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C - 57 AND C - 65

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DOUGLAS UNITED NUCLEAR MONTHLY REPORT

JUNE 1970

[REDACTED]

DOUGLAS UNITED NUCLEAR, INC.

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Verified By J E Savely 12-16-94

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C-57 & C-65

This document consists
of 35 pages. [REDACTED]

July 17, 1970

DOUGLAS UNITED NUCLEAR

MONTHLY REPORT

June 1970

DOUGLAS UNITED NUCLEAR, INC.

Richland, Washington

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

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68. DUN Record

TABLE OF CONTENTS

	<u>Starting Page</u>
<u>SUMMARY</u>	A-1
<u>REACTOR PLANT OPERATIONS</u>	
KE Reactor	B-1
N Reactor	BN-1
<u>FUEL AND TARGET FABRICATION</u>	
KE Reactor	C-1
N Reactor	CN-1
<u>TECHNICAL ACTIVITIES</u>	
KE Reactor	D-1
N Reactor	DN-1
<u>IRRADIATION SERVICES</u>	E-1
<u>ADMINISTRATION - GENERAL</u>	F-1
<u>APPENDIX</u>	
A. Project Status Summary	G-1
B. Employment Summary	G-4

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SUMMARY

REACTOR PLANT OPERATIONS

Production Statistics

	<u>KE</u>	<u>N</u>
Input Production - Pu (KMWD)	81.5	0
Time Operated Efficiency - %	71.9	0
Steam Availability to WPPSS - %	-	0

KE Reactor

Equilibrium power level was restricted to 3910 MW by brittle fracture tube power limits. There were two outages: one scheduled for charge-discharge, and one caused by a thorium fuel element failure.

N Reactor

N Reactor was shut down all month for planned maintenance.

FUEL AND TARGET FABRICATION

Production Statistics (tons)

	<u>For KE</u>	<u>For N</u>
Billets Extruded	-	89.2
Finished Fuel Produced	200.2	12.2

KE Fuel

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

N Fuel

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

TECHNICAL ACTIVITIES

KE Reactor

A final report of the KE Reactor proof test concludes that (1) it is safe to remove the interim riser pressure limits and return to a normal TORP of 460 psi, (2) at least 1000 additional pressure cycles can be incurred safely

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without fear of brittle fracture initiation, and (3) the question of water hammer effects does not compromise the conclusions.

The analysis of data from PTA-176, "Half-Plant Low Dichromate Evaluation at KW Reactor", was completed and the results indicate that 0.5 ppm dichromate concentration in the coolant is adequate to control groove and ledge corrosion when fuel surface temperatures are below 120 C and/or exposures are less than 75 operating days. At temperatures above 120 C coupled with exposures exceeding 75 operating days, a 1.0 ppm dichromate concentration is recommended.

One monitor column of the 10 kilogram PuAl irradiation was discharged to provide material with an intermediate Pu-240 content for chemical analysis.

Studies were completed as a basis for authorizing charging C3E fuel in KE Reactor to supplement C2N and O3N fuel previously approved.

Calculations of expected shielding requirements and dose rates from the single column irradiation of americium were completed during the month.

N Reactor

Additional test results completed on the Chase-fabricated tube 0758 were consistent with previous results obtained from Harvey tubes 1756 and 2455. The hoop stress required for burst failure at room temperature for irradiated tube specimens was 130 to 137 ksi with 2-3 percent elongation in the hoop direction. Burst test results at 300 C indicate that the 300 C failure strength of the N tubes is still increasing with exposure without any detectable loss in ductility.

A total flux value of approximately 1.7×10^{14} n/cm² sec. was determined for a full loading of 94 Metal at the unclassified power level of 4000 MW, to be considered for use as an unclassified total flux when discussing potential special isotopic production rates in N Reactor.

Although the use of Hanford 230 kv line No. 3 to relieve loads on lines No. 1 and No. 2 is no longer required, recommendations were made to continue leasing the line, based upon operability, reliability and maintainability factors.

The wear tester for the tube scratching investigation has produced Zircaloy wires of the larger size known to occur in reactor with the variable speed transmission installed.

Planning, tooling, and procedural preparation have been essentially completed in the program to upgrade the N Reactor ball 3X graphite channels. Actual work is planned during the current N outage.

A design criteria and project proposal have been submitted to the AEC-RL to provide a more reliable power source for certain critical instrumentation at N Reactor.

A program has been initiated to minimize the potential for river debris entering

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the circulating raw water system. Initial work involved improvement in the trash rack system at the 181-N building.

GENERAL

A five-year extension of the DUN prime contract with the Atomic Energy Commission was signed. Under the contract terms the Company will continue to operate Hanford's fuel fabrication facilities and production reactor complex for the Commission.

