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DOUGLAS UNITED NUCLEAR MONTHLY REPORT

MAY 1970

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June 18, 1970

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MONTHLY REPORT

May 1970

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69. DUN Record

TABLE OF CONTENTS

SUMMARY

REACTOR PLANT OPERATIONS

Production Statistics

	KE	N
Input Production - Pu (KMWD)	84.0	43.2
Time Operated Efficiency - %	71.7	45.5
Steam Availability to WPPSS - %	-	42.9

KE Reactor

Equilibrium power level was at the 4000 MW administrative limit associated with inlet piping brittle fracture considerations until mid-month when brittle fracture tube power limits become restrictive. There were two outages: one scheduled, for charge-discharge and maintenance, and one caused by a Panellit trip on an incompletely seated charge of target elements.

N Reactor

Reactor power level was limited to 3600 MW by abnormal back-leakage through the CV-2 check valves, which in turn reduced the amount of high pressure injection water available for primary loop makeup. However, main steam header pressure during equilibrium operation sufficed to sustain 800 MWe generation by WPPSS.

There was one scheduled outage for refueling and maintenance, and seven unscheduled outages. Causes of the latter were: two flow monitor trips, three fuel failures (one unconfirmed), one zone temperature monitor trip, and a manual scram on May 24 to correct a primary loop water-steam hammer. The reactor remained down through month end, initiating the summer outage.

Steam generator retubing in Cell 1 continued on schedule.

FUEL AND TARGET FABRICATION

Production Statistics (tons)

	For KE	For N
Billets Extruded	-	106.0
Finished Fuel Produced	253.7	5.2

KE Fuel

AlSi canning continued on the basis of day-shift operation, five days per week, with two lines for six days and one line for fourteen days.

N Fuel

The accelerated extrusion schedule begun in April was continued through May.

TECHNICAL ACTIVITIES

KE Reactor

Development of a digital code for solving pressure and flow transients in the primary coolant piping, using the theory of elastic medium, is nearing completion. This code will be used in determining front crossheader flow and pressure transients for the K reactor dual riser failure case.

An automated program for calculation of post-accident voiding patterns has been completed, and test cases have been run using flow and pressure transients used for the development of present tube power limits.

A 2x2 column cluster of NpO_2 thin-annulus target elements was charged into KE Reactor for 3 months of irradiation. Ten targets contain a mechanically separable, extruded graphite core. The other ten provide for passage of coolant through the center of the element.

Analyses of the final set of Zircaloy coupon samples removed from KW Reactor showed that hydriding is substantially reduced in anodized aluminum holders when electrical contact is maintained between the holder and the coupon.

Final examination of the two Zircaloy process tubes removed from KE Reactor in March showed that the anodic etch treatment of one tube with 1.2 percent ammonium fluoride following stripping with 3.0 percent nitric acid had effectively removed all surface hydride even though the tube had as much as 19 mils of case layer. A layer of dispersed hydride platelets up to 2 mils thick remained on the surface of the tube treated only with 3.0 percent nitric acid.

A hand and shoe counter that compensates for building background fluctuations has been developed and field tests indicate successful performance.

N Reactor

To provide a consistent basis for N Reactor high power level protection studies, a combination of computer codes is being reviewed. Results of these code evaluations will be utilized in evaluating rod and ball system reliability requirements.

Reliability analyses of the ball safety system trip and gate mechanisms indicate that the probability that no more than five hoppers will fail due to trip mechanism failure, on a demand trip of 105 hoppers, is 0.995 at 95 percent confidence level. On the same bases, the probability due to gate mechanism failure is 0.94. The combined hopper trip and gate mechanism reliability is 0.91.

A program to update and complete boiling burnout protection evaluation for N Reactor fuels confirmed earlier findings that, for all situations assumed under which the potential for burnout would exist, the ratio of local heat flux to burnout heat flux would be within acceptable limits.

Results of initial tests using long-lived foam developed in the noble gas confinement program to encapsulate air containing noble gas appear to have been very successful. The krypton gas, tagged with Kr-85, seems to have a permeation rate through a 10-foot foam layer of less than 10% per 15-day period.

Testing with the wear machine to investigate the process tube scratching problem has started. Small wires and shoe buildup similar to that occurring in reactor charge-discharge tests have been produced.

Further graphite distortion measurements and inspection showed that two additional ball channels have liner block separations, the side-to-side traverse channel in layer 41 has widened from 2 inches to about 4 inches at its center, and process tubes 3151 and 3158 have subsided 1.68 and 2.00 inches, respectively, at their centers.

A summary report (DUN-6807) was issued on the laboratory phase of the program to develop a process for applying oxidation-resistant coatings to the graphite moderators. By circulating a coating reagent with the gas atmosphere, a thin coating of predominantly silicon carbide is deposited on the graphite surfaces which reduces graphite oxidation rates in the N Reactor atmosphere at 900 C by a factor of seven.

A packing material designated as "Grafoil" has performed successfully in V-11 valve tests and is being adopted for use in this service.

Construction has been started on a test facility for use in evaluating an ion exchange process for reducing the level of activity in N Reactor effluent.

IRRADIATION SERVICES

One hundred sixty cc (STP) of Xe-128 have been shipped to Argonne National Laboratory as part of the Xe-128 development work requested. The purity of this gas was 99.8 mole percent.

Oak Ridge National Laboratory has requested that all Be_3N_2 targets be discharged since the AEC has stopped production of C-14 in its facilities. Ten columns of Be_3N_2 targets were discharged from KE Reactor process tubes and test facilities to comply with this request.

GENERAL

The "Equipment Management" audit was completed in accordance with the program furnished by the AEC Assistant Controller for Auditing through AEC-RL. All major contractors of the Commission participated in this AEC-wide audit.

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The audit of purchasing activities (required annually by the AEC Manual) is being directed to areas outside those appraised by the AEC, and includes compliance with Government procurement regulations, Company procurement procedures and policy guides, and reporting requirements. No significant exceptions have been noted.

A surprise count and reconciliation of the three cash funds was made, and the procedures for controlling cash were reviewed. No discrepancies were noted.

The record of purchases for cancellation of 130,000 sets of aluminum caps and cans, because of the KW Reactor deactivation, was approved by AEC-RL. The cancellation charges were $20,559, an amount $9,840 less than originally requested by Alcoa.

AEC-RL was provided information on changes in costs between FY-1970, FY-1971, and FY-1972. The need for more detailed explanatory information of this type became apparent during the presentation to T. M. Yakimchick, Assistant Director for Administration, AEC-HQ, in November 1969.

AEC-RL informed the Company that D and DR Reactor Plants will not be reclassified to Abandoned-In-Place at this time. These reactors are to be maintained in Standby status.

Force reductions resulting from the shutdown of KW Reactor continued throughout the month. The total reduction thus far stands at 303. One hundred ninety-nine of the layoffs were involuntary, 70 were voluntary, and the remaining losses were for miscellaneous other reasons. Job search assistance to laid-off employees has resulted in the placement of 76. The Company's force at month end was 1,384.

A disabling injury was sustained on May 3 by a power operator who slipped and fell while performing housekeeping duties in Bldg. 190-KE. The employee sustained a blow to the head which has required him to remain at home under the care of a private physician.

No radiation exposures exceeded operational control.

for Charles D. Harrington
President

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DUN-6849

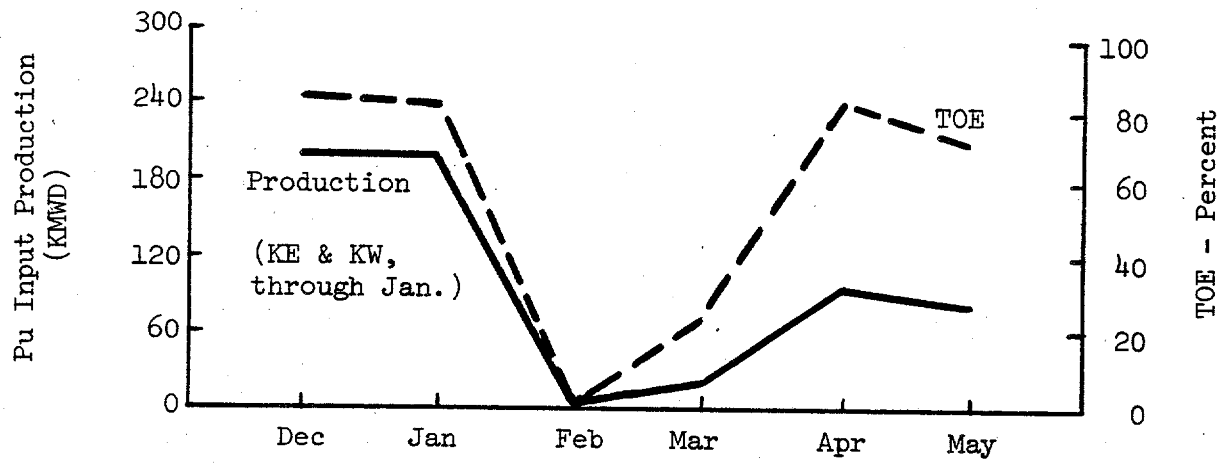
REACTOR PLANT OPERATIONS - KE

PRODUCTION

General

Production (largely weapons grade Pu), power levels, efficiencies and related statistics for the KE Reactor are tabulated below.

