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MONTHLY REPORT

MAY, 1968

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RICHLAND, WASHINGTON

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June 14, 1968

MONTHLY REPORT

MAY 1968

DOUGLAS UNITED NUCLEAR, INC.

Richland, Washington

Work performed under Contract No. AT(45-1)-1857 between the Atomic Energy Commission and Douglas United Nuclear, Inc.

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SUMMARY

REACTOR PLANT OPERATIONS

C & K Reactors

Input production (Pu) was 263.9 KMWD, totaling 202.8 at the two K reactors and 61.1 at C. U-233 input production was 4,068 equivalent MWD. Overall time operated efficiency was 81.1 percent, averaging 78.0 at KE and KW and 87.4 at C. The three reactors had operated concurrently for 25.4 days when the first May outage occurred. KE achieved 48.8 days of continuous operation, a new record for a K reactor, before its scheduled outage late in the month.

The production of nondefense plutonium containing 8-12 percent Pu-240 is continuing, but the program for making 27 percent Pu-240 material in depleted uranium is being completed. The E-D load in KE and about half of the E-D load in KW were discharged; the remainder in KW is scheduled for removal in early June. Thoria loads are replacing the depleted metal in both K reactors.

N Reactor

Input production was 73.9 KMWD. Time operated efficiency was 72.8 percent and steam availability was 71.4 percent. Reactor power level continued to be limited administratively to 3,500 MW to restrict Mark II fuel stress severity; this limit was further reduced to 3,460 MW late in the month because the confinement system vent valves operated slowly in a test performed during the scheduled outage. Modification of the valve actuating system is planned.

Cell 3 was returned to service following the completion of fire cleanup work and steam generator retubing. Cell 4 was removed from service, and steam generator 4B was successfully decontaminated in preparation for its retubing.

Boilout of the two new backup steam boilers was completed, and each unit achieved its full steam capacity of 200,000 pounds per hour. Further operational testing is scheduled to follow the June outage.

FUEL AND TARGET FABRICATION

C & K Reactors

Fuel production totaled 200.0 tons of 94 Metal elements and 14.3 tons of natural uranium elements. Canning line time operated efficiency was 99.4 percent. This is the fifth consecutive month with 99-plus efficiency. Thoria canning continued; a new high yield of 95.8 percent was achieved.

At month end, fuel core inventory was 975 tons, a 4.8 months' supply; finished fuel inventory was 1,112 tons, a 4.2 months' supply. These totals exclude all B Reactor fuel except the 94 Metal.

N Reactor

Input production totaled 304 extrusions, representing 141.2 percent of forecast. Extrusion of Mark IV fuel began on May 16. Output of finished fuel assemblies totaled 34.6 tons, which was 101.8 percent of forecast. The Mark IC conversion program is exceeding the planned rate, and all of the 300 Series 94 Metal assemblies have been converted except the 12-inch fuel.

TECHNICAL ACTIVITIES

C & K Reactors

The results of analyses of 14 Zircaloy process tubes removed from the K reactors after 55 to 59 months of exposure, some with and without anodized spacers, indicate that anodized spacers have not stopped hydrogen entry into the base metal. It is now suspected that the base metal hydriding may be caused by pickup of molecular hydrogen dissolved in the water. Laboratory studies of this mechanism are in progress, and more K tubes are to be removed for analysis and testing.

Studies continue on the K5E fuel model to determine the cause of the "beehive" corrosion in the spires which was experienced in late 1967. Evidence to date suggests that a higher-than-normal spire temperature is conducive to this type of corrosion.

Target element criteria, schedules and costs are being established in preparation for the irradiation of three test targets containing curium-244.

Except for the determination of Pu-236 content, data collection and analysis of the recently completed neptunium irradiation are finished and the results are being documented. A complete re-measuring and re-evaluation program is in progress to test the validity of Pu-236 measurements, past and current.

Engineering specifications covering uranium contamination in incoming thoria were revised to optimize product purity in the current U-233 production program.

An economic study of segmental charge-discharge at the K reactors showed a substantial incentive for the use of this technique if the refueling rate for natural uranium fuel elements can be made to equal current rates.

N Reactor

Inspection of Mark IV test fuel discharged from three tubes in March and April revealed no signs of malperformance. Assemblies from positions 3 and 4 from the downstream end of two tubes were sent to BNW Radiometallurgy for detailed examination. To date, the assemblies from one tube have been inspected and no evidence has been found of accelerated corrosion at the point of contact between the solid supports and the cladding. Definite conclusions cannot be reached until fuel elements subjected to higher power levels have been examined.

Presentations on the N Reactor licensability program were made to the AEC divisional staffs of Reactor Licensing, Operational Safety and Production on

[redacted]

April 23 and 24 at Bethesda, Md. Presentations covered subjects raised since the June 1967 meeting with the ACRS on N Reactor safety. A study of the AEC-DRL staff report on the Effluent Control Project and licensability was received after the Bethesda meeting. This study indicates that N Reactor can become licensable following implementation of the Effluent Control Project and resolution of some specific solvable problems.

Closed-circuit TV examination of two process tubes showed no significant changes in tube wall appearance between these and previously monitored tubes. Maximum depths measured showed a 1-mil scratch and a 11-12 mil fret mark. Both tubes showed heavy film from the upstream fuel piece to the center of the column, and light film from the center to the rear nozzle junction.

ADVANCED OPERATIONAL PLANNING

In cooperation with the Washington State Office of Nuclear Energy Development, the Warm Water Irrigation Proposal has been rewritten in prospectus form.

Updating of CAGE for the Mod 2 version is proceeding as rapidly as possible, with completion now set for August 1.

A preliminary study is nearly complete to estimate premium cost assignable to 2 percent Pu-240 plutonium. [redacted]

FEATURE REPORT

The appended summary report describes the Quality Program being used in the manufacture of fuel elements for the DUN-operated reactors. Recent in-reactor performance data for these fuels are included to illustrate Program effectiveness.

GENERAL

Employee Relations

A decision was received from the Federal Circuit Court of Appeals on the appealability of the District Court's earlier ruling on the question of contract coverage for N Reactor employees. The Company was advised that its appeal will not be entertained, and that the earlier ruling to submit the question to arbitration stands. Further courses of action open to the Company are being studied.

There were no disabling injuries, and no radiation exposures exceeded operational control.

Charles D. Harrington

Charles D. Harrington
President

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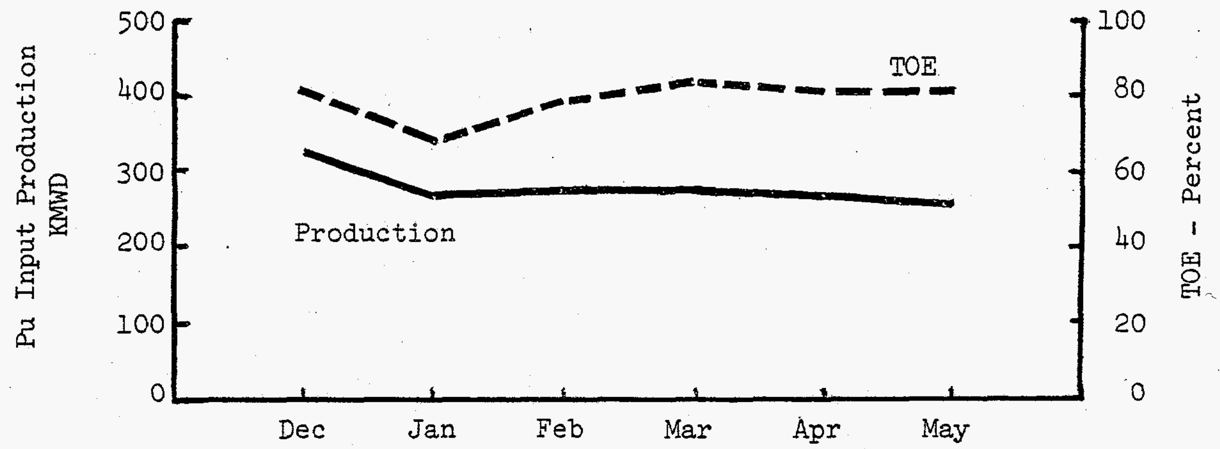
REACTOR PLANT OPERATIONS - C AND Ks

PRODUCTION

Reactor production, power levels, efficiencies and related statistics are tabulated on the next page.

Overall reactor input production and time operated efficiency for the past six months are charted below.



OPERATING EXPERIENCE

Reactor Loadings

Front face maps showing the loadings of the C, KE, and KW Reactors are reproduced on the three pages which follow page B-6. The tonnages listed are approximate; actual fuel totals are given on page B-2.

The special program for producing nondefense (high Pu-240) plutonium continued at the C and K reactors. At KW, following the failure of two depleted uranium elements (both in columns exposed to above 2600 MWD/T), about half of the depleted load was discharged to reduce the rupture potential, and was replaced with thoria target elements and supporting enrichment.

The KE Reactor was shut down May 25 for discharge of the depleted uranium load and the charging of a thoria load. The discharge in June of the remaining depleted metal in KW, and the charging of thoria there, will complete the program to produce plutonium containing 27 percent Pu-240, and will finish the initial charging of thoria loads in KE and KW to assure the delivery of 360 kg of "clean" U-233 as scheduled.

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PRODUCTION REACTOR STATISTICS — MAY 1968

REACTOR	C	KE	KW	TOTAL
INPUT PRODUCTION – PU – KMWD	61.1	102.6	100.1	263.9
– U-233 – EQUIV. MWD	1567	983	1518	4068
POWER LEVEL (MW) – MAXIMUM	2335	4325	4400	11 060
– AVERAGE	2258	4250	4135	10 643
TIME OPERATED EFFICIENCY – %	87.4	77.9	78.1	81.1
OUTAGE TIME ALLOCATION – %				
CHARGE – DISCHARGE	6.8	9.8	8.1	8.3
FAILED FUEL REMOVAL	0	0	2.4	0.8
WATER LEAKS	2.4	0	0	0.8
TUBE REPLACEMENT	0.4	0	0	0.1
OTHER MAINTENANCE	1.5	6.5	7.6	5.2
STANDARDS CHECK	0.9	1.5	0.9	1.1
PRODUCTION TESTS	0.6	3.9	2.8	2.4
PROJECT WORK	0	0	0	0
OTHER	0	0.4	0.1	0.2
TOTAL	12.6	22.1	21.9	18.9
NUMBER OF OUTAGES	1	1	3	5
NUMBER OF STARTUP INTERRUPTIONS	1	0	1	2
WATER LEAKS – TUBE	1	0	0	1
– VAN STONE	0	0	0	0
NEW TUBES INSTALLED	1	0	0	1
FUEL CHARGE – (TONS) – NATURAL URANIUM	140.3	288.0 (1)	263.1 (1)	691.4
– ENRICHED URANIUM	75.6	156.5	168.6	400.7
FUEL ELEMENT FAILURES	0	0	3	3
HELIUM CONSUMED – M CU. FT.	229.3	70.6	185.3	485.2
WATER TO REACTOR				
NORMAL OPERATING FLOW – GPM	107 000	196 000	196 000	499 000
pH	6.58	6.50	6.51	–
DICHROMATE – PPM	0.48	0.50	0.90	–

(1) Includes 45.3 tons (at KE) and 25.9 tons (at KW) of special depleted uranium in the E-D loadings (PITA-048).

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The irradiation of fuel in the C Reactor one-inch overbore block is progressing satisfactorily with no ruptures experienced to date. Seven charges with an average exposure of 516 MWD/T were discharged on a planned basis for inspection.

Power Levels

Power levels at the C and K reactors were restricted by the 95 C bulk outlet water temperature limit until May 27, when KW was restricted by the 4400 MW administrative limit. Five-pump operation continued at both Ks until mid-month when KW resumed six-pump operation.

Reactor Outages

The five reactor outages are summarized below:

Date Down	Reactor	Outage Hours	Remarks
May 15	KW	58.0	Removal of a failed 125 Metal fuel element (PTA-067) from tube 4365. A small charge-discharge also was completed.
May 18	KW	48.1	Removal of a failed depleted uranium fuel element from tube 1786 (PITA-048). Some miscellaneous maintenance work was completed, including gas leak repair and VSR work.
May 21	C	89.7	Scheduled charge-discharge and maintenance, including replacement of leaking tube 2563, EMS work, and inspection of Ball 3X entry piping.
May 22	KW	55.9	Removal of a failed depleted metal fuel element from tube 1680 (PITA-048). In addition, 131 other columns of depleted metal were discharged.
May 25	KE	164.5	Scheduled charge-discharge and maintenance, including replacement of two zirconium and one aluminum tubes, shrouding of three Ball 3X bellows, installation of a flat HCR in No. 5 channel, and changing orifices as required for the core thoria load which was charged. The outage continued through month end.

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EQUIPMENT EXPERIENCE

B Plant Deactivation

Completion of some phases of the B Plant deactivation work, such as the export water systems and the ventilation system, is being delayed by the Minor Construction forces strike.

Effluent Retention Tanks - 107-C

Repair to the 107-CW effluent tank which was leaking through the bottom was started May 31. This work is scheduled for completion by June 14; however, a delay caused by the Minor Construction strike is expected.

Confinement System - C and K Reactors

The 12 "A" cell absolute filters at C Reactor, damaged by water late in March, were replaced. A test showed that three additional filters were defective and these three also were replaced. Replacement of the charcoal filters in "A" cell also is planned before placing this cell in operation.

The charcoal filters in "B" cell at KW Reactor were found with voids and mechanical damage, which may have occurred prior to or when the filters were installed. This cell was taken out of service and a method of repair is being developed. It is expected that repairs will be made before June 15.

Horizontal Control Rods - KE Reactor

The original track blocks were removed from HCR channel No. 5 at KE, and flat track blocks were installed. A cooled flat HCR then was installed in this channel.

Charge Machines - KE Reactor

The K5 charge machines were successfully used at KE during the May 25 outage to charge K4 fuel, thus eliminating a need for maintaining the old K4 charge machines. This use of the K5 machines required the purchase of 515 magazines for the K4 fuel, and the fabrication of charge machine barrels and connector assemblies for use on aluminum tube type nozzles.

Ball 3X Bellows - K Reactors

Three ball 3X bellows were shrouded at KE. This completes the scheduled shrouding of ten bellows at KE and ten at KW.

Tube Replacement - KE Reactor

Zirconium process tubes 1672 and 1677 were replaced at KE to permit examination of the old tubes for hydriding effects. An aluminum tube (4456) also was replaced at KE.

High Speed Scanning Systems - K Reactors

The notably troublefree operation of the HSS system on the KE Reactor continued. Several minor problems have arisen in operation of the HSS system at KW, however, none of these have had significant impact on the continuity of operation.

Scaling amplifiers have been failing in approximately 3-6 months in both the KE and KW scanner systems. The most recently vendor-repaired scaling amplifiers were not repaired properly, and the units are being returned for rework.

PROCESS ASSISTANCE AND CONTROL

Process Physics

At KW Reactor, the replacement of 131 of the 300 columns of depleted uranium with 61 of thoria and 70 of 94 Metal, represented a partial early transition to the 94 Metal E-Q load scheduled for early June. The depleted fuel discharged had reached goal exposure. The present mixed E-D and E-Q lattice represents no operational physics problems.

Speed-of-control limits for the high exposure depleted uranium and other fuel required the use of additional helium to limit graphite temperatures for the final days of the operating period at KE Reactor. Speed-of-control and total control requirements for the E-Q loads at the K reactors will not be nearly as restrictive as for the past E-D loads. Speed-of-control requirements will not be limiting, and total control will be such that the minimum outage times will be somewhat reduced.

The 44-tube one-inch overbore block at C Reactor is running at the intended power. There are now 74 columns of 94 Metal and 13 columns of thoria constituting the surrounding buffer zone, which serves to lower the neutron density in the overbore block itself.

Flattening efficiencies will be reduced by the E-Q loads in the K reactors, but they should continue to be high at C Reactor.

Operational physics data of interest are summarized below:

	Reactors		
	C	KE	KW
Effective Central Tubes (ECT)*	1600	2279	2155
Flattening Efficiency** - May	0.85	0.70	0.70
- 12-mo. Average	0.83	0.71	0.70
Maximum Operating Time Permitting Scram Recovery - hours***	11	9	10

*Reactor power level divided by the average power of the ten most productive tubes which are representative of the reactor loading.

**ECT divided by the number of power generating tubes.

***The maximum operating time subsequent to a cold startup following which a scram recovery could be made using the currently approved startup procedures.

Production Fuel Performance

There were no failures of standard production fuel elements during the report period. Descriptions of three Production Test elements which failed are given under R&D Missions 1 and 5 in Section D of this report.

The following table shows failure frequencies, as number/million elements discharged, for the 3-, 12-, and 24-month periods ending April 30:

		3 Months	12 Months	24 Months
Small Reactors	- Natural U	0	40	25
	- 94 Metal	0	26	12
K Reactors	- Natural U	0	7	6
	- 94 Metal	19	9	8
C Reactor	- 1 Inch Overbore	0	-	-

Reactor Effluent Activity Data

The following table shows the total kilocuries/month discharge for six significant radionuclides in the reactor effluent during April:

Reactor	As-76	P-32	Zn-65	I-131	Cr-51	Np-239
C	2.6	0.28	0.35	0.017	5.4	2.2
KE	3.9	0.60	0.52	0.025	4.1	3.3
KW	3.1	0.46	0.33	0.055	3.5	2.9
Total	9.6	1.34	1.20	0.097	13.0	8.4

The progress of experimental work on effluent activity reduction is described in Section D of this report, under R&D Mission 10.

REACTOR PERSONNEL TRAINING

All 55 of the Nuclear Reactor Control Operators, and two Nuclear Reactor Operators, completed written recertification examinations. There were no failures. This completes the periodic three-year recertification examinations for the operating personnel in the single-pass reactors.

