates during replacement of the heat exchanger were lower by a factor of two than those experienced during the L:4-5 outage. This improved condition was attributed primarily to a new evaporator which was installed in purification approximately two months ago.

Control rods with long reactor exposure were broken in the dry cave. These control rods showed considerable swelling and warping; outgassing was observed from the slugs after breaking and storing under water. Tritium concentrations as high as 0.1 μc/cc were observed when bubbles burst at the surface of the disassembly basin water outside the dry cave. No positive tritium bio-assay samples were encountered.

K AREA

Disassembly of three special rods irradiated during the K:2 cycle and discharged from the reactor approximately four months ago was completed in the emergency basin. Bubbles of radioactive gas reading as high as 5 rad/hr burst at the surface during the disassembly operation. Gamma scans of the gas indicated primarily Kr-85.

Disassembly of three failed Mark VII-A fuel elements in the emergency basin continued, with water contamination as high as 7.2×10^{-2} μc/ml, nonvolatile beta. Smearable contamination on tools as high as 800 mrad/hr, and as high as 20 mrad/hr from surfaces in the work area necessitated the use of plastic air supplied suits.

The reactor was shut down for the scheduled K:3-4 charge-discharge operations. Radiation exposure rates were comparable to those experienced during the K:2-3 outage. Whole body exposure rates of 1000 mr/hr were experienced during replacement of thermocouples in the -40' level pin room. Tritium air concentrations of 7.3×10^{-3} μc/cc necessitated the use of plastic suits. Unplugging of septifoil, sparger and transducer lines was carried out in radiation fields averaging 1 r/hr. This necessitated a large number of authorizations to exceed the daily control limit. Seals were changed on the overflow tank room pump with exposure rates as high as 500 mr/hr.

C AREA

Work was completed on the disassembly of two of the four failed fuel elements stored in the emergency disassembly basin. Plastic suits were worn for major portions of the work because of high air activity. Floor contamination of 5 to 10 mrad/hr at 3" was measured on paper towel smears. Ropes and hoists removed from the basin were contaminated to 1500 mrad/hr at 3" and required replacement or decontamination. Activity in the basin water reached a maximum of 3.1×10^{-2} μc/ml.

Leakage through the stop logs to the disassembly basin and thus to the effluent was reduced by pumping from the monitor basin to the seepage basin, but only intermittent operation was possible without overflowing the seepage basin. An estimated 55 curies were pumped to the seepage basin and 30 curies released to the effluent.

A heat exchanger removed from L Area was repaired in the stack area. The Stokes vacuum dryer was used effectively prior to work and no significant tritium air activity was encountered as the work progressed, eliminating the need for plastic suits. However, particulate air activity necessitated the continued use of assault masks. On one occasion the floor of the work area was contaminated to 1100 mrad/hr at 3" when the heat exchanger heads were removed. Detection was immediate and contamination was not spread outside the immediate work area.

400 AREA

Installation of the evaporator was completed in the 420-D rework facility. Air activity reached a maximum of 9.5×10^{-3} μc H-3/cc during tie-in of this equipment and required the use of plastic suits. Repairs to leaking pump seals caused air activity problems during the report period. Nine Maintenance and Production Department personnel had tritium assimilations above 5 μc/ℓ, which is the maximum uptake normally experienced during routine operations. The maximum uptake experienced under the controlled working conditions was 18 μc/ℓ.

SEPARATIONS AREAS

GENERAL

The special product program resulted in significantly higher radiation levels (up to 5 r/hr) in H-Area warm canyon, B-Line, and the solvent handling system.

More than an estimated 0.1 gram of plutonium was released to the control hut in the transfer of a precipitator between H and FB-Lines.

Leakage from the solvent trailer resulted in contamination of a narrow strip of roadway between the burial ground and H Area.

Difficulty in operation of the mixer settlers in Building 221-F caused radiation levels to increase above normal levels by factors as high as 30.

CANYONS

An inadequate air pressure differential, between the hot canyon and hot canyon sample aisle, and the high gamma activity of process streams were of primary concern to Health Physics following the startup of Building 221-F.

On more than thirty occasions, airborne fission product contamination in the hot canyon sample aisle exceeded the maximum permissible concentration (MPC) for personnel without respiratory protection; the maximum was approximately 2×10^{-8} μc/cc. Repeated tests indicated that when the railroad tunnel door was opened, the air pressure in the canyon became positive with reference to the sample aisle; hence, air flowed from the canyon up through the samplers into the sample aisle. No air activity was experienced in the sample aisle after air flows were adjusted to maintain the canyon at a negative pressure with respect to the aisle.

As a result of erratic mixer-settler operation, gamma activity in process streams was abnormally high during the latter part of this report period. The increased radiation levels in Building 221-F facilities were as follows:

Location	Normal Level, mr/hr	Current Level, mr/hr
Warm crane cab	1	6-10
Warm crane walkway	3-10	100-150
2nd level unshielded solvent lines, during transfers	45	900
2nd level center, Sections 12-18	1	10
WGVC adjacent to low activity waste system	1	7
Tank 901, solvent washer, through concrete shield	3	30
Recovered-acid run tanks	2	60

Air activity in the F-Area warm crane walkway increased to 60 times MPC when liquid leaked from a faulty connection on the low activity waste reboiler. The walkway was contaminated to 7000 c/m beta and 10,000 d/m alpha per square foot. On another occasion, when the 1D mixer-settler feed tank was open for agitator repairs, air activity on the walkway was 10 times MPC for a two-day period.

The F-Area centrifuge to centrifuge-run-tank jumper (11.1c-11.2) was placed in the hot shop for repairs. A dosage rate of 200 rad/hr at 1' required its transfer to the decontamination cell where the contamination was reduced to 45 rad/hr at 1'. Attempts to shield the connectors in the hot shop were finally deemed impractical and the jumper was returned to the hot canyon. Follow-up surveys of the hot shop floor, after rinsing with water, showed contamination up to 5 rad/hr at 3'. The floor was decontaminated successfully.

Fission product contamination was detected in the Building 221-H hot canyon crane cab after a crane operator found his gloves contaminated. The loose contamination ranged from 150 c/m per 100 square centimeters to a maximum of 60,000 c/m per 100 square centimeters. The source of contamination was traced to the ceiling where the north optic tube housing enters the cab. Radiation levels up to 100 mrad/hr at 2" were observed at this location. Analyses of bio-assay samples submitted by crane operators showed a fission product assimilation (1% MPA) by one operator. The latter was working in the cab at the time HM recorder data indicated that the incident occurred. Radioanalysis of loose contamination from the crane cab showed the presence of radioisotopes of strontium, zirconium-niobium, ruthenium, cerium, and cesium.

