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MONTHLY REPORT

AUGUST, 1968

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September 16, 1968

MONTHLY REPORT

AUGUST 1968

DOUGLAS UNITED NUCLEAR, INC.

Richland, Washington

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TECHNICAL ACTIVITIES

C & K Reactors

Radiometallurgy Laboratory examination of the overheated fuel and tube from flow-stoppage channel 3560 of KW Reactor is proceeding satisfactorily. Surprisingly, two fairly central elements still had a large amount of unmelted and unreacted aluminum cladding, due presumably to their proximity to control rods. The process tube had numerous splits and cracks, not all of which appear to have occurred during tube removal. The metallurgy work is expected to be completed and reported by November. A document exploring the nuclear safety implications of 3560-KW observations relative to previous technology is also being prepared.

The loading pattern design for the 10 kg plutonium irradiation test has been changed from a compact configuration to a slab pattern. Although the total number of enrichment and poison columns in the flux-adjusting buffer zone will be increased, the slab configuration provides definite advantages in greater coverage by vertical safety rods and splines, reduced localized coefficient effects, and greater calculational simplicity. Canning of the Pu-Al fuel cores fabricated by BNW is progressing well in the Transuranium Pilot Plant. Reactivity testing has confirmed predictions.

The shipping containers assigned to DUN for off-site shipments of radioactive materials have been evaluated, and modifications have been designed to bring these containers into compliance with the requirements of AEC Manual Chapter 0529. The HAPO-13, -14, and Reed College casks have been modified and are ready for use. AEC-RL is obtaining the required special permits from the Department of Transportation.

N Reactor

In-core thermocouple measurements of Mark IV subchannel temperature rise have indicated that the fuel performance is very close to design predictions. The center channel temperature rise will be slightly less than that in the bulk process tube coolant; this is a beneficial effect in view of the extreme sensitivity of center channel imbalance relative to variations in inner channel dimensions.

Irradiation experience to date on Mark IV test fuel has been generally satisfactory with minor anomalies noted in swelling data. Preliminary swelling performance observations over an observed range of exposures from 590 to 1,480 MWD/T show that (1) a slight increase in clad thickness does not appear to affect fuel swelling, (2) comparable 125 Metal fuels have a slightly lower swelling rate than 94 Metal fuels, and (3) uranium with iron and silicon added (alloy 503) has swelled at a lower rate than that with iron and aluminum added (alloy 601).

Inspection and testing of the meteorological monitoring system for the 100-N Area tower is underway at the vendor's plant. An addendum to the vendor's acceptance test proposal has been prepared by DUN and includes a requirement

that the system be operated continuously in the 5-second scan mode for at least 500 hours prior to final acceptance.

ADVANCED OPERATIONAL PLANNING

DUN manpower and cost inputs have been reviewed and approved for the new CAGE model. The new numbers reflect escalation to 1968 prices, better definition of manning levels for reactor operating levels spanning one to six plants, and consideration for a viable organization at the reduced levels.

IRRADIATION SERVICES

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GENERAL

The Company began participation in a Hanford-wide Training-Orientation-Placement Program (TOPP) with the employment of four minority "hard-core unemployables." In this new program, participants will spend part of each work day receiving basic classroom educational instruction administered by the Richland School District. The remainder of each day will be devoted to on-the-job training. The first group of TOPP trainees started the educational phase of the program on August 5. Classes for the second group started on August 19.

Company personnel had no disabling injuries in August, and no radiation exposures exceeded operational control.

Charles D. Harrington
Charles D. Harrington
President

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REACTOR PLANT OPERATIONS - C & 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 C, KE and KW Reactors are reproduced on the three pages which follow page B-8. The tonnages listed are approximate; actual fuel totals are given on page B-2.

The production of nondefense plutonium in the 8-12 percent Pu-240 range is continuing with both natural and enriched uranium.

The central zones of the K reactors remain charged with loads of thoria and 94 Metal, as required for the assured delivery of 360 kg of U-233.

Performance of the one-inch overbore test block at C Reactor continues to be incident free. No fuel failures have occurred in the six months since its initial loading in February.

PRODUCTION REACTOR STATISTICS — AUGUST 1968

REACTOR	C	KE	KW	TOTAL
INPUT PRODUCTION — PU — KMWD	47.9	71.7	112.4	231.9
— U-233 — EQUIV. MWD	1113	9152	12 248	22 513
POWER LEVEL (MW) — MAXIMUM	2100	4100	4285	—
— AVERAGE	1940	3933	4070	9943
TIME OPERATED EFFICIENCY — %	79.6	58.8	89.1	75.8
OUTAGE TIME ALLOCATION — %				
CHARGE — DISCHARGE	4.1	17.9	1.4	7.8
FAILED FUEL REMOVAL	0	1.7	0	0.6
WATER LEAKS	7.1	0	5.3	4.1
TUBE REPLACEMENT	0	0	0	0
OTHER MAINTENANCE	5.8	15.9	3.5	8.4
STANDARDS CHECK	3.3	2.7	0.6	2.2
PRODUCTION TESTS	0.1	1.9	0	0.7
PROJECT WORK	0	0	0	0
OTHER	0	1.1	0.1	0.4
TOTAL	20.4	41.2	10.9	24.2
NUMBER OF OUTAGES	2	4	1	7
NUMBER OF STARTUP INTERRUPTIONS	0	2	1	3
WATER LEAKS — TUBE	2	0	0	2
— VAN STONE	2	0	1	3
NEW TUBES INSTALLED	4	0	1	5
FUEL CHARGE — (TONS) — NATURAL URANIUM	146.9	40.2	111.4	298.5
— ENRICHED URANIUM	70.6	332.3	274.0	676.9
FUEL ELEMENT FAILURES	0	1	0	1
HELIUM CONSUMED — M CU. FT.	309.4	122.6	210.0	642.0
WATER TO REACTOR				
NORMAL OPERATING FLOW — GPM	106 400	207 600	209 800	523 800
pH	6.59	6.52	6.52	—
DICHROMATE — PPM	0.50	0.50	0.90	—

B-2

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Power Levels

Power levels at the C and KW Reactors were restricted by the bulk outlet water temperature limit of 95 C. At KE the power level was similarly restricted until the scheduled outage at mid-month; after startup, tube temperatures became limiting.

Reactor Outages

Seven outages were experienced as summarized below. In addition, two reactors were down a total of 72.9 hours on outages initiated in July.

<u>Date Down</u>	<u>Reactor</u>	<u>Outage Hours</u>	<u>Remarks</u>
August 4	KE	48.8	Failure of a bumpered 94 Metal element in tube 5073. Major work accomplished consisted of repairing RTDs, traversing 3Z test hole, removing an unserviceable graphite thermocouple stringer, and installing a flexible VSR in channel 29.
August 5	KW	41.5	The reactor was shut down when the reactivity held in the control rods was insufficient to adequately control flux distribution. Five tubes of thoria were replaced with natural uranium to provide the needed additional reactivity. Miscellaneous maintenance was accomplished during the outage.
August 8	C	56.9	Process water leaks in two process tubes. A rear Van Stone flange leak was found in tube 2177, and tube 4461 was leaking internally. Both tubes were replaced. Miscellaneous maintenance work and EMS checks were performed.
August 9	KE	66.4	Unexplained Panellit trip on tube 2355. Principal outage work accomplished consisted of charging 78 tubes of thoria and 403 tubes of 94 Metal in the buckled zone, and installing (1) Zone 16 orifices for thoria tubes, (2) a new graphite thermocouple stringer, and (3) a flexible VSR in channel 22.

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<u>Date Down</u>	<u>Reactor</u>	<u>Outage Hours</u>	<u>Remarks</u>
August 18	KE	158.0	Scheduled outage for charge-discharge. Major work accomplished consisted of charge-discharging 1,483 tubes, installing new venturi screens on tubes on headers 30 and 34, inspecting crossheader screens and replacing as needed, recirculating balls in nine 3X hoppers, repairing HCR Nos. 12 and 15, and taking a vertical traverse on tubes 1870 and 3870.
August 23	C	61.0	Process water leaks in two process tubes. Tube 2762 was removed because of an internal leak; the channel was left empty due to an obstruction. Tube 3956 was replaced because of rear Van Stone flange leakage.
August 30	KE	32.2	Technical high-limit trip on the high speed scanner caused by a faulty RTD on tube 1457. Outage work accomplished included RTD repair, nine loading changes to improve reactivity balance, and adjustments to the Panellit system and HSSS limits

EQUIPMENT EXPERIENCE

High Speed Scanner - KW Reactor

A significant number of failure lights were experienced on the isolated logic program synchronizer of the KW high speed scanner. As a result, continuous tapes are being run in an attempt to locate the source of the problem. Prime suspects are stray voltages and noisy amplifiers.

Process Water Pump - KE Reactor

A five-year EMS check was given to the No. 1 low lift pump at 190-KE. Slight cavitation was noted in the suction bowl and the unit was replaced.

Confinement Filters

The charcoal filters for B cell at C Reactor were repacked and satisfactorily passed the tests for halogen removal. The cell was left in standby status.

This completes the renovation program for the operating single-pass reactor confinement filters. A development test has been written to keep a quarterly

check on all installed charcoal. For this purpose, specially fabricated sampling filters have been designed and a prototype built to determine the iodine removal efficiency and study the pollutant uptake.

HCR-VSR Maintenance

At C Reactor attempts to remove the blockage in VSR channel 36 were unsuccessful and the rod remains out of service. Horizontal rods 1 and 6 continue to bind during rod withdrawal.

At KE Reactor, HCR No. 15 was replaced to correct a binding problem. Inspection of the No. 12 rod at KE, which had been sleeved in January, revealed that at least one-third of the sleeves were broken. New sleeves were installed, but some binding is still being experienced.

Exhaust Fans - KW Reactor

Maintenance work was required on both of the emergency diesels which drive the 105-KW exhaust fans. A cylinder failed and was replaced on No. 1 unit. The No. 2 unit was completely overhauled when it was found that the cylinder pressure was considerably lower than the manufacturer's specifications.