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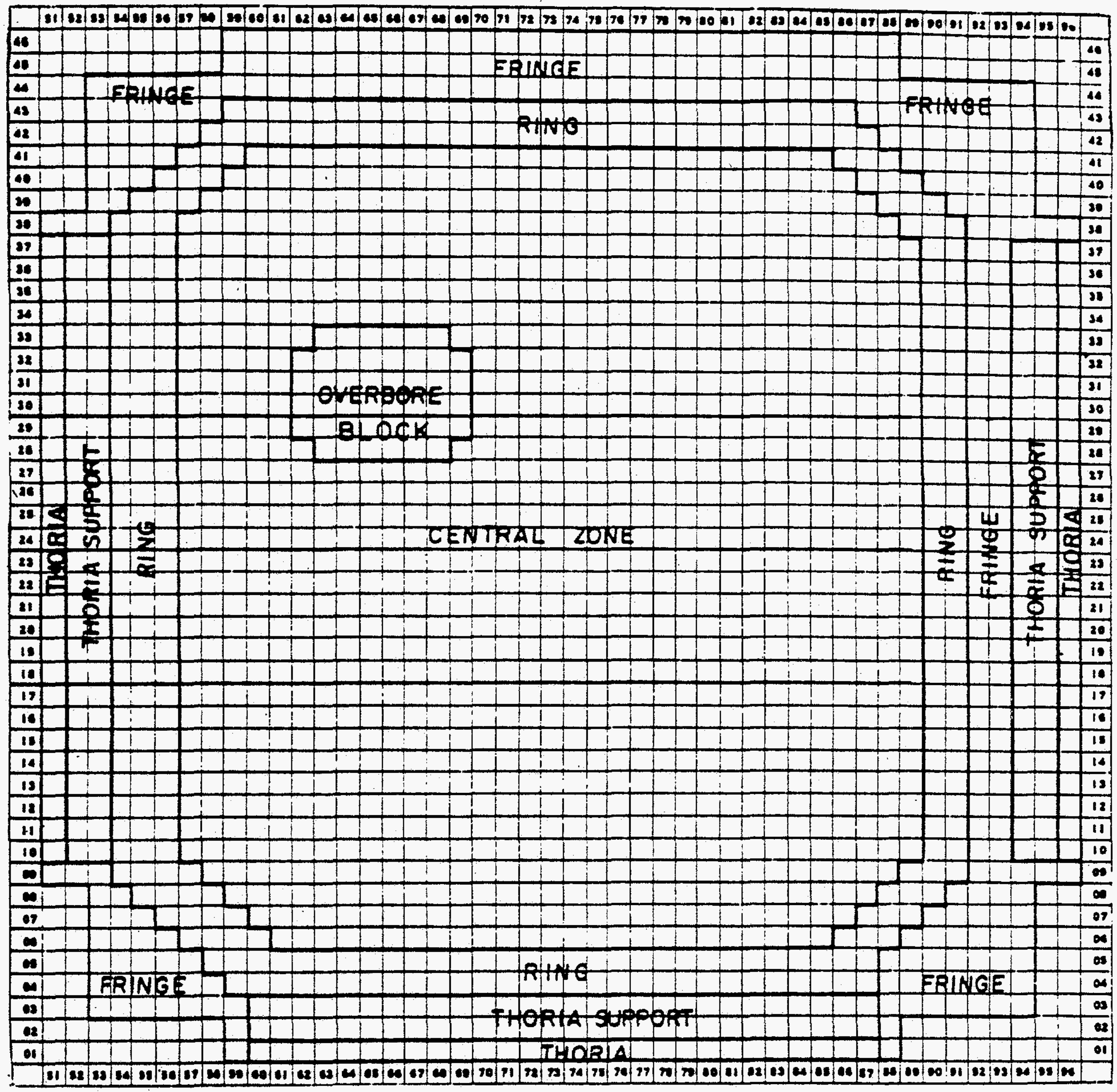

Zone	Tons	Material
Central	101	Natural Uranium
	40 (Includes 14 in overbore block)	94 Metal
Ring	12	Natural Uranium
	19	94 Metal
Fringe	25	Natural Uranium
Thoria Support	19	94 Metal
Thoria	4	Thoria

Loading Pattern - C Reactor

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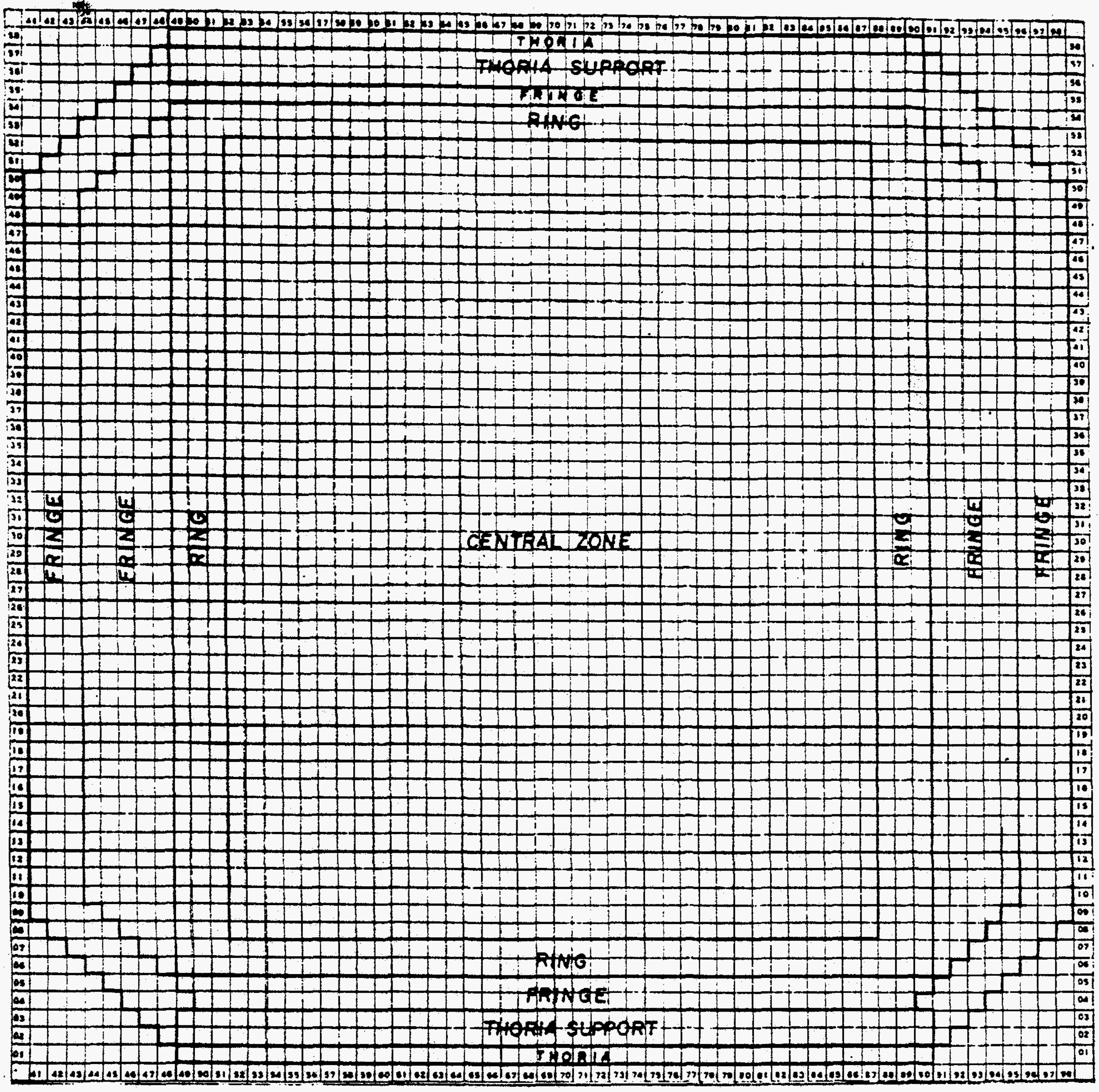

Zone	Tons	Material
Central	*45	Special Depleted Uranium (PITA-048)
	102	94 Metal (for depleted uranium support)
	84	Natural Uranium
Ring	32	94 Metal
	26	Natural Uranium
Fringe	133	Natural Uranium
Thoria Support	22	94 Metal
Thoria	4	Thoria

*The discharge of the depleted uranium load will not be shown until June since the reactor was down at month end.

Loading Pattern - KE

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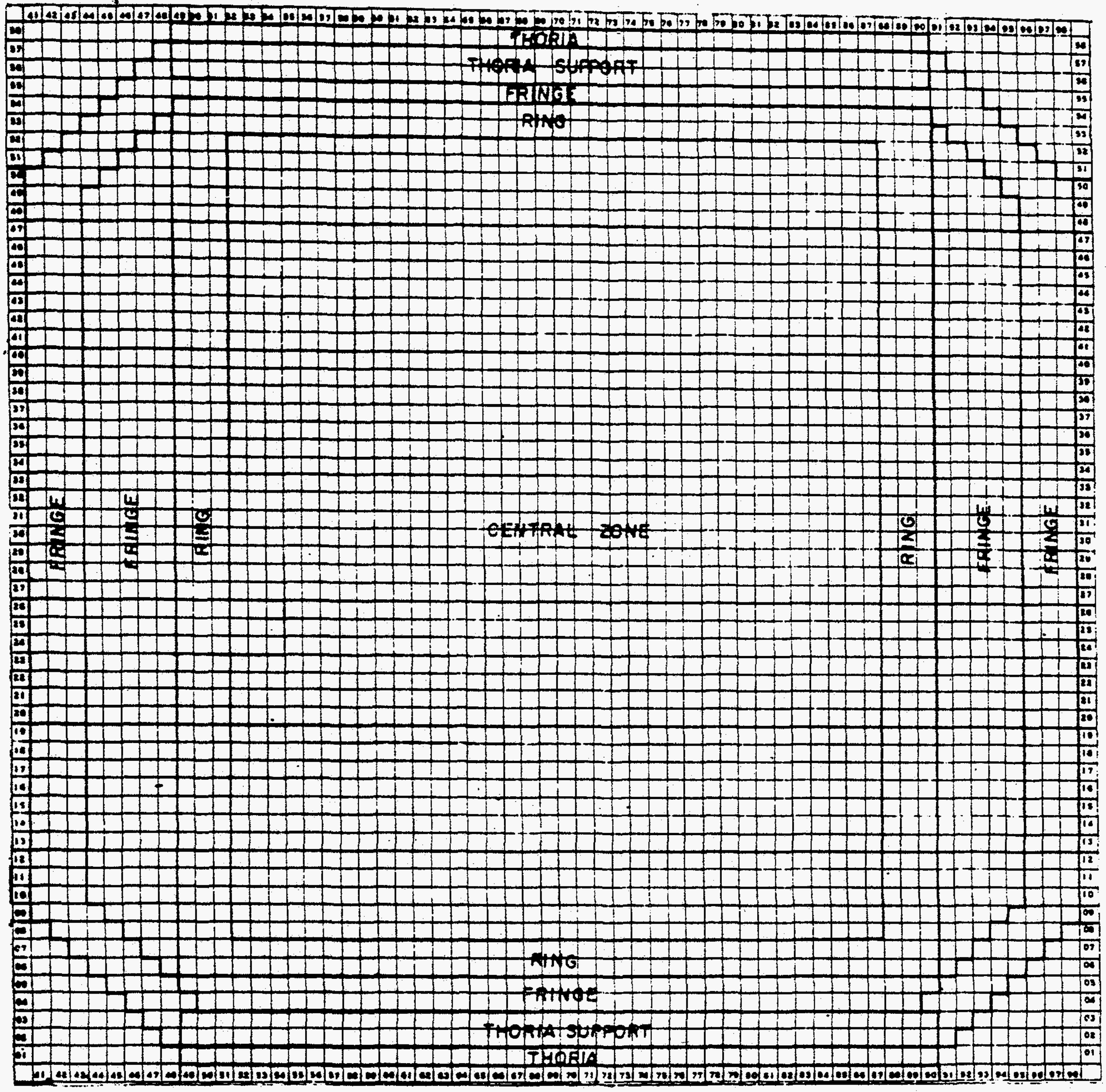

Zone	Tons	Material
Central	26	Special Depleted Uranium (PITA-048)
	114	94 Metal (for depleted uranium support)
	82	Natural Uranium
Ring	33	94 Metal
	25	Natural Uranium
Fringe	130	Natural Uranium
Thoria Support	22	94 Metal
Thoria	9	Thoria

Loading Pattern - KW Reactor

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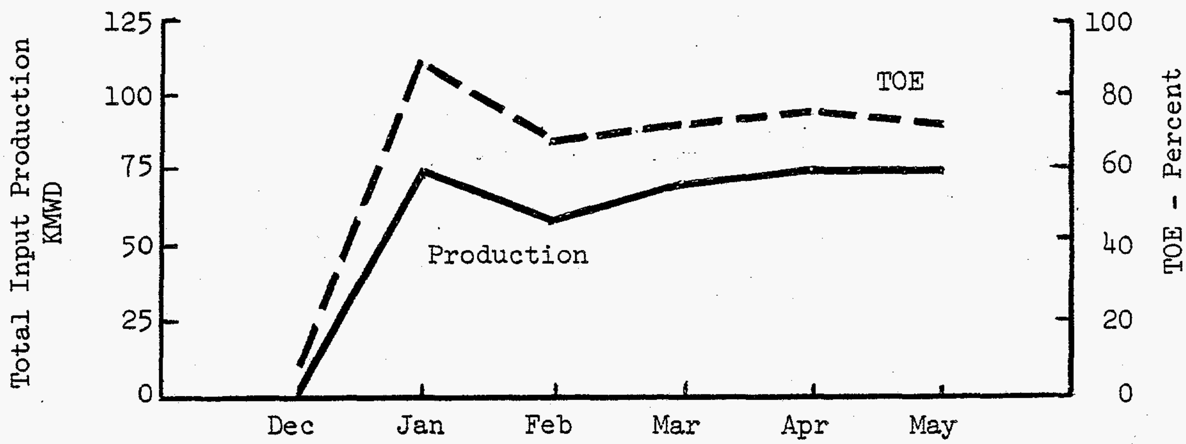
REACTOR PLANT OPERATIONS - N

PRODUCTION

General

Reactor production, power level, and related statistics are tabulated below. Input production and time operated efficiency (TOE) for the past six months are shown on the following chart:



Statistical Summary

Input Production (KMWD)	- Total	73.9
	- Coproduct	42.7
Power Level (MW)	- Maximum	3500
	- Average	3275
Time Operated Efficiency	- %	72.8
Steam Availability	- %	71.4
Number of Shutdowns	- Scheduled	1
	- Unscheduled	4
Fuel Failures		0
Fuel Charge (Tons)	- 94 Metal	171.9
	- 125 Metal	38.4
	- 210 Metal	93.2
	Total	303.5

[redacted]

Helium Losses - M cu. ft.	268.4
Fuel Oil Usage - bbl	8,327

OPERATING EXPERIENCE

Reactor Loading

The reactor loading at month end is detailed on the front face map which follows page BN-5. The footnote has been expanded to indicate the Nuclear Fuels Services (NFS) irradiation program. This program, which began in FY 1966 and is to extend through FY 1970, is for a total of 470 metric tons of nonrepresentative N Reactor fuel (the Pu-240 content of the plutonium being less than 7.8 percent or greater than 13.2 percent).

Power Level

Until the scheduled outage on May 16, reactor power level continued to be restricted administratively to 3500 MW due to Mark II fuel stress considerations. After that outage, the power level was further limited to 3460 MW because of nuclear safety considerations relating to the confinement system vent valves.

During an integrated confinement system test, which was performed during the scheduled outage, the vent valves were discovered to operate too slowly in tandem. Their slow operation is judged to be due to an inadequately sized oil return line on the valve actuating system. An equipment modification procedure has been written and material procurement has been initiated. When the material is available the modification to the oil line will be performed. It is hoped that the material can be obtained in time to permit making this modification during the June scheduled outage.

Reactor Outages

The five reactor outages and their principal causes were as follows:

Date Down	Outage Type	Outage Hours	Cause
May 14	Scram	1.6	A primary loop low-pressure trip occurred following a WPPSS double turbine trip-off (see below).
May 16	Scheduled	185.6	Charge-discharge and plant maintenance.
May 23	Unscheduled	12.2	Low flow indication on three process tubes and inspection of drive turbine #3.

[redacted]

Date Down	Outage Type	Outage Hours	Cause
May 25	Unscheduled	2.2	Fuel failure indication in process channel 1854. Indication went away after shutdown, and has not reappeared.
May 26	Unscheduled	1.0	Oil leak on primary pump #3.

WPPSS Load Rejection

The failure of a circuit breaker in the WPPSS switchyard resulted in a complete WPPSS load rejection at 5:00 a.m. on May 14. The resulting reactor coolant system transients caused a reactor scram about one minute after the load rejection.

In previous instances, the reactor has ridden through complete load rejection without scramming, although the margin from scram has been small. The underlying cause for the scram in this instance is not yet known. Possibilities include: dump condenser deflooding capability reduced due to one condensate line being out of service, insufficient base load on dump condensers (attempts have been made to maximize steam available to WPPSS), and maladjustment of the primary loop pressure control system.

EQUIPMENT EXPERIENCE

Cell 3 Restoration

Cell 3, which has been out of service since June 1967 for fire damage correction and steam generator retubing, was put into operation during reactor startup following the scheduled outage. At month end it was performing satisfactorily.

Cell 4 Deactivation

Cell 4 was removed from service when Cell 3 was put back on the line. Chemical decontamination was completed successfully, and retubing work on the steam generators will be started by Combustion Engineering early in June.

Resistance Temperature Detectors

The repair of only 13 RTDs during May is in accord with the plan to allow the backlog of RTD work to increase until the major outage in July.

Thirty-eight all-tube temperature monitor RTDs are inoperable on the central data logger system but readable on the auxiliary scanner, which is an increase of two over the previous month end. There are no unreadable RTDs, versus 11 last month end.

Thirteen zone temperature monitor RTDs are bypassed, an increase of one over the previous month.

Reactor Crate

Inspection of the right rear gas seal on the reactor showed no new cracks, and the RTV sealant over the buckled area was intact. No leakage was detected. Gas loss averaged more than 6,000 cubic feet per day during equilibrium operation.

Horizontal Control Rod System

The testing of fire-resistant hydraulic fluids continued. The water-glycol fluid test loop has been running on recirculation for over 450 hours. This test is scheduled to run for approximately 5000 hours to check for fluid degradation and system component deterioration. The 1705-N test control rod has been converted to a water-in-oil emulsion hydraulic fluid. Thirty-five hours of operation have been accrued comprising over 1800 rod scrams. The pumping unit has since been placed on recirculation for the remainder of the expected testing period of 5000 hours. There has been some difficulty in accurately determining the water content in samples of this fluid.

PROCESS ASSISTANCE AND CONTROL

Reactor Physics Support

Rod 103 was returned to service during the startup on April 28, although its flow indication is still not entirely normal. This increased the limit for total control with the rod system to 61.6 mk with the present loading. The ball system status includes two ball hoppers, 17 and 31, out of service and an effective strength of 53.9 mk. Both values are sufficiently high that the total control limits are satisfied with no impact on reactor operation.

On May 14, the scram recovery was accomplished in 1 hour 36 minutes from scram to first indication, and from first indication to 2000 MW in another 1-1/4 hours. Excess reactivity during the restart decreased almost to zero. The time available to accomplish a restart will continue to decrease as the Mark II fuel is phased out, until it will be impossible to restart after an equilibrium scram with the Mark I-C loading.

The Mark II fuel has operated failure-free since February 3. This presumably reflects (1) the success in selecting and discharging columns containing damaged fuel, and (2) the mild operating conditions maintained since the problem was identified.