Processing of special product material in the H-Area canyons resulted in continued higher than normal radiation levels throughout Building 221-H. For example, during a transfer of special product material from the cake dissolving vessel (12.1) in the hot canyon to the 2A bank feed (12.5) vessel in the warm canyon, the following radiation levels were observed:

- Section 11 (Crossover line embedded in the concrete from hot canyon to warm canyon).

  3rd Level Tank Gallery - 1500 mr/hr at 2" from floor

  Hot Canyon Sample Aisle - 3500 mr/hr at 2" from floor

  Warm Canyon Sample Aisle - 5000 mr/hr at 1' from floor

- Warm Canyon Walkway

  One foot over and above parapet in Section 11 - 2000 mr/hr

  Center of warm canyon walkway in Section 11 - 200 mr/hr

Similarly, the processing of special product waste streams in low activity waste evaporator 6.8E resulted in radiation levels up to 700 mr/hr at 1' above the warm canyon walkway parapet and 180 mr/hr in the center of the walkway.

Pluggage occurred in the cake dissolving vessel (12.1) sampler causing a delay in the special product program. During replacement of the sight-glass and needles, and in the rodding of lines, radiation levels were noted as high as 50 rad/hr at 3". Four employees received authorized body doses of 200 mrad, or hand equivalent, and eighteen employees received authorized body doses of 100 mrad, or hand equivalent, while performing the repair work. The maximum body exposure rate to personnel was 1700 mrad/hr and the maximum hand exposure rate 22 rad/hr.

During circulation of the rerun aqueous sampler, 18.1H, air activity greater than 100 × MPC for fission products was detected downwind of the sampler. It was discovered later that, due to a leak, loose contamination as high as 10 rad/hr per square foot was deposited between the sampler quonset shielding and the concrete wall. As a consequence, it was necessary to replace the sampler sightglass assembly and Christmas Tree while encountering body exposure rates as high as 2 rad/hr and hand exposure rates as high as 50 rad/hr.

Approximately 100,000 pounds of water was released from an open cooling water nozzle in Section 6 of the Building 221-H hot canyon and Sections 5, 6, and 7 of the hot canyon were flooded to a depth of approximately 24". As a result of the flooding of the cells, contaminated water seeped through the Section 5 and 6 expansion joint to the 1st level clean corridor. Radiation levels of 45 mrad/hr at 2" and loose contamination of 20,000 c/m per square foot were detected at the expansion joint. Some of the contaminated water apparently channeled along the floor expansion joint and came to the surface under the asphalt tile in the regulated locker room. Although there was no smearable contamination detected, radiation levels up to 12,000 c/m were detected over the tile-covered expansion joint.

JB-LINE

The following radiation levels were experienced during a typical product run:

Location	Maximum Radiation Level, mrem/hr
3" from the precipitator cabinet	15 neutrons
6" above filter boats and drying pans	235 (200 mrem/hr neutrons plus 35 mr/hr gamma)
3" from the final product	495 (480 mr/hr gamma plus 15 mrem/hr neutrons)

Numerous process leaks were experienced inside the JB-Line cabinets during startup. On twelve separate occasions, personnel were required to remove coupling cabinet panels into contamination control huts which were erected in the maintenance room. Outside the control huts, airborne plutonium contamination increased to a maximum of 65 × MPC for personnel without respiratory protection when waste was packaged near a control hut entrance. Personnel involved were wearing air-supplied suits.

Through an omission, CWS filters were not installed in the mechanical line boat dumper, pan dumper, and pressure chamber cooling cabinets. Approximately 40 milligrams of plutonium were released to the exhaust system secondary filters. There was no significant release of activity from the secondary filters to the Building 294-F sand filter.

B-LINES

In FB-Line, liquid containing approximately 1 milligram of plutonium leaked from the BP-4 precipitator cabinet onto the cabinet supports. The leak was discovered after an operator who was working at the cabinet detected contamination to 100,000 d/m on his coveralls and gloves. Plutonium airborne contamination increased to 9 × MPC during the successful decontamination of the affected area.

Installation of an air sparge line in the F-Area recovery recycle hold tank required several entries into the cabinet. Approximately 1.2 milligrams of plutonium were released to the contamination control hut during this work. An air sample taken 10 feet from the work site indicated airborne plutonium contamination to 5500 × MPC during removal of the hut. Four other air samples taken during the same period indicated the spread of airborne contamination throughout recovery. All personnel in the room were wearing assault masks during the hut removal. Nasal smears indicated no contamination; however, bio-assay samples were requested as a precautionary measure.

High level alpha contamination was released to a control hut during removal of an HB-Line precipitator from the mechanical line for use in FB-Line. Loose contamination up to 7.6×10^7 c/m and airborne alpha activity up to 4000 × MPC for plutonium were detected. An estimated 0.1 gram of product was released to the control hut before this work was completed. Contamination was immobilized effectively through the use of polyvinyl acetate emulsion applied to the walls and floor of the control hut. In the course of the job a Maintenance mechanic contaminated his left hand to 7500 d/m alpha when his grossly-contaminated gloves were torn in removing the precipitator agitator. Decontamination of the employee's hand was effected.

During the concentration of special-product-bearing streams on resin columns in HB-Line, higher than normal radiation levels were observed in the feed tank area and in the coupling facility. The radiation level at the cabinet panel housing the JT-22, JT-62, and BP-A vessels increased to a maximum of 80 mr/hr at 3". The radiation level at these locations is normally no greater than 6 mr/hr. The radiation level of the vessel containing the final special-product-concentrate was 400 mr/hr at 3" and the level from the outer shipping container was 150 mr/hr at 3".

A-LINE, BUILDING 211-H

Purex process solvent was transported by tank truck to Building 221-F for reuse and to the burial ground for storage in underground tanks. During one of the trips to the burial ground, contaminated solvent leaked from the forward top hatch of the tank truck. A strip of the roadway up to 18" wide and a mile long was contaminated to an average of 2500 c/m beta-gamma from the railroad crossing west of 200-H Area to the burial ground. Radiation levels up to 10 mrad/hr at 3" were observed from pools of the contaminated solvent at the burial ground and near the railroad crossing. Since the usual methods of decontamination were ineffective, the contaminated strip was excavated and repaved. Special Hazards Incident Investigation Number 100, "Contamination of 200-H Roadway to the Burial Ground," provides additional information.