Force reductions resulting from the shutdown of KW Reactor are essentially on target at the end of FY 1970. During the six-month period, January 1, 1970 to June 30, 1970, 325 employees were removed from the rolls; 109 were placed with other employers.

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

Charles D. Harrington

Charles D. Harrington
President

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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)	81.5
- U-233 (Equiv. KMWD)	.6
Power Level (MW) - Maximum	3,910
- Average	3,777
Time Operated Efficiency - %	71.9
Number of Outages	2
Number of Startup Interruptions	0
Operating Coolant Flow - 1000 gpm	200.7
Fuel Charge (Tons) - 94 Metal	0
- Natural U	6
Fuel Element Failures	1 (Thoria)
Helium Losses - 1000 cu.ft.	150.2

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OPERATING EXPERIENCE

Reactor Loading

The front face map showing the loading in KE is reproduced on page B-5. The tonnages listed are approximate; actual fuel charge totals are tabulated above.

Power Level

The power level at KE Reactor was restricted by brittle fracture tube power limits during the month.

Reactor Outages

<u>Date Down</u>	<u>Outage Hours</u>	<u>Remarks</u>
June 2	39.7	Thoria rupture 0174.
June 24	162.5	Scheduled charge-discharge

Fuel Failures

A thoria fuel element (lot TC-021 H, charged 10/29/69) failed in process tube 0174 on June 2. The previous thoria fuel element failure occurred on November 29, 1969. All shield thoria (rows 1 and 58) were replaced as a result of the failure in tube 0174.

Utilization of Surplus Type O and C Fuel

Authorization was received to charge the surplus type O and C fuel in specified sections of KE Reactor. During the June 24 outage, 1,548 pieces of O3E green end fuel, 410 pieces of C2N green end fuel, and 211 C3 water mixers were utilized in the fuel charges of KE Reactor. It is planned to use approximately 46,000 pieces of surplus fuel in KE Reactor during the next fiscal year.

EQUIPMENT EXPERIENCE

Vertical and Horizontal Rods

Nos. 37 and 45 VSRs were replaced with universal flexible rods. Presently, there are 33 universal flexible rods and 8 rigid rods in service.

All HCRs are serviceable.

Resistance Temperature Detectors (RTDs)

Six RTDs were replaced during June, bringing to 74 the total that have failed since the rear face hardware decontamination was performed in February.

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Front Face Connectors

One hundred ninety front face connectors that did not meet EMS requirements were replaced. A waiver has been issued to remove and inspect an additional 500 (estimated) front face connectors as conditions permit.

Boiler Modifications

The No. 3 165-KE boiler was returned to service on June 25 following the completion of the exception list items associated with the boiler modification program. No. 2 165-KE boiler has been removed from service to complete similar modifications.

Boiler Steam Temperature

An engineering review of K boiler operating characteristics is being made in an effort to determine what course of action, if needed, should be taken to reduce steam temperatures of boilers. EMS 209 specifies a steam temperature limitation of 750 F. This temperature has normally been exceeded and covered by waivers to permit temperatures to 800 F. The steam temperatures are not a recent development. A review of the original ATP (January 26, 1955) reveals average steam temperatures of 820 F, 775 F, and 780 F on boilers 1, 2 and 3, respectively, during the eight-hour boiler capacity test run at 53,000 to 58,000 pounds steam-per-hour demand.

PROCESS ASSISTANCE AND CONTROL

Operational Physics

During the unscheduled outage of June 2, 1970, the top and bottom thoria shield protection loading was replaced with alternate tubes of natural uranium and lithium. Observed shield temperatures and reactivity effects were as expected.

A total of 36 loading changes in the PuAl test block were completed during the June 24 outage. Natural uranium replaced 29 tubes of 94 Metal, 6 tubes of thoria, and 1 tube of PuAl.

Approximately 140 splines are needed to satisfy total control during scheduled outages. With 140 splines in the unit, six nonadjacent VSRs may be out of service and 16 HCRs must be fully inserted.

Operating during the report period has been stable, requiring a minimum of physics assistance.

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

Effective Central Tubes*	2186
Flattening Efficiency** - June	0.712
- 12-Month Average	0.716

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Maximum Operating Time Permitting 10
Scram Recovery - Hours***

Average Critical Prediction Error - June (c-mk) 145
- 12-Month Average 132

*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.

Thoria Target Element Failure

The thoria target element which failed in fringe process channel 0174-KE on June 2, was changed October 29, 1969, with 38 powdered thoria elements of the K6T model. The failure location in the target column is unknown. Column exposure at the time of failure was 1145 MWD/T (based on adjacent column exposure) with product estimated to contain approximately 3 ppm U-232. A cursory examination revealed a longitudinal split in the clad with the end cap being swollen, a condition similar to that of previous failures caused by water entry through a cap weld penetration. No further examination of the element is planned.