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



Statistical Summary

Input Production	- Pu (KMWD)	84.0
	- U-233 (Equiv. KMWD)	1.6
Power Level (MW)	- Maximum	4,000
	- Average	3,778
Time Operated Efficiency - %		71.7
Number of Outages		2
Number of Startup Interruptions		0
Operating Coolant Flow - 1000 gpm		200.1
Fuel Charge (Tons)	- 94 Metal	86.8
	- Natural U	350.0
Fuel Element Failures		0
Helium Losses - 1000 cu.ft.		126.3

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OPERATING EXPERIENCE

Reactor Loading

The front face map showing the loading in KE is reproduced on page B-4. The tonnages listed are approximate; actual fuel charge totals are tabulated above.

Power Level

The power level at KE Reactor was restricted by the 4000 MW administrative limit associated with inlet piping brittle fracture considerations until mid-month when brittle fracture tube power limits became limiting.

Reactor Outages

Date Down	Outage Hours	Remarks
May 9	180.8	Scheduled charge-discharge and maintenance.
May 18	30.0	Panellit trip caused by an improperly seated neptunium charge (PTA-209).

Fuel Element Lodged on Rear Face

While charge-discharging during the May 9 outage, a 94 Metal element from tube 4947 became lodged between a protruding perf in tube 4847 and another 94 Metal element protruding from tube 4947. The stuck metal was removed by downseating tube 4847. However, during the seating operation, a 94 Metal element was thrown on the near rear concrete pad and approximately three hours were required to dislodge it. The seating operation will be investigated since apparently the ram seating force is not achieving the desired fuel trajectory.

Panellit Trip - May 18

A check of the neptunium test charge position in tube 3077 (charged during May 9 outage) revealed that it lacked six inches from being seated against the rear cap.

EQUIPMENT EXPERIENCE

Vertical and Horizontal Rods

Following the replacement of No. 4 HCR with a flat cooled rod, all HCRs are serviceable. However, the round rod in No. 3 HCR channel, from which the track blocks have been removed, remains to be replaced with a flat rod.

All VSRs are serviceable with an average hot drop time of 1.79 seconds.

Resistance Temperature Detectors (RTDs)

Twenty-seven RTDs were replaced during May, bringing to 70 the total that have failed since the rear face hardware decontamination was performed in February.

Emergency Coolant Backup - Seven-Day Supply

The tie line between the 183-KE sedimentation basins and the filtered water clearwells has been installed and tested. With the valve about 40 percent open, a flow to the clearwells of over 16,000 gpm is attained. This more than meets the requirements for emergency coolant during total electrical black-out periods. The intertie was accepted on May 14.

PROCESS ASSISTANCE AND CONTROL

Operational Physics

Operation during the first half of the month required continual monitoring of the tubes used to calculate the flattening efficiency to assure compliance with the 2300 ECT upper limit established in the brittle fracture tube power limit memorandum.

Following startup from the May 18 outage, hot spots developed in several portions of the reactor. A power cut of 50 MW (from 3850 MW to 3800 MW) provided immediate relief to the problem areas and allowed stable operation within the brittle fracture tube power limits. After attainment of an optimum rod and spline configuration, power level was slowly increased.

Some operational physics parameters of interest are shown in the following table:

Effective Central Tubes*	2208
Flattening Efficiency** - May	0.712
- 12-Month Average	0.722
Maximum Operating Time Permitting Scram Recovery - Hours***	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 procedure.

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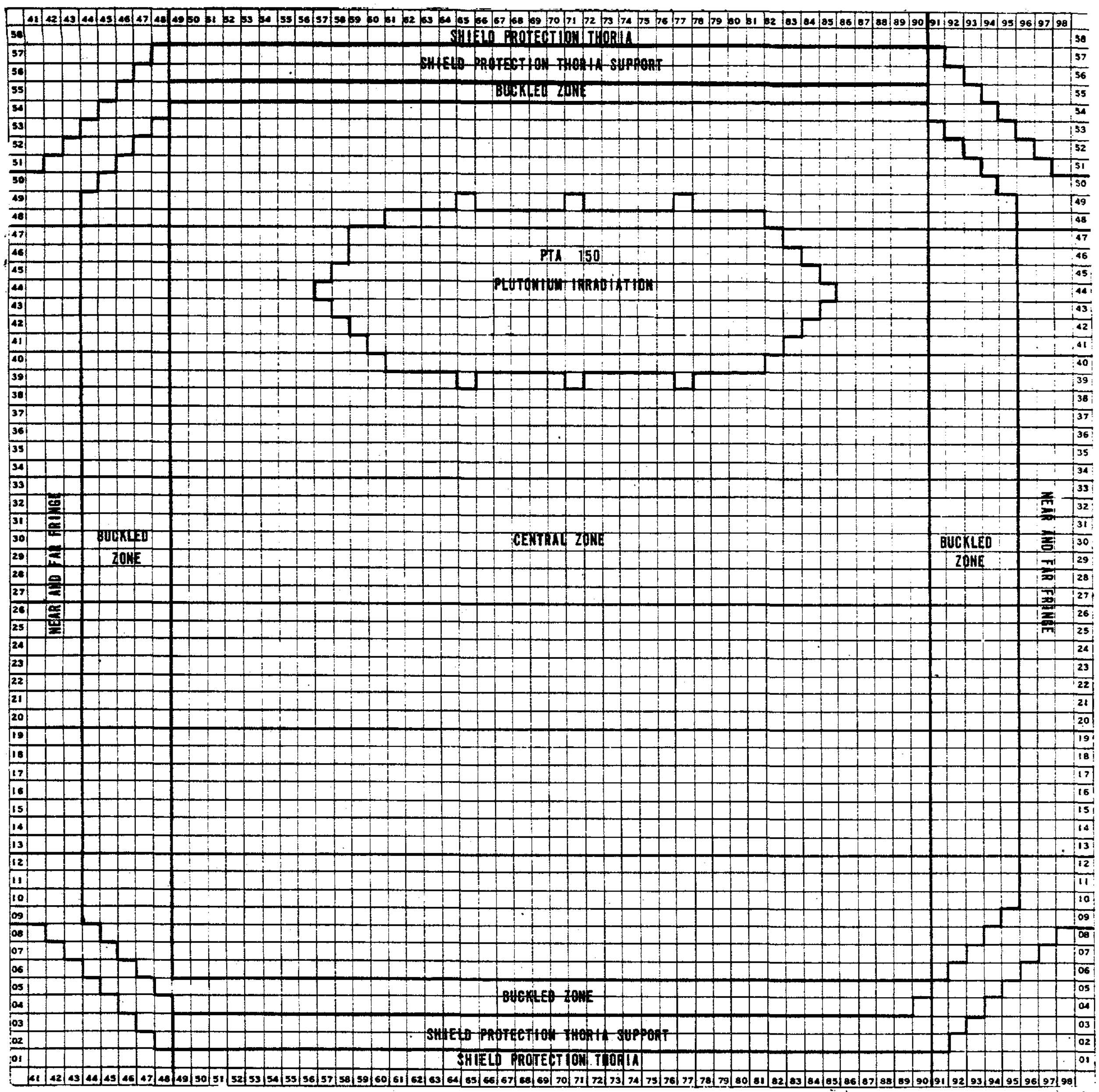

Zone	Tons	Material
Central	219 65	Natural Uranium 94 Metal
Buckled	82	Natural Uranium
Near and Far Fringe	48	Natural Uranium
Shield Protection	23 6	94 Metal (Thoria Support) Thoria