PROCESS ASSISTANCE & CONTROL

Operational Physics

The KW Reactor shutdown on August 5 was due to a lack of excess reactivity in the samarium dip. The reactor had been shut down previously (July 31) for the same reason, but the lack of reactivity at that time was assigned to water in the graphite stack and no loading changes were made. Although the water collection rate was in fact abnormal, conditions at the time of the second shutdown confirmed that the reactor was under-enriched, particularly in the top far side. Although the 94 Metal E-Q loading pattern was almost identical to that of a previous successful loading, five thoria columns were displaced with natural uranium, and the reactor was restarted. Subsequent operation has shown that the flux distribution and excess reactivity are optimum.

KE Reactor experienced significant cycling problems after the August 7 startup with high exposure fuel. The reactivity oscillations were eliminated in about 1.5 days. The cycling potential is relatively high with the E-Q load due to the high graphite temperature reactivity coefficient, and is highest when the fuel is ripe. The fuel exposure (and cycling potential) was considerably reduced by charge-discharge in the late August outage.

The C Reactor startup of August 10 progressed well, considering that a large water leak from the center of the tube 4461 (top near side) reduced the excess reactivity by about 8 mk. The tube had been pierced during the outage when an adjacent VSR channel was being cleared. The far side tubes were brought up to powers corresponding to turnaround level, and the water gradually was driven out of the shutdown near side. Near side tube outlet temperatures, which initially were practically at inlet, began to rise significantly after about 14 hours. As the near side tubes were allowed to approach powers corresponding

to turnaround level, the far side powers dropped about 20 percent. Control rod and spline movements were difficult to coordinate with demands at this time, as insertion was required for the near side (turnaround) and withdrawal for the far side (xenon buildup). A xenon-induced reactivity cycle ensued, but was damped-out in about one day, and the reactor was subsequently brought to equilibrium power in the normal manner. Three days of continuous physics coverage was provided for the startup.

Operational physics data of interest are summarized below:

	Reactors		
	C	KE	KW
Effective Central Tubes (ECT)*	1604	2146	2043
Flattening Efficiency** - August	0.85	0.69	0.73
- 12-mo. Average	0.84	0.71	0.71
Maximum Operating Time Permitting Scram Recovery - Hours***	8	11	11

*Reactor power level divided by the average power of the 10 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.

K Reactor RTD Problems

Thoria charges irradiated to date operate considerably cooler than uranium charges. This cooler operation has caused a significant increase in the failure rate of immersion-type RTDs associated with thoria charges at KE Reactor. Temperature sensor failures mainly result from low resistance to ground due to corrosion products and moisture penetration into the integral-cable RTD pin cavity. Moisture accumulation is minimized on high heat generating tubes, and conversely is aggravated on low heat generation tubes. The rate of RTD failure on thoria tubes is presently about 15 per month at KE Reactor, and may increase since the buckled zone and the remaining fringe thoria have been charged.

An addendum to PTA-129, "On-Reactor Evaluation of RTD Modifications," has been issued authorizing the installation of up to 300 additional development RTDs with epoxy-potted connectors. Results will be analyzed using the HSSTAP program noted below.

High Speed Scanner Temperature Analysis Program (HSSTAP)

The base calibration record tape required by this computer program for accuracy and noise computation was reconstructed from a backup tape after its

inadvertent destruction at the computer.

Assistance was provided in running HSSTAP to determine the effectiveness of a stop-gap development scaling amplifier in improving the temperature measuring capabilities of the KW scanner. The new amplifier was shown to have good accuracy and noise characteristics.

Criticality Studies

Updated criticality calculations for "C" and "J" aluminum-oralloy elements have given values of 219 and 93 elements, respectively, for a minimum critical size array; this corresponds to slightly over 3-1/2 kg of U-235 in each case (see Figure B-1, below). The "J" element value corresponds to the same figure previously determined experimentally for an 8 percent higher U-235 concentration, thus indicating the calculations to be conservative. The "safe mass" of 40 elements utilized in Process Standards will not be changed as a result of these calculations. Calculations have also been initiated on dry oralloy element arrays which, although not expected to limit handling procedures, may theoretically have the potential for criticality.

Figure B-1. Criticality Calculations for "C" and "J" Elements

Irradiation of Surplus B Reactor Fuel in C Reactor

A large number of tubes in C Reactor are charged with mixed B and C fuel elements, and there appear to be no problems associated with this type of charge. A supplement to the applicable production test authorizes the charging of full-length B metal charges in fringe zone tubes. It is expected to charge some of these during the next outage.

Production Fuel Performance

The production fuel element failure which occurred at KE Reactor is described as follows:

<u>Tube No.</u>	<u>Charge Date</u>	<u>Failure Date</u>	<u>Exposure</u>	<u>Failure Type</u>
5073-KE	5/25/68	8/4/68	763 MWD/T	Manufacturing Defect

Based upon preliminary examination, failure was attributed to poor weld integrity. The element did not exhibit extensive ledge or groove corrosion, nor was there any appreciable swelling.

The following table shows production fuel failure frequencies, as number per million elements discharged, for the 3-, 12-, and 24-month periods ending July 31:

	<u>3 Months</u>	<u>12 Months</u>	<u>24 Months</u>
Small Reactors - Natural Uranium	0	46	27
- 94 Metal	127	64	21
K Reactors - Natural Uranium	0	6	7
- 94 Metal	0	9	8
C Reactor - 1-Inch Overbore	0	-	-

Product Accountability

Numerical tables for yield and quality of plutonium as a function of exposure of depleted uranium were published and incorporated in the DUN accountability document for K5D elements.

Reactor Effluent Activity Data

The following table shows the total kilocuries per month discharge for six significant radionuclides in the reactor cooling water effluent during July:

<u>Reactor</u>	<u>As-76</u>	<u>P-32</u>	<u>I-131</u>	<u>Cr-51</u>	<u>Np-239</u>	<u>Zn-65</u>
C	1.7	0.22	0.018	5.5	2.1	0.2
KE	2.2	0.12	0.022	3.1	3.3	0.26
KW	0.1	0.05	0.002	1.2	0.4	0.0
Total	4.0	0.39	0.042	9.8	5.8	0.46

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<u>Zone</u>	<u>Tons</u>	<u>Material</u>
Central	107 36 (Includes 14 in overbore block)	Natural Uranium 94 Metal
Ring	12 19	Natural Uranium 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	14	Thoria ("X" designates tubes)
	174	94 Metal (for thoria support)
	21	Natural Uranium
	2	Special Depleted Uranium (PITA-048)
E Ring	48	94 Metal
	6	Natural Uranium
	1	Thoria ("X" designates tubes)
Fringe	10	Natural Uranium
Thoria Support	111	94 Metal
Thoria	16	Thoria

Loading Pattern - KE Reactor

B-B

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<u>Zone</u>	<u>Tons</u>	<u>Material</u>
Central	14	Thoria ("X" designates tubes)
	174	94 Metal (for thoria support)
	21	Natural Uranium
	2	Special Depleted Uranium (PITA-048)
E Ring	47	94 Metal
	6	Natural Uranium
	1	Thoria ("X" designates tubes)
Fringe	82	Natural Uranium
Thoria Support	53	94 Metal
Thoria	11	Thoria

Loading Pattern - KW Reactor

B-C

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REACTOR PLANT OPERATIONS - N

PRODUCTION

General

Reactor production, power level, and related statistics are tabulated below. Operation was resumed on August 12, following the scheduled outage which began on July 17.

Input production and time operated efficiency (TOE) for the past six months are shown on the following chart:

Statistical Summary

Input Production (KMWD)	- Total	53.1
Power Level (MW)	- Maximum	4000
	- Average	3417
Time Operated Efficiency	- %	50.1
Steam Availability	- %	48.5
Number of Shutdowns	- Scheduled	0
	- Unscheduled	7
Fuel Failures		0
Fuel Charge (Tons)	- 94 Metal	300.7
	- 125 Metal	65.2
	Total	365.9

Helium Losses	- M cu. ft ³	250.2
Fuel Oil Usage	- bbl.	15,959

OPERATING EXPERIENCE

Reactor Loading

The reactor loading at month end is shown on the front face map which follows page BN-7. All coproduct fuel has been discharged. Fifteen additional columns of 125 Metal fuel were charged at the request of WPPSS. Compensation for this spike is being negotiated by AEC-RL.

Power Level

Power level was administratively limited at 4000 MW throughout the month. The main steam header pressure was maintained at approximately 124 psig, being enhanced by the increased spike purchased by WPPSS (which permitted better flattening).

Reactor Outages

Seven outages were experienced, as summarized below. In addition, the reactor was down 285.6 hours for the scheduled outage initiated on July 17.

<u>Date Down</u>	<u>Outage Type</u>	<u>Outage Hours</u>	<u>Cause</u>
August 12	Unscheduled	7.8	Flow monitor low-flow trip apparently due to a leak on the ECW low-trip Barton-1653.
August 13	Unscheduled	3.2	Flow monitor trip due to a leaking toggle valve-3368.
August 17	Unscheduled	43.2	Manual shutdown to repair a severe water-wall leak in Foster Wheeler boiler.
August 19	Unscheduled	0.8	Flow monitor trip - power supply transfer.
August 19	Unscheduled	1.1	Flow monitor low-flow trip - 3268; toggle valve replaced.
August 20	Unscheduled	3.5	Flow monitor low-trip due to 5 amp special action relay trip which caused variation in primary pump speeds.
August 27	Unscheduled	25.7	Flow monitor low-trip. Trip of CN3 breaker caused by failure of FAR M2 relay in the post scram action circuit.

EQUIPMENT EXPERIENCE

Conversion of Final Acting Relays to DC Operation

EMP-261 authorized the conversion of all discontinuous and final action relays to 125V DC operation. This change was initiated because of an A bus transient caused by a lightning strike resulting in a transfer from A bus to B bus of the 120V AC panel DR. The DR panel feeds all of the discontinuous and final action relays and as they become momentarily deenergized, they opened the steam generator primary coolant bypass valves, reduced primary pump speed to 900 RPM, and reset the main steam pressure. This caused a severe upset on the primary system and scrambled the reactor.

This EMP was completed; however, there have been two subsequent relay failures in less than a week. The failures were attributed to bad relays. Investigation of the two failures revealed that the entire group of new relays drew excessive current due to low coil resistance. To prevent this from recurring, relays that did not meet the criteria established by testing were replaced. In the future, all Struthers-Dunn 125V DC relay coils used in N Area critical circuits will have to meet the newly established criteria.