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<u>Zone</u>	<u>Tons</u>	<u>Material</u>
Central	219	Natural Uranium
	65	94 Metal
	1	Thoria
Buckled	82	Natural Uranium
Near and Far Fringe	48	Natural Uranium
Shield Protection	23	94 Metal (Lithium Support)
	6	Natural Lithium Support
	.6	Lithium

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)	.00
Electrical Generation (KMW) - WPPSS	.00
- 184-N	.76
Total	.76
Power Level (MWt) - Maximum	0
- Average	0
Time Operated Efficiency - %	0
Steam Availability - %	0
Number of Shutdowns - Scheduled	0
- Unscheduled	0
Fuel Failures	0

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Fuel Charge (Tons) - 94 Metal	322.53
- 125 Metal	63.18
- Natural Uranium	<u>0.37</u>

Total

Helium Losses (1000 cu. ft.)	0
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Fuel Oil Usage (bbl.)	15,832
-----------------------	--------

OPERATING EXPERIENCE

Reactor Loading

The reactor loading at month end is shown on page BN-4. A total of 199 fuel columns were discharged and 186 charged during the month; the 13 empty channels will be charged later in the outage.

Reactor Outages

The reactor remained in scheduled shutdown status for the entire month.

EQUIPMENT EXPERIENCE

Primary Loop

A total of 171 V-11 valves were worked on by construction forces. Twenty-four valves had manual operators only installed; the remaining 147 were repacked and had manual operators installed.

A total of 18 V-12 valves were replaced by construction forces during the month. In addition, five Grayloc hubs on four valves were reworked by plant forces.

A total of seven primary loop leak repairs were completed by plant forces.

Combustion Engineering work on Cell 1 was completed. Final acceptance of construction work awaits official eddy current test results of steam generators 1A and 1B.

Flow Monitor Sensing Lines

Four flow monitor sensing line Inconel tubing bundles were installed by construction forces in vertical columns 58, 59, 67, and 68. Two bundles of the old stainless steel sensing lines were removed in preparation for the replacement. Fabrication by plant forces of all but four tubing bundles was completed.

184-N Turbine Generator

Disassembly and inspection of the turbine was completed. The rotating assembly

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was shipped to Portland, Oregon, for resurfacing of the shaft journals. The work was completed and the unit returned. Reassembly of the generator is now in progress.

Resistance Temperature Detectors

Thirty-one RTDs in the all tube temperature monitor circuit were repaired. New repair techniques and procedures were used.

Transfer Area Cranes

The low bridge crane was removed after salvaging the motor, circuit breaker, and relays. Eight runway rails were installed for the replacement cranes; four turntables are in place and checks are being made.

Ball Safety System

Ninety-seven ball hoppers were functionally tested by tripping the circuit and allowing the balls to enter the reactor. Seventeen of these ball hopper trip mechanisms have been given an overhaul that includes disassembly, inspection, cleaning, replacing of "O" ring seals, and reassembly.

Equipment Modifications

The following equipment modifications were completed:

MDC-N-70-30, "Pressure Gages in HCR Pump Suction Lines in Left Rod Room" provided for installation of pressure gages in the suction lines at 2 and 4 inlets to allow observation of the pump suction pressure and thus determine the operating characteristics of the pumps.

DC-3014, "Exhaust Air Monitor - Power Supply Revision", authorized revisions to minimize the risk of damage to the exhaust air monitor scintillation tube.

PROCESS ASSISTANCE AND CONTROL

Maintenance work during the outage has required that the control rod cooling water be shut off for short periods. Current total control calculations verified that the loss in control rod strength due to coolant shutoff does not significantly affect capability to meet nuclear safety requirements.

No operational physics parameters are reported due to the extended outage.

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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							E		E	F	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E							33	
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31							E	E	E	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	E	E	E					31
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29							E	E	E	X	X	94	X	F	G	G	F	G	G	G	G	G	G	G	G	G	X	X	X	X	E	E	E			29
28							E	E	E	X	X	G	G	G	G	G	G	G	F	F	G	G	F	F	F	F	G	X	X	X	E	E	E			28
27	E	E	E	X	X	X		G	F	G	G	G	G	G	G	G	G		F	F	G	G	G	G	G	G	X	X	X	X	E	E	E		27	
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18	E	E	94	X	X	G	G	G	G	G	G	G	G	G	G	G	G	F	G	G	G	G	G	G	G	G	G	G	X	X	X	E	E		18	
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16	E	E	X	X	X	G	G		N	F	G	G	G	G	G	G	G	G	G	F	F	G	F	G	F	G	G	G	X	X	X	E	E		16	
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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		