Loading Pattern - KE Reactor

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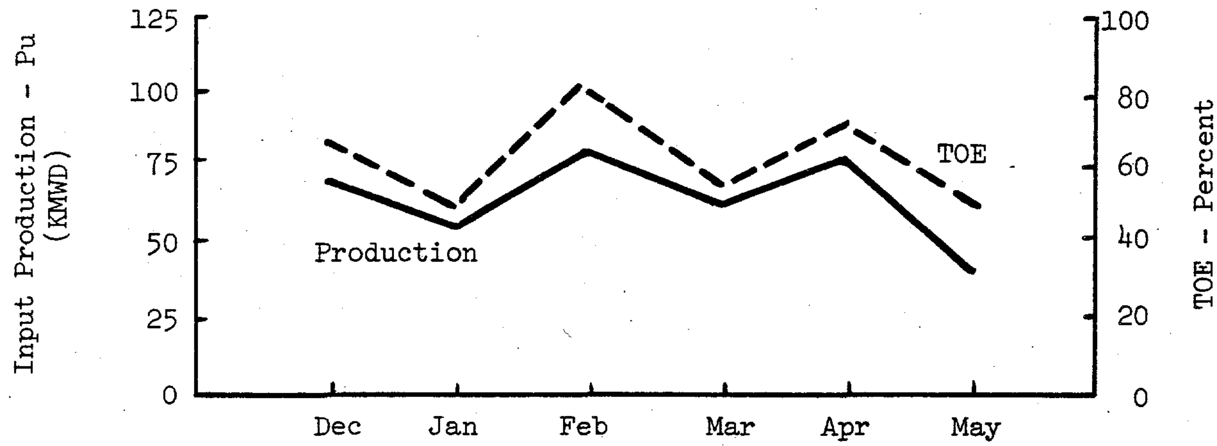
REACTOR PLANT OPERATIONS - N

PRODUCTION

General

Reactor production (largely fuel grade Pu), 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 - Pu (KMWD)		43.18
Electrical Generation (KMWH)	- WPPSS	222.01
	- 184-N	4.60
	Total	226.61
Power Level (MWt)	- Maximum	3,600
	- Average	3,061
Time Operated Efficiency - %		45.5
Steam Availability - %		42.9
Number of Shutdowns	- Scheduled	1
	- Unscheduled	7
Fuel Failures		3 (1 unconfirmed)

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Fuel Charge (Tons)	- 94 Metal	327.28
	- 125 Metal	65.28
	- Natural Uranium	0.37
	Total	392.93
Helium Losses - 1000 cu.ft.		374.32
Fuel Oil Usage - bbl.		20,951

OPERATING EXPERIENCE

Reactor Loading

The reactor loading at month end is shown on page BN-6. Three fuel columns were charged during the month, replacing columns discharged as a result of fuel failures.

Reactor Power Level

The 3600 MW maximum power level achieved during May was lower than the maximum authorized due to reduced availability of high pressure injection water for primary loop makeup (see page BN-3).

The main steam header pressure was maintained at 124 psig during equilibrium reactor operation; this permitted WPPSS to achieve an electrical generation rate slightly in excess of 800 MWe.

Reactor Outages

Start-up from the scheduled outage initiated April 26 occurred on May 5. The ensuing outages are tabulated below:

Date Down	Outage Hours	Remarks
May 5	9.9	Flow Monitor trip resulting from sensing line toggle valve leak.
May 6	34.3	Fuel failure in tube 2448.
May 12	38.8	Fuel failure in PT tube 1465 (unconfirmed).
May 19	40.5	Flow Monitor trip resulting from drive turbine steam reduction.
May 21	10.2	Fuel failure in tube 2361.
May 22	2.2	Zone Temperature Monitor trip resulting from electrical grounds.

Date Down	Outage Hours	Cause
May 24	170.4*	Manual scram to investigate primary loop water-steam hammer.

*The reactor remained down through month end, initiating the summer outage.

EQUIPMENT EXPERIENCE

Primary Loop

The primary loop leak rate, which was about 600 gpm during the first operating period, varied thereafter from a high of about 725 gpm to a low of about 500 gpm. It was about 540 gpm during the final operating period.

A total of forty-six V-11 valves was repacked in the leak repair effort. Leak repairs also were made to four other primary loop valves.

The cause of the water-steam hammer that necessitated the May 24 reactor shutdown was identified as a leakage of primary coolant through the left-side primary loop pressure relief valve. This leakage resulted from a loose position indicator on the valve.

Cell 1 retubing work continued on schedule. Both steam generator units are retubed; the secondary end caps have been installed and stress relieved. A leaking tube was identified and plugged in steam generator 5B.

Graphite Cooling System

The No. 2 graphite cooling system heat exchanger was removed from the auxiliary cell in preparation for transportation to 189-F for decontamination and retubing. No. 4 heat exchanger was installed and ready for acceptance testing at month end.

Primary Shield

Higher than normal primary shield coolant outlet temperatures were observed. Inspection of thermocouples did not reveal a fault and flow checks indicated normal flows. No change from normal temperatures in the crate sections were noted during the period of apparent high coolant outlet temperatures.

Emergency Raw Water Cooling System

The CV-2 check valves in the safety flow control leg of the emergency raw water supply piping were functionally tested; all valves operated satisfactorily.

The back leakage through these check valves (a design feature) was higher than desired. As a result, the amount of high pressure injection water that would be available to the primary makeup in case of an emergency cooling system header break was reduced. This in turn required that reactor power level be limited to 3600 MW.

Boiler Experience

Several leakage problems developed with the oil-fired boilers during May, as follows:

Date Down	Boiler	Remarks
May 6	FW	Leak identified on number 6 tube in row 4. Boiler returned to service on May 8.
May 21	CE-2	Three pinhole leaks were found on a tube about six inches from the steam drum on the end of the boiler next to the vertical gas shield. Boiler returned to service on May 25.
May 26	FW	Number 33 roof tube was found leaking and one other tube was leaking near the mud drum. Unit returned to service on May 28.
May 27	CE-1	A leaking tube was identified near the mud drum. Repairs were in progress at month end.

Equipment Modifications

The following equipment modifications were completed:

DC-3019, "CRW Pump Flow Monitor Revision - 181-N Building," provided test valves for each ΔP gauge and a new protective screen for ready access to the gauges for testing and calibration.

MDC-N-69-45, "High Pressure Injection Pump Pressure Gauge Panel," provided for relocation and panel mounting of the suction and discharge pressure gauges for the five high pressure injection pumps to facilitate gauge reading and replacement.

MDC-N-70-7, "Nitrogen Supply Low Pressure Alarm Switch Replacement," provided installation of a new low pressure alarm switch with a more suitable adjustable alarm point.

MDC-N-69-74, "Turbine Generator Emergency Seal Oil Pump Circuit," provided a pilot relay to control the highly inductive DC motor starter circuitry, and thereby to reduce the load on switch contacts and increase switch life.

PROCESS ASSISTANCE AND CONTROL

Operational Physics

The several startups were completed with no unusual reactivity problems.

Operational physics parameters of interest are shown in the following table:

Effective Central Tubes*	822
Flattening Efficiency* - May	0.82
- 12-Month Average	0.81
Maximum Operating Time Permitting Scram Recovery - Hours*	24

*For definitions see page B-3.

Production Fuel Performance

The fuel failure on May 6 (serial No. 57) was a Mark IV element charged on March 17. Penetration occurred on the inner surface of the outer element at the point where a solid support contacted that surface. Preliminary evidence indicates the failure mechanism to be wire fretting.

The element which failed on May 21 (No. 58) was a Mark IV charged on 11/3/69. Its failure was end-associated; Radiomet examination is scheduled.

Inconel-600 Steam Generator Tube Surveillance

The Inconel-600 tube removed from steam generator 6A in November 1969 has been visually inspected, eddy current tested, sectioned, and sent to BNW Metallography for examination.

Eddy current examination of the Inconel-600 tubes in steam generator 4A has been scheduled for the summer outage. The examination is desirable because of slight indications of a tube leak in 4A, and the increasing disclosure of failures in Inconel-600 tubed steam generators in commercial reactors.

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DUN-6849

	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							E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E							33
32						E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E						32
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28		E	E	E	X	X	X	F	G	G	G	G	G	F	F	G	G	F	F	G	F	F	F	F	F	G	X	X	X	X	E	E	E		28
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14	E	E	94	X	X	G	G	G	G	F	F	F	G	G	G	G	G	G	F	G	G	01	G	G	G	G	F	G	G	X	X	X	E	E	14
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12	E	E	X	X	X	G	F	F	F	F	F	G	G	G	G	G	G	F	G	F	G	F	G	G	F	G	G	G	G	X	X	X	E	E	12
11	E	E	X	X	X	G	F	F	F	F	G	G	F	F	F	F	G	G	F	G	G	G	F	F	G	F	F	G	F	X	X	X	E	E	11
10	E	E	X	X	X	G	F	G	F	F	F	F	F	G	F	F	F	G	G	G	F	G	F	F	G	G	F	G	G	X	X	X	E	E	10
09	E	E	X	X	X	X	G	F	F	G	G	G	F	G	G	G	F	F	F	G	G	G	G	G	F	G	G	G	X	X	X	X	E	E	09
08	E	E	E	X	X	X	F	F	G	G	G	G	G	G	G	G	F	G	G	G	G	G	G	G	G	F	G	F	X	X	X	E	E	E	08
07		E	E	E	X	X	X	X	G	G	G	G	G	G	F	G	G	G	F	G	G	G	G	G	G	F	X	94	X	X	E	E	E		07
06			E	E	E	X	X	X	X	G	G	G	G	G	G	G	G	F	G	G	G	G	G	F	F	X	X	X	X	E	E	E			06
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02							E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E							02
01																																			01
	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	

Fuel Code	No. Tubes	Description
E	224	Mk-IC (94 Metal - Fringe)
F	201	Mk-IV (94 Metal - High U-236)
G	333	Mk-IV (94 Metal - Central)
N	1	Mk-IB (Natural U)
X	231	Mk-IA & Mk-IV (125 & 94 Metal)
	990	Total
	14	Total PTs
	1004	Grand Total

PT-NR No.	No. Tubes	Description
94*	8	Mk-IV Demonstration
01	5	Mk-I & Mk-IV from Direct-Cast Billets
■	1	Graphite Samples Channel
	14	Total PTs

*Includes Mk-IV high U-236 content fuel and 6 tubes with Mk-IV-AA 125 Metal and Mk-IV 94 Metal.