Allis Chalmers Circuit Breaker Undervoltage Trip Service Replacement

The oil dashpot undervoltage trip devices have frequently failed to operate properly, and on many occasions have failed to trip the breaker upon loss of bus voltage. Failure to trip-off the loads fed by these breakers results in a very high starting current when power is restored.

Under EMP-289, replacement of the undervoltage trip device on the Allis Chalmers LA-25 breakers was completed. As each breaker was removed from service for scheduled maintenance, the undervoltage trip device was replaced with a time delay relay.

V-11 Valves

Mark II operators were installed on a total of 16 valves, five of which were repacked on the unit.

V-4 Valves

Major efforts were made to correct leaks in the V-4 valve bodies and to remove weld material around various body plugs. No work was done to correct bonnet leaks or welds as necessary tooling is not available. All valves were re-insulated with blanket insulation.

Circulating Raw Water (CRW) Pumps

The Class "A" overhaul of No. 1 pump was completed. It was necessary to metalize two sections of the shaft and replace the bearing sleeves on the third section. The bearing in the transition section and one spider bearing was replaced. Damage to the bowl liner was repaired by welding. A new bowl liner has been ordered but will require about six months for delivery.

On August 5, the No. 2 CRW pump motor sustained a short to ground on C Phase during an operational check. The motor was delivered to Tinling & Powell Co., Spokane, Washington, for disassembly to allow detailed inspection prior to going out for a repair contract bid. Failure of this motor resulted in operation of the plant with three CRW pumps throughout the month. The motor from No. 1 unit was moved to No. 2 unit to provide two B bus powered pumps, thus avoiding a power level restriction.

Foster-Wheeler Boiler

Third-party inspection of this boiler was completed. Repairs were made to a cracked skin plate and to the burner throats.

A severe water leak developed later in a three-inch water-wall tube high on the east side of the boiler. The leak was caused by a structural tack-weld pulling a hole approximately 3/8" in diameter in the tube wall. Repairs were made by welding the hole closed.

Traveling Wire Flux Monitor

The TWFM system has been performing remarkably well during the report period. All wires are currently operating following the replacement of No. 6 wire during the extended outage.

Rod Coolant Flow

Over a period of several months the coolant flow rate on several of the control rods decreased to near a minimum flow condition. Test PT-NR-103 was approved to permit chemically cleaning the coolant passages with a cold 10 percent chromic acid solution. Four rods were so cleaned successfully, with results as shown below:

<u>Rod No.</u>	<u>Flow Prior to Cleaning</u>	<u>Flow After Cleaning</u>	<u>Original Flow New Rod</u>
04	38	57	67
25	39	55	65
42	38	64	64
103	35	68	62

The above flow rates are in percent of flow (0 - 100% + 0 - 23 gpm) with the supply valve wide open, and the supply pressure at 218 psig. The original flow shown for new rods was obtained at 234 psig.

Tube Temperature Monitor

During the July-August scheduled outage nine inoperable all-tube temperature monitor RTDs were repaired, and 50 were removed from the local scanner to normal service. Twenty zone temperature monitor RTDs were repaired. At month end there were 21 all-tube temperature monitor RTDs failed with five nonreadable. Six zone temperature monitors were failed.

Rear Face Gas Seal

The rear face gas seal was found torn open again. Repairs were effected by cutting out the damaged section and installing a sheet of silicone rubber in its place. Silicone sealant was used to repair cracks in the seal section. Silicone rubber was injected into the full length of the cracked Omega seal covering the joint between crates No. 3 and 4. Following repairs the gas system was pressure tested at 2" H₂O. A decay test run from 2" H₂O pressure indicated a pressure decrease of 1" H₂O in 55 minutes. Gas usage rate at equilibrium conditions averaged 5,300 cubic feet per day.

PROCESS ASSISTANCE AND CONTROL

Operational Physics

The 15 columns of Mark IA spike added to the spike ring during the July-August outage have contributed favorably to optimum flattening and excess reactivity requirements for operation at 4000 MWt and 800 MWe with five cells. Operation at these levels was achieved, but unscheduled outages have precluded a sustained long run to date.

The reactor is being controlled with the inboard rods as for the more stable Mark II load. PT-NR-106, "Exploration of the Flattening Capability of N Reactor with a Tube-In-Tube Fuel," was written to authorize this rod configuration over the normal outboard Mark I pattern. The purpose of this effort is to attempt to prove that axial flattening can be achieved without axial cycling problems with Mark I type fuel loadings before this flattening technique is actually required for either higher powers or new fuel models. The inboard rod configuration has been successful to date in providing good control flexibility and axial flux flattening. While some small amount of flux shifting has been noticed, no significant shifts have occurred.

The five-cell operation continues to cause uneven flow distribution such that, although the tube outlet temperature flattening is very good, the tube powers on the right side of the reactor are somewhat higher than on the left side. This tube power imbalance leads to a deceptively low calculated flattening efficiency, while the outlet temperature distribution is actually quite flat in the central zone.

Operational physics data of interest are:

Effective Central Tubes (ECT)	796
Flattening Efficiency	0.79
Maximum operating time permitting scram recovery - hours	24*

*This was incorrectly stated as 2 hours in last month's report.

Enrichment for U-236 Compensation

As fuel is recycled to build up the U-236 content for enhancing Np-237 (and hence eventually Pu-238) production, the reactivity of the loading will decrease accordingly. Although the effect is not linear, and various computer codes appear to be affected differently by the resonance absorption effects, the calculated magnitude is in the range of slightly over 0.01 increase in U-235 enrichment required per 1000 ppm increase in U-236 concentration.

Chemical Waste Storage Tank - Inhibiting Corrosion

After the July decontamination of N Reactor piping, sodium hydroxide was added to the chemical waste storage tank within 4 hours of filling the tank with acid solution, giving a final pH of 10.1 two days later when the caustic addition was completed.

Corrosometer measurements show that no more than 0.2 mil corrosion occurred on the carbon steel piping during the time the acid solution was in the tank.

Hydraulic Fluids for Horizontal Control Rod System

The field of possible candidate hydraulic fluids for HCR system conversion has been narrowed to two types: the inverted emulsion (water-in-oil), and the water-glycol. Fluids of both types were selected to undergo simulated operation endurance tests. "Ucon Hydrolube 150-CP," the water-glycol fluid, was circulated in a test loop containing several HCR system valves and components. The 1705-N building HCR was converted to Dualube, Inc. "Protec-toLife" water-in-oil emulsion fluid.

Ucon Hydrolube 150-CP Test

Pump pressure in this test was maintained at 1100 psi, and relief valve pressure was varied from 750 to 1000 psi. Limits on fluid temperature ranged between 75 F and 140 F, again depending on the severity of orificing. The Marotta solenoid valve was timed to operate every three minutes to exercise the dynamic seals within.

The first problem was with pump cavitation, due to the suction strainer being plugged with sludge. Sludge collection plagued the rest of the test. A second failure occurred in the filter cartridge element when the fluid decomposed the element binder. The test was continued without filtration.

The test was terminated after 1375 hours of operation when the failure of solenoid valve elastomer seals allowed gross leakage of the hydraulic fluid. Disassembly and inspection of the valve disclosed eroded aluminum valve seats and extruded seals. The fluid manufacturer had indicated that it could have detrimental effects on unanodized aluminum and polyurethane elastomers.

These test results indicate that Hydrolube is not a likely candidate for the HCR hydraulic system fluid conversion.

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ProtectoLife Test

Coincident with loop testing of the Hydrolube, the testing of ProtectoLife (invert water-in-oil emulsion) was conducted in the converted 1705-N facility. Following a period of scram cycling during which the HCR was scrambled 1800 times, the high pressure scram pump was set to operate against a dead head at 3000 psi. This resulted in excessive heat build-up and necessitated a temporary shutdown for installation of heat exchanger coils. However, the cooling coils did not have sufficient capacity to remove the heat generated in the 3000 psi system; therefore, continuous operation has been limited to 1000 psi. At this pressure, a fluid temperature of 110-120 F was maintained.

Water content monitoring has continued with samples taken every 48 to 72 hours. In 1250 hours of operation, water content has decreased about 2-3 percent. Immediate plans are for a shutdown at 1500 hours for distillation, viscosity, and separation tests on the fluid, and a complete system inspection. Filters will be analyzed and the pump will be disassembled and checked for wear. During this down time, a larger heat exchanger will be installed to allow future operation at 3000 psi. The remainder of the test (500 hours) will be run at the scram system pressure.