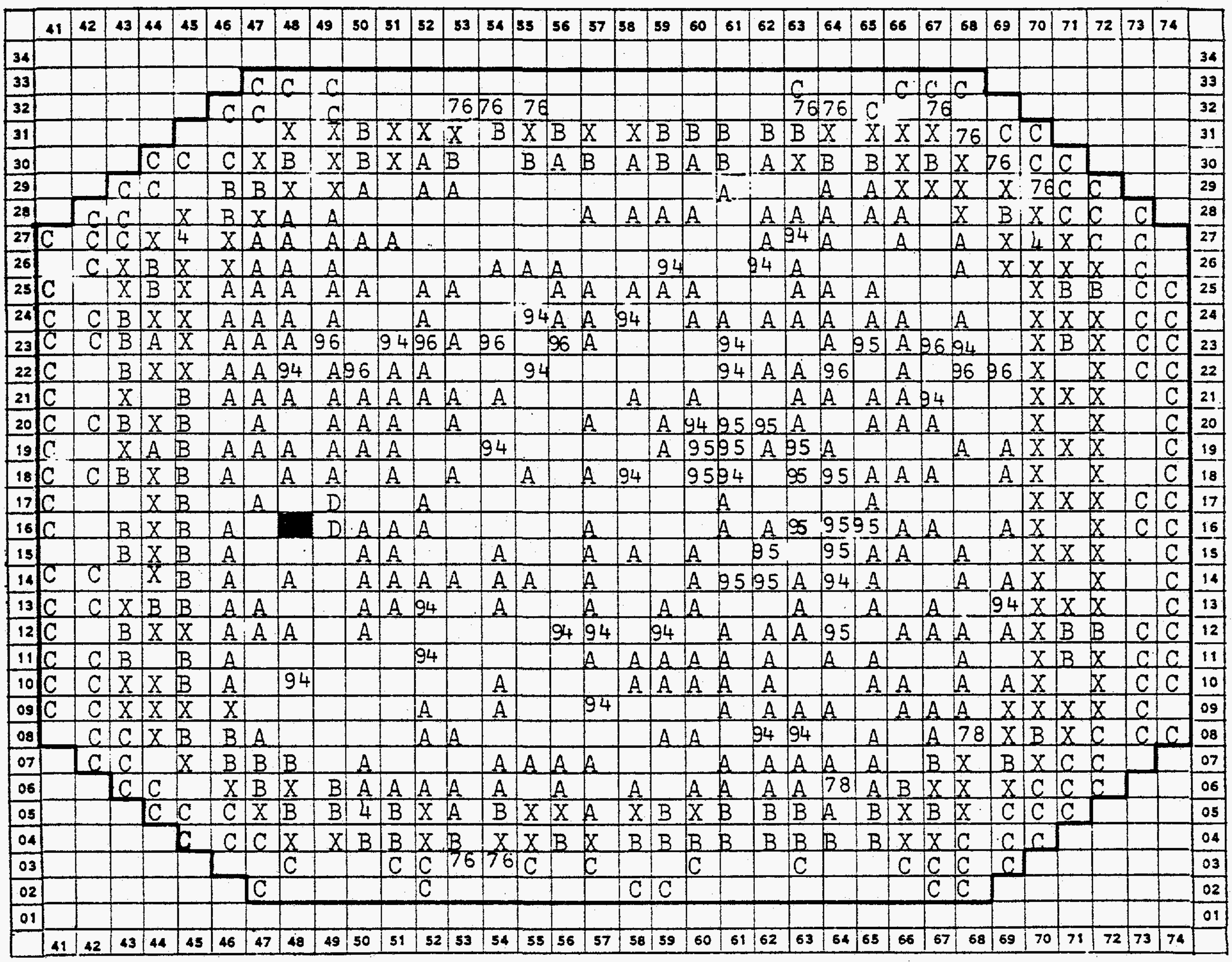
Target Element Performance

One or two coproduct targets failed during the month, as indicated by increases in tritium levels in the primary coolant. Peak tritium values of 1.7×10^{-2} µCi/ml and 9.2×10^{-3} µCi/ml were experienced on April 29 and May 11, respectively. Normal background is about 10^{-5} µCi/ml, and the level dropped to 5×10^{-4} µCi/ml between the two peaks.

A total of about 5600 curies of tritium has been released to the crib since December 1966. About half of this originated from the primary loop (failed targets in the reactor) and about half from the fuel storage basin (targets failing in basin from corrosion and mechanical damage) and continued leakage from failed targets. The rate of release of tritium, plus other radioactive materials, continues to be low enough that the limits of 10 CFR 20, which apply to the release of radioactivity from commerical facilities, are met at the point of discharge to the river.

Planning for August-November Operating Period

Reactor plans call for a 90 full-power-day operating period between August and November 15. Calculations of long-term reactivity transients confirm that adequate excess reactivity will be provided throughout this operating period by maintaining the current inventory of spike columns. Reactor behavior will be monitored during the remaining transition from Mark II to Mark I fuel to confirm this conclusion.



Fuel Code	No. Tubes	Description
(blank box)	334	MK-II Base (6% Pu-240)
A *	256	MK-IC (94 Metal Central)
B	84	MK-II Spike (6% Pu-240)
C **	128	MK-IC (94 Metal Fringe)
D	2	MK-II Poison (6% Pu-240)
X ***	130	MK-ICA (125 - 94 Metal Spike)
	934	Total
	70	Total PTs
	1004	Grand Total

* 50% NFS ** 100% NFS *** 94 Metal 25% NFS

PT-NR No.	No. Tubes	Description
4	3	Fuel Monitor Columns
76	11	Fuel for Meltdown Test
78	2	Increased Support Height
94	27	MK-IV Demonstration
95	17	High U-236 Content Fuel
96	9	MK-I From Upset-Forged Billets
(black box)	1	Blank Channel or Empty Tube
	70	

Loading Pattern - N Reactor

[redacted]

DECLASSIFIED

FUEL AND TARGET FABRICATION - C & K REACTORS

PRODUCTION

General

Production of AlSi-bonded fuel totaled 101 percent of forecast. Of the 214.3 tons produced, 198.1 tons (92.4 percent) were fuels with bumpers or self-support rails attached.

There was no production of hot-die-sized fuel. Personnel were moved to a fourth AlSi canning line May 13 to relieve a shortage of enriched fuel.

Thoria target element production was halted early in the month due to a shortage of oxide when two lots were determined to be out of specification. Production was resumed at a lower rate when some of the operators were moved to fill out the fourth canning line crew.

Acceptable Elements Produced

Type	Tons Input	Elements to Storage - Tons: Unrestricted Use	Upstream Use	Total	Yields - Percent: Current Month	FYTD
AlSi-Bonded, Natural U						
8" Regular	-	5.8	1.7	7.5	93.3	92.2
8" Bumper	50.3	-	-	-	-	92.9
8" Self-Support	-	6.2	0.6	6.8	96.6	91.5
AlSi-Bonded, 94 Metal						
6" Regular	-	7.7	1.0	8.7	94.8	97.4
6" Bumper	47.9	25.1	-	25.1	95.8	93.8
6" Self-Support	181.5	156.9	9.3	166.2	93.9	94.0
Thoria	28.0	-	-	33.0	95.8	-

Procurement and Inventories

Item	Tons Received	Tons Placed In Process	End-of-Month Stock: Tons	Mo's. Supply
Natural U Cores	-	50.3	678.0	9.8
94 Metal Cores	297.9	229.4	163.0	0.7
Thoria	17.0	28.0	16.1	0.4

[redacted]

OPERATING EXPERIENCE

AlSi-Bonded Fuel

Time operated efficiency was 99.4 percent. The downtime was charged about equally to operations and equipment. Canning line operation totaled 80 line-shifts, at the rate of three line-shifts for eight operating days and four line-shifts for 14 operating days.

The total manufacturing yield was 94.2 percent. Significant factors affecting yield were assembly, weld, and base-braze unbond. Of the total rejects in each category, the percent reclaimed or restricted to upstream use was: marred surface - 54 percent, closure weld - 91 percent, rail weld - 100 percent, AlSi slopover - 95 percent, and bond - 91 percent.

Thoria Target Elements

The total yield in thoria target element production, during the current campaign, has risen from 90.6 percent in February to 95.8 percent for May. Rejects have been reduced in all categories.

EQUIPMENT EXPERIENCE

The weld strength of the arch self-supports attached to fuel was determined to be less than desirable. Increasing the welding time from 0.3 to 0.6 seconds has yielded satisfactory weld strength, but has decreased throughput. This situation is believed to be equipment related, but this has not yet been confirmed.

PROCESS ASSISTANCE AND CONTROL

The fuels quality program is the subject of this month's appended Feature Report.

[redacted] DECLASSIFIED DUN-4018

FUEL AND TARGET FABRICATION - N REACTOR

PRODUCTION

Statistical Summary

Input (Billets Extruded)

Mark I, Inners, 94 Metal	201
Mark IV, Outers, 94 Metal	59
Inners, 94 Metal	35
I & E, 210 Metal	9
Total Billets	304
Tons - Total	48.0
- % of Forecast	141.2

Output (Finished Assemblies)

Mark I, Outers and Inners, 94 Metal	1,562
Mark I, Outers, 125 Metal Inners, 94 Metal	162
Total Assemblies	1,724
Tons - Total	34.6
- % of Forecast	101.8
Uranium Utilization - %	73.4

OPERATING EXPERIENCE

Extrusion Operations

The number of production extrusions exceeded forecast significantly, in anticipation of an extended outage of the press in July. Conversion to Mark IV fuel extrusions was effected on May 16.

Development work with the press included: one Zircaloy rod and three thin-walled Zircaloy tubing extrusions for Battelle Northwest, five uranium billet upset extrusions for C Reactor overbore fuel, one copper-silicon extrusion to produce billet stock, and one copper-silicon tube extrusion to test a butt shear technique.

[redacted]

Mark IC Conversion

Production on the Mark IC conversion program is exceeding forecast, and the conversion yield for May was 98.3 percent. All of the 300 Series 94 Metal assemblies in inventory with the exception of the twelve-inch material have been converted to Mark IC. The twelve-inch pieces will be completed in June.

White Oxide on Fuel

White oxide on fuel end closure welds and fuel supports has been observed on occasion in the past two months after autoclaving. Analysis of the manufacturing history of fuels showing this white oxide showed that they had been processed during the last few days of etch solution use. On one occasion the white oxide appeared on inner fuel in the top and bottom autoclave baskets, but was not present on outer fuel in the middle basket. It was found that the outer fuel had been etched just after a solution change, whereas the inner fuel had been etched just before solution change. A study is being made to redetermine the useful life of etch solution in relation to the number of pieces processed.

EQUIPMENT EXPERIENCE

Sleeve Flaring Expansion Tool

Conversion of Mark I copper components for use in assembling Mark IV outer billets was successful on 65 billets. Using pneumatic impact tooling, the inner sleeve is spin-flared to fit the taper of an end plate.

Mark IV Bond Tester

The first of the higher frequency bond testers, required for the thinner Mark IV clad, was received from BNW on schedule. To date, this tester has given excellent results, with only a few problems. A second tester is being built.

PROCESS ASSISTANCE AND CONTROL

Heat-Treat Fuel Temperature

Recent audits of the fuel temperature during heat treatment show that the cooling rate in air, during the transfer from the heat-treat bath to the quench bath, is higher for the audit fuel than for production fuel. The cooling rate difference could be a result of a higher emissivity of the audit fuel because of a thicker oxide on the clad. The heat-treat cycle transfer time for outers is based on a cooling rate of 2.5 C/sec. If the actual cooling rate is 1.7 C/sec, as recent results indicate, then the fuels are entering quench at about 700 C instead of the 685 ± 10 C specification. Investigation is continuing to determine whether the higher temperature is detrimental or an increase in transfer time is warranted.

Copper Evacuation Tube Embrittlement

Continuing investigation of copper evacuation tube embrittlement, as related to clad-to-core extrusion bonding problems, developed several causes and remedies. Embrittlement does not occur unless tensile stress is applied to the tube during the billet preheat cycle. Some tube failures have resulted from the use of graphite blocks which had off-center holes. When the blocks are wedged in the preheat can, an off-center hole (through which the evacuation tube protrudes) causes a bend in the tube. Embrittlement results from continual stressing of the tubes during the preheat cycle.

Embrittlement has been produced in types of copper tubing whereas copper-silicon tubing appears to be immune. Copper-silicon tubing would also be more readily welded to the copper-silicon end plate than is copper tubing. Further testing should determine whether copper-silicon could replace copper for these evacuation tubes.

TECHNICAL ACTIVITIES - C & K REACTORS

RESEARCH AND DEVELOPMENT

The progress on research and development work conducted by Douglas United Nuclear is reported by Mission number and title, here for the C and K Reactors, and for N Reactor beginning on page DN-1. Unless otherwise noted, the Missions are as defined in the July 1967 issue of this report. No coverage is accorded the two Hanford R&D Missions (6 and 9) in which DUN does not participate.

Mission 1 - Basic Production

1-A. Fuel Development

Self-Support Corrosion

A method devised for analyzing support corrosion involves shearing the supports from the fuel as cleanly as possible, and with a minimum deformation of the arch portion. The supports are then resheared to obtain a symmetric sample of the arch portion only. Preliminary testing has revealed that smaller standard deviation in weight can be obtained from unwelded supports after sample shearing than before. Tests are now being performed to ascertain the effect on sample weights for supports which have been removed from actual fuel elements, sample sheared, and then weighed. Sample weights will ultimately be converted to remaining support thickness. Actual measurements on the 50-mil Arch-Rail Support Test Fuel (PTA-105), which was discharged at a range of exposures up to 1700 MWD/T are scheduled to begin soon.

Tests were conducted with ribbed spacers to determine the actual slot spacing available for support rail row widths. Results indicated that support rail row widths of 300 mils, with angular tolerances of $91.5^{\circ}/88.5^{\circ}$, are acceptable on all current fuel models. Engineering Specifications have been revised to reflect these tolerances, and are being routed for approvals. A support study report including test details, results and analysis is being prepared.

Hot-Die-Sized Depleted Uranium Fuel Test

This Production Test, PTA-093, has been routed for approval. It authorizes continuation of the irradiation of 10 columns of hot-die-sized depleted uranium fuel in each of the K reactors to an exposure of 1800-2400 MWD/T. This test, which will take advantage of fuels charged in November and December, 1967, is designed to evaluate HDS depleted fuel performance at exposures higher than experienced to date.

Spire Corrosion

Study and tests are continuing on the K5E model which experienced the beehive corrosion in the spires late in 1967. Electron microprobe analysis is to be performed on an area near the spire-water surface of the spires where intergranular corrosion is prevalent. This will be done to determine the presence of higher than normal concentrations of nickel or iron, or possibly of other impurities such as copper. The evidence correlated to date suggests a higher than normal spire temperature conducive to the mechanism of beehive corrosion.

Occlusion Plating

Work continued on the occluding of Al_2O_3 and TiO_2 as a function of pH, using both direct current and pulsed current plating techniques.

The conditions used for the occlusion plating studies are as follows:

Bath - Watts bromide, 50 C
Current density=
Pulsed current - 1060 mA/cm^2, 500 μsec on and 1500 μsec off
Direct current - 360mA/cm^2
Agitation - rotating cathode at 60 rpm and dual air agitation at 35 ml/min.
Cell size - 140 ml
Particle size - about 2 microns
concentration particles: TiO_2 - 142 g/l; Al_2O_3 - 286 g/l

The results of the study (at a pH of 2, 3 and 4) show that the amount of material occluded in the deposit is independent of pH under direct current plating conditions for both Al_2O_3 and TiO_2. With pulsed current plating, the same effect was noted; the pH did not change the amount of Al_2O_3 or TiO_2 occluded.

In doing the above experiments, it was noticed that TiO_2 agglomerated, and these agglomerates were subsequently occluded in the nickel deposit. This phenomenon has not been noticed using the other materials. The diameter of the TiO_2 agglomerates ranged from 0.5 to 1 mil; they probably could be broken up using a deflocculating agent or a wetting agent in the bath.

1-B. Zircaloy Hydriding

The results of analyses of 14 Zircaloy process tubes removed from the K reactors after 55 to 59 months of exposure indicate that the use of anodized spacers has not stopped hydrogen entry into the base metal. These tubes had been operated for the past 12 to 18 months, some with and some without anodized spacers, specifically to make this evaluation.

At the time the test was started, coincident with the major shift to anodized spacers, it was believed that such a result would mean that the hydride case layers would have to be removed to assure tube life. However, it is now suspected that the base metal hydriding may be caused by pickup of molecular hydrogen dissolved in the water, because the hydrided surface is sensitive to hydrogen in laboratory tests with water at reactor temperatures. It is believed that the sandblasted surface is also sensitive and, therefore, there is some concern about a case layer removal program.

A summary of the present status of the hydride program from the BNW point of view is documented in BNWL-CC-1676, dated May 14, 1968.

Two tubes are to be removed from KE Reactor at the next opportunity to obtain further information on tube hydriding characteristics. In view of recent indications that the base metal is still picking up hydrogen, these tubes will be tested for physical properties before being cut up for hydrogen analyses. The data obtained will be used to develop estimates of the effects of base metal hydride buildup on tube life.

Laboratory tests were started to determine the effect of molecular hydrogen, dissolved in the aqueous phase, on total and base metal hydriding rates of Zircaloy samples as a function of surface pretreatment. The effect of hydrided and sandblasted surfaces is of particular interest.

1-C. Computational Techniques

HAMMER

The ETOG and FLANGE computer codes, which were developed by BNW to obtain the cross-section libraries for HAMMER from the ENDF/B library, are operational. Although a variety of flaws were discovered, ETOG now seems to be running correctly, and Pu-239, Pu-240, and Pu-241 have been successfully taken from the ENDF/B library. FLANGE has still to be tested before any extensive use of the program can be made.

EXTERMINATOR-II

This code has been placed on the FASTRAND drum where it will be available for all users. This eliminates the necessity of having to store the program on tape, and all related problems.

1-D. High Pu-240 Program

Depleted Uranium Analysis

Analysis of the depleted element recently discharged from one of the K reactors indicates that the measured plutonium isotopic content and the total plutonium yield match the calculated element exposure within five percent, thus indicating the validity of the production scheduling accountability curves. A document will be issued detailing these results.

Depleted Fuel Failures

The fuel failures in KW Reactor on May 19 and May 23 were K5D depleted uranium elements under irradiation as authorized by PITA-048. They represented the third and fourth failures in this program to date, and occurred in the 12th and 18th downstream positions, respectively. A visual examination of these failure elements indicated that both resulted from splitting of the uranium cores. The depleted loads have reached their goal exposure; the two columns containing the May failures had reached 2950 and 266Z MWD/T, respectively.

Mission 2 - Coproduct

Lithium-Aluminum Splines

PTA-136, which authorizes testing of lithium-aluminum splines, has been approved. It is planned to start this test at an early K reactor outage. The irradiation portion of the program will require about three months. However, a significant delay is expected before the tritium analysis from the irradiated splines can be made at Savannah River.

Mission 3 - Transplutonium Technology

3-A. Pu-Al Test Irradiation

Preparations are proceeding for the irradiation of 10 kg of plutonium. The capability for charging plutonium elements over splines is being developed and demonstrated. This capability is necessary to control tube powers over the first two to three months of the irradiation. No problems have arisen in the first tests with this concept. Also, locking devices are being procured for the rear caps. This is considered necessary to assure that the plutonium is not inadvertently discharged during the scheduled long irradiation period.