Loading Pattern - N Reactor

[redacted] [redacted]

FUEL AND TARGET FABRICATION - KE REACTOR

PRODUCTION

General

Production of AlSi-bonded fuel for KE Reactor was 115.3 percent of forecast. Canning continued on the basis of day-shift operation, 5 days/week, with two lines for six days and one line for 14 days. Seventy-seven percent of the elements had self supports attached.

Acceptable Fuel Elements Produced

Tons		253.7
Yields:	Current Month	95.0%
	FY to Date	95.6%

Month-End Inventories - Tons

Bare Uranium Cores		619*
Finished Fuel:	AlSi-Bonded	1,754*
	Hot-Die-Sized	19
Thoria Elements		14

*These totals include 73 tons of bare cores and 138 tons of finished fuel in sizes used in the smaller reactors.

OPERATING EXPERIENCE

Canning efficiency of the AlSi-bonding lines was 98.4 percent. Downtime was assigned 63 percent to equipment malfunctions and 37 percent to operations.

EQUIPMENT EXPERIENCE

Nothing significant to report.

PROCESS ASSISTANCE AND CONTROL

Nothing significant to report.

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FUEL AND TARGET FABRICATION - N REACTOR

PRODUCTION

Input Production

Total billets extruded	494
Tons extruded	106.0
Percent of forecast	1,060.0

Output Production

Total finished fuel assemblies	324
Tons output	5.2
Percent of forecast	43.3

Uranium Utilization - %	77.0

Month-End Inventories - Tons

Bare uranium billets	138
Finished fuel	169

OPERATING EXPERIENCE

An accelerated coextrusion schedule was begun the latter part of April and continued throughout May. Because of this emphasis on input, output production fell below the forecast.

EQUIPMENT EXPERIENCE

Nothing significant to report.

PROCESS ASSISTANCE AND CONTROL

Coextrusion Techniques

The use of a copper-silicon spacer plate at the rear of the billet and just ahead of the steel follower block has increased the usable extruded tube length and has cleared all uranium and Zircaloy from the sheared butt sections. The longer tube has made it possible to virtually eliminate the 12-inch fuel normally cut for the end piece.

X09 extrusions are now producing a 23-inch rear piece on 80% of the extruded tubes, with the remaining 20% cut to 17-inch lengths. The B09 extrusions are producing approximately 80% 17-inch rear pieces with the remainder cut to 12 inches. This means that 26-inch B09 pieces requiring rework due to a defect within three inches of the end can now be cut to fit the 23-inch X09, while inner warp rejects can be reworked to fit the 17- or 12-inch long B09 pieces.

The increased yield of a B09 extrusion is not as great as it is for the inner extrusion because the preshape angle is not ideal, and the larger ID allows copper-silicon to flow into the rear end tube defect a greater distance.

TECHNICAL ACTIVITIES - KE REACTOR

RESEARCH AND DEVELOPMENT

Basic Production

Water Hammer Study

The development of a digital code for solving the pressure and flow transients in the primary coolant piping of KE Reactor, using the theory of elastic medium, is nearing completion. The original code has been expanded and refined to include the process tubes and rear face piping, with the result that a full flow transient determination may be possible for single or dual riser breaks. Difficulty has been experienced in determining transition flows in the process tube model during initial voiding because of the necessity of including hydraulic resistance in the fuel column area. This difficulty is being examined at present and a plausible model is being developed.

Zircaloy Process Tube Hydriding

Coupon Irradiations

The final set of Zircaloy coupon samples were removed from the KW Reactor at the time of shutdown after 14 months of residence. These coupons were irradiated to investigate hydride mechanisms and the efficiency of case layer removal by the electrolytic process in arresting hydrogen buildup in the K process tubes. The coupons have been analyzed and the following observations appear to be supported by the data:

- All coupons hydride readily in bare aluminum holders.
- Hydriding is substantially reduced, but not completely eliminated, in anodized holders when electrical contact is maintained between holder and coupon.
- The only coupons in stainless steel holders which showed a definite increase in base metal content were those which had been laboratory hydrided so that there was a one mil case layer on the surface with no increase in the base metal content.
- The scatter in the base metal hydrogen content of coupons made from hydrided reactor tubes was so great that it will be difficult to draw firm conclusions as to whether hydrogen migration from the case layers did occur.

Hydride Removal Test Results

Examination of the two Zircaloy process tubes (2255 and 3056) which were removed from KE Reactor to evaluate the effects of the electrolytic hydride removal was completed with the following results:

- Tube 2255, which had been anodically treated in 3 percent nitric acid while in KE Reactor, showed essentially complete removal of the massive hydride case layer. A layer of dispersed hydride platelets up to two mils thick was left on the surface. Only two minor spots of localized corrosion were observed. One of these could not be related to the stripping process.

- Tube 3056, which had been treated with ammonium fluoride following the 3 percent nitric acid treatment, showed that the combined processes effectively removed all surface concentrations of hydride even though this tube had as much as 19 mils of case layer. A very uneven surface was left in the top of the downstream three feet of tubing which presumably resulted from a similarly nonuniform hydride layer in the tube before stripping.

Photographs of sections taken from tube 3056 are shown on page D-A (at the end of this section). The low-magnification view near the rear VanStone flange (Figure D-1) shows the effectiveness of the process in removing a layer of case and dispersed hydride about 19 mils thick. The plateau area at the left in the photo, where the case layer is shown intact, was shielded from the treatment. It can also be seen that very little, if any, excess metal was removed beyond the case and dispersed layer. Figure D-2 shows the treated surface near the rear VanStone flange at a higher magnification. No surface concentrations of hydride are visible.

Product Flexibility

Ten Kilogram PuAl Irradiation (PTA-150)

Only minor changes were made in the basic enrichment support within this test block during May. Specific powers in the PuAl columns, still conservative relative to assigned limits, can be easily maintained at the desired levels with splines as control rods are withdrawn from the local area.

Pu-238 Studies

A draft document on americium irradiation predictions is being rewritten to include effects of revisions in Am-241 and plutonium cross sections. During the report period, the cross section library utilized in the computer code was revised, and new production quantities were calculated. Extensive calculations to determine fission product buildup, and the post-irradiation heat generation rate due to both fission products and isotopic buildup, were also performed.

A 2x2 column cluster of NpO_2-Al thin annulus target elements was charged into KE Reactor on May 10, as authorized by PTA-209. This cluster, surrounded by ten columns of support enrichment, will be irradiated for about three months. Ten of the NpO_2-Al matrix targets contain a mechanically separable, extruded graphite core. The other ten targets provide for passage of process coolant through the center of the element. The test will provide information on the conversion efficiency of the two types of targets, and on the contamination level in Pu-238 at higher exposures in a 2x2 column cluster.

Environmental and Regulatory Technology

K Reactor Tube Power Limits

Work in progressing on development of K reactor tube power limits for increased top-of-riser pressure (TORP). As noted under Basic Production, a computer code is nearly complete for application of the elastic water column theory in determination of front crossheader flow and pressure transients for the dual riser failure case.

An automated program for calculation of post-accident voiding patterns has been completed, and test cases have been run utilizing the flow and pressure transients used for development of present tube power limits (based on rigid water column theory). The peak fuel temperature that would be reached in a specific case was calculated by the new code for comparison with results previously calculated using the manual evaluation procedure. The peak fuel temperature obtained was 60 C lower (520 C versus 580 C). This result stemmed from the ability to apply more rigorous analytical techniques than were feasible for use in the manual procedures.

Waste Management

Nothing to report.