Fuel Code	No. Tubes	Description	PT-NR No.	No. Tubes	Description
E	220	Mk-IC (94 Metal - Fringe)	94*	7	Mk-IV Demonstration
F	91	Mk-IV (94 Metal - High U-236)	01	2	Mk-IV From Direct Cast Billets
G	439	Mk-IV (94 Metal - Central)			
N	1	Mk-IB (Natural U)	■	1	Graphite Sample Channel
X	230	Mk-IA & Mk-IV (125 & 94 Metal)		10	Total PTs
	981	Total			
	23	Total PTs and Empty Tubes	□	13	Empty Tubes (Scheduled to be loaded at a later date)
	1004	Grand Total			

*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

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DUN-6850

FUEL AND TARGET FABRICATION - KE REACTOR

PRODUCTION

General

Production of AlSi-bonded fuel for KE Reactor was 82.7 percent of forecast. Canning continued on the basis of day-shift operation, 5 days/week, with two lines for one day and one line for twenty-two days. Ninety-nine percent of the elements had self supports attached.

Acceptable Fuel Elements Produced

Tons	200.2
Yields: Current Month	94.7%
FY to Date	95.5%

Month-End Inventories - Tons

Bare Uranium Cores	682*
Finished Fuel: AlSi-Bonded	1,950*
Hot-Die-Sized	19
Thoria Elements	14

*These totals include 73 tons of bare and 148 tons of finished fuel in sizes used in the smaller reactors.

OPERATING EXPERIENCE

Canning efficiency of the AlSi-bonding lines was 99.8 percent. Downtime was assigned 77 percent to equipment malfunctions and 23 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	545
Tons extruded	89.2
Percent of forecast	892.0

Output Production

Total finished fuel assemblies	542
Tons output	12.2
Percent of forecast	101.7

Uranium Utilization - % 89.0

Month-End Inventories - Tons

Bare uranium billets	94
Finished fuel	107

OPERATING EXPERIENCE

An accelerated extrusion schedule was begun the latter part of April and continued through May and June.

EQUIPMENT EXPERIENCE

Nothing significant to report.

PROCESS ASSISTANCE AND CONTROL

The vacuum on all billet assemblies is measured after the preheat and just prior to extrusion. If a certain level of vacuum has not been maintained, the billet assembly is not extruded for fear of an unbond. The rejected assemblies are then disassembled to reclaim the Zr-2 cladding and uranium. Occasionally (approximately 10 times per year) during disassembly, the inner cladding sticks on the uranium billet. Previous methods of removing the stuck clad have been pressing or machining it out. Both methods are undesirable. The cladding is always destroyed by machining and unusable about 50 percent of the time when a pressing operation is used.

It is possible to free the cladding by expanding the uranium away from it by taking advantage of the greater thermal expansion of uranium over that of

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Zr-2. This technique has been successfully demonstrated with no deleterious effects to the cladding or uranium. Performing this technique includes putting a mandrel through the cladding ID and a nut is screwed onto the mandrel. The nut OD is smaller than the billet ID but larger than the cladding ID; therefore, the weight of the billet is supported by the cladding resting on the nut. The assembly is then placed in a sieved container which is lowered into a salt bath furnace at ~600 C. After heating for 15-30 minutes the container is slowly withdrawn from the furnace with the molten salt flowing back to the bath through the holes. The container is placed on the floor and the mandrel containing the cladding is easily withdrawn. The billet is protected from oxidation by a thin film of salt and is water quenched after a short cooling period.

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TECHNICAL ACTIVITIES - KE REACTOR

RESEARCH AND DEVELOPMENT

Basic Production

Half-Plant Low Dichromate Evaluation

Analysis of data from PTA-176, "Half-Plant Low Dichromate Evaluation at KW Reactor", was completed. The primary purpose of this test was to provide a quantitative comparison of fuel element 8001 aluminum alloy cladding corrosion in cooling water containing 0.5 and 1.0 ppm sodium dichromate inhibitor. Another purpose was to provide information on cladding corrosion as a function of temperature and as a function of time at both inhibitor concentrations.

Equations were developed to describe and predict fuel element weight losses at both dichromate concentrations and the effect of dichromate concentration on localized corrosion was clearly established. The corrosion equations show that fuel weight losses can be expected to increase by a factor of 1.45 at a dichromate concentration of 0.5 ppm as compared to exposure at 1.0 ppm dichromate. Little effect of dichromate concentration on localized corrosion was observed when fuel cladding temperatures were below 115-120 C, but above these temperatures the localized corrosion was definitely more severe in coolant containing 0.5 ppm inhibitor. The localized corrosion almost always occurred on the bottoms of the fuel elements, since these surfaces operate at a higher temperature. The bottom rows of fuel element supports corroded away rapidly at the higher temperatures, causing a further increase in local temperatures due to flow imbalances which in turn accelerated the localized corrosion. Because the higher dichromate concentration reduced the amount of support corrosion, 1.0 ppm dichromate should be used to control localized corrosion whenever cladding temperatures are expected to be above 120 C for long periods of time.