3-B. Plutonium Fuel Fabrication

Development work is continuing on core casting techniques and closure welding of Pu-Al target elements. Isostatic pressing of the cladding on the core to assure core-cladding contact is being evaluated. Dummy target elements with four rows of 3/4 inch, 50-mil supports were prepared for flow and chattering tests. The latter will be performed for at least a month of continuous operation to try to determine the long-term chattering potential for the low-mass Pu-Al elements. Brief tests were made with an eddy current instrument to sense the movement of the upstream dummy with and without a spline in the tube. The result was a high frequency, low amplitude vibration which is not expected to cause chattering damage.

3-C. Americium Irradiation

Design criteria and specifications were prepared for target elements to be used for long-time irradiation of americium. The elements are designed with a range of 9.6-10.4 grams/foot of fuel to 19.2-20.8 grams/foot of fuel. Assuming a maximum specific power of 20 Kw/ft., the elements are calculated to have a maximum core temperature of less than 200 F.

3-D. Curium Irradiation

Target element criteria, schedules, and costs are being established in the preparation and irradiation of three test targets containing Cm-244. Estimates of target activity are being prepared since the long irradiation time (2-4 years) may necessitate discharge and recharging of the targets.

Mission 4 - Plutonium -238

Except for the determination of Pu-236 content, data collection and analysis of the recently completed neptunium irradiation are finished and the results are being documented. Recent measurements in ARHCO laboratories, as well as re-evaluation of the earlier BNW measurements, have indicated that the original Pu-236 measurements were high. These high values were possibly due to curium contamination. In any event, a complete re-measuring and re-evaluation program is in progress.

With this favorable change from the earlier conclusions in regard to Pu-236 contamination, the effect of the graphite diluent on reducing Pu-236 concentration even further is being re-examined. The final design of the planned small irradiation test to further minimize the Pu-236 concentration is awaiting the results of these new measurements.

Mission 5 - Other Isotopes

Uranium-233

Engineering Specifications covering uranium contamination of incoming thoria were revised to optimize U-233 product purity in the current thoria irradiation program. An outgrowth of thoria dissolution studies sponsored by ARHCO is a strong indication that thoria wafers may reduce uranium cross-contamination by reducing dissolution time. Thoria wafer specifications and quantities were established to permit evaluation of the irradiation behavior of target elements containing thoria wafers.

Radiometallurgical examination of irradiated thoria wafers indicated that the wafers were cracked in relatively large segments. Only a small amount of thoria powder was found in the targets, which is thought to be caused in part by handling and cutting the targets for wafer examination. Samples of irradiated thoria wafers have been made available for dissolution studies sponsored by ARHCO.

Mission 7 - Target Space Enhancement

7-A. High Power Density Fuel

150 Percent Power Demonstration Test PTA-067

The fuel element failure at KW Reactor on May 15 was in tube 4365 containing 125 Metal being irradiated under this test. Examination of the discharged fuel revealed that the element in position 12 had sustained an apparent inner cladding failure characterized by projections of uncertain origin extending into the hole region of the element. These projections appeared similar in nature to what has been termed as "beehives" seen in the hole regions of K5E elements.

Examination of elements from the downstream half of the failure charge revealed three other elements which also showed apparent beehives in the hole region. The outside jacket on all elements appeared to be in good condition, with only mild localized corrosion.

Corrosion work on aluminum in high temperature process water has resulted in a similar phenomenon, but the temperatures involved in these experiments were considerably above that expected for fuel element surface temperatures in the K reactors. Work is underway to characterize this new phenomenon, and to investigate the effect of inclusions in the component materials which might initiate or catalyze the reaction.

Due to the failed element, and the virtual achievement of the desired exposure (1965 MWD/T for the failure column), the remaining 125 Metal columns were discharged, thus terminating PTA-067. Preliminary results of the examination of six weighed and measured columns of 125 Metal discharged on March 26 at about 1300 MWD/T exposure indicate relatively mild corrosion with localized corrosion. Weight loss was less than 6 grams maximum. Diametral changes were moderate, with an average +6 mils on the OD and -4 mils on the ID. Length changes were greater than expected, abcut 30 mils. To date, no explanation for this has been uncovered.

Zircaloy-Clad Fuel

It is planned to charge 21 columns of aluminum-clad 170 Metal and 11 columns of Zircaloy-clad 150 Metal columns. Physics calculations are being made to determine the operating characteristics of these columns when used as thoria support. Modifications are being made to current K reactor venturi inserts to accommodate the higher coolant flows of these columns.

The K15Z Zircaloy-clad fuel for the planned 150 Percent Tube Power Test is undergoing final inspection. Tentative yield data indicate that one column of the 301 alloy and ten columns of the 801 alloy will be available for testing. Spare elements of each type will also be available.

A waiver of warp specification is being initiated to permit double-throw warp up to 12 mils. The original limit was established at 8 mils based on projected process capability; however, the short K15Z elements preclude the use of straightening procedures to meet the original specification. Since there is no basis for expecting problems from 12 mils of warp, the waiver is being prepared to improve process yield.

200 Percent Power Demonstration Test

The K21Z Zircaloy-clad fuel has been extruded for the 200 Percent Tube Power Test. The waiver mentioned above will also include a relaxation from 6 to 10 mils of double-throw warp for these fuel elements. Yields have not as yet been established, but no problem is anticipated in realizing the necessary five columns. One column is expected to be of the 301 alloy with the remainder being composed of the 801 alloy.

Alloy 801 has been specified for the AlSi portion of the 200 Percent Tube

Power Tests. The use of this alloy was based on the performance of this material in the 150 percent specific power core alloy test, and reconfirmed by data from six columns of the 801 alloy in the second segment of the 150 percent specific power demonstration discharged at 1300 MWD/T. The data were compatible with and confirmed performance from the 150 percent specific power core alloy test for the 801 alloy. No cracked cores were detected by ultrasonic testing of the six columns.

Fuel Support Corrosion

Calculated data on the increased pressure drop characteristics of the 50-mil thick, 280-mil wide arch-support versus the conditions for standard 32-mil thick, 235-mil wide arch self-supports has been compiled for the K6 (210 Metal) and the K7 (170 Metal) fuel geometries. Since support corrosion appears to be closely related to bulk coolant temperatures, the heavy supports will be used only on the downstream 10 fuel elements of the small scale E-N Demonstration and the 150 Percent Tube Power Test, which will utilize these respective fuel types.

Bulk coolant temperature considerations indicate that at least equivalent protection is afforded by a 32-mil support in the downstream position. Thus, while providing the maximum support corrosion protection possible, the effects of increased coolant temperatures on cladding corrosion are minimized. Work is proceeding to procure 50-mil thick, 235-mil wide arch supports. Data indicate that flow decreases associated with these narrower supports could be tolerated on a full column basis, which will eliminate the problems of working with two support types on a single fuel model.

Alternate Aluminum Alloys

Corrosion tests being conducted on candidate aluminum cladding alloys include a dual alloy cladding concept consisting of an internal layer of high silicon alloy with an external layer of 8001 alloy. Corrosion test specimens have been fabricated from this material. Specimens with the AlSi internally isolated were made by welding together, around the periphery, two of the duplex alloy specimens. Defects, both into and through the X8001 cladding, were then introduced into some of the test specimens. These specimens, along with control specimens, will be subjected to 360 C in a low-flow autoclave, and 130 C at 25 ft./sec. in a flow loop, for periods of 30 and 60 days.

Alloys exposed in the flow loop for 30 days show no excessive visually apparent corrosion. Detailed examination of these alloys is proceeding. Weight loss data on the 10-day autoclave specimens indicates no corrosion penetration greater than 1.5 mils. In general, the high silicon alloys showed higher corrosion rates. Also, the alloys with AlSi as a base showed higher corrosion rates than the comparable alloys made with a high-purity aluminum base.

A study was initiated on nondestructive tester capacity to test the candidate uranium alloys for grain size characteristics. The Nondestructive

Testing Section of BNW developed a program of evaluation to include the preparation of fuels standards and the calibration of test equipment.

7-B. Reactor Modernization

C Reactor Test Facility

The status of PTA-103, One-Inch Overbore Facility in C Reactor remains unchanged. Average column exposure at month end was about 550 MWD/T.

Preliminary hydraulic designs have been completed for a coproduct element using a lithium-aluminate target rod with a driver fuel tube of either 210 Metal or a fully enriched uranium-aluminum alloy. Designs were prepared for three rod sizes. These designs will be given further physics study in planning for a test coproduct loading in the C Test Facility.

Graphite Channel Preparation for K Reactor Overbore

The characteristics of K reactor stacks will require modification of the tooling and techniques for overboring which were used for the C test block. It is possible physically to remove sufficient graphite from the tube blocks with present tooling for the tube to accommodate existing distoriton without flattening. However, graphite expansion in the center of the stack will lead to tube flattening in regions of the channel distortion; i.e., bends will be generated between blocks which now contain no bends.

Numerous methods are being considered for K overboring to obtain a satisfactory tube life, including: excessive graphite removal from the tube blocks; oval-shaped holes in selected regions of the channel; use of reinforcing ceramic or high strength graphite sleeves around the process tubes to prevent tube flattening and limit block bending; and thick-wall Zircaloy tubes which may resist flattening and help support the stack.

7-C. Highly Enriched Fuel

PCTR Measurements

Measurements of the reactivity parameters of the oralloy fueled mixed lattices is scheduled to start in the PCTR about June 1. The J metal (7 percent oralloy in aluminum) is now being transferred to BNW.

Ex-Reactor Criticality Control

Results of critical mass calculations (HAMMER) for oralloy fuel rods, as a function of rod diameter and enrichment for clad fuel rods, are presented in the following table:

Wt% Oralloy in Aluminum	Maximum Buckling (μb)	Minimum Critical Spherical Mass (kg oralloy)	Minimum Critical Mass Inf. Cyl. (kg oralloy/meter)	Minimum Critical Mass Slab (kg oralloy/meter^2)
		For 1.00" Diameter Rod		
5	9,404.0	2.80	4.04	7.28
10	12,270.0	3.15	5.17	10.56
15	13,830.0	3.69	6.44	13.95
		For 1.35" Diameter Rod		
5	9,253.0	3.27	4.69	8.42
10	11,670.0	3.98	6.43	12.95
15	12,810.0	4.90	8.28	17.42
		For 1.50" Diameter Rod		
5	8,985.0	3.54	5.00	8.89
10	-	-	-	-
15	12,500.0	5.44	9.10	19.01

The results presented in the table correspond to the minimum critical mass for that particular rod enrichment and geometry with optimum moderation, and the critical values reported contain no significant safety factors. Calculations for the corresponding bare rods have also been performed and are presently being tabulated.

Before these results can be correlated with experimental measurements and reduced to their proper safe values, some discrepancies must be accounted for in the manner in which HAMMER handles leakage calculations.

Fuel Design

A uranium-aluminum fuel design (I&E) was developed for use in a K reactor test to define graphite heating and physics parameters. The oralloy content of the fuel is being developed from physics considerations to achieve tube powers of 2250 kw.

Mission 8 - Nuclear Safety

Nothing significant to report.

Mission 10 - Columbia River Studies

10-A. Deionized Water Studies

Operation of two single-pass tubes, charged with Zircaloy-clad fuel instead of aluminum, and cooled with deionized water, resulted in reduced concentrations of effluent radioisotopes. These reductions ranged from 60 to 90 percent of the concentrations obtained during operation with aluminum-clad fuel, and verified results obtained previously during similar operational sequences.

These results are significant because the additional Zircaloy surface did not provide similar reductions when process water was used to cool the tubes. This re-emphasizes the importance of the impurities introduced with process water in providing both the primary source of the parent isotope and the material from which the in-reactor film is derived.

10-B. Water Treatment Variables

"Cat Floc," a positively charged polyelectrolyte, has been used as the coagulant in the Water Treatment Pilot Plant for one full fuel cycle. The substitution of this material for aluminum sulfate flocculant did not result in any change in the effluent concentrations of Sc-46, Cr-51, Co-63, Zn-65, La-140, and Np-239 from either the aluminum or the Zircaloy tubes. The effluent concentrations of P-32 and As-76 were increased 3.5 and 4.1 times, respectively.

This loss of effectiveness is likely the result of removal of less of the parent material from the raw water. The two parents of interest are anionic, and form insoluble compounds with aluminum and/or are removed by association with hydrous aluminum oxide in the standard treatment process. The corresponding association with "Cat Floc" probably does not form such an insoluble compound and, therefore, arsenate and phosphate retention may be reduced by mass action effects of other more abundant anions such as sulfate. Conversely, if a strong bond is formed, removal is governed by the performance of the filters. A very short run, with bentonite added to the raw water to provide a floc builder, resulted in a decrease in filter effluent turbidity. This study will be continued during the next operating cycle to determine the resultant effect on reactor effluent radioisotope concentrations.

10-C. Water Filtration Studies

Studies were conducted to evaluate the effect of rapid mix intensity on system performance of the water treatment pilot plant. Tests were run at mixer propeller speeds from 170 to 500 rpm. Results obtained are presented in the following table:

Mixer Speed (rpm)	Filtration Time (min.)	Effluent Turbidity (JTU)	Total Headloss (ft. water)
170	300	5.5	3.5
340	300	5.0	3.7
500	270	2.8	7.1

More mixer speeds are being evaluated in an attempt to determine the mixing intensity required for best system performance.

ENGINEERING AND TECHNOLOGY - REACTORS

Economic Studies

Segmental Charge-Discharge at K Reactors

The K reactor study compared the effect of charge-discharge rate on the economics of operating with segmental charge-discharge and weapons-grade plutonium production. Each of the charge-discharge schemes studied, in which the refueling rate for natural fuel elements was considered equal to current rates, showed an economic incentive for the segmental discharge technique. The incentive varied from $260,000 to $275,000 a year. Consideration of the green-end fuel program would reduce this incentive by about $45,000.

The economic break-even point occurs when the charge-discharge rates are 80 percent of current rates. Thus, the application of this technique to reactor operation is dependent upon developing refueling procedures which are nearly as efficient as current practices.

C Reactor Tube and Fuel Redesign

Determination of the economic incentives for converting to a larger diameter fuel element in a correspondingly larger inside diameter process tube in C Reactor has been completed. The results indicate that the conversion ratio gain from the larger fuel element is offset by losses due to the shorter life of the thinner tube.

Use of B Reactor Fuel in C Reactor

Operating data from the mixed columns of B and C fuel currently under irradiation in C Reactor continues to confirm calculations and laboratory data. Currently, mixed B and C fuel columns are being charged on a routine basis. The first columns of mixed fuel were discharged (at weapons-grade exposure) during the May 21 scheduled outage. Supplement A to PTA-127, authorizing irradiation of full length columns of B Reactor natural and enriched fuel in the fringe (outer three lattice positions) of C Reactor, has been written and is routing for approval.

Confinement Studies

In the past month a review of the 70-Point Criteria for Power Reactors has been in progress to identify criteria requirements not satisfied with the existing low-pressure reactor systems.

A sample of front riser piping from KE Reactor was subjected to tensile tests and Charpy V notch impact tests. Results were as follows:

Yield strength	29,885 psi
Ultimate tensile strength	49,635 psi
Elongation	36 percent
Reduction of area	66.6 percent
Charpy impact strength @ 33 F (averages of 2 sets of 3 samples each set)	6.7 & 6.5 ft. lbs.

The impact values are substantially lower than desirable for service at the indicated temperatures, and a program for further evaluation and surveillance was initiated.

Rear Face Hardware - K Reactors

Replacement of rear hardware on the K reactors as a long-term solution to the rear face leak problems is being investigated. Initial findings indicate a rather wide-spread need for replacement of connectors and adapters. In view of potential tube channel enlargement of the central zones, new rear nozzles of a new design for an interim period do not appear justified.

Although the stock of original rear nozzles has been depleted, cast aluminum inlet nozzles removed during the Zircaloy retubing program are being modified for rear face use. Leak repairs may require a rather large stock of nozzles because of the high probability of damage occurring when trying to remove nozzle adapters, or the nozzles themselves, because of salt deposits on their fastenings.

Personnel Radiation Dosimeters

A prototype design for a pocket dose rate alarm device has been completed. Work is underway to integrate all circuit components into a packaged configuration which causes no interaction between the components, and which also meets space limitations. Discussions are underway with a mold designer relative to the pocket dose rate alarm case. Purchase requisitions have been issued for most of the required components to fabricate approximately 40 prototype units for on-reactor testing.

Operational Discharging of Np Target Elements

Studies for the redesign of the existing Operational Fueling Facility flapper cap actuator, to adapt it for use in the operational charge-discharge of neptunium target elements, is approximately 40 percent complete. The objective is to reduce the number of components and to simplify the actuator controls. A schematic sketch of the revised water and air system has been worked out. One of the innovations included is the interlocking of the control functions at the actuator instead of through remote relays as was done on the original machine.

Charcoal Filter Testing

Tests are in progress at the 189-D filter test facility to develop a method to determine, in place, the iodine retention efficiencies of the activated charcoal filter banks in the reactor confinement system. Two methods are being developed simultaneously, neutron activation and use of the modified oxidant detector.

Test procedures and other details are in preparation for the in-place filter bank tests. Both methods, neutron activation and modified oxidant analyzers, will be used. Present plans call for introduction of the iodine vapor into the exhaust airstream at the 119 Sample Building. Samplers will be placed so that the iodine retention efficiencies can be obtained across the particulate filter bank, the charcoal filter bank, and the over-all filter system.

Two charcoal filter units removed from the C Reactor filter cell following their water dousing incident were tested in the 189-D facility using the modified oxidant analyzers. The iodine retention efficiencies were 87.1 and 82.6 percent. Visual inspection of the filters showed voids in the charcoal beds. It is planned to repack the beds by adding additional charcoal to determine if the efficiencies can be increased.

Project Engineering - Reactor Facilities

Project Status Summary

The status of approved construction projects relating to B, C and K reactor facilities is summarized in Appendix A.

ENGINEERING AND TECHNOLOGY - FUELS AND TARGETS

Aluminum-Uranium Fuel Development

Process development work related to fabrication of uranium-aluminum cores is proceeding. Six billets are ready for extruding into I&E rod size for fabrication into 11-inch fuel cores. These cores will be utilized to establish process parameters for diffusion bonding by the hot-die-size process and braze bonding by the AlSi process.

A second approach to fabrication of aluminum-uranium alloy fuel is direct coating of the core, and cladding by the hot-die-size process. Several cores have been cast, machined, and clad using both nickel plated and bare cores. Dimensional and metallurgical data for the finished process are being gathered.