ENGINEERING AND TECHNOLOGY - REACTOR

Background Compensated Hand and Shoe Counter

The current model hand and shoe monitor used at Hanford the last 10 years for personnel monitoring is becoming obsolete from function, circuit, and hardware standpoints. Further, the need to take into consideration radiation background level fluctuations in the reactor buildings make these monitors less effective than desirable for detecting radioactive contamination on personnel. A literature search and market survey was made for hand and shoe monitors which would compensate for radiation background levels. The survey showed that there are numerous hand and shoe monitors that are acceptable instruments if the background radiation levels remain constant. Functionally, however, these instruments would not be an improvement over our present monitors.

Accordingly, a concept involving background compensation was formulated and used in a prototype built for field tests. These tests established that the monitor was reliable and that its features met user needs. A subsequent change incorporated an increase in the sensitivity of the monitor for beta radiation detection.

In general, the background-compensated hand and shoe monitor operates as follows: The count rate circuits continually monitor the background radiation and establish an alarm level that is as close to background as random count variations in the background will allow. When the monitor is activated by a person stepping onto the stand for shoe monitoring, a switch on the shoe monitoring section stops the accumulation of background counts and holds the count for use in comparator circuits. When the hands are inserted in the hand monitoring ports, microswitches are operated and a five-second counting cycle is initiated. If during this count cycle a count rate greater than the background count is obtained, the alarm is activated. Indicating and warning lights are similar to those used in the currently used monitor.

Plans are being firmed to equip each reactor site with the required number of monitoring stations using the above developed equipment.

Project Engineering

Completed Project Resume - DAE-518

Project DAE-518, Effluent Radioiodine Monitor - KE & KW Reactors, has been completed. Its purpose was to install instrumentation to measure the release of I-131 into the Columbia River as a result of fuel element cladding failures, in order to warn of an unusual release in time to put into effect appropriate control measures including the curtailment of downstream potable water intakes from the river. This instrumentation would measure the effluent water from each reactor and monitor it for specific I-131 activity.

The project Proposal was submitted to AEC-RL in May 1968. A Directive was received from the Commission in September 1968 with an interim funding to start design. Total project authorization was obtained in November 1968. Initially, the scope of the project included the KE, KW, and C Reactors; however, subsequent to the construction of the project, C Reactor was deactivated and construction effort started at the K Plants only. DUN was responsible for the procurement and on-site fabrication of special printed circuit modules developed particularly for this monitoring system. Construction started May 1, 1969, and was completed January 30, 1970.

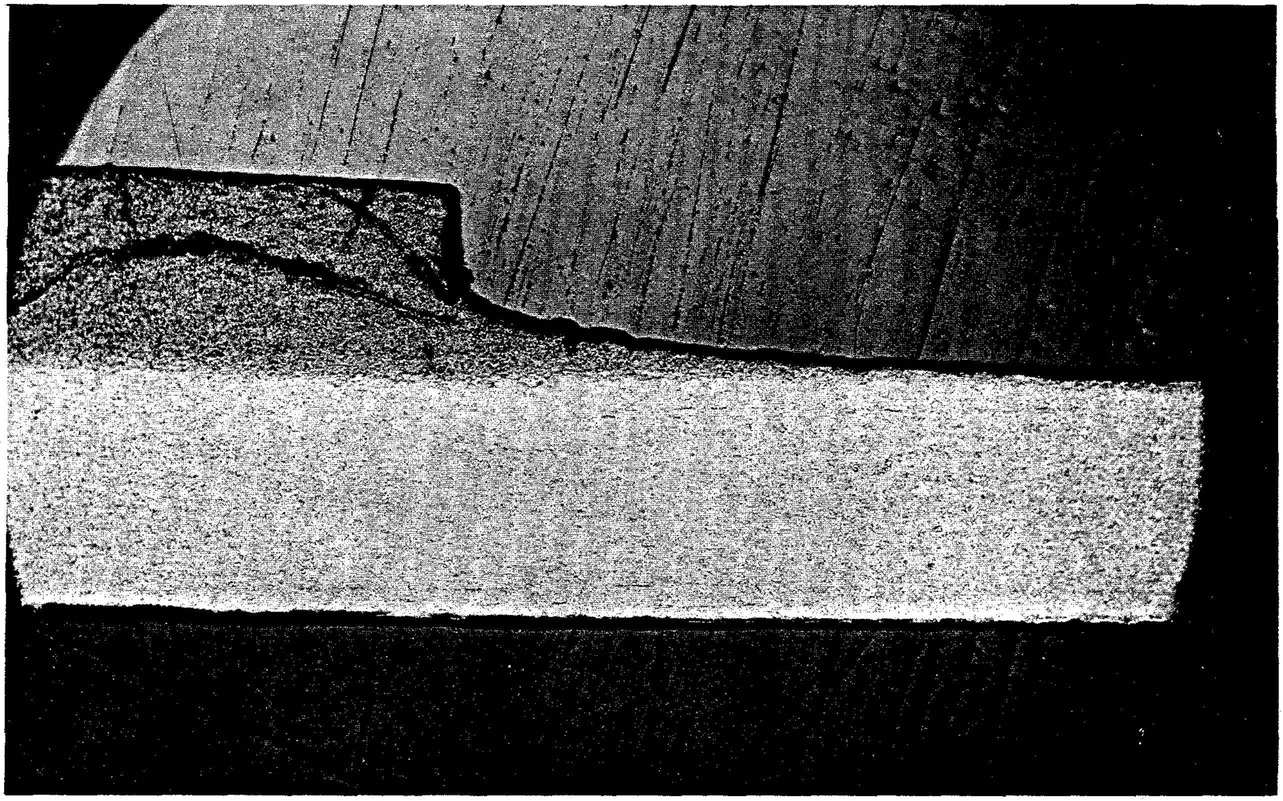
Upon completion of construction, deactivation of KW Reactor was announced. However, at AEC-RL request, acceptance testing of both the KE and KW systems was completed recognizing the fact that there was no immediate need for the KW system. The instrumentation is installed in the data monitoring building located at the 107-KE retention basin area. The equipment has performed in accordance with the Design Criteria and is in operation at the KE Reactor. This project was completed within the funds authorized and within the schedules provided.

Project Status Summary

The status of construction projects relating to K Plant facilities is summarized in Appendix A.

ENGINEERING AND TECHNOLOGY - FUELS AND TARGETS

Nothing significant to report.



50X

Figure D-1. Hydride Case Layer Removal - Tube 3056 Near Rear VanStone Flange (Plateau at left shows pretreatment case layer)

250X

Figure D-2. Hydride Case Layer Removal - Tube 3056 (Showing complete removal from tube section near flange under higher magnification)

AEC-GE RICHLAND, WASH.

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TECHNICAL ACTIVITIES - N REACTOR

RESEARCH AND DEVELOPMENT

Basic Production

Accountability Analyses

Dissolver sample analytical data from PT-NR-95 were received from ARHCO during the month. The isotopic ratios for both the ARHCO and the BNW analyses agree well with DCODE and MOFDA lattice code predictions. The ARHCO absolute production data indicated yields substantially greater than predicted, however, and results are being remeasured.

Nuclear Protection Studies

To provide a consistent basis for N Reactor high power level protection studies, a combination of computer codes is being reviewed. This combination includes a code for developing effective cross sections, a second code utilizing these cross sections to develop local rod strengths, and a third code for determining overall reactivity and flux distribution effects. The HAMMER-DTF-EXTERMINATOR chain will be utilized for the most part, with normalizations to well established critical rod configurations from N startup experiments. The EGGNIT and THERMOS codes for determining effective cross sections, and the DUNCAP (DUN version of FLARE) code for dimensional effects, will be utilized for backup and crosschecking. Results of these computer code evaluations will also be used in the evaluation of rod and ball system reliability requirements involving both random and localized failures.

A sample criticality case run on the updated KENO code showed good results. Both the 12,000 and 18,000-history Monte Carlo cases checked within 1 percent of the critical multiplication, requiring less than 1 minute of computer time per thousand histories.

Ball Safety System

To assess the adequacy of the ball safety system trip and gate mechanisms, reliability analyses have been performed. The probability that no more than five hoppers will fail due to trip mechanism failure, on a demand trip of 105 hoppers, is 0.995 at 95 percent confidence level. This reliability value, the probability of successful system operation, is based on the accumulation of trip tests from January 1969 to May 1970. Specifically, one hopper trip mechanism failed during the 290 tests performed during that period.

Three hopper gate mechanism failures were experienced during the April 1970 outage. The probability that no more than five hoppers will fail due to gate mechanism failure, on a demand trip of 105 hoppers, is 0.94 at the 95 percent confidence level. It is possible that these failures are directly related to the non-reworked condition of some hoppers.