While replacing a check valve in the 6.4D dissolver off-gas line on the east side of Building 221-H, a Maintenance mechanic contaminated his leg to 5000 c/m beta-gamma. The contamination occurred when approximately one half pint of liquid, having a radiation level of 5 rad/hr at 2", spilled from the line when the check valve was removed. The contaminated liquid penetrated two pairs of coveralls and taped canvas boot covers. One of the employee's personal shoes was contaminated to 15 mrad/hr at 2". Nasal smears indicated no contamination; decontamination of his leg was successful.

BUILDING 241-H

In February, the connectors on an overflow jumper in pump pit 4 were loosened to eliminate the possibility of the overflow of high activity waste to the Building 241-H catch tank. Because of extremely high radiation levels, the jumper was not removed. The overflow header was cut and blanked as an alternative. In order to retighten the connectors the cell covers were removed again. A maximum radiation level of 35 r/hr was observed directly over the open pit. Personnel were exposed to radiation levels of 500 mr/hr during the removal of the cell covers.

BUILDING 772-F

Data obtained from the analysis of bio-assay samples of personnel working in the Building 772-F Purex laboratories during a period of high activity (reported in last month's report) indicated that eight employees assimilated minute quantities of fission products. The highest assimilation was less than 1% of the maximum permissible body burden.

Upon investigation, it was concluded that the intermittent failure of the high level and/or low level exhaust blowers was the most logical cause of the high air activity (28 × MPC). Further information was reported in "Airborne Activity in Center Section (Purex Labs)," Special Hazards Incident Number 102, Laboratories Section, Building 772-F.

BUILDING 643-G (BURIAL GROUND)

Several pieces of equipment were decontaminated successfully by sandblasting at the burial ground. The maximum airborne contamination detected 50' downwind from the job site was 12 × MPC for fission products. All personnel involved in this work wore respiratory protection. The following table lists the respective contamination levels of equipment decontaminated using this technique:

Item	Maximum Contamination, mrad/hr	
	Before Sandblasting	After Sandblasting
H-Area dissolver slug chute rack	600 at 2"	15 at 2"
H-Area bucket storage racks	5000 at 6"	1 at 2"
Evaporator riser plates	300 at 3"	30 at 3"
Reboiler parts	50 at 2"	10 at 2"

While this is an effective decontamination method it suffers from the fact that the radioactive material removed is then disposed over the grounds.

BUILDING 232-H

An indication of effective controls in the miscellaneous products area was reflected in the single skin contamination case and the lone assimilation of only 0.2 MPA for the report period.

The processing of Mark VI-J target elements, irradiated in two reactor cycles, increased the radiation-contamination problems associated with work in the decanner area:

- Plastic suits were required for work inside the hood since vacuum cleaning failed to reduce product concentrations in air to less than 60×10^{-5} μc/cc. Concentrations of product in air as high as 600×10^{-5} μc/cc were noted. This was approximately three times the concentrations normally encountered.

- Body exposure rates averaged 100 mrad/hr as compared with a previous 60 mrad/hr.

- Concentrations of product in air increased by a factor of three for open scrap casks (7000×10^{-5} μc/cc), open crucible casks (5000×10^{-5} μc/cc), and open slug casks (200×10^{-5} μc/cc).

The north furnace was successfully decontaminated in place with 1% nitric acid following the failure of four Calrod heating elements. The acid was added through a plastic line from above the furnace hood, agitated by moving an empty crucible up and down inside the furnace, and siphoned into a 55 gallon drum for disposal. Subsequent water rinses were collected similarly. As summarized in the following table, sufficient decontamination factors were achieved:

Location	Before Decontamination, mr/hr	After Decontamination, mr/hr
Front of furnace	300 at 12"	50 at 3"
Bottom of furnace	1000 at 3"	70 at 3"
Back of furnace	800 at 13"	80 at 3"

The radiation level of the drum containing the acid solution was 4.3 r/hr at 3". Analysis of this liquid indicated the major component to be Zn-65 in a concentration of 6 μc/ml. Further analysis indicated 0.33 mc/ml or approximately 50 curies of product. The major steps in this operation were performed remotely to reduce radiation exposure to personnel and the maximum rate to the body was 2 r/hr.

BUILDING 232-F

Product concentrations in air greater than 5×10^{-5} μc/cc in the regulated corridor were noted on three occasions during this period. The maximum concentration was 80×10^{-5} μc/cc and occurred as a short peak. The source of activity was attributed to a partially closed damper on the No. 2 exhaust fan. A faulty solenoid valve caused the damper to be out of adjustment. Since repair, no increases above background were noted.

SPECIAL STUDIES

300/700 AREA AND MISCELLANEOUS

BUILDING 736-A CALIBRATIONS

	April	1959 to Date
Instruments Calibrated	7271	26,948
Instruments Decontaminated	106	610

INDUSTRIAL HYGIENE

Amercoat No. 33B - Toxicity and Flammability Hazards. The toxicity and flammability hazards associated with the use of Amercoat No. 33 HB were investigated at request of the Projects Department. Results indicate that toxicity problems will be similar to those of Amercoat 33, while flammability problems will be greatly reduced due to the higher flash point of Amercoat 33 HB.

RADIATION ENGINEERING

University of Georgia Co-60 Source. A 180-curie Co-60 source, belonging to the University of Georgia for the irradiation of small animals, was transferred from Building 773-A and installed in a device at the Bush House without incident.

Evaluation of Dosimeters. A group of 119 Bendix (200 r) dosimeters and associated chargers obtained from Civil Defense were evaluated for the Atomic Energy Commission. The pencils were acceptably accurate, but extremely difficult to charge; chargers were inadequate to charge pencils. A report on this evaluation was issued.

Neutron Calibration Equipment. A special A-1 linear amplifier and scaler built by Engineering Assistance Group was installed in the Calibrations Building. This equipment will be used in Hurst ion chamber testing (fast neutron) dosimeters and neutron flux measurements with "long counter."

Test procedures were established for the Hurst Chambers and calibration curves were obtained for the "long counter." The "long counter" will be used in conjunction with special studies relating to neutron dosimetry.