TECHNICAL ACTIVITIES - N REACTOR

RESEARCH AND DEVELOPMENT

Mission 1 - Basic Production

1-A. Mark IV Fuel Development

Five columns of Mark IV Production Test fuel (PT-NR-94) were discharged from the reactor during the May outage for observations and dimensional measurements. This left 27 columns of Mark IV test fuel in the reactor, as follows:

Enriched Metal	Clad Thickness	Alloy	No. of Columns	Predicted Average MWD/T by June Outage
94	Thick	601	5	900
94	Base	601	4	900
94	Base	503	10	1100
94	Base	601	2(Modified Design)	1300
125	Base	601	6	1250

A total of eight Mark IV test columns have been discharged to date, as follows:

Discharge Date	Tube No.	No. Pieces per Column	Enriched Metal	Alloy	Clad Thickness	Exposure MWD/T
3-3-68	1946	12	125	601	Base	560
4-1-68	0853	12	125	601	Base	770
	1646	12	125	601	Base	820
5-16-68	0852	14	94	601	Thick	545
	1065	14	94	503	Base	660
	1558	11	125	601	Base	850
	2158	11	125	601	Base	845
	2654	14	94	601	Base	520

Inspection of the fuel from tubes 0853, 1646 and 1946 in the reactor basin has revealed no signs of malperformance. Assemblies from positions 3 and 4 from the downstream end of tubes 1646 and 1946 have been sent to BNW (Radiometallurgy) for detailed examination; half of these have been inspected to date. No evidence of accelerated corrosion was found at the point of contact of the solid supports and the cladding; however, due to the mild operating conditions imposed to control the Mark II failure mechanism, definite conclusions about Mark IV performance are not yet possible.

Approximately 20 columns of Mark IV fuel will be charged in the reactor during the July outage under the authorization of PT-NR-94. The initial production quantities will utilize Yankee Reactor tails which contain a greater amount of U-236 than standard 500 series recycle uranium. Mark IV fuel with high U-236 content will be designated as 986 series for inner and 987 series for outer elements.

1-B. Flow Loop Testing

Life testing of two columns of fuel assemblies is being performed at Bldg. 189-D in the NPR PCE loop. The testing is being conducted by alternating each column in and out of the loop on approximately 10-day cycles. The types of fuel and support hardware being evaluated are as follows:

Fuel Type	Inner Support	No. of Fuels	Time in Loop (hr)
Mark I-C, 12-inch	Buggy spring	6	1600
Mark IV, 7.7-inch	W-spring	6	1025
Mark IV, 23.2-inch	W-spring with fully rounded solids	5	25
Mark IV, 26.1-inch	End spider	16	350
SDT, 23.2-inch	W-spring	9	1600

The Mark I-C, 12-inch fuel with "over-hardsized" buggy springs have not shown any signs of fretting. The fuel assemblies with W-springs and solid supports also have shown no signs of malperformance.

The end-spider supports have received a total of 7.5×10^7 cycles (60 cps) and appear to be free of any fatigue cracks as observed with the unaided eye.

1-C. Billet Fabrication Studies

Ten columns of Mark I-C fuel assemblies with outer elements fabricated from upset forged billets have been charged into the reactor under PT-NR-96. The fuel will be irradiated to at least 2100 MWD/T.

Fuels fabricated from direct cast outer billets and upset forged inner billets have been manufactured and will be charged in June upon approval of the Production Test (PT-NR-101). A maximum of 12 columns will be irradiated to goal exposure or more depending on the fuel performance.

Of the 20 columns of Mark IV fuel to be charged in July, seven columns will have outer elements fabricated from sleeved Mark I outer billets. These seven columns will essentially be a pilot run for the remaining 800 Mark I outer billets in inventory that will be sleeved in order to utilize them in fabricating Mark IV outer fuel elements. The technique of sleeving uranium billets was successfully used in fabricating the large number of 125 metal coproduct test fuels which were irradiated under PT-NR-8 up to exposures of 2500 MWD/T.

I-D. Mark I-C Reliability Assessment

The coproduct demonstration will be completed during July, and N Reactor will be fully loaded with the renovated Mark I fuel named Mark I-C. Study is under way to assess the reliability of Mark I-C fuel with respect to failure causes, mechanisms and rates.

Two areas of Mark I-C reliability have received major emphasis: (1) the fuel failure mechanism whereby buggy spring supports on an inner fuel element penetrate the inner cladding of the outer fuel element by a fretting-corrosion mechanism; and (2) the end-associated fuel failure mechanism. New findings in these areas are described in the following paragraphs.

The reactor operating history of the fuel columns that have contained 12-inch long fuel elements has been assembled to show coolant flow in pounds per hour. Variability in the flow was caused by either orificing or by the number of steam generator cells on the line.

This flow variability, and the fuel failure experience, are shown in the following table:

	Always Orificed		Partly Orificed		Never Orificed	
	Total Columns	No. of Failures	Total Columns	No. of Failures	Total Columns	No. of Failures
5 Cells	168	0	57	1*	54	0
5 & 6 Cells	90	0	40	1*	135	4
6 Cells	72	0	0	0	12	4

*Shown to have defective buggy springs; other failed elements did not have buggy springs that were cracked or broken.

Analysis of the performance of the 1391 12-inch long fuel elements irradiated in two types of column loadings (P-12 and Spike), and distributed as in Figure 1 confirms that the buggy spring fretting failure rate was dependent upon coolant flow rate. It is concluded that the low flow rates in orificed columns do not cause failures, whereas high flow rates in non-orificed columns do. Similarly, the high flow rates achieved during 6-cell operation cause failures whereas the low flow rates with 5 cells do not.

The effort on end-associated failures has been in two separate areas. In the first of these, the Reactor Information File has been interrogated to yield operating history. The raw data extracted will be used to fit a failure rate equation based upon tube power and operating days. In the second areas of study, good quality end closures from fuel elements that were rejected for other than end-associated defects are being examined microscopically.

The end welds on 28 inner fuel end caps have been progressively inspected in (1) the as-autoclaved condition, (2) the as-vapor blasted condition, and

(3) lightly etched in a nitric plus hydrofluoric acid solution. Many different types of defects were found. Some only detracted from the physical appearance of the welds, and cannot be considered deleterious to in-reactor performance, whereas others are considered potentially deleterious to in-reactor performance. Defects of the latter type are either those whose shape or constitution could lead to the entrapment of pickling acids, or those that are deep enough to be considered significant penetrations already.

The most serious type of defect found is one that has been tentatively identified as a "hot tear." Two end closures showing this type of defect have been metallographically prepared and straight-sided, oxide-lined cracks can be confirmed. These defects resemble those found in the braze layer of the closures that failed during irradiation.

1-E. Zircaloy Process Tube Evaluations

Stress-rupture testing continued on sections of irradiated process tube 1756 at hoop stresses of 59,000 and 57,000 psi loads and a temperature of 300 C. No failures have resulted after over 9000 hours of this testing.

1-F. Coated Graphite Studies

Another sample of silicon carbide coated TSX graphite (prepared by the Marquardt Corporation) is undergoing oxidation tests at four temperatures in the range 800 to 950 C in flowing helium containing 6000 ppm of water vapor. Preliminary analysis of the raw data indicates that greater oxidation protection is afforded by the coating at the higher temperatures. The ratio, oxidation rate of uncoated sample/oxidation rate of coated sample, increases with increasing exposure temperature.

1-G. DCODE

Further investigation of computer code accuracy continues even though application to actual study cases has started. Two significant features are under investigation:

- Adjusting the effective resonance heights of resonance absorbers (U-236 and U-238) to more accurately match experimental yield determination.

- Recomputing neutron streaming in lattice voids to more closely match startup and PCTR measurements.

These changes are being studied as a result of an effort to determine the required fuel enrichment adjustment as a function of U-236 feed concentration. Results of that study (Mission 4 work clearly suggest that DCODE (and FLEX) requires adjustment of the U-236 resonance self-shielding approximation. The MOFDA code results appear more reasonable, although identification of accuracy suffers from a lack of experimental data.

1-H. Reactivity Effect of Fuel Load Displacement

The change in shutdown margin caused by a displacement of the reactor's fuel charge from the core centerline was calculated using the EXTERMINATOR-II code. Results indicate the shutdown margin could decrease by 12.2 mk until the fuel charge approached the shields. The shutdown margin then increases, as shown in Figure DN-1, because the fuel reactivity decreases faster than the rod strength.

Mission 2 - Coproduct

Breached Target and their Detection

Coproduct target element stripping, shipping and extraction remain an area of concern. The stripping process occasionally breaches the aluminum cladding of a target element, allowing the tritium to escape and water to enter. Targets are breached during this operation by either cutting too deep around the end cap, so that the end cap is partially or entirely separated from the target, or by too deep a cut in the longitudinal direction.

The problem lies not with the tritium release, although undesirable, but with the water intake. Breached targets have the effect of diluting the tritium content of the targets with hydrogen, thereby reducing the efficiency of the tritium extraction process. A breached target subassembly contains about 20 grams of water which represent as many molecules of hydrogen as there are molecules of tritium produced in 90 subassemblies. Clearly, the dilution from just one breached target is quite significant.

The most promising methods of detecting breached targets are:

- Monitoring for tritium release during the stripping operation. At present, the gas release from a breached subassembly is unlikely to cause a tritium alarm as there is not a monitor between the target stripping area and the nearest exhaust port. The placement of a tritium detection system in the exhaust port, or between that port and the stripping area, could detect breached targets at the time of breaching. This tritium detection system, which is being further evaluated, would not need to be part of the radiation alarm system.

- Visual examination for gas release during the stripping operation. This has always been done, to some extent, but there can be no assurance that a high percentage of the breached targets will thus be detected unless someone is specifically assigned to look for leaking targets.

- The pickup of water by breached targets is substantial and weighing appears to be a feasible, although possibly time-consuming, method of detecting breached targets after the fact. The feasibility of this method is discussed below.

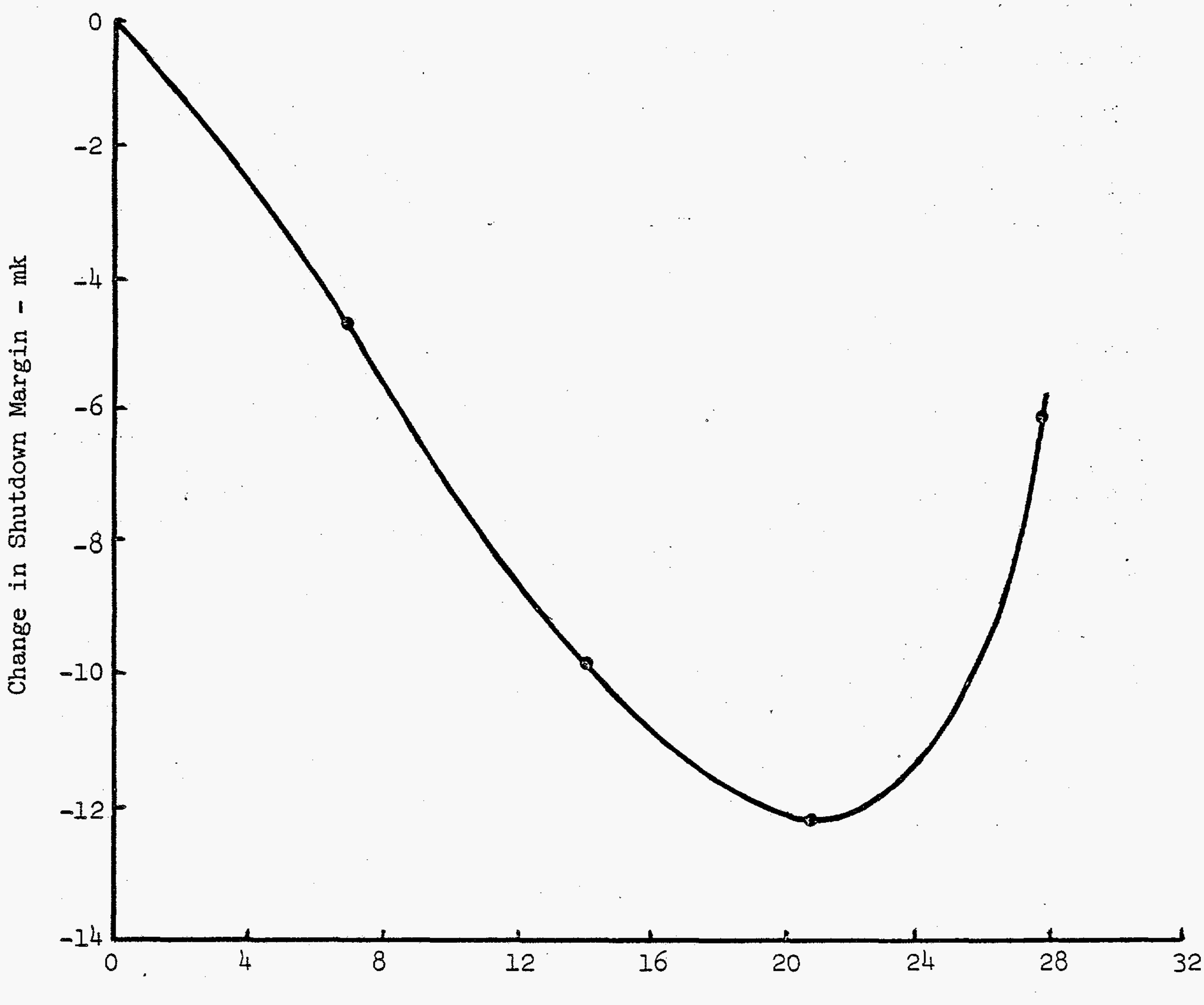


Figure DN-1. Change in shutdown margin caused by displacing the reactor fuel charge.

- Tritium detection just prior to shipping. It may be possible to place several subassemblies in a cask, then evacuate the cask with a roughing pump and check the exhaust for tritium.

Weighing of subassemblies after stripping appears to be a feasible method of locating breached targets. Figure DN-2 shows the distribution of target weights in water from a sample consisting of 100 targets. The sample mean was 313 grams, and the estimated standard deviation was 9.5 grams. Based on this relatively large sample, 2.5 percent of all targets would be rejected if the rejection weight were to be $\geq$ 332 grams. Based on the Table of Reference Dimensions in document RL-GEN-933, and a lithium aluminate density of 80 percent of theoretical, an acceptable target could weigh as much as 333 grams. From this, it would seem reasonable to lower the rejection point to $\geq$ 330 grams, since the conditions to achieve the higher weights seem rather remote.

Six irradiated breached targets were located and weighed and found to all be considerably heavier than 330 grams. Eight unirradiated targets were defected by drilling holes in the cladding and soaked overnight in water and weighed. Five of the eight weighed greater than 330 grams.

Twenty-four additional preweighed, unirradiated subassemblies were defected: Six by removal of an end cap, six by splitting, and twelve by drilling holes. Their weight gain in water then was determined as a function of time. Water pickup was very rapid during the first day After 24 hours, 13 of the 18 subassemblies defected by drilling or splitting exceeded 330 grams. The elements are continuing to increase in weight; however, the rate of increase is greatly reduced. One additional element has passed the 330-gram mark, and it appears that all of the 18 will eventually weigh more than 330 grams. The six subassemblies defected by end cap removal did not exhibit such large weights, primarily due to this treatment having removed about 12 grams of aluminum. Only one of these targets exceeded 330 grams. It should be noted that two of the breached irradiated targets had lost an end cap, but they both weighed greater than 330 grams.

The conclusion is that it is technically feasible to determine breached targets by weight, and the feasibility of weighing those targets already stripped is being evaluated. It is believed that implementation of other detection methods for those targets that have not yet been stripped will be as effective as weighing and further weighing will not be necessary.

Mission 3 - Transplutonium Technology

There were no significant developments on this program.

Mission 4 - Pu-238 Program

There were no significant developments on this program.

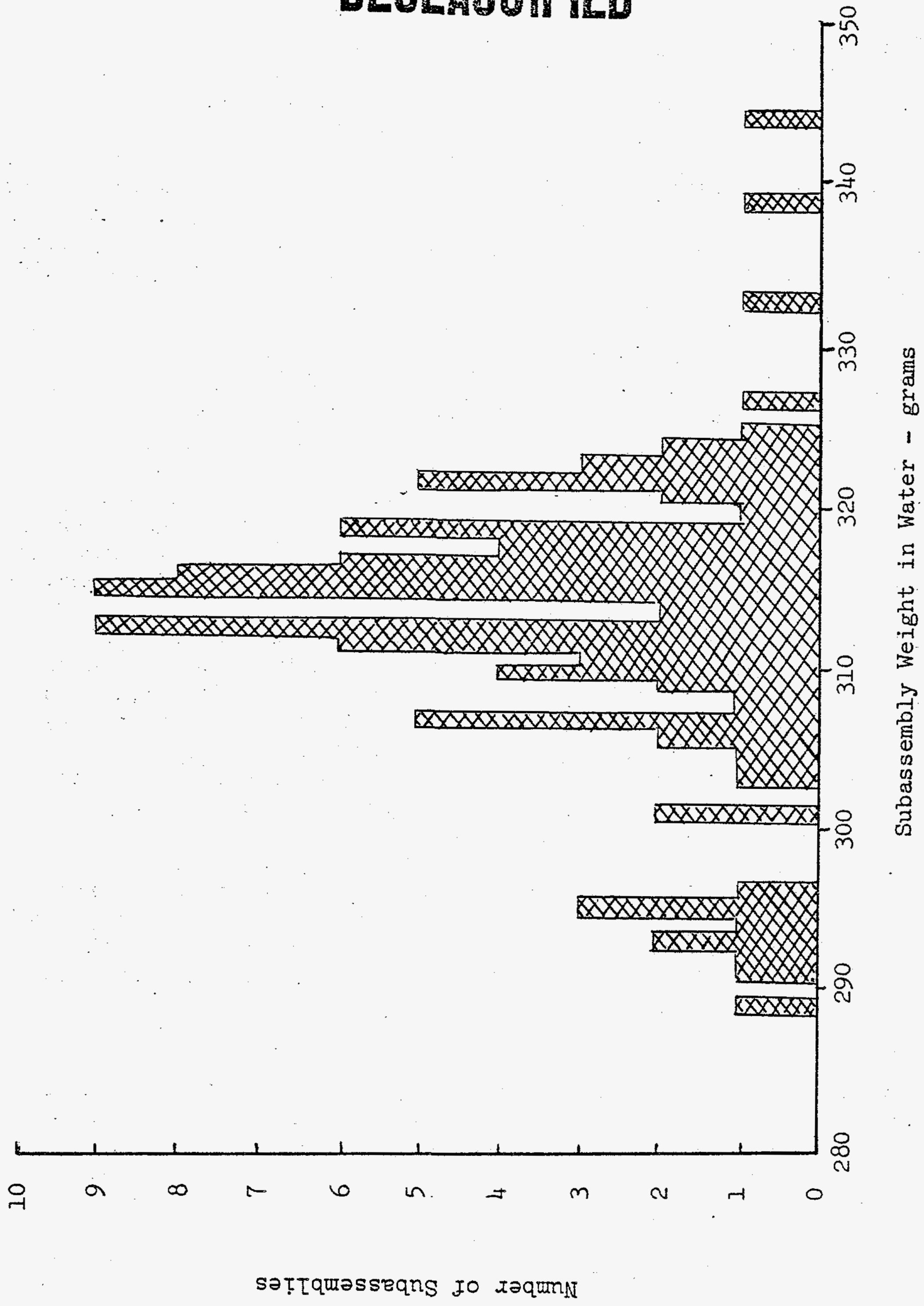


Figure DN-2. Distribution of Coproduct Target Subassemblies by Weight

Mission 5 - Other Isotopes

5-A. Thulium Test Irradiation

The Sanders Nuclear Corporation has expressed an interest in using the thulium oxide which is presently under irradiation in N Reactor graphite cooling tubes. Approximately 150 grams will have been irradiated to about 350 curies per gram activity by July 1. Technical details concerning the irradiation have been given to Sanders, which has also been in contact with Oak Ridge and Savannah River for a supply of material. This irradiation was intended to characterize the graphite cooling tube location as an irradiation facility; arrangements for use of the irradiated material will recognize the need for post-irradiation measurements.