The combined hopper trip and gate mechanism reliability, based on a total of four failures in 290 tests, is 0.91. However, if the apparent correlation of failures relating directly to non-reworked hoppers is assumed, a ball hopper system reliability of 0.98 would be achieved.

Graphite Distortion

During the April outage four additional ball channels were inspected, two process tubes were traversed for vertical subsidence, and two side-to-side traverse channels were inspected for graphite damage and shifting. The following facts were established as a result of these examinations:

- Two additional ball channels (48 and 57) were found with misalignment and separation between the second and third liner blocks. It was also found that the vertical liner block columns have contracted and subsided more than previously thought (≈5 inches total) and that the gaps between the second and third blocks may vary considerably from one channel to the next.

- The side-to-side traverse channel in layer 41 has widened from 2 inches to approximately 4 inches at its center. The filler layers have pulled apart about 2 inches at the center slip joint. There is some key breakage, particularly in layer 17, which is confined to the outer ends of the channel. About 35 balls were found in the layer 17 channel adjacent to ball channel 49.

- Tubes 3151 and 3158 have subsided 1.68 and 2.00 inches, respectively, at the center. Plots of these vertical height data show a peculiar "longhorn" shape not observed in the two tubes inspected in March and reported last month.

Graphite Coatings

The laboratory phase of the program to develop a process for applying oxidation-resistant coatings to the graphite moderators was completed and a summary report (DUN-6807) issued. The process consists of circulating a coating reagent, tetraethylsilane, tetramethylsilane, or hexamethyldisilane, with the gas atmosphere.

A thin coating of predominantly silicon carbide is deposited on the graphite surfaces exposed to the treated gas atmosphere at temperatures above 425 C. The coatings reduce graphite oxidation rates in the N Reactor atmosphere at 900 C by a factor of 7. It is concluded that oxidation-resistant coatings could be applied to the Richland reactor moderators with a high probability of success. The coatings could provide for the use of nitrogen in place of helium at the N Reactor, to effect operating economies, or could provide for an increase in N Reactor power level assuming continued use of the present helium-water vapor atmosphere.

Primary Coolant Neutral pH Test

PT-NR-119 SUP2, which attempted to demonstrate a reduction in primary piping radiation level by reducing coolant pH to 7, was conducted April 22-24. The test was terminated after three days of operation (with primary coolant pH ranging from 6.1 to 9.5) because of decreases of about 2 percent in reactor coolant flow and up to 5 percent in individual tube coolant flow because of crud buildup on the fuel surfaces. After returning the coolant pH to 10, tube flows stopped decreasing and began to return to normal.

Although coolant flow reductions with reduced coolant pH have been observed in power reactors, no such effect was anticipated at N Reactor with its large flow channel area.

Results of the test failed to show any significant improvement in primary piping radiation levels. The net result of three days operation at pH 7 appeared to have been to reduce the deposition of activated corrosion products at the front of the reactor and increase it at the rear. It should be noted, however, that the rear piping activity increases may have been caused by the fuel failure in the preceding operating period.

Preliminary radioisotopic analyses of the crib influent indicated that concentrations of I-131, Sr-90, and P-32 going to the crib were in the normal crib influent concentration ranges for these isotopes, after correcting for the variations between analytical laboratories. Only Sr-90 showed a systematic change in concentration with time during the test, increasing from 0.3 to 7.2 nCi/l.

Product Flexibility

Nothing to report.

Environmental and Regulatory Technology

Boiling Burnout Protection for Fuels

A program to update and complete boiling burnout protection evaluation for N Reactor fuels has been completed, including evaluations for fringe-loaded Mark IV fuels in orificed tubes. Results confirmed earlier findings that, for all situations assumed under which the potential for burnout would exist, the ratio of local heat flux to burnout heat flux would be within acceptable limits.

Three basic cases were investigated for each fuel type, namely: (1) reduction of tube flow to low trip, (2) reduction of flow to connector boiling caused by a leak in an inlet connector (the high flow trip case), and (3) increase in tube power to connector boiling (the condition for Zone Temperature Monitor (ZTM) trip). In all cases, the computer code BOTHER was utilized to calculate local heat flux and coolant enthalpy as a function of length along the fuel column. Theoretical chopped cosine axial flux profiles were used for the first two cases, while the third utilized a rear peaked profile corresponding to a double rod swap accident with a 2.16 peak-to-average heat flux ratio (removal of two control rods at the rear and insertion of two in the front, all on the same elevation).

The following table summarizes results of these studies:

[redacted] DECLASSIFIED [redacted]

Fuel Column	Max. Ratio of Local Heat Flux to Burnout Heat Flux
Mark I-C	
Central Zone	0.537
Orificed	0.555
Mark IV	
Central Zone	0.685
Orificed	0.665
Mark IV-AA	
Central Zone	<0.680
Orificed	0.570

In all cases, the maximum burnout ratio obtained was that corresponding to ZTM trip conditions. The limiting value of burnout ratio in the ZTM case is 0.70, which allows for a ±30 percent scatter in burnout heat flux obtained from laboratory experiments. Figure DN-1 (next page) is an example of the detailed results obtained from a computer output.

N Meteorological System

The N meteorological system was operational the past month. The problem of discontinuity between true wind direction and readout from wind changes through the 0-360 degree azimuth has not been rectified, but work on the problem continues.

A computer program is being checked and varied to obtain optimum deletion of incorrect wind direction data.

Foam Encapsulation of Air Containing Noble Gas

During the report period, tests were performed utilizing the long-lived foam developed in the noble gas confinement studies.

Under the concept of gas control with long-lived, high-expansion foam, escaping gases would be passed through a foam generator that would encapsulate the contaminated air. Radioactive decay and slow permeation of the gases through the foam would restrict the final release of the gases to within acceptable radiation control limits.

In the initial test, krypton gas, tagged with a small quantity of Kr-85, representing the quantity of krypton released in a full fuel melt at N Reactor, was introduced into the foam generator. The purpose of the initial test was to study krypton gas migration through the foam layers for a period of 15 days. This is the time period necessary to allow 95 percent of fission-produced noble gas to decay, leaving only the krypton-85 and some xenon-133.