Criticality Monitoring Program (Neutron Dosimeters). A small "pencil" containing several neutron-sensitive foils was developed as part of the Savannah River Plant criticality program. The purpose of these foils is to measure the neutron spectrum in the vicinity of personnel during a nuclear incident. By this means a better estimate can be made of the exposure received.

The pencil is 4½ inches long and ½ inch in diameter. It contains bare and cadmium covered indium foils, a cadmium covered copper foil, and a vial containing approximately 1 gram of sulfur. These materials can be used to estimate neutrons with energies between 0 to 0.4 ev, 0.4 ev to 1 Mev, and those above 2.9 Mev, respectively.

The pencil will also contain a small piece of gamma-sensitive cobalt glass and a vial of sodium fluoride. The cobalt glass has a range from 10^3 to 10^5 r. (The film packet presently used can measure gamma exposures up to 10^3 r.) The sodium fluoride may be useful in correlating the activation of body sodium to the moderating abilities of the body.

It is planned to mount these pencils throughout all critical areas.

At the present time 150 of these pencils are being fabricated for immediate coverage of all areas. Testing will continue to develop the most effective combinations of foils which can be housed in this package.

Heavy Water Components Test Reactor. Review of the current design status of the HWCTR was continued by the Radiation Engineering Group. The HWCTR HM system and spent fuel basin shielding were reviewed. The following conclusions were reached:

- HM System. The placement of Health Monitoring ionization chambers in the Reactor Building 770-U and associated Buildings 771-U, 774-U, and 735-U was studied. A list of recommended locations in the reactor and associated buildings was prepared. It was suggested that the HM chambers and the Beckman selector switches be grouped according to the expected radiation level in the area in which the chamber is located. The recommended HM chamber and Beckman multiswitch arrangements were also listed. A total of 32 chambers and 4 spares were recommended for the reactor building and associated facilities.

- Spent Fuel Basin Shielding. A study was made to determine the radiation levels expected in the pipe alley if a horizontal day old fuel element is placed on the ledge in the spent fuel basin directly above the pipe alley. Radiation levels of 2 r/hr at 2' from the ceiling and 110 mr/hr at head height in the pipe alley could be expected. Neither scattered radiation nor radiation from process piping was considered in determining the expected radiation levels. Existing shielding between the spent fuel basin and the pipe alley consists of two feet of ordinary concrete. An additional two inches of lead shielding on the spent fuel basin ledge above the pipe alley was recommended.

Inventory of D_2O, H_2O Using Reflected Neutrons. A test of the feasibility of checking sealed drums, to determine if they contained D_2O or H_2O by measuring neutrons reflected from a moderated Po-Be neutron source, was conducted for du Pont and Atomic Energy Commission accountability personnel. A 2.6-curie Po-Be source was placed in a paraffin moderator and moved next to drums of D_2O and H_2O. Readings were taken from the side of the drums 90° from the source location. Drums containing heavy water appear to reflect between 2 and 3 times as many slow neutrons as do similar drums of light water. Considerable more work is required to make a semiquantitative check, but feasibility was demonstrated.

Control and Methods

—REGIONAL SURVEY

	April	1959 to Date
Field Measurements	569	2,105
Samples Collected	2385	8,496
Samples Analyzed	4570	16,819
Routine	1243	
Special	2844	
Control	483	

STACK RELEASES

	3/16 - 4/15		1959 to Date	
	F Area	H Area	F Area	H Area
Alpha, mc	2.8	0.2	9.3	1.6
Nonvolatile Beta, mc	85	47	168	586
Radioiodine, c	0.8	0.1	0.8	34

ENVIRONMENTAL RADIOACTIVITY

Bomb Fallout. Deposition of nonvolatile beta on the Plant site during this report period was estimated to be approximately 50 curies, bringing the total deposition since the beginning of the October weapons tests to approximately 250 curies. Large concentrations of long-lived beta activity continued to be observed in environmental samples influenced by fallout of weapons tests debris.

Average concentration of filterable beta in air during the report period was 820×10^{-14} μc/cc, as compared to an average concentration of 760×10^{-14} μc/cc during the six months since the October weapons tests fallout was first detected (October 7, 1958 through April 21, 1959). Decay studies of beta activity collected on filters during the report period indicated an apparent half life of 75 to 120 days.

Concentrations of nonvolatile beta on vegetation reached the highest level observed since September 1957, but due to the new spring growth were lower during the latter part of the report period. Average concentrations ranged from 610×10^{-12} c/g in samples collected on March 25 to 130×10^{-12} c/g in samples collected on April 22.

RADIOACTIVITY IN WATER

Waste Released to Seepage Basins. The following table indicates radioactive waste released during this report period.

	3/19 - 4/15		1959 to Date	
	F Area	H Area	F Area	H Area
Alpha, mc	72	30	285	154
Nonvolatile Beta, c	2.2	2.2	7	18
Radioiodine, c	0.1	4.2	0.1	36
Tritium	33	22	33	1146
pH (range)	1.7 - 2.4	1.8 - 2.8	-	-

The pH of liquid in Basins 1 and 2 in both F and H Areas has been below 3.0 for the past two months. Soil adsorption of both Pu and Sr is greatly reduced below pH 3.0. pH is a major factor in soil retention. Radioactivity in the ground water has passed beyond the monitoring wells and it will be necessary to drill more sampling wells to define the contaminated area.

A summary of the pH values observed in the 200-Area seepage basin waste during the last two months is shown in the following table.

	pH of 200-Area Seepage Basin Water					
	F-Area Basin No.			H-Area Basin No.		
Date	1	2*	3*	1	2	3
2/25	6.5	-	-	2.4	2.9	9.4
3/4	2.3	-	-	2.5	2.8	9.6
3/11	1.3	-	-	2.7	2.9	9.6
3/18	1.2	2.3	9.2	2.5	2.7	9.6
3/25	1.8	-	-	2.3	2.7	10.0
4/1	2.0	-	-	2.2	2.8	9.7
4/8	1.8	-	-	2.2	2.6	10.1
4/15	1.5	1.9	3.5	2.2	2.6	9.8
4/22	1.6	-	3.1	2.2	2.6	9.9

* Sampled on a four-week cycle.