5-B. Cobalt Irradiation Tests

Two cobalt charges, one containing a single BNL strip, the second containing three strips, were charged into graphite cooling tubes during the May outage.

Mission 7 - Target Space Enhancement

There were no significant developments on this program.

Mission 8 - Nuclear Safety

8-A. Status of 324-D Fission Product Release Laboratory

Construction work on the laboratory is 99 percent complete, and setup of portable equipment will be started by BNW. The open access door (providing entry through a clean zone) through the side of the cell will be left open until the portable equipment has been set up inside the cell. The project will, therefore, not be officially closed out for several weeks. Initiation of hot tests is still predicted for early July.

8-B. Meteorology Studies at 100-N

The automated system is still on schedule and is expected to be operational by mid-July. Purchase of a strip printer for printout of meteorological data in the control room was approved, and a purchase order for it has been written. Permanent utilities for the meteorological tower have been installed.

A second rough draft of the software programming specification for the N Reactor meteorological data system was completed. Copies were distributed to Computer Sciences (CSC) for comments. Several meetings also have been held with CSC to discuss details of the system. A sample tape from the hardware manufacturer will be procured for use by CSC in testing the input portion of their program.

8-C. N Reactor Licensability

Work continued on the study of N Reactor licensability. A meeting was held at Bethesda, Md., with members of the staffs of the Divisions of Reactor Licensing, Operational Safety, and Production on April 23 and 24. At this meeting, presentations were made to cover subjects raised since the last (June 1967) meeting with the ACRS on N Reactor safety.

The presentations covered area seismicity, the emergency cooling system, including seismic design, brittle fracture properties, and results of failure analyses; the calculational models for primary loop blowdown and fuel heatup on loss of cooling; the confinement system as an engineered safeguard; and the proposed new design basis accident, which denies the credibility of total loss of coolant to the entire reactor. The presentations are summarized in document DUN-4215, dated May 2, 1968.

Preparations are now being made to address the Hanford Subcommittee of the ACRS here late in June.

ENGINEERING AND TECHNOLOGY - N REACTOR

Primary Pipe Surveillance

A meeting was held on April 29 to discuss with W. J. Love, General Electric Company's R&D Center, N Reactor's primary pipe surveillance program and, in particular, an ultrasonic indication in Cell 2 which was considerably larger than the established minimum of acceptable flaws. The nondestructive testing people who performed the ultrasonic examination described what had been detected, and the original x-rays of the questionable areas were reviewed. It was concluded that the ultrasonic indication does not represent a hazardous condition (at worst, minimum wall thickness is still preserved); however, re-examination of the area with ultrasonics to detect any spread was recommended.

The possibility of re-x-raying the area is being investigated to determine if a double-wall x-ray can produce a better indication than the original single-wall pass (which shows nothing detrimental). BNW has a representative section of 18-inch pipe for nondestructive testing. Current plans call for re-examination of the questionable area in Cell 2 during the June outage. Selected areas in Cell 5 will be ultrasonically examined during the same outage.

Process Tube Monitoring

Process tubes 2158 and 2654 were examined on May 20, using the closed-circuit TV tube monitoring system. There were no significant changes in tube wall appearance between these and previously-monitored tubes. The maximum depth measured in a scratch mark area was 1 mil, and the maximum fret mark depth measured was 11 to 12 mils. Both tubes had heavy (dark) film from the upstream fuel piece to approximately the center of the column; film was light from the center to the rear nozzle junction.

Decontamination Tank Inspection

The decontamination storage tank in Bldg. 1310 was examined internally, both visually and photographically. A remote camera was used to photograph about 70 percent of the tank surface. One series of photos with a telephoto lens revealed a questionable dark line which appeared to be a weld but was darker than the other welds. It was decided that this area could and should be ultrasonically tested from the outside, as it was about half-way up the tank. This will be done.

Visual inspection of the tank bottom revealed it to be in good condition but with some pitting (mostly 1/32-inch deep but with some approaching 1/16 inch deep and 1/4-inch across). Generally, these pits were 1 to 2 inches apart and not over the entire surface. Of the approximately 30 square feet scrutinized, only 40 percent of the area had pits. Small areas examined under 4 to 6 inches of sludge revealed no pitting. All welds examined under sludge and on bare surfaces showed no preferential attack or pitting.

Attempts to ultrasonically check the tank thickness below ground level from inside failed, evidently because the surface could not be adequately prepared.

Visual examination of the 10-inch recirculation line downstream of the recirculation pump showed the pipe, where it goes through the building wall, to have large pits 1/4 to 1/2-inch diameter and up to 3/16 inch deep (three-fourths of the way through the pipe wall). None of the other sections examined in the recirculation line showed pitting of this severity except the section which developed a leak subsequent to the through-reactor decontamination in October 1967. The corresponding piping upstream of the recirculating pump has not been examined.

Fuel Spacer Testing

Out-of-reactor testing was completed on a fuel spacer design modification. The principal result was that a perforated spacer modified according to present requirements (modified post-coproduct spacer) vibrated with less intensity than the original post-coproduct version. Additionally, it was found that, when spacers cocked into the outlet nozzle port, the flow reduction was considerably less than experienced with the old original design (precoproduct). In fact, in one angular position with respect to the nozzle port, the pressure drop increase was negligible.

The range of increases in pressure drop caused by spacer cocking was from 2 to 13.5 psi. The flow reduction in a reactor process tube due to spacer cocking would be about 7 percent at the most. In the majority of the cases, it is doubtful that spacer cocking would be detectable from flow maps. Presumably, slots punched alongside supports (to facilitate die-forming the supports) act as safety valves. These slots were not present in the spacer used with Mark I fuel.

Development testing is continuing on concepts for redesign of spacers to minimize vibration and tube scratching.

Fifty columns of Mark I-C fuel were charged during the May outage with modified post-coproduct spacers. Charging went unusually smoothly with no columns requiring high charging forces. Examination of the spacers from three of the columns showed no evidence of cocking and no Zircaloy wires or shavings.

Project Engineering - Reactor Facilities

Backup Boiler Facility

Boilout of both units was completed May 4. Boilers No. 1 and No. 2 achieved full steam capacity of 200,000 pounds per hour on May 20. However, some tube leaks were later discovered and it was necessary to reroll all 211 tubes in the upper drums on both units. The re-rolling has been completed and no further leaks have appeared. Further operational testing is scheduled after the June outage.

Project Status Summary

The month-end status of approved construction projects relating to N Reactor facilities is summarized in Appendix A.

ENGINEERING & TECHNOLOGY - FUELS & TARGETS

Follower-Block Tooling Development

One test extrusion using a copper billet was made to determine if an insert shear follower-block was feasible for butt shearing various types of extrusions, and as a test of the extrusion press mandrel moving device to determine if larger extrusions and extrusions of metals such as Zircaloy could be sheared with the limited mandrel mover ram capacity of approximately 300 tons. This first test extrusion was successful and more will be made.

The upset and running pressures were normal until the extrusion was almost complete. A slight rise in pressure was noted as the follower-block moved into the die cone, which indicated the estimated one-fourth inch butt cone thickness was approaching a stall condition. The pressure involved to perform the shearing operation was about 95 tons. The extrusion and the butt section indicated a clean, square cut with no distortion. Inspection of the liner indicated no metal galling or back extrusion.

Graphite Cutoff Block Dimensions

The graphite cutoff block serves two purposes and must be designed to fulfill both; namely, (1) to seal the area between the follower block and the extrusion, both at the liner and at the mandrel, and (2) to enter the die cone and force the extrusion butt through the die without equipment or tooling damage. Improper dimensional clearances may allow back extrusion, unequal compression pressure which can result in cracking or crumbling of a section of the cutoff, or may cause mandrel breakage.

Recent tests on 51 extrusions using inner traveling mandrels machined with O.D. and I.D. clearance 0.025 inches proved quite successful. No mandrel breakage occurred, which is unusual for small diameter traveling type mandrels. If tests continue to be as successful, a change in the graphite block fabrication dimensions will be made.

Uranium Billet Fabrication Development

The 36 Mark I upset forge billets coextruded last month were processed through finishing steps. Data obtained for these fuels show there is no significant processing differences between upset forge uranium when compared to uranium produced by the standard drill-machine fabrication route. The only abnormality encountered was two fuels that were rejected for I.D. unbond indications at the ultrasonic testers. These unbond areas, when sectioned and examined metallographically, were found to have been caused by uranium oxide inclusion. The source of this oxide has not yet been determined.

ADVANCED CONCEPTS AND PLANNING

MFCs AND OTHER PROPOSALS

MFC-12 - New Capabilities for Production Reactors

Nothing to report this month.

MFC-13 - Study of Application for Hanford-Produced Cobalt-60

A proposal is being prepared which requests funding for Douglas United Nuclear to design a conceptual irradiator based on the tubular cobalt source. The irradiator would be used to irradiate liquids and fluidized solids such as jet fuel, heating oil, sewage, and wastewater. Work would include calculating the utilization efficiency of the irradiator and preparing estimates of operating and construction costs. The utilization efficiency of an irradiator is defined as the ratio of radiation absorbed by the material irradiated to the total radiation emitted by the source.

Warm Water Utilization

DUN and AEC-RL representatives spent two days in Olympia, Washington, with the State Office of Nuclear Energy Development rewriting the Warm Water Irrigation Proposal in prospectus form. The prospectus is now complete so far as the technical portions are concerned; the State is responsible for preparing a section on administration of the program.

SYSTEMS ANALYSIS

CAGE Mod 2-A

Updating CAGE to the Mod 2 version is proceeding as fast as possible since it is becoming more and more apparent that an updated CAGE program is needed. A new look is being taken at all segments of the program in order to make improvements based on knowledge gained during the recent period of CAGE Mod 1 use, and to update the price levels to the mid-fiscal 1968 values.

Mod 2-A will be normalized to the fiscal 1970 budget level rather than fiscal 1969, since 1969 is considered to be an abnormal and austere year. Therefore, Mod 2-A will use the 1969 budget as an input directly with fiscal 1970 being the first really free study year. Expected completion date of Mod 2-A is now August 1. At that time, a set of standard cases will be run to serve as future reference cases which will permit a point of normalization and comparison for later studies. These standard cases will be chosen carefully to reflect a broad base of planning assumptions with respect to number of reactors and reactor modes.

Planning Estimates

The current series of planning estimate studies are being delayed because of lack of input information. In preparation for the 1968 planning estimates, a linear program computer model for the UNIVAC 1107 has been under development which will prepare the output for the planning estimates.

The program, called LOPER, which stands for Linear Optimizing Planning Estimate Recorder, includes in matrix form all the loads that are currently in CAGE and will be used as follows: after product requirements have been modified to reflect real time and inventory considerations, the computer program will seek solutions which will optimize the scheduling of the four Richland reactors over the planning period. Various optimizing possibilities are possible but it has not yet been decided which one to use. Constraints can be included which minimize, for example, the use of too high enrichments; also, some simple cost formulation could be used to optimize on a cost basis.

It is envisioned that this computer program will considerably speed up the preparation of planning estimates; however, it is not likely that it will be ready in sufficient time to do more than check the current planning estimates.

Plutonium-238 Incremental Costs

As part of AECOP Task 5, studies continue to be made on the feasibility and cost of producing Pu-238 at Hanford. The most recent studies indicate that, under present and future load conditions, it is feasible to campaign irradiated neptunium through the Purex plant every six months, rather than a continuous operation utilizing the now shutdown Redox plant. The K reactors with enhanced neutron flux would irradiate neptunium for three months out of every six-month campaign.

While there would be penalties resulting from not using the fast turnaround capability, and neptunium inventories would be higher, both capital costs and incremental operating costs of neptunium recycle are lowered substantially from the case using the Redox plant. It is planned to use Purex campaigns as the basis for costing in the future AECOP studies except possibly for cases in which extremely high production rates are required.

ADVANCED PLANNING

Future Plutonium Value

After further studies of the factors that influence the future value of plutonium, 1990 is still indicated as the probable time when plutonium values will increase to the levels (relative to U_3O_8 costs) shown below:

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Some authors have maintained the position, even recently, that the option of using U-235 fuel in fast flux reactors would limit the value of plutonium to a lower level than shown above, but the best available, practical, fast converter designs do not appear to bear this out.

The Battelle-Northwest linear program for studying the USA power generation to the year 2040 has indicated both low and high values of plutonium under widely different circumstances. DUN is now using this program to run cases to establish under what conditions the high values for plutonium are confirmed by the much more sophisticated linear program.

Plant Operating Alternatives

Briefing tables issued to describe various plant operating alternatives are being reissued with revisions and corrections. The new tables also include the 195 enrichment level for the coproduction of tritium and plutonium. This enrichment level has been added because of recent revisions in feedsite capital cost estimates which indicate a major break in capital requirements at this enrichment level.

Research and Development and DUN capital cost estimates for the 195 coproduct mode are not expected to be any different than for the 210 product modes. Allowing for uranium utilization factors, feedsite requirements for four Richland reactors on a 195 coproduct mode would closely approach or slightly exceed the 3000-ton/year level on which the Fernald capital costs are based. Tritium production capability is about 6 percent lower for the 195 coproduct mode as compared to the 210 mode.

High Quality Plutonium Production Study

A preliminary study is nearly complete to estimate the premium cost that would be assigned to weapons plutonium production if the assay were reduced to 2 percent plutonium-240. The study encompasses a requirement range up to 5000 kg over a three-year campaign.

The results of the study are being documented in DUN-AOP-122.

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WITH DELETIONS

[redacted]

IRRADIATION SERVICES

FUEL TECHNOLOGY

A series of irradiations is being conducted for NASA-LRC as part of a program to develop a fuel for aircraft and space vehicle propulsion reactors. One assembly of three capsules of uranium oxide fuel dispersed in molybdenum and clad in molybdenum was charged in tandem, with independent helium supplies and thermocouple trains, into a General Purpose facility at KW Reactor. Maximum operating temperature and pressure are 2500 F and 2000 psi; goal exposures may be as high as 10,000 hours. During the report period, a cladding failure occurred after approximately 1400 hours of irradiation in the capsule farthest in the reactor. This capsule was separated from the assembly, and the remaining two-capsule assembly was recharged into the test facility.

An additional molybdenum-uranium oxide cermet single capsule was charged into a General Purpose facility at KW.

MATERIALS TESTING

Boron Carbide Irradiation for Battelle-Northwest

A preliminary irradiation of boron carbide rods was conducted in the Snout facility at KW Reactor to determine the heat generation rates and heat transfer characteristics of a proposed long-term irradiation capsule. This test is in support of an FFTF control rod material study by BNW.

Corrosion Products Transport Studies for Battelle-Northwest

The first test in the Corrosion Products Transport Facility was successfully completed on May 14. A total operating time of 669 hours had been accumulated at operating conditions of temperature, pressure, and flow: 500 F test section temperature, 1200 psig, and 0.062 gpm total flow (0.031 gpm through each test section). The test specimens and ballast material were discharged from the facility, the facility was chemically cleaned, and the ballast tanks and test sections were charged. This second test is planned for 1000 hours.

ISOTOPE PRODUCTION

Two capsules containing strontium nitrate were shipped to NASA-Ames Research Center to provide Sr-85 radiation sources for ground tests associated with their biosatellite studies.

ROUTINE IRRADIATIONS

The following routine irradiation services were performed:

- Sixty-seven activation analysis capsules were irradiated in the KE, KW

[redacted]

and C Reactor Quickie facilities for Battelle-Northwest.

- A double-tandem uranium swelling capsule was charged into a front-to-rear General Purpose facility at KW Reactor.
- Two cooled tensile specimen capsules were discharged from the Snout facility at KW Reactor.

ADMINISTRATION - GENERAL

EMPLOYEE RELATIONS

A decision was received from the Federal Circuit Court of Appeals on the appealability of the District Court's earlier ruling on the question of contract coverage for N Reactor employees. The Company was advised that its appeal will not be entertained, and that the earlier ruling to submit the question to arbitration stands. Further courses of action open to the Company are being studied.

Calculation of retroactive pay for "red circle rate" employees affected by the arbitration ruling has been completed. These employees will receive a total of $12,704. Recalculation of employee pension benefits to include the retroactive increase will also be required.

EMPLOYMENT SUMMARY

Douglas United Nuclear employment as of May 31 is summarized in Appendix B.

SAFETY

Month end safety statistics were:

Disabling injuries:	May	0
	CY to date	0
Days since last disabling injury		340
Man-hours since last disabling injury		3,700,000

No radiation exposures exceeded operational control.

PRODUCTION COMPUTER DATA TRANSMISSION TERMINAL

As the result of a survey of the facilities and layout of the 1101-N Building, particularly the walk-in vault and the adjacent library space, consideration is being given to locating the production computer data transmission function in that 100-N Area building. Other advantages of this location over 100-D or 100-H include: better fire protection, lower site preparation costs, and closer proximity to the maximum number of future users of the services.

APPROVAL LETTERS

At the close of the reporting period, the following approval requests had not been acted upon by AEC-RL:

ATD Number	Subject	Date of Transmittal to AEC-RL
AP-39	Pension Plan portion of letter entitled "Pension Plan, Salaried Savings Plan and Wage Savings Plan"	January 12, 1966
ATD-92 Add. #1	Performance and Cost Effectiveness Program (FY 1969 Budget)	May 23, 1968
ATD-93 Add. #6	Plant and Equipment Adjustments	May 6, 1968
ATD-152	Merit Salary Increases Exempt Employees	February 29, 1968
ATD-155	Douglas United Nuclear, Inc., 1968 Annual Report	April 26, 1968
ATD-160	Approval of Suggestion No. NR 2209	May 22, 1968

MEDICAL CLAIM PROCESSING

Effective July 1, employee medical claims will be mailed directly to the Portland Claims Office of the Connecticut General Life Insurance Company for processing. Historically at Hanford, claims have been reviewed and processed locally by Benefit Plans groups and then forwarded to the insurance carrier for payment. No particular problems are anticipated in the transition.

BUDGET ACTIVITY

At AEC-RL request, various budget alternates were examined to determine the effects on the FY 1969 and FY 1970 budgets. The results are detailed in document DUN-4298.