An analysis of data indicates that the initial test was very successful. The

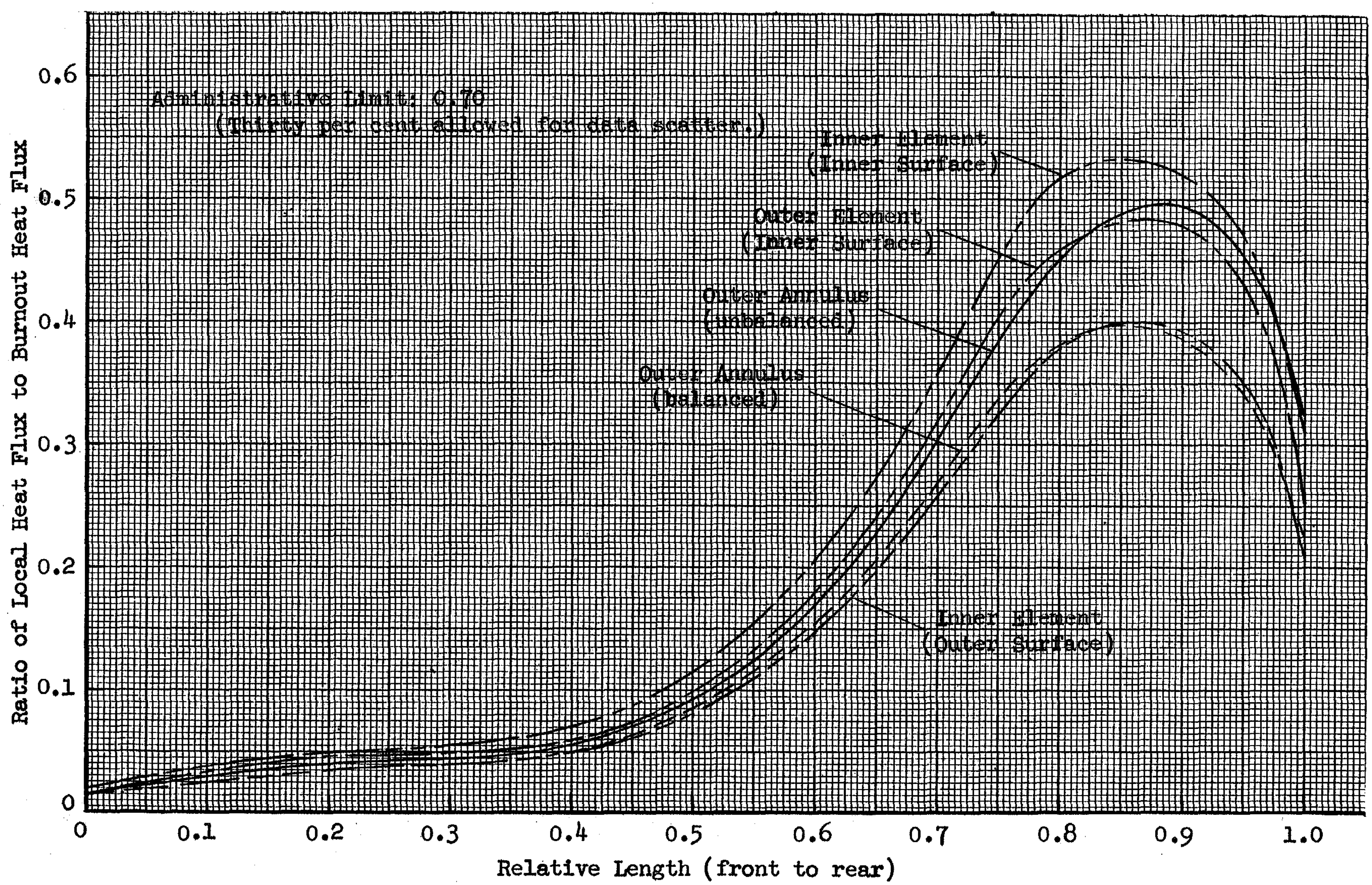

Figure DN-1. Burnout Heat Flux Ratios for Mark I-C Fuel in Unorificed Tube (With and without a 10.6 imbalance in temperature rise between top and bottom of the outer annulus due to eccentricity. ZTM trip conditions)

[redacted] DECLASSIFIED [redacted]

krypton was seen to move from the lower foam levels to the layers above, but very slowly. Although much data are still to be analyzed, it appears that the permeation rate is less than 10 percent per 15-day period through a 10-foot foam layer.

A second test using the same quantities of foam and krypton has been started. The foam generation rate was reduced, thus lowering the expansion ratio to 50:1 and giving a smaller bubble.

Preliminary data from the second test indicate no migration of krypton and no activity above background seen in the air space above the foam layer, giving assurance that there was no bypass of the foam generator and all air passing through the generator was encapsulated.

N Stack Radioactivity Release Monitor

Engineering has been completed in support of a project proposal to provide for quantitative measurement of radioactivity in the gaseous fission products discharged through the N Reactor exhaust system. Design Criteria have been prepared and submitted to AEC-RL (in DUN-5583) for review and approval.

It is proposed to provide equipment and instrumentation at the 116-N exhaust stack and in the 100-N Reactor plant that, in the event of a postulated severe incident, will: (1) measure and record the gross gamma activity of fission product noble gases released through the stack, (2) measure and record the radiation level of the radioiodines released from the stack, (3) monitor and record the volumetric discharge rate of the stack so that immediate quantitative data on the noble gas and iodine-131 release rates can be obtained, and (4) provide for operation of the instrumentation involved in the event of concurrent loss of A and B bus electrical power.

CCTF Pilot Plant

The Contaminated Coolant Treatment Facility is intended to treat all liquid waste streams from the reactor and reduce the levels of activity to within current nuclear safety guide lines. A two-stage ion exchange process has been proposed; however, the effectiveness of the process has not been demonstrated.

A pilot plant has been proposed to demonstrate feasibility of the ion exchange process, and $20,000 was allocated for construction during the current fiscal year. Construction was initiated in May. A sheet metal building is being moved to N Area from one of the deactivated reactors to house the equipment which will include filters, ion exchange columns, instrumentation and other normal laboratory equipment. The filters and ion exchange columns, which will remove radioactive ions from the waste water, will be located in a water-filled pit for shielding purposes. The pilot plant will determine the practicability of the basic process as well as the effectiveness of a variety of commercially available ion exchange resins.

Waste Management

Nothing to report.

 [redacted]

[redacted]

ENGINEERING AND TECHNOLOGY - REACTOR

Tube Scratching Problem

Testing with the wear machine was initiated at constant disc rpm. Small wires and buildups were produced on the first track run on an unscratched autoclaved Zircaloy-2 disc. The buildups appeared similar to those seen on shoes used in reactor charge-discharge tests. The wires were similar to the smaller sized wires produced in-reactor and probably were not large enough to cause fretting failures. This observation agrees with the fact that fretting failures were not observed after the initial loading of N Reactor.

Loads of 12.5, 25, 50, and 71 pounds were used on scratched tracks during repeat tests. It was found that the weight of buildup increased with load and that at 25 and 50 pound loads, the effective loading force of the buildup on the disc was approximately 25,000 psi. The effects of sliding velocity were not apparent from these tests. At constant disc rpm, each test begins at low velocities and continues up to 120 ft/min at the outer disc radius.

Testing has been temporarily halted due to imminent delivery of the transmission required for constant sliding velocity testing. Once the transmission is in place, tests will be conducted on reautoclaved scratched discs simulating the current scratched process tube surface conditions.

Circulating Raw Water System (CRWS) - Debris Monitoring

Development Test 218 is underway to collect and characterize debris which passes through the traveling screens into the CRWS. Representative CRWS streams, at points where flow can be measured, will be passed through fine-mesh screens to collect debris. A small stream (about 250 gpm) will be monitored at the 181 river pump house and a large stream (about 3500 gpm) will be monitored where raw water enters the water treatment plant coagulators. The test also provides for measurement of all accumulated material any time a part of the CRWS is opened up.

Arrangements have been made to conduct model studies at Washington State University's Albrook Hydraulic Laboratory to determine the effect of river bottom contour and diversion jetties or booms to reduce the debris carried into the pump intakes.

N Flow Monitor Sensing Line Replacement

Replacement of the original stainless steel flow monitor sensing lines on process tube row 64 with the new Inconel tubing, which was successfully completed during the April outage, was planned and performed as a lead installation to permit checkout of procedures and tooling prior to the full-reactor sensing line replacement program scheduled for the summer outage. Although the work progressed smoothly, there were several areas in which improvements could be made to speed up the work or provide an easier method of accomplishing a specific operation. Efforts to provide solutions to these items are underway.

Of particular interest was the very effective performance of the water bag shielding concept which resulted in an exposure reduction of about 50 percent.

V-11 Valve Packing

For a number of months a packing material designated by its manufacturers as "Grafoil" has been under test with the intent of determining its potential as an improved packing for use in the V-11 valves. A variety of tests have been run on this material, including: installation of a Grafoil packed valve in the NPR loop and in an environmental chamber setup, determination of compatibility with reactor decontamination materials, reaction to radiation, and the potential for galvanic action. In all cases, the results indicated that Grafoil is a superior packing. Production Test PT-NR-138, authorizing the on-reactor installation of 25 Grafoil packed valves, was initiated in March; no evidence of leakage of these valves has been noted to date.

Briefly, Grafoil is a soft packing material fabricated from 99 percent pure graphite. It contains no binders or stiffeners and, therefore, has little or no tendency to undergo dimensional changes or to harden. Because of these characteristics, if leakage through the packing does occur, it can be easily stopped by taking up slightly on the gland follower. Also, since Grafoil is a relatively inert material, its resistance to the effects of both radiation and decontamination solutions is excellent.

On the basis of this information, a Design Change designating Grafoil as the standard material to be used in repacking the V-11 valves is being prepared.

Nuclear Flux Monitor System

Engineering has been completed in support of a project proposal to upgrade the nuclear flux monitor system at the N Reactor. Design Criteria have been prepared and submitted to AEC-RL (in DUN-6854) for review and approval.

The modifications proposed for the flux monitor and described in the design criteria include replacement of primary detectors and signal conditioning equipment for each of the three subsystems. It is also proposed that new mechanical positioners be provided for the startup range primary detectors.

High Pressure Injection System

Engineering has been completed in support of a project proposal to upgrade the control system for the high pressure demineralized water injection system at N Reactor. Design Criteria have been prepared and submitted to AEC-RL (in DUN-6865) for review and approval.

The changes proposed for the control system and described in the design criteria include modifications of the injection pump speed control system, the pressurizer level and injection valve control system, and the compressed air and electric power supply system to provide redundancy.

Project Engineering

The status of contrruction projects relating to N Plant facilities is summarized in Appendix A.

ENGINEERING AND TECHNOLOGY - FUELS AND TARGETS

Salt Bath Annealing

Copper and copper-silicon tubes have been successfully annealed using the salt bath facility located in Building 3716. The technique involves immersion of the tube in the molten salt at a temperature of 1200 F for 2-3 minutes followed by an immediate water quench. It has been found that not only is this method extremely fast, but the tubes can be etched without a blasting operation (a step found necessary when annealing in air). Salt which adheres to the copper surface will be removed in nitric acid, causing no preferential etching to occur.