The tests indicated that corrosion occurring at exposures of 66 operating days with clad temperatures of 120 C or above would not seriously limit fuel integrity at 0.5 ppm inhibitor, but extensive pitting corrosion would be expected for exposures of 105 operating days. However, it was found that at the longer exposure periods and higher cladding temperatures operation at 1.0 ppm dichromate would reduce but not completely eliminate localized corrosion.

Brittle Fracture Program - K Inlet Piping

Proof Testing of KE Piping

A final report of the KE Reactor proof test is almost complete. The report includes a description of the test details and results, but the main body is concerned with interpretation and evaluation of the test. Although some of the numbers have changed since Safety Analysis Report (HW-74095 SUPP5) was issued, the basic assumptions, methods of calculation and conclusions,

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UN-6850

are essentially the same. Significant conclusions from the test include:

- It is safe to remove the interim riser pressure limits and return to a normal TORP of 460 psi.
- At least 1000 additional pressure cycles can be incurred safely without fear of brittle fracture initiation.
- The question of water hammer effects does not compromise the conclusions.

Product Flexibility

Ten Kilogram PuAl Irradiation (PTA-150)

The eleven 28-piece PuAl columns operated during the month at an average tube power of approximately 1030 kw and approached an exposure of 17,000 MWD/T (peak Pu-240 is calculated to occur at approximately 28,000 MWD/T). The shorter charges in the two monitor columns (containing six and eight PuAl pieces respectively) had an exposure of approximately 20,000 MWD/T. In order to have material at an intermediate Pu-240 content available for chemical analysis, one of the monitor columns (No. 4481) was discharged during the outage in late June. In addition, six thoria and 29 E-metal columns were replaced with natural metal to compensate for the decreasing PuAl tube powers, and thus maintain an average tube power of 1000 to 1100 kw.

Pu-238 Studies

Calculations of expected shielding requirements and dose rates from the single column irradiation of americium were made during the month, essentially completing the calculations necessary for the technical report on the americium irradiation.

The 2x2 column array of NpO₂-Al thin shell target elements charged in KE Reactor continued to operate satisfactorily. A 3-month irradiation is planned, and a 3-month cooling period will be required before the targets will be processed for recovery of the neptunium and plutonium-238.

Environmental and Regulatory Technology

Nothing to report.

Waste Management

Effluent Isotope Concentrations

Calculations were made to determine the effect of 45- and 90-minute holdup times on the decay of the 20 most significant radioisotopes contained in the KE Reactor effluent water. Except for Mn-56, Si-31, and Sr-92 (half-lives less than three hours), there was no significant reduction in concentration. Concentrations in the Columbia River, assuming complete mixing and appropriate decay times, were calculated for normal maximum and minimum

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DUN-6850

river flows at the K reactor release site and Ringold. Concentrations were below the 10 CFR 20 limits at both locations; however, it cannot be assumed that complete mixing occurs instantaneously at the K reactor release site.

Routine analysis of KE effluent water for Mn-56 was initiated on June 10 following an informal request from AEC-RL.

ENGINEERING AND TECHNOLOGY - REACTOR

Utilization of B and C Reactor Fuel

The April 1970 report indicated that thermal-hydraulics studies were completed to determine the feasibility of fringe and central zone loading of B and C Reactor type fuels in KE Reactor. Those studies considered C2N and O3N fuels in mixed and/or full-length columns.

Additional study has been made of the C3E fuel, and the required safety analysis was completed. Formalized documentation is being issued to authorize the following column charges in the specified tube locations.

1. Full-length (35 element) C2N columns in central zone aluminum tubes.
2. Full-length (43 element) C3E columns in locations outside of the enrichment ring.
3. Mixed C2N - K4N columns in central zone aluminum tubes.
4. Mixed O3N - K4N columns (up to 12 O3N elements in upstream position) in the outer three lattice units.

Project Status Summary

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

ENGINEERING AND TECHNOLOGY - FUELS AND TARGETS

Precision Extruded Cores

There are economic and technical incentives for testing fuel cores precision extruded to K5E dimensions utilizing the N extrusion facilities. These cores cannot be fabricated to the same precision dimensions as machined cores and could present fabrication problems in aluminum jacket assembly and clad testing. For this reason seventy K5E extruded cores were AlSi canned and tested through UE-1 tester.

There was some difficulty at the AlSi canning bath in seating the aluminum spires. This problem was caused mainly by the difference in chamfer on the extruded cores from that on the regular cores. There were no bond test rejects, and examination of the bond revealed nothing unusual in the bonding layers.

Five tubes of these K5E cores will be charged in KE Reactor on a production test to complete data for this program.