Radioactivity Released to Basins and Streams Due to Fuel Element Failures. Twelve fuel element failures occurred in the reactor areas during the period April 1 through April 26. An estimated 87 curies of nonvolatile beta activity were released to Plant streams from the disassembly basins. While the policy of isolation of failed fuel elements in "harps" or emergency basins has aided in reducing radioactivity releases following the discharge of the failed elements, the radioactivity released to Plant streams as a result of C&D operations following a rupture is very evident. The effect of fuel element failures in each area on releases of radioactive materials to Plant streams is shown in the following paragraphs.

K Area. Two ruptures occurred in K Area, the first on April 1 and the second on April 4. Nonvolatile beta activity in disassembly basin water increased by a factor of 10 during the discharge of the second failure but returned to normal after the element was isolated in the emergency basin. C&D operations began on April 20, 16 days after the last known failure.

Nonvolatile beta activity in disassembly basin water increased by a factor of 1000 following discharge of the irradiated load. Gamma pulse height analysis of basin water indicated that >90% of the gamma activity was due to Np-239. An estimated 30 curies of nonvolatile beta activity were released to Pen Branch from the disassembly basin. This represents a tenfold increase in released radioactivity when compared to previous discharges of K-Area reactor loads in which no rupture had occurred.

C Area. Three fuel elements failed in the C reactor during the three days April 21 through April 23. No appreciable releases of radioactivity were associated with the discharge of the failed elements. Approximately 30 curies of nonvolatile beta activity were released to Four Mile Creek during the period March 26 through April 25. This release was due to bleed-off of radioactivity from the disassembly basin after discharge of a reactor load on March 22 when C&D operations immediately followed the discharge of a series of failed elements.

Approximately 175 curies of nonvolatile beta activity were released in the past two months to Four Mile Creek as a result of this series of failures and the subsequent discharge of the reactor load. Gamma pulse height analyses of disassembly basin weir water during the period March 26 through April 25 indicated that approximately 85% of the radioactivity released was Np-239. The remainder was attributable to I-131, Ba-La^{140}, Ce-141, and radiostrontium in descending order of abundance.

R Area. There were two fuel element failures in R Area, the first on April 18 and the second on April 21. Discharge of the reactor load started on April 23. Approximately 24 curies of nonvolatile beta activity were released from the disassembly basin to Par Pond through April 25. Disassembly basin water radioactivity remained at 1×10^{-11} μc/ml during discharge of the failed elements, but increased to a maximum of 1.5×10^{-5} c/ml following discharge of the full load. Gamma pulse height analysis of the basin water indicated that about 90% of the gamma activity was Np-239.

P Area. Three fuel elements failed in P Area. The first occurred on April 3, the second on April 16, and the third on April 21. Disassembly basin water activity increased to a maximum of 5×10^{-11} c/ml during discharge of a failed element on April 21 but decreased to 10^{-12} c/ml within 24 hours. Approximately two curies of nonvolatile beta activity were released from the basin to Steel Creek during the report period. Discharge of the reactor load is not anticipated until the middle of May. Nonvolatile beta activity in the emergency basin increased to a maximum of 1.3×10^{-7} c/ml following disassembly of the failed element of April 21; stoplog leakage was held to a minimum by pumping emergency basin water to the seepage basin.

L Area. Two fuel element failures occurred in L Area, the first on April 1 and the second on April 26. Both elements were placed in "harps" and stored in the disassembly basin. Approximately six curies of nonvolatile beta activity were released from the basin to Steel Creek. The full reactor load was not discharged during the report period so the full effect of the failures was not seen.

Stream Water. Concentration of nonvolatile beta in the water of each reactor effluent stream and the estimated number of curies flowing past each stream sample location is shown in the following table.

	Nonvolatile Beta in Water, 3/27 - 4/23		
	Concentrations, 1×10^{-15} c/ml		
	Weekly Max	Avg	Estimated Discharge, curies/four weeks
Four Mile Creek at Road A	1000	520	12
Pen Branch at Road A	6000	1600	37
Steel Creek at Road A	170	130	6
Lower Three Runs at Par Pond Precooler	150*	140*	3*

* R-Area release discussed above was not covered by sampling period.

A water sample collected on January 23 from Mill Creek was found to contain tritium with a concentration of 13×10^{-3} $\mu c/\ell$. Additional samples were collected from Mill Creek on February 19 and February 26 at several locations over a distance of about six miles. All samples contained detectable amounts of tritium. In each set of samples, the maximum concentration was observed in the sample nearest the headwater of Mill Creek.

A summary of these results is shown in the following table.

		H-3, 1×10^{-3} $\mu c/\ell$		
Date	No. of Samples	Max	Min	Avg
2/19	5	30	7	21
2/26	13	32	7	18

Ground water seeping into a stream at the headwater contains a larger percentage of water from recent rainfall than ground water seeping in further downstream. Thus, rainfall appears to have contributed the tritium found in Mill Creek.

For comparison with these results, samples were collected on March 19 from 13 headwaters that feed the five major Plant streams. The maximum concentration of tritium observed in these streams was comparable to the Mill Creek samples. A resample of the same locations (13 headwaters) on April 20 indicated that the average concentration was only 55% as high as it had been on March 19. Rainfall for the two weeks previous to the April 20 samples was only 26% of the two-week rainfall previous to the March 19 samples.

A summary of these results is shown in the following table.

Date	No. of Samples	H-3, 1×10^{-3} μc/ℓ			Rainfall, inch*
		Max	Min	Avg	
3/19	13	31	9	18	2.3
4/20	13	18	<4.5	10	0.6

* Two weeks previous.

Savannah River. The flow of radioactivity past sampling locations upstream and downstream from the Plant during the period from March 25 through April 21 is shown in the following table.

Location	Radioactivity, curies/four weeks	
	Nonvolatile Beta	H-3
Upstream From Mouth of Upper Three Runs (control)	12	<1800
Highway 301 Crossing	24	5500

ZOOLOGICAL SPECIMENS

Par Pond. Radiostrontium concentrations in the bones of Par Pond fish (38 specimens collected) remained in equilibrium with the radiostrontium content in Par Pond water and, for the first time since December 1958, the radiocesium concentrations in the fleshy tissues did not increase. A comparison with concentrations found in March is shown in the following table.

Nonvolatile Beta, 1×10^{-12} c/g					
Bone			Flesh		
Max	Avg	March Avg	Max	Avg	March Avg
685	225	220	130	60	60

Par Pond waterfowl (16 ringed-neck ducks and one coot) contained slightly lower concentrations of nonvolatile beta in the bones and the same concentration in the flesh as reported in March.