To date this technique has been applied only to the removal of cold work between drawing or sizing steps used to make small quantities of tubing for extrusion cladding. However, it might also be applicable to the annealing of inner sleeves for production coextrusion assemblies.

[redacted]

DECLASSIFIED

IRRADIATION SERVICES

ISOTOPE PRODUCTION

Xenon-128 for Argonne National Laboratory

A quantity of 160 cc(STP) of Xe-128 has been shipped to ANL as part of the Xe-128 development work requested. The purity of this gas was 99.8 mole percent. The final report on this program is in draft stage and will be issued shortly.

C-14 for Oak Ridge National Laboratory

ORNL has requested that all Be_3N_2 targets be discharged since the AEC has stopped production of C-14 in its facilities. Ten columns of Be_3N_2 targets were discharged from KE Reactor process tubes and test facilities to comply with this request.

MATERIALS TESTING

Creep Rate Measurement Program - Battelle-Northwest

The first creep rate measurement capsule was charged into the outlet end of the 2A General Purpose facility at KE Reactor.

Corrosion Product Transport Facility - Battelle-Northwest

The first in-reactor test has been completed, the test section has been removed for analysis, and a new test section installed.

ROUTINE IRRADIATIONS

Sixty-six activation analysis capsules were irradiated in the KE Quickie Facilities.

Three cooled tensile specimen capsules were discharged from the Water-Cooled Magazine facility.

NASA fuel development capsule 34 experienced a fuel cladding failure and the experiment was pulled into the reactor biological shield.

GENERAL

Exposure & Flux Considerations

Two documents associated with prediction of test irradiation exposures and with the flux tailoring concept were issued: (1) DUN-6909, "Power Generation Predictions Using HAMMER-HAMMER," and (2) DUN-6930, "Neutron Flux Classification Aspects in a Tailored Thermal Neutron Environment."

[redacted]

ADMINISTRATION - GENERAL

AUDITS

The "Equipment Management" audit was completed in accordance with the program furnished by the AEC Assistant Controller for Auditing through the AEC-RL Director of Finance. All major contractors of the Commission participated in this AEC-wide audit. The particular aspects of equipment management audited were (1) the accountability records maintained, and (2) procedures and practices for identifying, justifying, retention, and disposition of idle equipment.

The audit of purchasing activities (required annually by the AEC Manual) was directed to areas outside those appraised by the AEC, and included compliance with Government procurement regulations, Company procurement procedures and policy guides, and reporting requirements. No significant exceptions were noted.

A surprise count and reconciliation of the three cash funds was made, and the procedures for controlling cash were reviewed. No significant exceptions were noted.

CANNING COMPONENTS CANCELLATION

The record of purchases for cancellation of 130,000 sets of aluminum caps and cans because of the KW Reactor deactivation, was approved by AEC-RL. The cancellation charged were $20,559, a lesser amount than originally requested by Alcoa. The AEC requested that the Office of Defense Contract Agency, Pittsburgh, Pennsylvania, perform an audit of the charges. As a result of the audit, $9,840 was deducted from Alcoa's claimed loss on caps.

SUMMARY OF BUDGET CHANGES

AEC-RL was provided information on changes in costs between FY-1970, FY-1971, and FY-1972. The need for more detailed explanatory information of this type became apparent during the presentation to T. M. Yakimchick, Assistant Director for Administration, AEC-HQ, in November 1969.

D AND DR REACTOR PLANTS STATUS

AEC-RL informed DUN that D and DR Reactor Plants will not be reclassified to Abandoned-In-Place at this time. These reactors are to be maintained in Standby status.

EMPLOYMENT

Employment Summary

DUN personnel totals and employee allocations as of April 30 and May 31 are shown in appendix B.

Force Reduction

Force reductions resulting from the shutdown of KW Reactor continued throughout the month. The total reduction thus far stands at 303. One hundred ninety-nine of the layoffs were involuntary, seventy were voluntary, and the remaining losses were for miscellaneous other reasons.

Job search assistance to laid-off employees has resulted in the placement of seventy-six persons.

SAFETY

No personnel radiation exposures exceeded operational control.

Month-end safety statistics were:

Disabling injuries - May	1
- CY to date	2
Days since last disabling injury	28
Man-hours since last disabling injury	187,000

The disabling injury was sustained on May 3 by a power operator who slipped an and fell while performing housekeeping duties in the basement of the 190-KE building. The employee sustained a blow to the right forehead and a sprained wrist, and has been required to remain at home under the care of a private physician.

APPENDIX A

PROJECT STATUS SUMMARY - REACTOR FACILITIES

AUTHORIZED PROJECTS

Number & Title	Authorized Funds - $	Percent Complete Design	Percent Complete Construction	Remarks
Single-Pass Reactors				
DAP-516, Storage Building Addition - 105-KE & KW	142,000	100	0	Closed at present scope because of deactivation of KW Reactor.
DAE-518, Effluent Radioiodine Monitor - KE & KW Reactors	100,000	100	100	Acceptance testing of the 105-KE system has continued as outage time and manpower allowed. Except for final calibration with radioiodine samples, which is in process, the project has been completed (see resume on page D-4).
DAP-526, Deactivation of Hanford Production Reactor C	105,000	100	44	Redesign of the 105-C fire protection system to a fire alarm system has been completed (the change resulting from scheduled deactivation of the metal exam facility).
DCP-535, Sodium Sulfite System - 182-K Building	25,300	100	94	Critical piping tie-ins were completed.
N Reactor				
GCP-406, Improved Safety Platforms and Accesses - 100-N Area	300,000	100	88	Water Quality Lab escape hatch to roof completed May 1. Cell 1 snubber work deferred to the N summer outage.

Number & Title	Authorized Funds - $	Percent Complete Design	Percent Complete Construction	Remarks
GCP-411, Effluent Control Program - 100-N Area	1,830,000	100	75	Section II - All pipe ordered for this section has been received. Section IV - Held in abeyance pending approval of increased funding requested in Rev. 2 to the project proposal. Section V - Insulation of dump tank completed. Backfilling started May 18.
DCE-519, Replacement of Bridge Crane and Hoist System with New Crane System - 105-N Storage Basin Area	465,000	99	4	The crane vendor's sub-suppliers are on critical path for crane delivery in June. Minor structural work progressing in the storage basin.
DCP-528, Fire Protection System Improvements - 100-N	40,000	5	0	Design activity suspended by AEC-RL. $40,000 authorized of $290,000 requested.

PROJECT PROPOSALS AWAITING AUTHORIZATION

Number & Title	Funds Requested	Date to AEC-RL
GCP-411, Rev. 2, Effluent Control Program - 100-N	$2,010,000 (new total)	9/24/69
DCP-525, Fire Protection Improvements - KE	225,000	5/2/69
DCP-527, Graphite Cooling & Fog Spray - N	97,000	5/9/69
DCP-528, Rev. 1, Fire Protection System Improvements - 100-N	394,000 (new total)	4/1/70
DCP-529, Gravity Drainage System and Disposal Basin - 100-N	200,000	7/10/69

Number & Title	Funds Requested	Date to AEC-RL
DAP-530, Upgraded Electrical Services and Lighting - 1100-N and 1101-N Buildings	78,000	10/2/69
DAP-531, Establish 1102-N Office Building - 100-N Area	45,000	8/19/69
DCE-532, Isolation of Process Coolant Risers - K Reactors	270,000*	9/25/69

*Additionally, this project requests the use of $240,000 of operating funds.

PROJECT PROPOSAL PREPARATION

Number & Title	Design Criteria	Project Proposal
DCP-527, Rev. 1, Graphite Cooling & Fog Spray - N	Completed	In approval status
Export Water System Backup - 182-D (for 200 Areas)	Completed	In approval status
Stack Monitoring Improvements - 100-N Plant	Completed	In preparation
Heat & Ventilation System Improvements - 105-N & 109-N	Completed	In approval status
High Pressure Injection System Improvements - N Plant	Completed	In approval status
Upgrade Flux Monitor System - N Reactor	Completed	In approval status
Smoke Density Monitors for DUN-Operated Facilities	Completed	In approval status
Reactor Instrumentation Critical Power Supplies - N Reactor	Completed	In preparation

APPENDIX B

EMPLOYMENT SUMMARY
(with employee allocations)

	4/30/70	5/31/70
CONTRACT PERSONNEL		
02 Programs		
Douglas United Nuclear	1353	1297
Assisting Other Contractors	19	18
Total - 02	1372	1315
Other Programs Under AEC Contract		
Assisting Other Contractors and WPPSS	36	35
Special Irradiations	10	8
Other Programs - Standards	1	2
Total - Other Programs	47	45
Total Contract Personnel	1419	1360
COMMERCIAL ACTIVITIES PERSONNEL	20	24
TOTAL FORCE	1439	1384