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	41	42	43	44	45	46	47	48	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64	65	66	67	68	69	70	71	72	73	74		
34																																			34	
33							C	C	C												C	C	C			C	C	C							33	
32						C	C		C				76	76	76	76			C				76	C	C		C	C	C						32	
31							X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	C	C					31	
30				C	C	C	X	X	X	X	X	A	X	E	X	A	X	A	X	A	X	A	X	X	X	X	X	X	C	C					30	
29			C	C		X	X	X	X	A	E	A	A	A	A	A	E	E	E	E	E	E	E	A	A	X	X	X	X	C	C				29	
28		C	C		X	X	X	X	A	A	A	A	A	E	E	E	A	A	A	A	E	A	A	A	A	A	A	E	X	X	X	C	C	C	28	
27	C	C	C	X	X	X	X	A	A	A	A	E	E	A	A	A	A	E	E	E	E	E	A	94	A	E	A	A	X	X	X	X	C	C	C	27
26	C	C	X	X	X	X	A	A	A	A	A	A	A	A	A	A	E	A	94	E	E	94	A	A	94	94	94	A	X	X	X	X	C	C	26	
25	C		X	X	X	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	E	E	A	E	X	X	X	C	C	25	
24	C	C	X	X	X	A	A	A	A	A	A	A	A	A	A	94	A	A	A	94	A	A	A	A	A	A	A	96	A	E	X	X	X	C	C	24
23	C	C	X	X	X	A	A	A	96	E	94	96	A	96	E	96	A	A	94	A	94	A	A	A	A	95	A	96	A	E	X	X	X	C	C	23
22	C		X	X	X	A	A	94	A	96	A	A	E	96	94	A	E	96	96	96	94	A	A	96	E	A	96	96	96	X	X	X	C	C	22	
21	C		X	X	X	A	A	A	A	A	A	A	A	E	94	A	E	A	E	A	94	E	A	A	A	A	94	94	A	X	X	X		C	21	
20	C	C	X	X	X	E	A	E	A	A	A	A	A	A	A	E	A	E	A	94	95	95	A	E	A	A	A	E	E	X	X	X		C	20	
19	C		X	X	X	A	A	A	A	A	A	E	E	94	E	E	E	E	A	95	95	A	95	A	95	E	E	A	A	X	X	X		C	19	
18	C	C	X	X	X	A	E	A	A	E	A	E	A	E	A	E	A	94	E	95	94	E	95	95	A	A	A	E	A	X	E	X	C	C	18	
17	C	C	X	X	X	E	A	A	A	A	A	A	E	E	A	94	94	94	94	A	94	A	95	A	95	A	95	95	X	X	X	C	C	17		
16	C	C	X	X	X	A	A	D	A	A	A	E	E	E	A	A	E	E	E	A	A	95	95	95	A	A	A	A	X	X	X	C	C	16		
15		C	X	X	X	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	95	A	95	A	A	A	A	A	X	X	X		C	15	
14	C	C	X	X	X	A	A	A	A	A	A	A	A	A	A	A	A	94	A	95	95	A	94	A	A	94	A	A	X	X	X		C	14		
13	C	C	X	X	X	A	A	A	A	A	A	A	A	A	A	94	A	94	A	A	E	94	A	E	A	94	A	E	94	X	X	X		C	13	
12	C		X	X	X	A	A	E	A	E	A	E	A	A	A	94	94	E	94	E	A	A	A	95	E	A	A	A	X	X	X	X	C	C	12	
11	C	C	X	E	X	A	E	E	E	E	A	E	E	E	E	E	A	A	A	A	E	A	A	E	A	E	E	A	E	X	X	X	C	C	11	
10	C	C	X	X	X	A	E	94	E	E	E	E	E	A	E	E	E	A	A	A	A	E	E	A	A	E	A	A	X	X	X	X	C	C	10	
09	C	C	X	X	X	X	A	E	E	A	A	E	A	A	A	94	94	94	A	A	A	A	A	94	A	A	A	X	X	X	X	C	C	09		
08	C	C	C	X	X	X	X	E	A	A	A	A	A	A	A	A	E	A	A	A	94	94	E	A	E	A	X	X	X	X	C	C	C	08		
07		C	C		X	X	X	X	A	A	A	A	A	A	A	A	A	E	A	A	A	A	A	A	E	X	X	X	X	C	C	C		07		
06			C	C		X	X	X	X	A	A	A	A	A	A	A	A	A	A	A	A	A	A	E	A	X	X	X	X	C	C	C		06		
05				C	C	C	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	A	X	X	X	X	C	C	C		05			
04					C	C	C	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	C	C	C		04			
03							76	C			C	C	76	76	C		C	C	C	76	C		C		C	C	C	C					03			
02							C	C	C			C	C	C		C	C	C	C	C	C	C	C	C	C	C	C	C					02			
01																																			01	

Fuel Code	No. Tubes	Description	PT-NR No.	No. Tubes	Description
	57	MK-IC(94 Metal Fringe Normal)	76	9	Fuel Meltdown Test
A *	349	MK-IC (94 Metal Central)	94 +	46	MK-IV Demonstration
C **	158	MK-IC (94 Metal Fringe)	95	22	High U-236 Content Fuel
D	1	MK-IC (71 Metal)	96	15	MK-I from Upset-Forged Billets
E	114	MK-IC 94 Metal (High U-236)		1	Blank Channel or Empty Tube
X ***	232	MK-ICA (125-95 Metal)			
	911	Total		93	
	93	Total PTs			
	1004	Grand Total			
* 50% NFS ** 100% NFS *** 94 Metal 25% NFS			+ Includes 23 tubes MK-IV High U-236 content fuel.		

Loading Pattern - N Reactor

BN-A

FUEL AND TARGET FABRICATION - C & K REACTORS

PRODUCTION

General

Production of AlSi-bonded fuel elements for the C and K reactors was 102.3 percent of forecast. Of the 254 tons produced, 200 tons (79 percent) were fuels with bumpers or self-support rails attached. There was no production of hot-die-sized fuel during the month.

Thoria target element production continued on schedule.

Acceptable Elements Produced

	<u>Input Production</u>	<u>Finished Production</u>	<u>Yield - Percent Current Mo.</u>	<u>FYTD</u>
AlSi-Bonded Elements (tons)	282.7	253.6	95.1	95.0
Thoria (pieces)	11,355	12,242	91.2	92.0

Inventory

Bare Uranium Cores	886 tons
AlSi-Bonded Elements	1,458 tons
Thoria Elements	21,855 pieces

OPERATING EXPERIENCE

Overall operating efficiency of the canning lines during August was 98.9 percent. Downtime was distributed 25 percent to equipment malfunctions and 75 percent to operations. Canning line operation continued throughout the month at the rate of 4 line-shifts per day until August 12, and 3 line-shifts for the balance of the month.

The overall manufacturing yield for the month was 95.1 percent.

EQUIPMENT EXPERIENCE

Nothing significant to report.

PROCESS ASSISTANCE AND CONTROL

Fuel failures in the K reactors caused by corrosion of the 32-mil thick aluminum supports welded on the elements inspired a program to evaluate the use of thicker stock for arch self-supports. This program resulted in modification of the three welders to use supports fabricated from 50-mil thick stock for K-5 fuel elements.

FUEL AND TARGET FABRICATION - N REACTOR

PRODUCTION

Statistical Summary

Input Production

Total billets extruded	62
Tons extruded	12.1
Percent of forecast	100.8

Output Production

Total finished assemblies	1557
Tons output	26.1
Percent of forecast	87.0

<u>Uranium Utilization</u>	27%

Inventory

Bare uranium billets	329 tons
Finished fuel in storage	268 tons

OPERATING EXPERIENCE

Assembly of finished fuel was discontinued on August 14 to investigate the problem of cracked outer supports reported under Engineering and Technology, on page DN-6.

EQUIPMENT EXPERIENCE

Auditing of the heat treatment operation following brazing showed that the new liner recently installed in the salt-bath furnace improved the in-furnace and transfer fuel temperatures. A thermocoupled audit fuel indicated a marked reduction in the temperature gradient between the top and bottom of the salt-bath.

PROCESS ASSISTANCE AND CONTROL

Braze Performance

Production braze performance was closely monitored during the past month in an effort to improve process control. A run of 125 Metal outer fuels was brazed using cast rings with tack-ons. Two small tack-ons were used instead of a single large one, which seemed to promote a more uniform melt-in of the

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additional braze metal necessary to meet the weight requirements. No significant problems were encountered throughout the entire run. The reject rate at braze was 0.84 percent for 2,628 ends brazed. The reject rate at end-test was 0.92%. The reject rate for all braze related rejects was 1.75 percent during the run of 125 Metal outers.

The braze stations were converted to inner fuels and a run of 94 Metal inners commenced on August 5. A new lot of powder braze rings was used at that time. Data calculated to August 19 shows that the braze reject rate was 1.29 percent for 5,014 ends brazed, and the end-test reject rate was 2.45 percent. This represents a total braze-related reject rate of 3.30 percent.

The use of the powder braze rings was reviewed to assure that the braze stations were properly adjusted and that strict adherence to the Quality Control Standards was maintained. Removal of the press flash from the powder braze rings was continued to provide the closest possible contact of the braze ring to the end cap, thus providing a more uniform rate of heating during the braze operation.

TECHNICAL ACTIVITIES - C & K REACTORSRESEARCH AND DEVELOPMENTMission 1 - Basic Production1-A. Analysis of Overheated Fuel from Process Channel 3560-KW

The fuel and process tube sections from flow-stoppage channel 3560 of KW Reactor are being examined in detail, first in the KW technical viewing pit and then in BNW's Radiometallurgy Laboratory, to provide empirical bases for temperature calculations and nuclear safety predictions.

Fuel elements from positions 1 through 5 (downstream) and 44 through 51 (upstream) have been examined in Radiomet. The element from position 1 showed AlSi melting (approximately 580 C) near its upstream end. An area near the midpoint in the element from position 3 operated in the beta range of uranium (668 to 774 C). A metallographic sample of the cross-sectional area at the midpoint of element No. 3 revealed two visually distinct inter-action layers on the outer surface of the element and two within the bore. The inner bore was approximately 90 percent blocked by reaction products at this particular cross-section.

The process tube sections were arranged side-by-side in the order they were located in-reactor, and then photographed to provide a record of relative tube condition. Numerous splits and cracks were noted in the process tube, not all of which appeared to have occurred during removal. Photographs were also taken of tube sections which were unusually or badly distorted; the process tube was generally bulged at the areas corresponding to the ends of fuel elements.

In an effort to remove these elements from the tube, the tube was cut into sections containing two or three elements, with cuts between elements. A red, rust-colored powder was found in the tube along with silvery metallic-looking flakes and particles believed to be UAl_3 . The elements were not removed without first removing nearly all of the loose material surrounding the element. This material was judged to be uranium-aluminum intermetallics, but this has not yet been confirmed.

As of report time, 19 fuel pieces had been removed from the process tube--from positions 13 through 15, 22, 25 through 35, 39 through 41, and 43, not including 13 elements already in Radiomet. Nearly every element had a considerably smaller outside diameter than undamaged elements. Surprisingly, two of the elements, those from positions 40 and 41, revealed unreacted cladding. The rail supports on element 40 were still intact, and much of the cladding of element 41 did not appear to have reacted with the uranium, whereas elements 43 through 47 showed evidence of extensive aluminum-uranium reaction. Element 40, along with elements 13, 22, 29 and 35, will be sent to Radiomet for further examination.

A cut between elements 17 and 18 did not reveal any powder as did previous cross-sectional cutting between elements. Instead, a solid, metallic-looking material was seen. This material is probably either UAl_3 or metallic aluminum. These elements could not be removed from the process tube and will be sent to Radiomet for tube removal.

Random samples from a test run on unirradiated aluminum-clad fuel in Zircaloy-2 tubing, to simulate what occurred in channel 3560-KW, revealed both UAl_3 and UAl_4 . The former is silvery, whereas the latter is a black powder.

1-B. Non-Uniform Corrosion of Aluminum-Clad Fuel

Testing of the 18 aluminum alloy cladding candidates in the autoclave and TF-18 flow loop is essentially complete. This phase of testing was to compare the uniform corrosion rates of the alloys and screen out those with poor performance. The best performers were then to be evaluated with respect to non-uniform corrosion. The flow loop test, which is considered representative of reactor conditions, showed significant corrosion difference between the alloys. None could be eliminated on the basis of this test.