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TECHNICAL ACTIVITIES - N REACTOR

RESEARCH AND DEVELOPMENT

Basic Production

Accountability Analyses

Remeasurement by ARHCO and recompilation of the detailed operating history in the high U-236 test (PT-NR-95), combined with RIBD code analyses, have failed to resolve the discrepancy between expected plutonium yields and experimental results. Additional comparisons with BNW sampling data and large-scale separations data from both NFS and Hanford will be required before prediction renormalization for accountability purposes would be warranted.

Average Total N Reactor Neutron Flux

This subject was reviewed to determine the representative flux associated with the unclassified power level of 4000 MW. A total flux value of approximately 1.7×10^{14} n/cm²/sec was determined in a full loading of 94 Metal. DUN-7067 has been issued to document the calculation and the flux values which may be feasibly considered as (1) an unclassified total flux when discussing potential special isotopic production rates in N Reactor, and (2) classified fluxes at specific energies and peaking conditions.

Nuclear Protection Studies

The Mark I fuel lattice has been run on HAMLET and EGGNIT to obtain four-group cross sections. These cross sections were then used as input into DTF and HFN to determine the infinite multiplication of the lattice. The EGGNIT cross section set gave a multiplication on the order of 20 mk higher than the HAMMER cross section set.

Progress was made in the development of a consistent and complete system of physics computer code treatment of N Reactor reactivity control for accidents initiated at high power level. The HAMMER code was found to converge on actual control rod geometry with special treatment used for the accompanying THERMOS code. Additional effort can now be spent on determination of fuel-rod configurations best utilizing the code methods which can now describe spectrum effects for control rod absorptions in place of the gross parameters otherwise necessary to approximate "rodded fuel" regions.

MOFDA calculations on inner channel voiding of Mark IV fuel give a reactivity effect of 0.35 percent $\Delta k/k$ on k_{∞} and 0.3 percent $\Delta k/k$ on k_{eff} . DCODE or HAMMER runs will be made to check this result.

Quarter-core mockup runs with the DUNCAP version of the FLARE 3-D code have been completed, and progress is being made on the full-core mockup. Currently, the full-core model is being adjusted to an N Reactor factor map, assuming no rod perturbations. Following normalization to empirically tested

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conditions, the code is to be employed in developing an extensive calibration of the N Reactor control rod system.

Zircaloy Process Tube Monitoring

Additional test results completed on the Chase-fabricated tube 0758, which was removed from N Reactor in February 1969, were consistent with previous results from Harvey tubes 1756 and 2455. The hoop stress required for burst failure at room temperature for irradiated tube specimens was 130 to 137 ksi with 2-3 percent elongation in the hoop direction. These results are well above the failure strength of unirradiated tubing and fall within the scatter band of previous room temperature test results for irradiated PRTR and N Reactor tubes.

Burst test results at 300 C indicate that the 300 C failure strength of the N tubes is still increasing with exposure without any detectable loss in ductility. Failure stress at 300 C for irradiated specimens was about 90,000 psi with 20-30 percent elongation. Unirradiated specimens fail at about 60,000 psi and 20-30 percent elongation at 300 C.

Fatigue test results were also in good agreement with previous tests and indicate that the critical length of a through-wall sharp flaw is about two inches (or a wall thickness of 0.25 inches).

Through-Reactor Decontamination

The N Reactor primary system piping was decontaminated for the fourth time by the single-pass application of inhibited phosphoric acid. The decontamination operation went smoothly with excellent control of flow, concentration and temperature.

Results of the decontamination followed the trend established by previous decontaminations with final measured dose rates higher than those measured after the previous decontamination. Table 1 (next page) shows before and after decontamination dose rate readings for the four decontaminations performed to date. Activity levels in the front and rear pipe spacers were reduced by a factor of three.

Graphite Cooling System Heat Exchanger Decontamination

The Graphite Cooling System Heat Exchanger 2 was decontaminated at the 189-F facility in preparation for retubing. The procedure was the same as that used for the 1969 decontamination of Heat Exchanger 4, an alkaline permanganate pretreatment followed by an oxalic-citric acid cleaning step. Final dose rate readings on the tubes were less than 1 mR/hr.

Product Flexibility

Nothing to report.

TABLE 1

Activity Reduction From N Reactor Decontaminations

		Average Dose Rate Readings, mR/hr									
		Front					Rear				
<u>Date</u>	<u>Time</u>	<u>Elev.</u> <u>Center</u>	<u>Elev.</u> <u>Apron</u>	<u>Nozzle</u>	<u>Connec-</u> <u>tors</u>	<u>Pipe</u> <u>Space</u>	<u>Elev.</u> <u>Center</u>	<u>Elev.</u> <u>Apron</u>	<u>Nozzle</u>	<u>Connec-</u> <u>tors</u>	<u>Pipe</u> <u>Space</u>
10/3/67	Before decon	140	200	290	1000		110	210	440	280	
	After decon	12	24	33	20		32	80	150	50	
7/21/68	Before decon	47	82	74	308		93	203	252	317	
	After decon	13	35	34	34		35	75	110	70	
7/24/69	Before decon	132	192	234	573		148	390	419	550	
	After decon	29	44	103	59		57	113	220	58	
6/2/70	Before decon	102	162	192	318	301	131	227	533	455	444
	After decon	41	62	100	63	118	124	215	217	93	122