APPENDIX A

PROJECT STATUS SUMMARY - REACTOR FACILITIES

Number & Title	Authorized	Percent Complete Design	Percent Complete Construction	Status
C & K Reactors				
DCE-505, Boiler Control Improvements - 165-KE and KW.	$281,000	99	24	Equipment installation on the No. 3 KW boiler is nearing completion. Control system acceptance testing, which had been scheduled for May 27, has been rescheduled for June 3 at the request of Republic for convenience in scheduling their field engineer for boiler startup assistance. The No. 3 KW primary power supply and the backup supply were acceptance tested May 17.
DAP-510, Discharge Chute Clearing Equipment - K Reactors.	$190,000	100	3	J. A. Jones Construction Company is fabricating conveyors in their shop.
DAE-512, Replacement of Turbine with Diesel Drive - 181-B Pump.	$ 87,000	100	64	Work on schedule. Assembly of reducer gear awaits arrival of new gears.
DAP-513, Deactivation of Hanford Production Reactor (formerly 509)	$ 80,000	100	44	Electrical work at 190-D and the steam line conversion at 185-D are progressing.

PROJECT STATUS SUMMARY - REACTOR FACILITIES (contd)

Number & Title	Authorized	Percent Complete Design	Percent Complete Construction	Status
C & K Reactors				
DAP-515, Replacement Steam Generating Facility - 100-D Area.	$140,000	25	0	Foundation drawings are being firmed and construction start is imminent.
N Reactor				
GAP-401, Upgrading Fire Protection - 100-N.	$150,000	100	82 (est.)	Contractor proceeding with work in Room 6 and on CO_2 piping system. This work expected to be completed during June outage.
GCE-405, N Reactor Temperature Monitoring System Improvements.	$189,000	100	57	No change in status.
GCP-406, Safety Platforms and Accesses.	$300,000	100	60	Installation of platform system for north end of Cells 3 and 5 completed. This completes the platform installation for all cells except No. 6.
GCE-408, W, C, D Elevator Safety.	$ 90,000	100	0	Selection of a power sensing switch has been made for elevator level detection. Drawing revisions to reflect this selection are in progress.

PROJECT STATUS SUMMARY - REACTOR FACILITIES (contd)

Number & Title	Authorized	Percent Complete Design	Percent Complete Construction	Status
N Reactor (contd)				
GCP-409, Fuel Handling Improvements.	$310,000	80	27	AEC-RL approval for Rev. 2 to the Project Proposal not yet received. Testing of alarm monitoring system is being performed and an underwater dump test is scheduled to be conducted in Bldg. 189-D.
GAP-410, Decontamination Waste Loading Facilities.	$ 65,500	100	30 (est.)	Pipe excavation work by J. A. Jones' subcontractor, HUICO, has been completed. Fabrication of railway unloading station by Jones is proceeding.
GCP-411, Effluent Control Program - 100-N.	$590,734	96	82 (est.)	Construction estimates made on the various segments of this project indicate that the original contingency of $218,000 has now been reduced to $80,000. Vitro/HES has been requested to issue a revised design schedule. $1,079,266 of authorized funds held in reserve by AEC.
DCP-514, Air Conditioning, Water Quality Lab., 109-N.	$ 37,500	100	85	The last purchase order for furnishing and adjusting the automatic controls was placed with Johnson Service Co. on May 10. Installation work now in progress.
DCP-517, Additional Storage Basin Cubicles, 105-N.	$ 37,000	70	0	All major components for this work are on order. Arrangements being made for having the concrete tested at the C Reactor.

APPENDIX B

EMPLOYMENT SUMMARY
(as of 5-31-68)

	Exempt	Non-Exempt	Total	Change from 4-30-68
Operations Division				
General	1	1	2	0
Manufacturing	190	584	774	-16
N Reactor	71	194	265	- 2
Fuels Section	117	320	437	+ 3
	379	1,099	1,478	-15
Technical Division				
General	1	1	2	0
Research & Engineering	71	37	108	- 1
N Research & Engineering	47	17	64	- 4
Advanced Concepts & Planning	12	3	15	0
Facilities Engineering	54	14	68	- 2
N Project Engineering	54	14	68	+ 2
	239	86	325	- 5
Finance & Administration Division				
General	1	1	2	0
Finance & Administration	67	72	139	- 2
Employee & Community Relations	17	10	27	+ 1
	85	83	168	- 1
Company General				
Management	6	1	7	0
TOTAL	709	1,269	1,978	-21

FEATURE REPORT

THE QUALITY PROGRAM IN FUELS MANUFACTURE

INTRODUCTION

High integrity fuel elements are essential for efficient reactor operation, and a Quality Program designed to assure this integrity has had a key role in fuel manufacture at Hanford since production reactor operations began in 1944. The Program has been revised and expanded over the years to meet the higher demands on fuel quality which resulted from increasingly more rigorous reactor operating conditions.

The purpose of this summary report is to describe briefly the Quality Program presently used by the Fuels Section of Douglas United Nuclear in manufacturing fuel elements for the DUN-operated reactors. Recent in-reactor performance data for these fuels are included to illustrate the effectiveness of this Program.

FUELS SECTION ORGANIZATION

The function of the Fuels Section is to develop and produce high quality fuel and target elements as required for irradiation in the C, KE, KW, and N Reactors. Responsibility for product quality in this Section is assigned to all of the several Subsections as follows:

<u>Fuels Manufacturing Subsection</u> - Fabricate fuel and target elements within process specifications to obtain a consistently high quality product.

<u>Quality Control Subsection</u> - Provide and guide the overall Quality Assurance and Control Programs for the various fuel fabrication processes.

<u>Engineering Subsections</u> - Provide fuel design and process specifications for new and existing fabrication processes to ensure product manufacture within the required quality criteria.

<u>Plant Services Subsection</u> - Contribute to product and process specification conformance by careful maintenance of all equipment and facilities used in fuel manufacture and testing.

This organizational structure provides for independent action by the Quality Control component at the same level as the Fuels Manufacturing component; this relationship is essential for an effective Quality Program.

Management is expected to exercise its authority and organizational freedom to identify and evaluate quality problems and to initiate, recommend, or provide solutions.

THE QUALITY PLAN

General

The fuels quality plan stresses the importance of each operator auditing and controlling his machine or process. Quality Control provides the necessary tools for use by the operators in meeting their responsibilities. These tools take the form of process audit check sheets, machine setup checks, first-piece inspection, capability studies, control charts, hourly data feedback, etc. Quality Control performs nonroutine audits of process, product, and procedures to determine conformance to the quality plan. Additional features of the plan are described separately below.

In the design and conduct of the fuels Quality Program, special consideration has been accorded the excellent information contained in two basic reference documents: (1) Department of Defense Specification MIL-Q-9858A, titled "Quality Program Requirements," and (2) the "AEC-ALOO Quality Program Specification" issued in 1966.

Design

Quality is designed into the product at the outset. Design Engineering starts with a determination of the basic characteristics required to make a quality fuel element, and then incorporates these into the product specifications. A Process is then developed which will manufacture the product within the specifications.

Although Design Engineering is primarily responsible for all quality-related design, the coordinated efforts of other groups such as Material Handling, Shop Operations, Purchasing, Financial, and Quality Control are essential for a quality-designed product. Design reviews are held so that all groups concerned with the product and process will be informed of the design and can contribute to it. For example, Quality Control reviews the design to ensure that the functions of inspection, control, and test can be performed effectively.

Incoming Materials

To preserve the required as-received quality level, active liaison is maintained with the various vendors that supply materials for the fuel fabrication process. Vendors are kept informed as to why tight tolerances are needed, where their materials are used in the process, and the consequences of poor quality. Acceptance inspection criteria are established for component parts and these become a part of each procurement contract.

All incoming material used in fuel manufacture is sampled, measured, and tested for compliance with purchase specifications. The test reactor, analytical laboratory, and receiving inspection facility are used as appropriate in measuring the quality of incoming material. New or substitute materials are analyzed or evaluated for fuel process suitability before use.

Figure 1, appended, shows the dimensional checking of incoming aluminum spires prior to use in the AlSi canning process employed in fabricating fuel for the C, KE, and KW Reactors. Figure 2 (on the same page) shows incoming uranium billets being checked for enrichment level prior to coextrusion with Zircaloy cladding to form fuel tubes for N Reactor; billet wall thickness is measured at six points, as chalk-marked.

Gage Control

A gage control program is employed to protect against damaged gages, to assure dimensionally correct gages, and to maintain gages in accordance with applicable drawings and specifications.

All adjustable and fixed-limit gages are calibrated, using standards having known valid relationships to national standards, and are certified prior to use in the process. All certified gages are identified with a recertification due-date to ensure rechecking on a routine basis. A gage control laboratory in which both temperature and humidity are controlled, is maintained for gage storage, calibration and certification.

Process Control

The process is controlled to ensure that the product meets all specifications during manufacture, and to keep scrap and rework cost within economic bounds.

Process parameters are routinely audited and documented. Work instructions and procedures are provided for all operations which affect quality. Reject-rate feedback is provided routinely to each operating area that generates defects, and control limits are established for each defect characteristic. Control charts are used where tighter process control is required. Process solutions are analyzed routinely to assure conformance with specifications.

Process and equipment capability studies are made to determine the amount of inspection and process control needed to assure product quality. Defect analyses are performed to establish cause-and-effect relationships for process improvement and better control. Prompt corrective action is taken with respect to any assignable condition which is adverse to quality.

In-Process Inspection and Testing

Fuel cores and elements are inspected and tested at key points in the process for compliance with specifications. Inspection stations are located as closely as possible to areas where defects occur, in order that corrective information feedback can be provided with minimum delay. A sequential-type inspection is performed at machines used in making the product to ensure best control. Visual displays and/or physical standards are provided at all inspection stations to illustrate various types of defects. Test equipment is standardized routinely, and tester precision is calculated at regular intervals. Samples of defective fuel elements are evaluated metallographically in the Analytical Laboratory. In addition, all finished elements are autoclaved as a final check on closure integrity.

Several photographs are appended to show some of the principal in-process inspection and testing activities. These are:

Figure 3 - After pickling, and before AlSi canning, the bare fuel cores are visually inspected for cracks and cleanliness.

Figure 4 - After facing, the AlSi-bonded elements are checked for conformance to length specification. (The facing lathe appears at the left in the photo.)

Figure 5 - Every AlSi-bonded element is tested ultrasonically for clad-to-core bond continuity and clad thickness.

Figure 6 - Every coextruded Zircaloy-clad element is similarly checked ultrasonically for bonding and for clad thickness.

Figure 7 - In the final inspection and assembly of coextruded fuel tubes, the inner supports are sized to give desired compressive strength, and steel shoes are crimped onto the outer tube supports. (Inner and outer fuel tubes are visible on carts behind the operators.)

Figure 8 - After final assembly of the N fuel, the compressive strength of the inner tube supports is measured.

Figure 9 - Close visual inspection (using magnification) is given the closure weld of every AlSi-bonded element.

Quality Information

Complete and reliable records essential to the economical and effective operation of the Quality Program are maintained. Reject rates for each operating area in the process are compiled and published daily, with control limits indicated for each type defect. Data concerning critical quality characteristics of finished fuels are plotted, and inspection efficiency is calculated on a daily basis and published monthly.

Figure 10, appended, is a photo of the large display board which is centrally located in the AlSi process shop to show quality information data that are updated daily.

Disposition of Nonconforming Materials

Incoming materials which do not meet an acceptable quality level are identified as unacceptable and removed from the process. Data concerning such material are documented and presented to a review group which makes a recommendation regarding disposition of the material. Review group recommendations that substandard material be used require concurrence by subsection management.

Similarly, in-process fuel found to be outside of process specifications, or fuel of questionable quality, is removed from the process stream. Disposition of such material is recommended by a review group for subsection management approval. Fuel which does not meet product specifications is placed in a "hold" category, and is released for reactor use only if waivered by reactor management.

Quality Assurance

The last step in the program to assure high-quality reactor fuel is a final evaluation of the quality level of each fuel element lot, using an appropriately-sized sample randomly selected after in-process inspection and testing. The sample is measured for conformance to specifications. If these measurements indicate that the lot does not meet an acceptable quality level, the entire lot is given an additional 100 percent in-process inspection for the characteristic(s) in question. All lots must meet an acceptable quality level before they are released for reactor use.

In addition to affording a final check on fuel quality level, the Quality Assurance program generates data used to measure in-process inspection efficiency and provides feedback information leading to the improvement of equipment and process control.

Product Surveillance

All reactor complaints on fuel quality are investigated and documented. Defect analyses are performed on defective or substandard fuel elements detected at the reactors, to determine why the faults were not detected during manufacture or before shipment. Fuel lots defined as failure-prone, because of poor performance under irradiation, are investigated to determine if anything unusual occurred while the lot was being fabricated that could have affected quality adversely. In-reactor fuel element failures attributed to fabrication-type defects are investigated to determine ways to prevent further such failures and improve fuel quality.

QUALITY COST ANALYSIS

As a management tool to measure the cost of the Quality Program for AlSi-bonded fuel elements, a quality cost analysis is performed and published quarterly. This cost analysis represents expenditures incurred in the following quality cost categories:

Prevention - Efforts directed toward the prevention of future defects. Included are development of inspection and testing methods and equipment, capability studies, defect analyses, personnel training, etc.

Appraisal - The work of measuring the quality characteristics of current production to determine conformance to specifications. Includes all inspection and testing.

Failure - Costs resulting from failure of the product to meet established quality criteria.

During FY 1967, the total quality cost per AlSi-bonded element averaged $0.59, and represented 17 percent of fuel Shop cost. The breakdown of this total by cost categories was: Prevention - 22%, Appraisal - 53%, and Failure - 25%.

Appraisal costs are high with this product due to the large amount of 100 percent inspection and nondestructive testing performed throughout the process. The reactor operational difficulty and production loss attending a fuel element failure justifies this very extensive inspection. However, the cost of in-reactor fuel failures is not included in these quality cost analyses. The 25% assigned to failure cost represents only those costs which are internal to fuel manufacture (e.g., scrap and rework).

Quality cost analysis is planned for extension to cover also the coextruded fuels made for N Reactor.

IN-REACTOR FUEL PERFORMANCE

AlSi-Bonded Fuel

In fuel manufacture, the prime measurement and most meaningful index of product quality is the in-reactor performance of the fuel elements. In the 2-year period 1964-1965, the failure rate of regular production fuel (excluding test material) was 34 per million elements irradiated. This period typifies prior failure experience and, indeed, was a reduction in rate over many of the previous years. In the 2-year period 1966-1967, the fuel failure rate on a comparable basis was only 10 per million elements irradiated. This 3-fold reduction in failure rate (and associated reactor outages) has the added significance of having been achieved during a period when the fuels were being subjected to longer irradiation times.

One of the major contributions to this improvement in fuel performance was the attachment of self-supports to the fuel elements; this markedly reduced the incidence of "side-hot-spot" failures. Other factors contributing to the improved fuel quality include better specifications, increased inspection efficiency, more rigid control of nondestructive testing equipment, and establishment of the product Quality Assurance Program. In the most recent year of record, 1967, less than half of the fuel failures were attributed to fabrication defects. Studies of these defects indicate that most involved the end closure.

Coextruded Fuel

In 1965 through 1967, the failure rate of regular N Reactor fuel (excluding test material) was 0.65 per thousand fuel assemblies irradiated. This failure rate is not directly comparable with that of fuel in the single-pass reactors, because of the more severe operating conditions in N Reactor and the more complex design of its fuel.

Thirty-one percent of the N fuel failures occurred near the ends of the tubular elements and were probably related to closure defects. Sixty-six percent were caused by fretting corrosion, and 70 percent of these failures were associated with the inner fuel supports. Failures of the latter type should be eliminated, or at least reduced significantly, as the reactor loading is converted to Mark IV fuel with its improved support system. A number of improvements have been effected in the end closure, including a chevron-shaped end cap and a double-welding procedure; these steps also should reduce the number of failures.

PROGRAM OUTLOOK

The Fuels Quality Program in Douglas United Nuclear is subjected to continual review and refinement. Flexibility will be maintained to assure ready adaptability to new or modified fuel and target element designs.

Presently, the Quality Control approaches for N Reactor fuel and single-pass reactor fuel are being integrated to provide optimum utilization of the best features.

With increased reactor emphasis on the production of nonweapon-grade materials, and the resulting increase in fuel irradiation severity, tighter acceptable quality levels may be necessary to keep fuel failures within economic bounds.