Nonvolatile Beta, 1×10^{-12} c/g					
Bone			Flesh		
Max	Avg	March Avg	Max	Avg	March Avg
35	15	20	65	25	25

Lower Three Runs. A localized fish-kill six miles below the Par Pond dam during the latter part of March hampered fish collections and caused the downstream migration of some fish. The illegal fish-kill was apparently due to the use of chemicals by persons unknown in a backwater section of the stream. Three of the 10 fish collected 14 miles below the dam contained flesh concentrations of radiocesium that ranged from 4 to 15 times the average flesh concentration for this location. One of these fish collected on March 25 contained 600×10^{-12} c/g of radiocesium in the flesh, the highest flesh concentration found so far downstream to date. A comparison with the concentrations found in March is shown in the following table.

Miles Below Par Pond Dam	No. of Samples	Nonvolatile Beta, 1×10^{-12} c/g					
		Bone			Flesh		
		Max	Avg	March Avg	Max	Avg	March Avg
1	21	1200	520	385	765	155	165
6	2	770	645	475	80	65	205
14	10	360	155	110	600	145	40

Steel Creek. Concentrations of radiostrontium in the bones of Steel Creek fish (2 specimens), collected one mile from the Savannah River, were slightly lower than found in March. Trace concentrations of radiocesium were found in the flesh. A comparison with the concentrations found in March is shown in the following table.

Nonvolatile Beta, 1×10^{-12} c/g					
Bone			Flesh		
Max	Avg	March Avg	Max	Avg	March Avg
35	30	55	10	10	10

Four Mile Creek. Ten fish collected from Four Mile Creek, six miles downstream from C Area on March 25 contained generally higher concentrations of nonvolatile beta in the bones than were found on March 11. No significant change was noted in flesh concentrations as a result of the fuel element failures in C Area. A comparison of the concentrations before and after the incident in C Area is shown in the following table.

Nonvolatile Beta, 1×10^{-12} c/g					
Bone			Flesh		
3/25		3/11	3/25		3/11
Max	Avg	Avg	Max	Avg	Avg
110	45	30	10	10	10

Savannah River. A fish collected from the Savannah River near the mouth of Lower Three Runs on April 7 contained 110×10^{-12} c/g of radiostrontium in the bones and 35×10^{-12} c/g of radiocesium in the flesh, the highest flesh concentration found in Savannah River fish to date. Six additional fish from this location and nine fish collected sixty miles downstream at Stokes Bluff did not contain significant concentrations of radioactivity in the flesh. A fish collected near the mouth of Four Mile Creek on April 20 contained 115×10^{-12} c/g of nonvolatile beta activity in the bones, the highest bone concentration found at this location to date. A comparison with the bone concentrations found in March is shown in the following table.

Location	No. of Samples	Nonvolatile Beta in Bone, 1×10^{-12} c/g		
		Max	Avg	March Average
Upper Three Runs	7	10	10	10
Four Mile Creek	4	115	50	15
Steel Creek	3	20	15	10
Lower Three Runs	7	110	30	No Samples
Highway 301	1	10	10	10
Stokes Bluff	9	35	15	10

Miscellaneous. A turtle collected near H-Area seepage basin on April 3 contained 1000×10^{-12} c/g of nonvolatile beta in the bones and shell and 95×10^{-12} c/g in the flesh. Gamma pulse height analysis of the whole specimen revealed Cs-137 to be the primary gamma emitter.

CHEMISTRY METHODS

SPECIAL STUDIES

Plutonium Bio-Assay. Following revision of the evaporation-anion exchange procedure, the ratio of the tracks obtained from a 0.005 d/m/150 ml and a 0.01 d/m/150 ml spike was less than 1.2. In order to distinguish between these two levels, the procedure was evaluated using 250 ml and 300 ml samples. Preliminary data indicate that the number of tracks obtained corresponding to 0.05 d/m/1.5 ℓ and 0.10 d/m/1.5 ℓ is sufficiently above the blank activity to give the proper ratio when the larger sample volumes are used.

Neptunium Bio-Assay. Final evaluation of the ammoniacal phosphate coprecipitation-anion exchange procedure is in progress. This method includes:

- Coprecipitation of neptunium on an ammoniacal phosphate precipitate from raw urine.
- Separation of neptunium from macroquantities of other inorganic ions on Dowex-2 anion exchange resin.
- Concentration of neptunium on a small area by electrodeposition.
- Autoradiography and track counting.

Environmental Strontium-90. A second status report was prepared summarizing all data through March. Composite soil samples from all perimeters were extracted with 6N HCl for a "total" calcium assay. Zero to fifty percent greater calcium was extracted than was obtained previously by the "available" ammonium acetate method. Strontium-90 and Cs-137 assays will also be made on these samples.

Tri-Carb Interferences. Eleven different fission products were observed by the Tri-Carb liquid scintillation counter. Studied were $Sr\text{-}Y^{90}$, I-131, Ru-103, Fe-59, $Zr\text{-}Nb^{95}$, Ce-144, Ru-106, Cs-137, Cr-51, Co-60, and Zn-65. Results show that if these are abundantly present, they interfere with the determinations of tritium in urine.

Tritium Loss to Container. After a period of two months no loss of tritium was noted where wax or polyethylene containers were used to hold low level tritium oxide solution (1.45 μc/ℓ).

Cesium-137 in Urine. A method for the quantitative determination of cesium in urine was completed. A recovery of 91 ± 6.7% was obtained. The sensitivity of the method is 1.5 ± 0.1 d/m using a low background beta counter. A determination of the cesium content for a group of ten employees was made. An average value of 39.5 μμc of Cs-137/g potassium was found using an average of 2.5 grams of potassium per 1500 ml of urine (average daily void). No potassium estimation per sample was made.

Trace Levels of Tritium in Urine. The study of the background amount and the nature of tritium in urine was concluded. An average total tritium concentration of 0.021 ± 0.005 μc per liter of urine was obtained from the urinalysis of personnel known not to have inhaled or injested any radioactive materials other than those present in food, water and air. A comparison between the amount of tritium found by the combustion of urine, 0.019 ± 0.003 μc/ℓ and that found in the distillate fraction of urine 0.006 ± 0.003 μc/ℓ indicates the presence of bound tritium in urine. Electrolytic concentration of the activity in the distillate verified the presence of tritium.