An earlier erosion-corrosion study of 8001 and 1245 aluminum alloys (HW-74358 REV, April 1963) had indicated 8001 to be more susceptible to erosion-corrosion, or grooving, and that an exposed area of AlSi braze was several times more erosion-corrosion resistant than 8001. Further testing of the 18 alloys will be conducted to compare their erosion-corrosion with that of 8001. These tests will utilize an impingement-type device similar to that used in the referenced studies.

A standardized uranium core for both hot-die-size (HDS) and AlSi processes will soon be tested in-reactor. The universal core will have a thicker aluminum cladding for the HDS fuel, hopefully reducing the HDS failure rate from ledge and groove corrosion. The test will provide striped columns of K5E (standard AlSi), K5AE (old HDS) and K5AE-A (standard HDS) elements for comparing erosion-corrosion behavior of AlSi and HDS fuel.

1-C. Uniform Cladding Corrosion Studies

Weight-loss data from fuel elements irradiated at 130 and 160 percent specific power conditions in the K reactors have been satisfactorily described as exponentially proportional to temperature. Using the new equation, in place of the corrosion-rate model incorporating heat flux directly, it has been possible to predict the average cladding corrosion within 0.2 to 0.3 gram of that measured. The new relationship will be tried against other production test and specific power data.

1-D. Zircaloy Process Tube Hydriding

Flexure tests of two hydrided tube sections from KE Reactor have been run to 5×10^5 cycles without apparent failure; hoop and tensile stresses will also be applied.

Although the average hydriding in C Reactor 0.550 inch overbore tubing appeared lower than in K tubes, one tube has been found with a 25-mil case layer and over 5000 ppm total hydrogen, the thickest case layer discovered to date. A correlation is being attempted with hydride content and fabrication history, whether by drawing or by tube reduction.

The total hydrogen content of Zircaloy tubes with 52-58 months' exposure and equipped with unanodized spacers was observed to be lower than concentrations in similar tubes with 42 to 44 months' exposure. This result is anomalous in that base metal concentrations were higher in the 52 to 58 month case, implying partial loss of the hydride case which has not been observed.

1-E. Reinforcement of Reactor Piping

The use of a fiberglass-reinforced plastic as a method for reducing the failure potential of in-service reactor piping is being examined. A "spray up" technique, whereby a mixture of resin, catalyst, and chopped glass fibers can be coated to any desired area and thickness, appears especially attractive.

Equations relating to the stresses in the steel pipe wall and fiberglass coating to the coating thickness, application temperature, coefficient of thermal expansion and creep characteristics indicate potential stress reductions of 15 to 40 percent, depending on coating thickness and the time following coating application.

1-F. Physics Code Improvements

Two discrepancies discovered in HAMMER program use have been resolved. The first, in the THERMOS section, was not discovered sooner as it affected only very small systems with inputted bucklings. The second problem was associated with an inconsistent normalization of fluxes when the five energy groups are combined from the THERMOS and HAMLET programs. Fortunately, the results of the particular problems run on these programs prior to observation of the discrepancies were not significantly affected by these discrepancies. Further HAMMER code work included the continuing addition of resonance parameters to its cross section libraries as pertinent to DUN uses.

The GEM III MONTE CARLO code was set up for an uranium rod water lattice using the square hole routine. This type of code could be more versatile for handling criticality and other extreme geometry cases. The problem was run successfully for a limited number of neutron histories. The measure of reactivity was within 7 percent of the experimental answer for 1000 neutron histories.

Mission 2 - Coproduct

Lithium-Aluminum Splines

The testing of lithium-aluminum splines, fabricated by Nuclear Metals, was begun in KE Reactor. The splines are to be irradiated for 20 days prior to the first inspection to determine irradiation effects.

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Mission 3 - Transplutonium Technology

3-A. Plutonium Irradiation

Fuel Fabrication

A total of 325 machined plutonium-aluminum cores have been fabricated by BNW and delivered to DUN for canning in the Transuranium Pilot Plant. Canning operations are progressing rapidly with welding and helium leak checking essentially complete. Magneforming is complete on one-half of the elements. The only canning operations remaining are weld radiography and reclaiming of weld rejects. Testing of completed elements in the Hanford Test Reactor confirms reactivity predictions. A rough draft of the production test is to be issued for comment shortly.

Physics Planning

Design of the loading pattern in a K reactor for the 10 kg plutonium irradiation for transplutonium data is being based on a slab configuration for both the PuAl columns and the adjacent buffer regions. Although a somewhat larger loading of enrichment and target material is required for flux tailoring around the PuAl columns than for the previously-considered cylindrical plateau concept, the horizontal slab approach has several potential advantages:

- Greater calculational simplicity and hence accuracy.
- Increased availability of vertical safety rod and spline tube coverage with less severe effects per VSR or spline within the zone.
- Dispersion of local effects of the larger graphite coefficient resulting from the PuAl columns.
- Greater flexibility for removal of some of the PuAl columns as desired without upsetting the buffer geometry.

Although about 50 percent more enrichment and target columns will be required in the initial slab buffer region for creating a flux depression than for the cylindrical buffer, the difference in the flux enhancement buffer toward the end of the loading would be closer to 20 percent. In actual practice, the larger fuel and target buffer will simply replace a larger number of displaced enrichment and target columns in the E-Q loading during the first year or irradiation.

A RIBD code case to determine the fission product inventory for a PuAl column representative of the PuAl loading has been run and the results have been made available for nuclear safety considerations.

Mission 4 - Pu-238

In addition to the 180 grams of americium of mixed isotopic composition

recently authorized for irradiation in the Hanford reactors, the possibility has been raised that some pure americium-241 might also be available. Order-of-magnitude calculations have been initiated to estimate the potential improvement in prediction accuracies which might be attained with pure americium relative to data from the mixed americium (78 percent 241, 21 to 22 percent 242). Results of these calculations will be considered in relation to potential prediction improvements versus added costs to determine Hanford interest in the pure americium-241.

Mission 5 - Other Isotopes

Uranium-233

Sintered thoria wafers are being considered as an alternate to thoria powder as target material for producing U-233. To more closely define temperatures, a series of metal alloy temperature monitors are being fabricated for inclusion in a production test of thoria wafers. (See also wafer development, on page D-7).

Mission 7 - Target Space Enhancement

Relative heat generation data have been calculated for the planning of high power density tests utilizing 1.5 and 1.7 percent enrichments with both aluminum and Zircaloy cladding. Although the columns will have to be shortened somewhat to accommodate the expected 30 percent increase in specific power (relative to adjacent 94 Metal), and charging procedures will have to ensure against the loading of a 20-column, critical-size region, the proposed tests will not impose added speed-of-control or total control restrictions.

Mission 8 - Nuclear Safety

Technical Evaluation of Tube 3560-KW Flow Stoppage

Data from the 3560-KW tube flow stoppage incident, particularly the physical effects on the fuel and the process tube (see page D-1) are being examined closely for nuclear safety implications. The document to be issued following the study will evaluate the validity of theoretical models for predicting post-accident events, including the aspects of temperature transient analysis, fuel and tube behavior under accident conditions, fission product release, accident probabilities and future prevention.

Mission 10 - Columbia River Studies

10-A. Analysis of Columbia River Water

Activation analysis of a sample of Columbia River water defined the existing concentrations of various metals. In general, the observed concentrations were comparable with those measured in 1963, except that zinc, cobalt and scandium were higher by respective factors of 4, 2 and 10.

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10-B. Film Studies

The analysis of film removed from two downstream spacer elements from a KW Zircaloy process tube showed that the activity per unit surface area was lower by one to three orders of magnitude than a previous analysis of film from spacers irradiated in an aluminum tube had shown. It is postulated that the difference is due in part to the difference in process tube material.

ENGINEERING AND TECHNOLOGY - REACTORS

EMS Computer Program

The four sections of the Equipment Maintenance Standards (EMS) computer program have been completed and checked-out with a wide variety of trial data. The first draft of a writeup of the program has been completed.

Smoke Density Monitors

To meet the criteria set forth under Title 42 CFR 76, which implements Executive Order 11282, "Control of Air Pollution Originating from Federal Installation," smoke density monitors are required on all fossil fuel burning facilities.

A general design for the installation of smoke density monitors at all DUN fossil fuel burning facilities is progressing. The design incorporates photo-electric type monitors with integral alarm and recorder components. A procurement specification and a request for purchase of the monitors are being processed.

Shipping Container Evaluation

The shipping containers assigned to DUN, used for off-site shipments of large quantities of radioactive and fissile material, have been evaluated and modifications have been designed to allow these containers to meet the requirements of AEC Manual Chapter 0529.

Document DUN-4250, "Evaluation of Shipping Containers for Off-Site Shipments of Radioactive Isotopes," was distributed together with a letter requesting AEC-RL to obtain DOT special permits for HAP0 13, 14, and Reed College casks. The evaluation has been accepted and AEC-RL has prepared Certification-Approvals to request the DOT special permits. The modifications to the containers have been completed, and they are ready for use as required.

Uncooled Control Rod

The uncooled developmental control rod being manufactured by General Electric at Nuclear Systems Programs (formerly NMPO - Cincinnati) is largely complete and expected to be ready for examination in September. The possibility of improved cermet creep strength, hopefully to be achieved by vendor funding using appropriate additions of molybdenum and/or tungsten, will also be discussed.

K Reactor Secondary Coolant System Reliability

A technical bases document is being issued to support a Process Change Authorization to remove the steam turbine pump from service until the logic circuit is improved. A recent reliability study on the K secondary system disclosed weaknesses in the logic circuit controlling the backup steam turbine pump (utilizing steam directly from the boiler) which degrade the overall system reliability; the secondary backup is provided principally by operation of the low-lift pumps using AC power from turbine-generator sets driven by oil-fired boilers.

Confinement Studies

An engineering study of the Hanford K Reactors to determine the modifications required to comply with 10 CFR 100 and the Nuclear Power Reactor Siting Criteria continues.

Vitro/HES is continuing work on a series of layouts of improved effluent disposal facilities, including piping, diversion valves, quench tank, and ground disposal crib. Considerable time has been spent the past month in coordinating the efforts of the Vitro work.