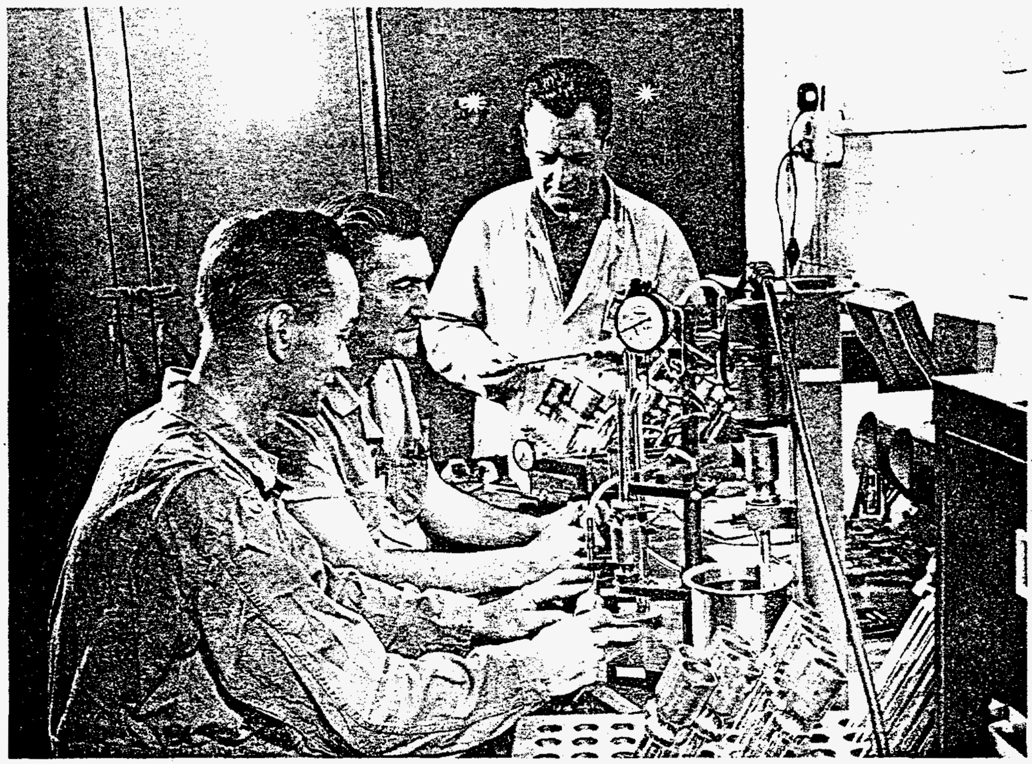

Figure 1. Inspection of Incoming Spires

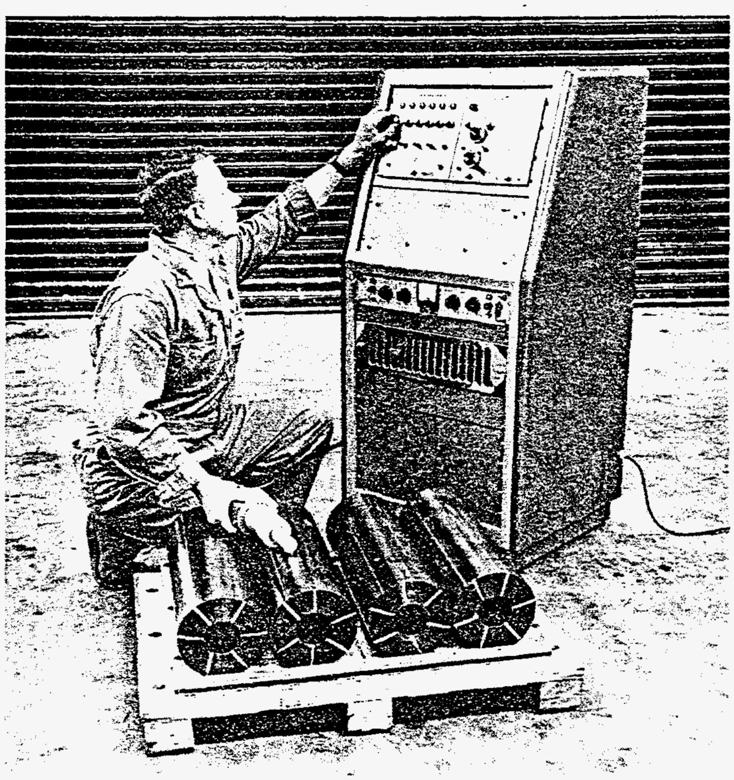

Figure 2. Inspection of Incoming Billets

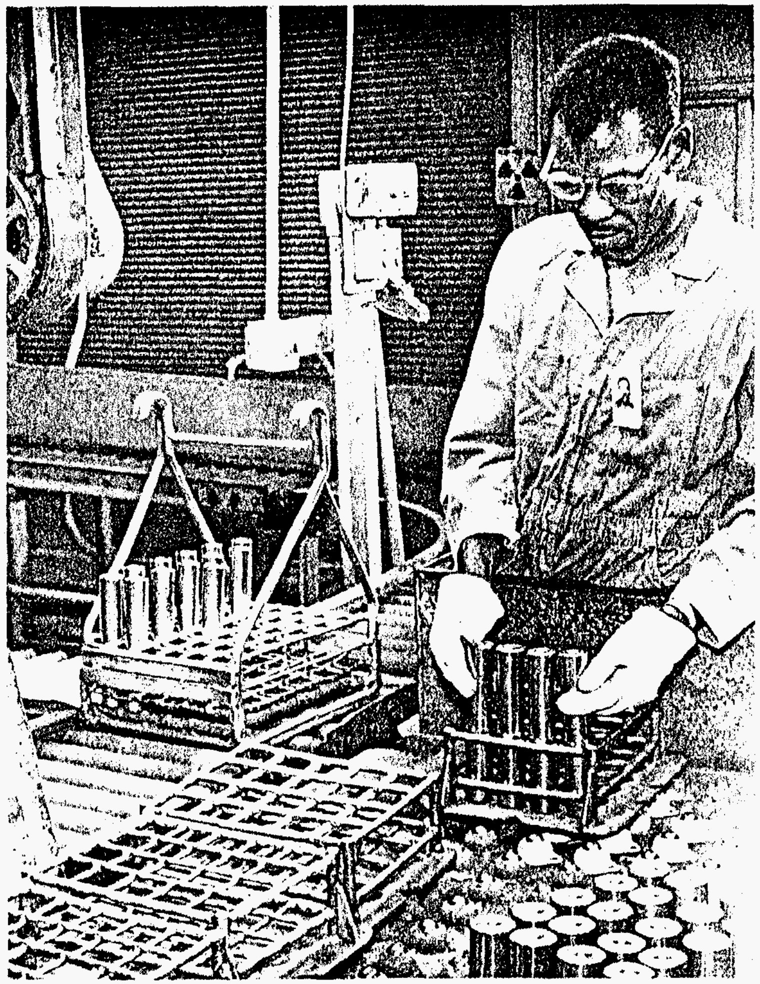

Figure 3. Inspection of Cores

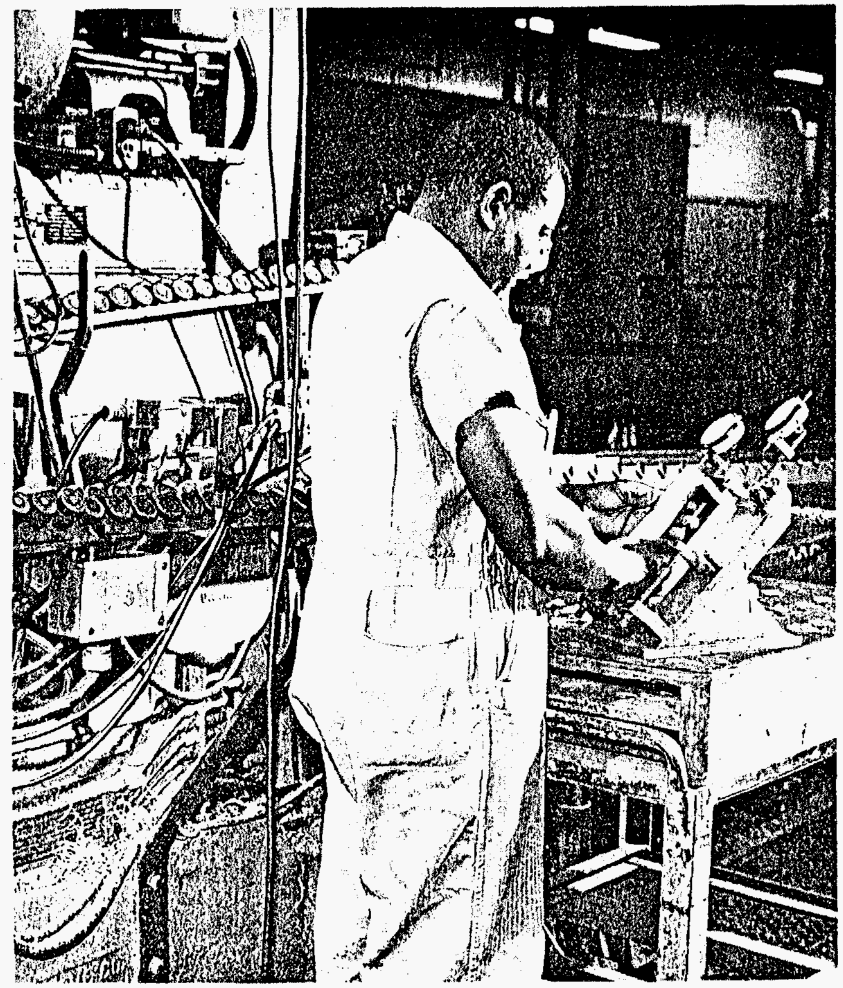

Figure 4. Length Check, after Facing

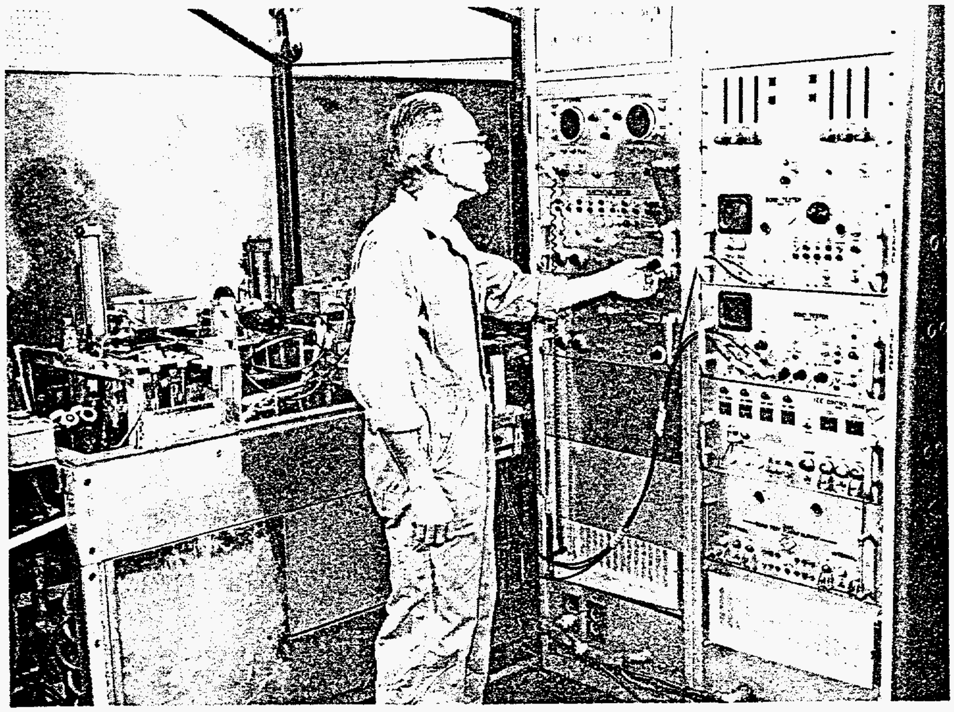

Figure 5. Ultrasonic Clad & Bond Test (AlSi Process)

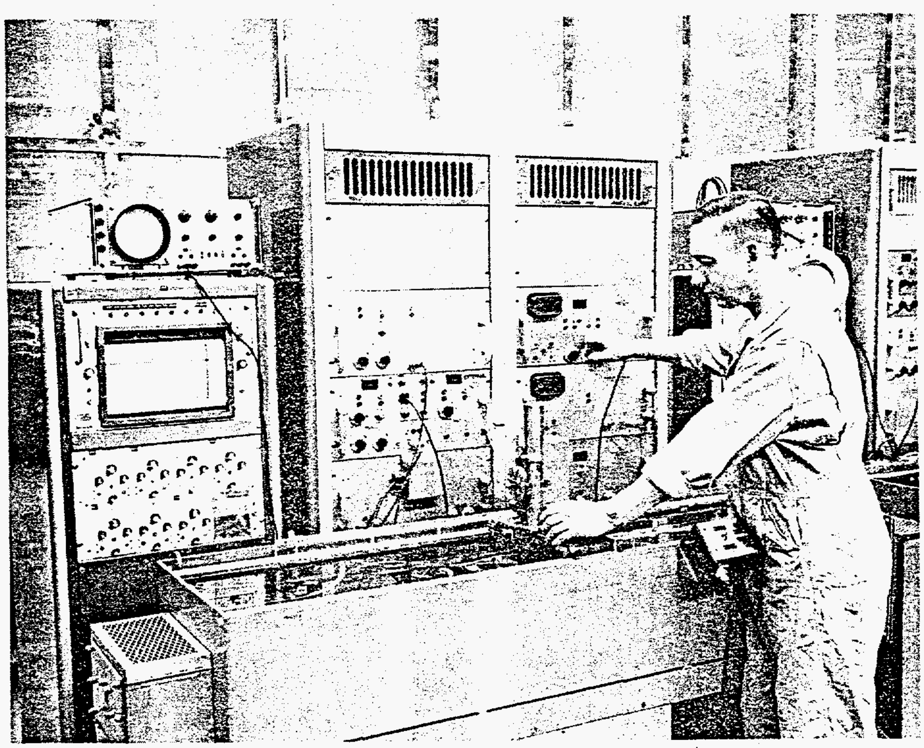

Figure 6. Ultrasonic Clad & Bond Test (Coextruded)

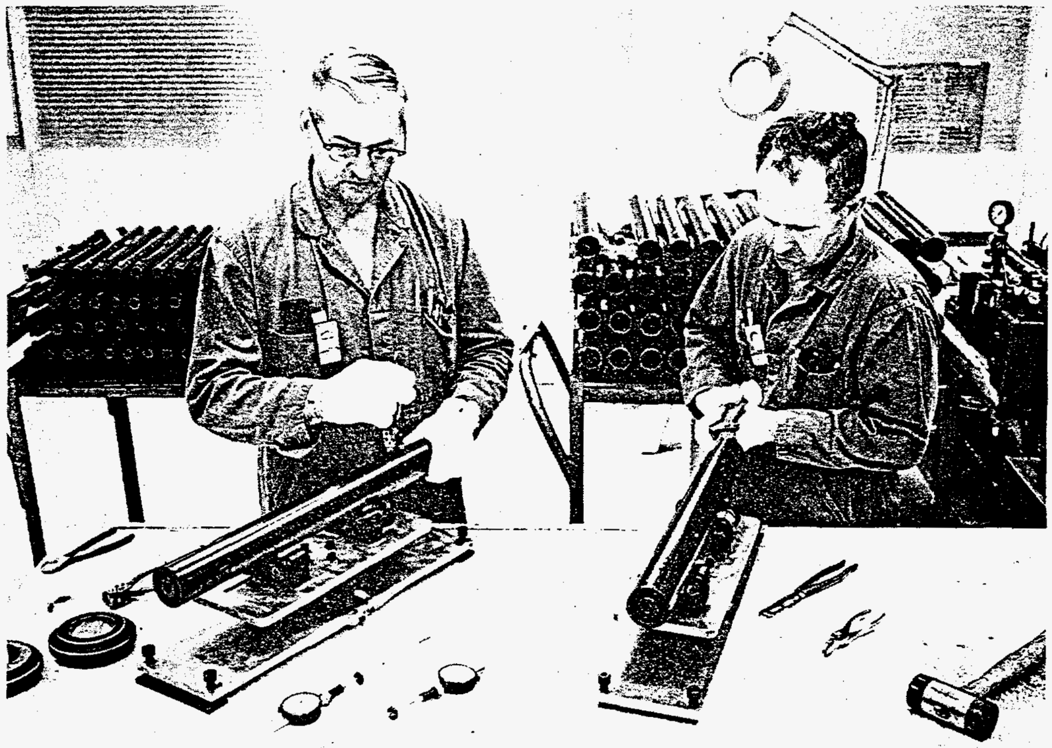

Figure 7. Sizing Supports & Crimping-on Shoes (N Fuel)

Figure 8. Inner Support Strength Test

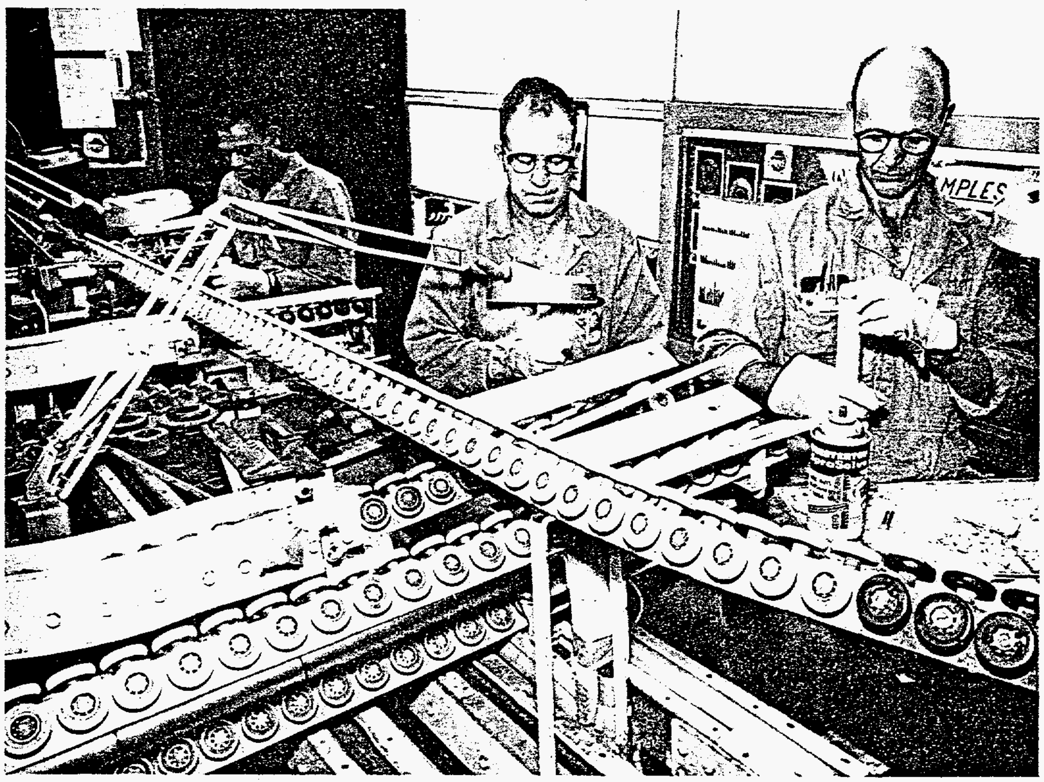

Figure 9. Inspection of Closure Welds (AlSi Process)

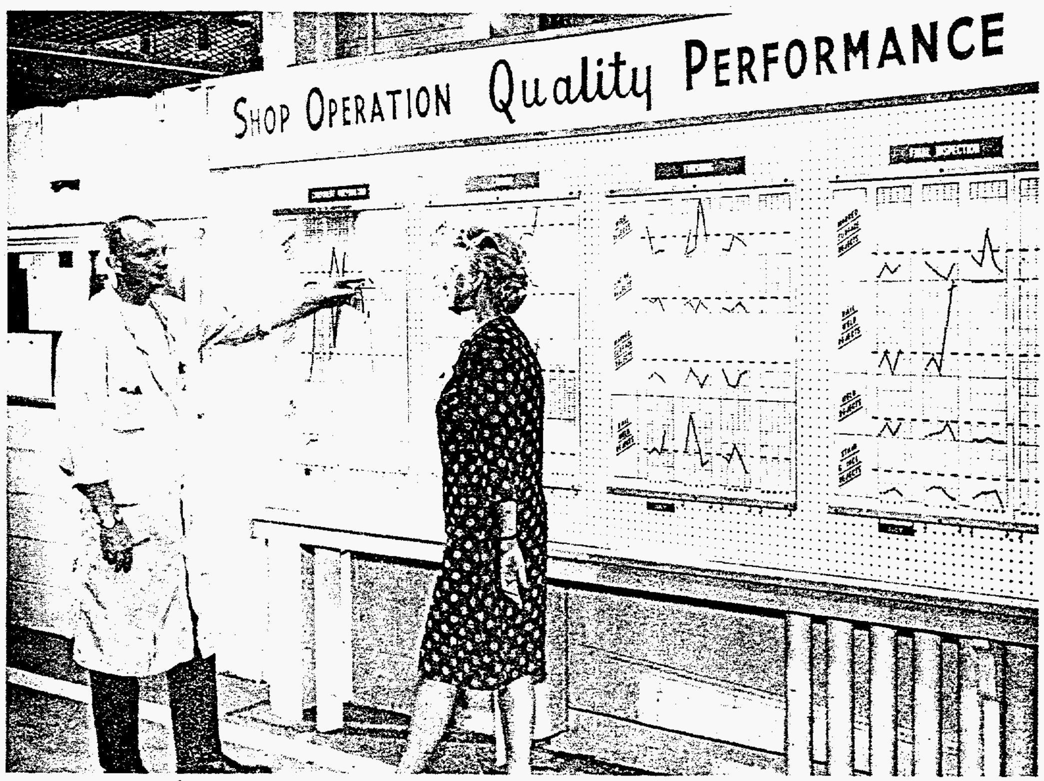

Figure 10. In-plant Quality Posting (changed daily)