PERSONNEL MONITORING

A routine bio-assay sample submitted by a Separations Area mechanic indicated a fission product uptake (it was later confirmed by a resample). This was the first fission product uptake determined by routine sampling this year.

Tests indicate plutonium-238 can be recovered through either (coprecipitation or ion exchange) plutonium bio-assay procedure with the same 0.05 d/m/1.5 ℓ threshold as currently obtained with plutonium-239.

BIO-ASSAY

Bio-assay results are shown in the following table.

	Totals	Positive Results	Maximum Results	1959 to Date
Samples Analyzed	2659	-	-	10,477
Tritium	1964	137	27 μc/ℓ	7,937
Uranium	147	0	<5 μg/ℓ	496
Plutonium	365	95*	1.10 d/m/750 ml	1,479
Fission Products	125	22	932 d/m/225 ml	462
Enriched Uranium	58	2	16 d/m/1.5 ℓ	103

* Fifty-one suspected to be due to contamination of the samples during analyses.

STATISTICS

Film Badge*

Area	Badges Processed	Lost Results	300 to 600 mrad	Over 600 mrad
F	3,559	6	12	0
H	2,675	10	6	0
M	1,422	2	7	0
D	189	0	0	0
Tech	2,515	6	3	1**
A	2,802	2	0	0
R	1,234	6	2	0
P	994	4	0	0
L	995	5	1	0
K	884	1	0	0
C	1,517	1	2	0
Misc	1,172	6	0	0
Control	2,476	-	-	-
Total →	22,434	49	33	1
Total Year to Date →	69,274	162	127	2

* Cycles 6, 7, and 8 of 1959.
** Over 300 mrad in one week.

Investigation of lost film results showed no possibility of overexposure.

Miscellaneous Film

Processed	April	1959 to Date
Neutron Film	154	510
Finger Rings	31	97
5 × 7 Radioautographs	3	6
14 × 17 Radioautographs	163	450

	April	1959 to Date
Lost film results		
Film lost from badge while being worn	7	30
Film exposed to light or fogged	3	13
Lost complete badge	34	106
Defective film	0	0
Film contaminated and destroyed	4	9
Film damaged by moisture	1	4
Lost by Personnel Meters	0	0
Badge sent through laundry	0	0
Total →	49	162
Special badge pulls	23	88
Visitor badges worn by personnel who have permanent badges	1174	3099

LABORATORIES SECTION

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Analytical Summary

Control Laboratories	Samples	Determinations
Raw Materials	1,400	2,808
Essential Materials	128	167
Separations	7,018	16,561
Heavy Water (400 Area)	4,094	4,931
Reactor Moderator (100 Areas)	2,173	5,279
Power Department (100 and 400 Areas)	209	1,056
Metallurgical	8,300	8,300
Total →	23,322	39,102[1]

Reactor Moderator Control

Mass spectrometer and pycnometer cross-checks of moderator continued in good agreement. The average difference observed in 12 samples was 0.012 mol %; the maximum difference was 0.03 mol %.

An analytical procedure has been developed for the determination of mercury in moderator at concentrations in the ppb range. This method, developed at the request of the Reactor Technology Section under RTA 435, is a dithizone extraction procedure, sensitive to 10 ppb of mercury. A precision of approximately 10% (relative) has been obtained.

Metallurgy Control

The status of ESF element inspection facilities is as follows.

BORESCOPES

Modifications in L and C Areas will be scheduled after modifications in K Area are completed.

COMPARATORS

Underwater nitrogen pressure gages have been installed on comparators in K, L, and C Areas.

The comparator for P Area has been crated and placed in storage.

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[1] In last month's report this value was reported as 37,309; this should have been 32,309.

On all comparators the scale for measuring the distance between scribe marks is reversed. This can be corrected but would involve changing the optical system, a major and costly modification. Instead, personnel will be retrained to take the reverse image into consideration when reading the scale.

PERISCOPES

The monocular periscope in P Area was installed in the P-Area monitor basin at the request of Production and will be used for slug inspection.

The binocular periscope in C Area was installed, but the top of the periscope was returned to the vendor for resilvering of the optics.

GAGE ASSEMBLIES

Modifications of the gage assemblies in C and L Areas have not been completed as Construction has not received the replacement parts from the Federal Products Corporation.

Separations Control

25 PROCESS

Installation of equipment in the three junior caves modified for 25-Process control was completed on April 10. Shift personnel are being trained in the use of this equipment.

A colorimetric method using hematoxylin as a color reagent was found satisfactory for the determination of 0.1M aluminum (concentration expected in dissolver heel hold tank 8.2). The 2 sigma precision for 20 determinations on a 0.1M $Al(NO_3)_3$ solution was ±14%; the bias was -1.0%.

The 25-Process synthetic quality control data for the period March 22 to April 23 are as follows:

Sample Code	Analysis	Method	Standard Value	Number of Determinations	2-Sigma Precision, %, ±	Bias, %
8.3	Uranium	Polarograph	3.63 g/ℓ	63	7.9	-0.9
8.3	Uranium	Colorimetric	3.63 g/ℓ	33	15.0	-2.0
11.8	Uranium	Fluorophotometric	2×10^{-3} g/ℓ	34	12.4	-1.4
8.3	Mercury	Colorimetric	0.002M	51	17.0	0.0
161	Mercury	Titrimetric	0.03M	22	0.66	-0.3
8.3	Free Acid	Conductometric	-	32	16.8	-
902	TBP	Acid Saturation	2.39%	25	6.5	-0.4
904	TBP	Acid Saturation	7.40%	16	4.3	+0.07

Shift personnel have been trained on all 25-Process analytical methods except the determination of plutonium by the lanthanum fluoride - TTA method and the determination of aluminum by the colorimetric - hematoxylin method. Specification analysis group personnel are being trained on the polarographic determination of uranium.

The four chemists assigned to shifts to assist in 25-Process startup are being trained in the process and analytical procedures.

Conversion is completed of the CEC 21-220A mass spectrometer from a solids to a gaseous instrument. One gas standard has been prepared and others have been ordered to cover the expected range of samples. Inadequate resolution of the isotopic peaks required realignment of the ion source and resulted in downtime which hindered checkout of the instrument. However, synthetic samples have been prepared and shift personnel are being trained in sample preparation.

PUREX

All raw metal solution samples are being analyzed for both nonextractable and extractable plutonium. A gross alpha analysis is used to determine total plutonium and a TTA analysis to determine extractable plutonium.