A draft report by A. K. Postma of Battelle Northwest entitled "Atmospheric Release of Noble Gas Isotopes from a Coolant Disposal System Following a Reactor Accident," was reviewed and commented upon. This document provides a basis for evaluating the improved confinement system and will be issued after certain rework.

Project Engineering - Reactor Facilities

The status of approved construction projects relating to single-pass reactor facilities is summarized in Appendix A.

ENGINEERING AND TECHNOLOGY - FUEL & TARGETS

Production Test Analysis (PTA) System

A combination users manual and systems definition of the PTA system was completed for publication. For this computer code system, designed to analyze the operating characteristics and performance of production test fuels in the single-pass reactors, the manual explains in detail the system input requirements sources of input data, methods of calculation, and output reports.

Thoria Wafer Development

Fabrication of the thoria wafer target elements for irradiation testing was completed. The thoria used to produce these wafers was shipped from Hanford to NLO in the form of thorium nitrate. The uranium was removed by solvent extraction, a sol gel oxide was prepared by the precipitation process, and sintering at 1250 C produced the material used for wafer preparation at Hanford.

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In spite of this extensive processing and three years of aging, fission product activity in the form of Ru-106 was detected during the wafer preparation in Building 3722. However, no unusual personnel exposures were encountered during production of the 2000-pound test run. All material for the reactor test met specifications except 100 pounds which had expanded beyond the 1.499-inch upper diameter limit.

"AlSi Slop" Study

In the AlSi canning process the aluminum can is protected from the molten AlSi braze solution by inserting the can in a steel sleeve. When thermal expansion is complete, the upset metal ring on the can forms a mechanical block which prevents molten AlSi from flowing between the can and the sleeve. In the event molten AlSi passes beyond the upset metal ring to a point where it cannot be removed by a facing operation, this reject is called an "AlSi Slop."

Comparing the variability of can dimensions from two vendors indicated that one vendor consistently fabricated cans slightly below nominal diameters. These cans resulted in a 10 percent reject for AlSi Slop, as compared to a 1 percent reject rate for cans slightly above nominal dimensions. Vendors have been requested to keep the OD dimensions of cans above nominal, if possible.

TECHNICAL ACTIVITIES - N REACTOR

RESEARCH AND DEVELOPMENT

Mission 1 - Basic Production

1-A. Mark IV Fuel Development

Irradiation Testing

As of reporting time, N Reactor contained 23 columns of production test Mark IV fuel and 23 columns of production model Mark IV fuel. These 46 columns, being irradiated under PT-NR-94, include 25 monitor columns which are to be discharged at various exposure levels for determining fuel performance.

To date, 11 production test-fuel monitor columns have been discharged for evaluation. Preliminary swelling performance observations on those columns (over the observed range of exposures from 590 to 1480 MWD/T) are summarized as follows:

- A slight increase in clad thickness has not appeared to have affected fuel swelling.
- Comparable 125 Metal fuels have shown a slightly lower swelling rate than 94 Metal fuels.
- Uranium with iron and silicon added (alloy 503) has swelled at a lower rate than that with iron and aluminum added (alloy 601).

Swelling rates to date for each type of Mark IV fuel and a comparison with Mark I are summarized below:

<u>Type Fuel</u>	<u>Maximum Swelling Rates</u> (% ΔV /1000 MWD/T)	
	<u>Inners</u>	<u>Outers</u>
Mark I	0.75	1.00
Mark IV		
Thick clad (nominal 30/25)	0.80	0.70
Base clad (nominal 25/20)	0.70	0.80
Base clad (125 Metal)	0.50	0.75
Base clad (with alloy 503)	0.35	0.50

Flow Loop Testing

Life testing is continuing on Mark IV fuels with end-spider inner supports, with fully rounded solid supports, and on short Mark IV fuel elements. To date, there have been no indications of any support system malperformance. The nine end-spider supported fuel elements under test have experienced a

total of 2.9×10^8 cycles, approximately 30 percent of their expected in-reactor service life.

Physics Aspects

A document on calculated lattice heating with the Mark IV fuel has been completed in final draft stage. As might be expected, the proportionate heat generation in the moderator is calculated to be slightly less in the Mark IV case than in the Mark I loading with its slightly lower uranium per foot content. Calculations are proceeding on Mark IV speed-of-control calculations for hazards analysis purposes, and the desired physics features of a spike element for support of Mark IV loadings at levels above 4000 MW are being determined.

Channel Flow Balance Data

In-core thermocouple measurements of Mark IV subchannel temperature rise have indicated that the fuel performance is very close to design predictions. As shown in the table below, which compares the averages of reactor measurements with predicted values, the in-reactor observations show the center channel temperature rise to be slightly less than that in the bulk process tube coolant. This will be beneficial in terms of elimination of concern over the extreme sensitivity of center channel imbalance to variation of dimensions of the inside diameter of the inner fuel cylinder.

MARK IV ENTHALPY IMBALANCE DATA

(Ratio: Enthalpy Rise in Subchannel
to Enthalpy Rise in Fuel Column)

	<u>In-Core Measurements</u>		<u>Design Prediction</u>
	<u>Average</u>	<u>Deviation from Average</u>	
Center channel	0.954	+ 3.5%	0.992
Inner annulus	1.034	+ 0.3%	1.053
Outer annulus			
Bulk	0.977	+ 0.3%	0.983
Bottom	1.048	+ 1.4%	1.052

1-B. Mark I-C Flow Balance Data

Mark I-C flow split measurements, obtained with thermocouple trains installed during the recent long outage, agree precisely with those obtained with the first Mark I-C prototypes.

Flow split measurements with 12-inch Mark I-C fuel with spring-stop support systems indicated that the center channel enthalpy imbalance would be 15 percent, which is 5 percent higher than for Mark I-C equipped with buggy springs. This resulted from the much lower pressure drop across the spring-stop system, causing the inner annulus to rob flow from the other two sub-

channels. This imbalance in enthalpy, together with an increase in flow resistance (because of the large number of supports and junctions in the flow stream) will require selective loading of full-length columns of 12-inch Mark I-C elements.

1-C. Zircaloy Process Tube Studies

Three process tubes were borescopically examined in-reactor during the recent long outage. No significant damage was observed, although wear rings about 1 mil deep and 2 inches wide were found in a process tube which contained a variable orificing device being tested.

Mission 2 - Coproduct

Breached Target Detection

As a result of an inspection for breached elements, employing a weighing procedure, 246 lithium target subassemblies were separated as suspect from approximately 2400 pieces. The suspect pieces were visually examined and subsequently reweighed using a slightly expanded range for acceptable subassembly weight based upon a review of the pellet lot densities associated with the subject targets. The reject targets are to be stored in a separate basket for offsite shipment for processing.

Mission 5 - Other Isotopes

Thulium Target Irradiation

The three thulium target elements in graphite cross-coolant channel No. 1689 were discharged during the July outage at N Reactor. Calorimetric examination of the discharged samples is planned following their shipment to the C Area basin.

Mission 7 - Target Space Enhancement

Coating Formation on Graphite

The effect of temperature on the coating of graphite with tetramethylsilene was investigated. Graphite samples were held in tetramethylsilene for 5 hours in the same apparatus at temperatures of 650 C, 700 C and 750 C. Preliminary data show that (1) the thermal stability of the reactant gas appeared to be reasonable for the intended application, and (2) coatings can be produced at 650 C. The ability of this coating to inhibit oxidation has yet to be shown, however.

Mission 8 - Nuclear Safety

8-A. Meteorology Data System Preparations

An addendum to Packard Bell's proposal of an acceptance test for the N meteorological data system has been prepared. Included in this addendum are requirements that the system be operated continuously in the 5-second scan mode for at least 500 hours prior to final acceptance; this interval

was chosen for service testing on the basis of previous Hanford experience with mean-time-to-failure of tape recorders.

The computer program for storing, editing and reducing data from the N meteorological tower has been written and is now in the debugging stage. A test magnetic tape from the vendor (Packard Bell) has been shown to be comparable with CSC tape units on the UNIVAC 1108 computer.

Inspection and testing of the system hardware at the vendor's plant in California prior to shipment is in progress. Completion and use of this system by October is expected.

8-B. Metal-Water Reaction and Fission Product Release Tests

The gas mass flow meter, which is used to measure the amount of hydrogen generated during metal-water reaction experiments, has been successfully repaired on site and utilized in a series of experiments to measure the Zircaloy-steam reaction rate with unirradiated N Reactor process tubes in the range of 800 to 1200 C. Results of these measurements, although not yet correlated with empirical reaction models developed elsewhere, are shown by Figure DN-1 (next page) to be self-consistent and precise. Tests with irradiated fuel are planned for the near future.

8-C. Soil Studies

An attempt to measure ground water travel time from the 100-N crib to the river by using a pulse injection of irradiated sodium dichromate did not yield definitive results. Plans are proceeding for an alternate test utilizing dye.

ENGINEERING AND TECHNOLOGY - N REACTOR

Low-Flow Printout for Tests

A code for the ISI-609 computer to monitor 24 process tubes for low flow conditions during work on the V-4 valves is now operational. A digital display of each of the 24 flows is printed out 2 to 3 times per minute.

Machined Parts Cask

The design details, arrangements, and supporting calculations for using the Savannah River casks as containers to ship coproduct target elements have been documented in DUN-4633. This document contains information required to support the Safety Analysis, and includes final design data for the proposed method. A request for a shipping license is to be made.

Supplemental Control

Work has been initiated to scope the feasibility of using a ball system as a supplemental control for N Reactor. The system visualizes a ball system comprising an aluminum ball for spacing and possibly flux monitoring and a slightly larger ball of an aluminum-boron carbide mixture. The two types of balls could

Figure DN-1. Hydrogen Produced in Zircaloy-Steam Reaction

be used to charge any configuration of poison to any selected tube (using 3/8-inch diameter Zircaloy-2 tubes inserted in a selected number of the 3/4-inch diameter Zircaloy moderator cooling tubes). For total control, only the aluminum-boron carbide balls would be used; for flux monitoring, the aluminum balls would be used.

Procurement requests have been initiated to provide mock-ups of the proposed system for use in demonstrating the concept and obtaining data for system design.

Process Tube Replacement

The tube replacement procedure has been revised and is now in rough draft form. This procedure is intended to contain all of the information which would be needed for the replacement of all N Reactor process tubes.