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Environmental and Regulatory Technology

230 kv Hanford Line No. 3

A proposal has been made by ITT/Federal Support Services for early termination of the lease with Bonneville Power Administration for Hanford transmission line No. 3. The use of this line primarily to relieve loads on lines 1 and 2 is no longer required as a result of curtailment of reactor operations during the past five years. A review was made of the Hanford transmission system operability, reliability, and maintainability with respect to the nuclear safety and economical aspects of removal of line No. 3.

To provide a basis for a management decision regarding the continued leasing of line No. 3, a reliability study was made of the Hanford transmission loop with three different line No. 3 configurations: (1) with the line connected; (2) with the line removed; and (3) with the line available but open at K Reactor (standby). The reliability of all three configurations was found to be satisfactory with a slight edge for the line No. 3 standby configuration.

In the final analysis, however, factors other than electrical power reliability were included in making the decision. The proposed requirement of operating steam electrical backup facilities when the transmission loop is open would result in an economical balance between the annual incremental cost for additional fuel versus the annual lease cost for line No. 3. For this reason, recommendations were made for continued leasing of line No. 3.

Accident Analysis Codes

The RACER code has been modified to include the capability for calculation of personnel dose on the basis of the "finite-cloud" calculational model. In addition, Pasquill stability parameter input has been streamlined. Efforts will now be directed to incorporating INDOSE into the RACER code for calculation of personnel dose from exposure to radiohalogens.

Waste Management

Nothing to report.

ENGINEERING AND TECHNOLOGY - REACTOR

Tube Scratching Problem

The variable speed transmission required to produce a constant lineal disc velocity on the wear tester has been installed and constant linear velocity testing using 1010 steel shoes has started. The constant disc velocity commensurate with normal charging velocity and a load of 75 pounds, which is characteristic when fuel coking occurs, has produced wires of the larger size known to occur in-reactor. These wires are considered capable of causing fretting failures and were produced during tests conducted with 1010 steel shoes.

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The generation of these wires appears to be dependent upon the condition of the zirconium surface. A previously scratched surface appears to be a pre-requisite for wire generation. Testing of 17-4 PH stainless steel and chrome-plated 1010 steel is being conducted in an effort to determine the zirconium surface qualities required to produce the large wires.

The decrease of approximately 15 percent in required charging pressures noted after the June decontamination is considered as a further indication of the importance of surface qualities. Shoes from one green fuel column which required discharge are being examined to investigate this observation.

Circulating Raw Water System (CRWS)

Debris Monitoring

Data have been obtained from the test screens which were installed under Development Test 218 to collect and characterize debris which passes through the traveling screens into the CRWS. Data to date show that the main constituent of the debris which passes the traveling screens at the river and which is caught on the 1/4 inch mesh test screens is grass and bits of bark or twigs. In addition an occasional small fish (2-inch) or small stick is found. On one typical run, approximately one cupful of grass was taken from the test screen after about 8 million gallons of water had passed.

River model studies were performed at WSU on June 3 and 4, to find a way to reduce the debris flowing to the pumps but no obvious solutions were found. A report will be published shortly.

A-B Electric Bus Independence

Engineering efforts have continued to determine the reasons for CRWS flow instability which could cause the loss of B bus electrical power because of system upset resulting from the loss of the A bus powered CRWS pumps. Much data analysis and discussions with W. G. Love, a General Electric Company consultant, have confirmed that the general approach being taken to solve the problem is appropriate. Additional tests are planned in an attempt to identify the causes. As part of the additional testing, viewing ports will be installed in the 184 Building turbine generator (TG) condenser, air-trapping baffles will be installed in the CRW supply and effluent lines at the TG, and more exacting TG flow measurement will be provided.

Reactor Instrumentation Critical Power Supplies

Engineering in support of a project proposal to provide a more reliable AC power source for certain instrumentation at N Reactor has been completed. It is proposed that static inverters be provided to assure continuous supply of power to those instruments considered essential during an event in which both A and B bus power supplies are lost. The instruments selected to be supplied from the inverter source are those which provide information related to reactivity and temperature in the reactor and radiation conditions within and adjacent to the reactor building.

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Design Criteria DUN-6851, and Project Proposal DCE-544 have been approved and submitted to AEC-RL for approval.

River Pumphouse Sluice Gate Grating

On a number of occasions debris has been found at several locations within the Circulating Raw Water System indicating a potential for plugging of some important cooling subsystems. A study of the river pumphouse, the primary point of raw water cleanup, was undertaken to develop a corrective course of action.