Difficulty has been experienced with the Mettler balance density apparatus located in the JB mechanical line cabinet. Although malfunctioning components have been repaired and the balance checked out by the analytical balance servicing agency, good agreement between successive weighings is difficult to obtain. This is primarily due to the sensitivity of the balance to off-center positioning of the button. Although the balance is inferior in precision to the H-Area balance, it is expected to meet the Separations Department's requirement of measuring density to a precision of ±0.1 unit. The analytical procedure for operation of the cabinet equipment and determination of product density is being forwarded for approval.

An analytical method has been developed for the determination of uranium in organic samples similar to 1AP; this is a TBP - thiocyanate colorimetric procedure adapted from a 25-Process method.

NEPTUNIUM RECOVERY

The scheduled laboratory-scale bismuth phosphate precipitation tests were completed on April 17. In the test period, a total of 50 samples were received and 110 neptunium determinations made. During production runs, some difficulty was encountered in dissolving the $BiPO_4$ cakes and several use-test experiments were therefore made for Separations Technology.

MISCELLANEOUS PRODUCTS

Three CEC 21-620 mass spectrometers required ion source changes during the month due to electrical shorts or loss of sensitivity. Nine source changes were required to repair the three instruments. As a result, mass spectrometer 3H has been returned to operation to facilitate checkout of ion sources, electronic drawers, and peak selectors, prior to their installation in spectrometers by the Instrument Department.

Raw Materials Control

Analytical procedures for determining the composition of baths and for assaying lithium-aluminum and uranium-aluminum alloys were all under statistical control during the month.

The scheduled comparison program of analyzing Mark VI-J alloy samples by both tin-fusion and vacuum-fusion was started during the first week in April. Fourteen samples have been analyzed by both methods.

For this comparison study, three samples are taken from a billet; equal portions of each of the three samples are combined and analyzed for hydrogen as a single vacuum-fusion sample. A portion of each sample is also analyzed by tin-fusion, resulting in triplicate hydrogen analyses by this method. The results of this study are as follows:

Billet No.	Tin-Fusion, vol % H_2	Vacuum-Fusion, vol % H_2
366L	32.0	12.6
502L	15.7	4.9
592L	13.2	7.0
602L	22.3	11.6
500L	23.6	29.0
560K	16.7	31.8
535K	28.1	32.1
596L	28.4	20.2
365J	21.4	21.0
583L	35.0	11.8
594L	34.6	38.5
594J	25.0	15.0
594K	21.8	10.8
497L	38.2	15.4
Avg →	25.4	18.7

These data follow the previously observed pattern in that tin-fusion values are generally higher than those by vacuum-fusion and have somewhat less scatter.

The moisture meter assembly (P_2O_5 conductivity cell) was successfully used to make preliminary moisture determinations on gas from the tin-fusion outgassing equipment. Before any quantitative measurements could be made, the equipment was recalled for use in the 200 Area.

The accelerated program to analyze aluminum-magnesium alloys for magnesium content has progressed satisfactorily during this report period. Five aluminum-magnesium standards (ranging from 1.46% to 5.15% magnesium) were received from Alcoa. Emission spectrographic analyses of these standards indicate that the 1.46% magnesium standard can be analyzed with a precision of ±0.05% (absolute). The precision increases to ±0.50% (absolute) with the 5.15% standard. Refinement of the spectrographic technique should improve this precision. A wet chemical technique is also being used for the magnesium analysis. This method, an ammonium sulfide separation and 8-hydroxyquinoline precipitation, is time-consuming but appears to be more precise than the emission spectrographic method. A precision of ±0.1% absolute is being obtained on Alcoa standards. The flame spectrophotometric method has been disappointing to date. Extremely high backgrounds from the burning fuel gases have made the magnesium content difficult to assess. Modifications to the photomultiplier unit of the instrument to reduce background interferences may reduce some of these difficulties. Routine production samples of aluminum-magnesium alloys are being received and analyzed by the wet chemical method. Suitable Alcoa standards are being analyzed with each group of routine samples as controls and emission spectrographic analyses of these samples are being used as secondary checks.

Essential Materials Control

All essential material shipments passed the acceptance tests and no waivers were requested.

Heavy Water Control

Pycnometer and mass spectrometer cross-checks on Extraction Area samples show satisfactory agreement:

Method	Deuterium Oxide, mol %
Mass Spectrometer	7.39
Pycnometer	7.27

During April the following cross-checks on samples of heavy water were reported by the 100-Area and 400-Area laboratories:

Deuterium Oxide, mol %

100 Area		400 Area	
Pycnometer	Mass Spec	Pycnometer	Mass Spec
99.73	99.70	99.70	99.70
82.82	-	82.87	-
7.76	-	7.91	7.92

The CEC 21-620 mass spectrometer has been installed in Building 772-D and the various components tested. Two sources, acceptable for use, have been supplied by the manufacturer. The instrument is being used for routine analysis.

REPORTS AND TECHNICAL PAPERS
ISSUED BY WORKS TECHNICAL DEPARTMENT

Separations Technology

Inman, G. H. and Mason, J. E., Jr., "Hydrostatic Test and Inspection, GS Unit 22 - Sixth Round," DPSPU 59-11-8, April 1959.

Myrick, R. E., "Heavy Water Extraction Units - Effect of Humidifier Purge Rate on Hot Tower Temperature," DPSPU 59-11-6, April 1959.

Shulko, R. M., "Factors Influencing Diffuser Recirculation Time Cycle," DPSP 59-25-5, April 1959.

Engineering Assistance

Bancroft, L. C., "Chalk River Papers," DPSPU 59-30-4, April 1959. Approved for presentation at Tripartite Instrumentation Conference, Chalk River.

Kelker, J. W. and Mousel, R. M., "Horizontal Double-End Slug Welding," DPSP 59-25-4, March 1959.

Kennedy, W. R., Jr., "B-Line Precipitator - Hi Volume," DPSP 59-25-8, April 1959.

Mousel, R. M., "Irradiated Mark I and Mark VIII Slugs - Inspection Data for R-2 Cycle," DPSP 59-25-9, April 1959.

Mousel, R. M., "Irradiated Mark I Slugs - Inspection Data for P-3 Cycle," DPSP 59-25-10, April 1959.

Health Physics

Sanders, S. M., Jr., "Power Functions Relating Excretion to Body Burden," DPSPU 59-30-2, April 1959. Approved for publication in "Health Physics."

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