The double-ended tube mandrel, intended for pushing an old tube from the reactor with the replacement tube, has been fabricated and tested. This tool has been returned to 189-D for slight modification.

Project Engineering - Reactor Facilities

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

ENGINEERING AND TECHNOLOGY - FUELS & TARGETS

Brittle Fracture of Outer Supports

One lot of outer supports for Mark I fuel showed a tendency towards brittle fracture in the area of the support shoulder. A similar condition was encountered with the Mark IV outer supports used for the irradiation test. At that time, the problem was felt to be a direct result of incorrect male-female die dimensions during fabrication; the male die was too large to allow complete filling of the female die without severe shear stressing of the support shoulder. This excessive deformation was evidenced by internal grain distortion and cracking in the stressed areas. Metallographic studies indicated that the Mark I outer supports had been subjected to the same type stressing as in the case of the Mark IV supports.

Since this condition existed for some time, preliminary findings indicate that fuels with cracked supports have been successfully irradiated. Investigations continue to determine the total number of similar fuels successfully irradiated. To eliminate the possibility of encountering similar incidents, the Quality Control inspection plan is being revised to include both bend testing and metallographic sampling of future support lots.

Extrusion Press Tooling Development

Mandrel costs are a major contributor to the overall cost of N fuels fabrication tooling, hence a search is under way for a more reliable type of tool steel. High-speed tool steel (T-1) has been used exclusively for production mandrels;

however, this steel is extremely brittle at low temperature, is difficult to heat treat, is expensive to rework, and its initial cost is high compared to steel used by other extrusion press users. Mandrels of H-13 tool steel have been tested to extrude such material as Zircaloy, copper-silicon and aluminum. On the Zircaloy billet, which was extruded from a 7.6-inch liner at 1250 F, the mandrel being preheated to 350 F, the mandrel appeared to be on the soft side; however, no dimensional or hardness changes occurred and this type of steel is ideal because of its excellent shock resistance. The copper-silicon and aluminum extrusions, using this mandrel, were quite successful. Tests using this mandrel are continuing.

Copper-Silicon Alloy

A development program has been initiated in Building 3722 to investigate the feasibility of producing copper-silicon alloy from the existing inventory of pure copper scrap. This alloy is used for casting billet preshapes, spacer rings and extrusion billets.

The first step was to prepare a master alloy of 10 percent silicon by adding pure, elemental silicon to molten 3 percent Cu-Si at about 1200 C. In the second step, appropriate amounts of this master alloy are added to molten copper at the same temperature. Prevention of oxidation and slag losses was accomplished by melting the copper scrap under 2" of granulated charcoal. This prevents the formation of copper oxide, removes the oxygen present in the scrap, and virtually eliminates slag formation during the alloying operation.

ADVANCED CONCEPTS AND PLANNINGSYSTEMS ANALYSISCAGE Mod 2-A

All new CAGE manpower and cost input has been reviewed and approved within DUN. Manpower levels by contractor and for reactor levels spanning the range of six to one are ready for Production Program Council review and transmittal to AEC-RL. Cost data for AEC review will be furnished by actual CAGE output.

In general, cost and manpower levels are higher in the new simulation due to escalation to FY 1968 prices, better definition of manning levels (particularly in the few-reactor cases), and consideration for viability. Marginal and residual unit costs will not be greatly different from those of Mod 1 for the 3-6 reactor level cases (mainly escalation effects), but substantial changes appear in the 1-2 reactor level cases. A document reconciling differences between Mod 1 and Mod 2 is being prepared.

Combined Operational Planning & Evaluation (COPE) StudiesFirst Priority Study Cases

All Priority 1 COPE cases have been completed. The cases submitted to date cover the following nondefense plutonium equipment assumptions: (Earlier submitted Cases 1 and 2, noted in June report, are now obsolete.)

Case 3: ZPPR, SEFOR, NPTF, and miscellaneous.

Case 4: ZPPR, SEFOR, NPTF, miscellaneous, plus FTR and three LMFB demonstration reactors stated as 12% Pu-240.
(Considered as base case.)

Case 5: Same as 4 except FTR and FBDB requirements stated as 25% Pu-240.

Case 6: Same as 4 except FTR (at 12%) requirements met only through FY 1973 and only one FBDB (at 12%).

The detailed requirement and return schedules are summarized in Table E-1 (next page). With four reactors, Hanford has ample capacity to meet all the requirements of the four cases except (limited to 94 Metal enrichment for E-D loadings) a 6,300 kg deficit exists for the 25% Pu-240 plutonium in Case 5. Use of a higher enrichment, 111 Metal, which is essentially an available technology, reduces the deficit to 3,800 kg. The higher enrichment case is being submitted as Case 5*.

Reactor assignments for Case 4 are shown in Figure E-1, Page E-3. The letter designations in the case numbers indicate tritium and isotope schedules which in these instances do not affect Hanford's loading assignments. The

TABLE E-1. 5-68 COPE NONDEFENSE Pu SCHEDULES (In Kilograms)

Program	Case 3		Case 4 (Base Case)		Case 5		Case 6	
	Minimum Nondef. Requirements		Min. + FTR + 3 FBDP as 12%		Min. + FTR + 3 FBDP as 25%		Min. + FTR Thru FY 73 + 1 FBDP as 12%	
	Require.	Return	Require.	Return	Require.	Return	Require.	Return
ZPPR	3800	300	3800	300	3800	300	3800	300
SEFOR	1000	1100	1000	1100	1000	1100	1000	1100
NPTF	1100	200	1100	200	1100	200	1100	200
MISC.	3700	6100	3700	6100	3700	6100	3700	6100
FTR			7400	3900	8900	4700	3000	2800
FBDP I			3200	1000	3800	1200	3100	1000
FBDP II			2600	300	3100	300		
FBDP III			3400	300	3400	300		
TOTAL	9600	7700	26200	13200	28800	14200	15700	11500
Stockpile at RL		2600		2600		2600		2600
Net Requirement	-700		10400		12000		1600	

REACTOR	PRODUCT	FY-69 ^(a)	FY-70	FY-71	FY-72	FY-73	FY-74	FY-75	FY-76	FY-77	FY-78
C	Pu(6% 240)			504 kg	504 kg	504 kg	504 kg	504 kg	504 kg	504 kg	504 kg
	Pu(9% 240)	125 kg	491 kg								
	Pu(12% 240)	347 kg									
	U-233	11 kg	11 kg	11 kg	11 kg	11 kg	11 kg	11 kg	11 kg	11 kg	11 kg
KE	Pu(6% 240)	376 kg	980 kg	982 kg	1033 kg	1130 kg	1130 kg	1130 kg	1130 kg	565 kg	1130 kg
	Pu(9% 240)	455 kg								540 kg	
	Pu(12% 240)	65 kg									
	Pu(27% 240)			125 kg	125 kg						
	U-233	159 kg	58 kg	11 kg	11 kg	11 kg	11 kg	11 kg	11 kg	11 kg	11 kg
KW	Pu(6% 240)	376 kg	980 kg	982 kg	1033 kg	1130 kg	1130 kg	1130 kg	1130 kg	565 kg	1130 kg
	Pu(9% 240)	455 kg								540 kg	
	Pu(12% 240)	65 kg									
	Pu(27% 240)			125 kg	125 kg						
	U-233	159 kg	58 kg	11 kg	11 kg	11 kg	11 kg	11 kg	11 kg	11 kg	11 kg
N ^(b)	Pu(6% 240)	57 kg								911 kg	1074 kg
	Pu(12% 240)	370 kg	885 kg	890 kg	896 kg	901 kg	906 kg	910 kg	910 kg	303 kg	
	Tritium	0.4 kg									

Note: Shaded area represents nondefense assignments.

- (a) The length of each major product block indicates only the relative annual share of the total annual production, not the absolute amount produced nor the portion of the year in which it was produced. The absolute amount in kilograms is indicated by the number next to or in each major product box.
- (b) Does not include Pu destined for shipment to NFS of 133 kg in FY-69 and 31 kg in FY-70.

Figure E-1. Reactor Assignments for 5-68 COPE Cases A-4-A,B,C; B-4-A,B; C-4-A,B.

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REACTOR	PRODUCT	FY-69 ^(a)	FY-70	FY-71	FY-72	FY-73	FY-74	FY-75	FY-76	FY-77	FY-78
C	Pu(6% 240)			504 kg	464 kg	464 kg	464 kg	464 kg	474 kg	504 kg	504 kg
	Pu(9% 240)	125 kg	419 kg								
	Pu(12% 240)	347 kg									
	U-233	11 kg	11 kg	11 kg					2 kg	11 kg	11 kg
	Tritium				1.3 kg	1.3 kg	1.3 kg	1.3 kg	1.0 kg		
KE	Pu(6% 240)	376 kg	980 kg	982 kg	986 kg	1015 kg	1015 kg	1015 kg	1044 kg	565 kg	1130 kg
	Pu(9% 240)	455 kg								540 kg	
	Pu(12% 240)	65 kg									
	Pu(27% 240)			125 kg	125 kg						
	U-233	159 kg	58 kg	11 kg					3 kg	11 kg	11 kg
KW	Tritium				2.5 kg	2.9 kg	2.9 kg	2.9 kg	2.1 kg		
	Pu(6% 240)	376 kg	980 kg	982 kg	986 kg	1015 kg	1015 kg	1015 kg	1044 kg	565 kg	1130 kg
	Pu(9% 240)	455 kg								540 kg	
	Pu(12% 240)	65 kg									
	Pu(27% 240)			125 kg	125 kg						
N ^(b)	U-233	159 kg	58 kg	11 kg					3 kg	11 kg	11 kg
	Tritium				2.5 kg	2.9 kg	2.9 kg	2.9 kg	2.1 kg		
	Pu(6% 240)	57 kg								911 kg	1074 kg
	Pu(12% 240)	370 kg	885 kg	890 kg	896 kg	901 kg	906 kg	910 kg	910 kg	303 kg	
	Tritium	0.4 kg									

Note: Shaded area represents nondefense assignments.

(a) The length of each major product block indicates only the relative annual share of the total annual production, not the absolute amount produced nor the portion of the year in which it was produced. The absolute amount in kilograms is indicated by the number next to or in each major product box.

(b) Does not include Pu destined for shipment to NFS of 133 kg in FY-69 and 31 kg in FY-70.

Figure E-2. Reactor Assignments for 5-68 COPE Cases B-4, 104-C; C-4, 104-C.

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Linear Optimizing Planning Estimate Reporter (LOPER)

Debugging of the linear program (LP) portion of LOPER is complete.
Some portions of the output logic still require programming effort.

LOPER should assist materially in the future in preparing planning estimates, all the way through from scheduling the reactors (using the LP

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portion) to printing out the detailed output required for the COPE studies. The matrix used for the LP covers an 11-year period, quarter by quarter, beginning with fiscal year 1969; it is described in DUN-AOP-27.

After the linear program solutions are obtained, the LOPER program will use auxiliary matrices of data and various equations to calculate, shift to approximate real time, and report the following: products made, product assays including Pu-241, products delivered, fuels required, fuel U-235 assays, targets required, fuel scrap returned, fuel scrap assays, UO₃ returned, and U-235 assay of UO₃ returned.

ADVANCED PLANNING

Market Analysis

To assist in keeping track of nondefense plutonium requirements and returns, a computer program has been written for mechanizing the bookkeeping and report generation for such requirements and returns. The program generates reports in two major formats: (1) year-by-year summaries of requirements and returns of all projects arranged in order of project number, and (2) complete campaign histories of each project. The latter report appears to be quite useful in revealing flaws in the input data. At present, the code requires separate requirement and return input for each plutonium-using project; in the future it is hoped that the demand-return relationship can be developed so that each plutonium-using project will need only one input card. Then simply removing that card from the input deck will permit computation of the effect of that program's deletion.

IRRADIATION SERVICESFUEL TECHNOLOGY

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Uranium Swelling-Program - Battelle-Northwest

A pressurized uranium swelling capsule was charged into a side General Purpose test hole at KE Reactor. The necessary control and data-recording equipment has been installed. Operating conditions will be 1000 psi and 450 C. Delays in irradiations for this program because of the KW extended outage required that the additional testing space be utilized.

Uranium-Plutonium Fuel Irradiations, - Battelle-Northwest

The first in a series of capsules containing fully enriched uranium-plutonium foils was irradiated in a Snout facility in KE Reactor. These irradiations are part of a study to determine the isotopic content of the irradiated fuel samples.

MATERIALS TESTINGSerpentine Concrete Irradiation - Battelle-Northwest

The third and last serpentine concrete capsule was removed from a side General Purpose facility at KE Reactor. This material is being considered for the top shield of the Fast Flux Test Facility.

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ROUTINE IRRADIATIONS

Fifty-four activation analysis capsules were irradiated in the K and C Quickie facilities.

Seven cooled tensile specimens along with a new holder were charged into the KW Reactor Snout facilities.

Two triple-tandem capsules and a single capsule containing UO_2 -Mo cermet plates were charged into General Purpose facilities at KE and KW Reactors in support of a nuclear space development program for NASA-Lewis Research Center.

ADMINISTRATION - GENERAL

EMPLOYEE RELATIONS

DUN began participation in a Hanford-wide Training-Orientation-Placement Program (TOPP) with the employment of four minority "hard-core unemployables." In this new program, participants will spend part of each work day receiving basic classroom educational instruction administered by the Richland School District. The remainder of each day will be devoted to on-the-job training. The first group of TOPP trainees started the educational phase of the program on August 5. Classes for the second group started on August 19.

SAFETY

Month end safety statistics were:

Disabling injuries: August	0
Year to date	0

Days since last disabling injury	432

Man-hours since last disabling injury	4,650,000

No radiation exposures exceeded operational control.

EMPLOYMENT SUMMARY

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

FOURTH "PACE" PROGRAM

The fourth campaign in the Company's Performance and Cost Effectiveness (PACE) program was begun. The subject is Personal Development, with the emphasis to be on helping each employee gain a better understanding of his or her job, and thus to improve individual job performance.

PAY INCREASE - NONEXEMPT PERSONNEL

The four percent pay increase to nonexempt personnel was included in checks delivered on August 23. Payment of the retroactive increase will be made early in September.

B AND D DEACTIVATION STUDY

Considerable effort has been devoted to an AEC-HQ request for information related to the B and D Reactor deactivations, including details concerning the

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numbers and types of people affected, severance payments, shutdown costs, and comparisons to CAGE estimates.

APPROVAL LETTERS

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

<u>ATD Number</u>	<u>Subject</u>	<u>Date of Transmittal Date to AEC-RL</u>
AP-39	Pension Plan portion of letter entitled "Pension Plan, Salaried Savings Plan and Wage Savings Plan"	January 12, 1966
ATD-83 Add. #1	Fidelity Bond Coverage	August 29, 1968
ATD-90 Add. #2 Supp. #1	Douglas United Nuclear, Inc. Participation in AEC Combined Operations Planning Committee (L. D. Gustafson)	August 29, 1968
ATD-167	Study of Mark IV Fuel Support Design (\$49,000 Mark IV Study by DWD Labs)	August 9, 1968

AUDITS

Discussions were held internally and with AEC-RL personnel regarding areas for fiscal 1969 audit coverage, and an audit schedule was prepared. In addition to the required audits, primary attention is to be given to operational audits where cost reductions can be realized.

The quarterly report on the status of audit findings and recommendations was transmitted to AEC-RL on August 30. This report indicated that audit findings and recommendations were being satisfactorily resolved on a current basis.

APPENDIX A

PROJECT STATUS SUMMARY - REACTOR FACILITIES

<u>Number & Title</u>	<u>Authorized</u>	<u>Percent Complete</u>		<u>Status</u>
		<u>Design</u>	<u>Construction</u>	
<u>Single-Pass Reactors</u>				
DCE-505, Boiler Control Improvements - 165 KE & KW	\$281,000	100	38	Construction continued on the No. 3 KE installation and is about 70 per- cent complete. Additional flame transfer testing (from main-to-alternate burner) was performed on the No. 3 KW system dur- ing the last reactor outage. This project-provided feature remains a problem and further work and testing will be required to develop the proper technique. Project schedule and cost will be re- estimated upon completion of the No. 3 KE installation.
DAP-510, Discharge Chute Clearing Equipment - K Reactors	\$190,000	100	30	Conveyors are 50 percent complete at Jones' shops. Bracket and support installation started at KE on August 13.
DAE-512, Replacement of Turbine with Diesel Drive- 181-B	\$ 87,000	100	100	Complete with minor exceptions which are being corrected.
DAP-513, Deactivation of Hanford Production Reactor (D)	\$ 80,000	100	95	Project closed-out on scheduled date of August 15 with accruals and exceptions.

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PROJECT STATUS SUMMARY - REACTOR FACILITIES (cont'd)

Number & Title	Authorized	Percent Complete		Status
		Design	Construction	
DAP-515, Replacement Steam Generating Facility - 100-D Area	\$140,000	97	35	The boiler has been set on its new foundation, and the new boiler building erected. Boiler repairs started and in progress.
<u>N Reactor</u>				
GAP-401, Upgrading Fire Protection - 100-N Area	\$150,000	-	91	A tentative punch list was prepared, even though the contractor has not completed the work. AEC-RL is reviewing the list for deletion of items not included in the contract before presenting a list of exceptions to the contractor.
H-2 GCE-405, N Reactor Temperature Monitoring System Improvements	\$189,000	100	57	Project in hold status awaiting test results on new method of connecting RTDs. A four-month time extension has been requested for better test evaluation.
GCP-406, Improved Safety Platforms and Accesses - 100-N Area	\$300,000	100	80	Cell 1 has been completed, and fabrication is completed for Cells 2 and 4. The balance of materials have been ordered and fabrication started on the remaining 105-N items.
GCE-408, W, C, & D Elevator Safety Improvements - 105-N	\$ 90,000	100	0	Scheduled shipment of one overtilt pressure switch for test slipped to September 15. Since a delay to October in ordering project quantities would likely delay the project schedule, all required switches were placed on order, completing procurement activity other than stores items.

PROJECT STATUS SUMMARY - REACTOR FACILITIES (cont'd)

Number & Title	Authorized	Percent Complete		Status
		Design	Construction	
GCP-409, Improved Irradiated Fuel Handling System - 105-N	\$310,000	90	47	Outage work on installation of the fast cart portion of project completed August 8.
GAP-410, Decontamination Waste Loading Facility - 100-N Area	\$ 79,000	100	97	Catch tank pump was received and installed August 26. Level indicators are only exception to construction completion.
GCP-411, Effluent Control Program - 100-N Area	\$1,670,000	92	0	The current estimate, using a 15 percent contingency, indicates that the project can be completed for \$1,780,000 with the elimination of the last three items in the project proposal. Design completion of the first phase of the project is scheduled for September 6. Directive HW-572 Mod. 2, dated July 29, authorized total project funds of \$1,670,000.
DCP-517, Additional Storage Basin Cubicles - 105-N	\$ 37,000	100	0	Concrete components for slabs received. Cubicle covers on order; delivery scheduled for September 15.
DCE-519, Replacement of Bridge Crane and Hoist System with New Crane System - 105-N Storage Basin	\$269,000	0	0	Initial layout drawings (preliminary) were received August 13 and comments obtained.

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APPENDIX B

EMPLOYMENT SUMMARY (as of 8-31-68)

	<u>Exempt</u>	<u>Non- Exempt</u>	<u>Total</u>	<u>Change from 7-31-68</u>
<u>Operations Division</u>				
General	1	1	2	0
Manufacturing	200	565	765	- 8
N Reactor	60	198	258	+ 1
Fuels	<u>113</u>	<u>316</u>	<u>429</u>	<u>- 4</u>
	374	1080	1454	-11
<u>Technical Division</u>				
General	1	1	2	0
Engineering	93	24	117	- 1
Process & Programs	69	44	113	0
Technology	58	12	70	0
Advanced Concepts & Planning	<u>14</u>	<u>3</u>	<u>17</u>	<u>0</u>
	235	84	319	- 1
<u>Finance & Administration Division</u>				
General	1	1	2	0
Finance & Administration	68	70	138	0
Employee & Community Relations	<u>14</u>	<u>10</u>	<u>24</u>	<u>0</u>
	83	81	164	0
<u>Company General</u>				
Management	6	1	7	0
Summer Employees	<u>1</u>	<u>22</u>	<u>23</u>	<u>- 8</u>
	7	23	30	- 8
TOTAL	699	1268	1967	-20