The river water passes through a trash rack, designed to prevent the passage of large objects, and then through the traveling screens designed to prevent the passage of smaller debris. It was found that debris could enter at the top of the trash racks through the existing sluice gate slots, 11-1/2' x 1-1/2' openings, which provide access for isolation sluice gates between the trash racks and traveling screens. Each year during the spring runoff when the river is most heavily debris laden, it rises several feet above these openings and there is a fair probability for potentially damaging objects to reach the traveling screens.

Heavy grating with 4" x 1" openings has been installed to prevent debris from entering the openings. The installation was completed on May 15, 1970 prior to the 1970 spring runoff. This completes the first of several steps to improve the raw water cleanup at the 181 river pumphouse.

Ball 3X Graphite Channel Modifications

During 1969 some distortion and misalignment of the ball channel liner due to shifting of the graphite layers in the moderator stack were discovered. This misalignment caused concern relative to the potential for lost balls during a ball 3X trip. A corrective course of action including the following major steps was developed: (a) removal of the upper section of the separated ball channel liners, (b) installation of a new graphite liner sleeve having end keying to maintain the ball passage at the sleeve junctions, and (c) modification of the ball entry funnel to make it longer and more flexible to accommodate angular and horizontal misalignment of the graphite in relation to the horizontal shield opening. Preparation for the work included construction of a graphite mockup to simulate the ball hopper casing, thermal shielding, and 7.5 feet of graphite stack for tool testing and procedure development prior to on-reactor activities.

At present all major tooling fabrication is nearing completion and mockup testing and procedure development are in progress in preparation for later use in the current N outage.

Project Engineering

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

ENGINEERING AND TECHNOLOGY - FUELS AND TARGETSOnsite Uranium Billet Fabrication

Two SS Material Requests were issued to AEC-RL in the past month as part of the program to demonstrate the onsite capability to fabricate uranium billets by primary extrusion and upset forging.

SS Material Request DUN-N-12 (DUN-7021) calls for delivery of 20 outer primary uranium ingots by July 17, 1970. These large ingots (10.95" OD x 2.58" ID x 21.5" length) will be primary extruded on the 333 Building extrusion press and upset forged to finish billet size. The 20 ingots should yield 40 finished billets of the new longer length (23.8 in., 521 lbs.) for a design uranium utilization of approximately 80 percent. SS Material Request DUN-N-13 requested 40 inner uranium billet "mults" (an unfinished raw billet) to be delivered by August 14, 1970. As it was felt that sufficient bare uranium primary extrusion technology could be gained on the outer size, the inner mult order will require only upset forging to finish billet dimensions with no onsite primary extrusion work.

The present timetable for completion of this demonstration work is (1) ingot and mult delivery in July and August, (2) primary extrusion and upset forging in August and early September, (3) coextrusion and fuel fabrication in September and October and (4) reactor charging in November 1970. This schedule should allow for complete fuel fabrication and irradiation evaluation by the end of FY-1971.

Tube Sizing - 3716 Building

The LaPointe Broach, obtained from Excess and installed in the 3716 Building, has been used recently to size copper and copper silicon tubing for special extrusions of clad shells which require non-standard size components. The technique consists of using the power of the broach to push a hardened steel plug through the ID of a stationary tube, expanding the tube to a larger size. Using a series of plugs and annealing the tubes between sizing steps, tubes have been expanded up to .200 inch with little accompanying change in wall thickness but with considerable decrease in length. Excellent surfaces have been obtained using Moroline petroleum jelly smeared on the sizing plug as the only lubricant. Annealing the tubes before sizing and between sizing steps is necessary to prevent extreme warp. The salt bath has been used to accomplish the annealing. The 60-inch stroke available on the broach limits the length of any piece to be sized to 30 inches which is sufficiently long for any extrusion billet that might be considered.

The advantages of being able to convert standard size tubing, which can readily be purchased on a short lead time basis, into usable billet cladding to nearly any size in a matter of hours are obvious. The offsite purchase of odd size tubing, especially in small quantities, is not only expensive but requires lead times of several months. The LaPointe Broach (combined with the Fenn draw-bench) provides a great deal of flexibility for producing such items onsite in a matter of days.

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DUN-6850

IRRADIATION SERVICES

ROUTINE IRRADIATIONS

Sixty-nine quickie activation analysis capsules were irradiated in the KE Quickie Facilities for BNW and the University of Washington.

NASA fuel development capsule 34 was discharged from a side general purpose facility and sent to Radiometallurgy for examination. Capsule 38 was recharged in its place.

A (U-Pu) O₂ fuel development capsule was irradiated in the KE Reactor Snout Facility in support of the LMFBR program.

ADMINISTRATION - GENERALPRIME CONTRACT

A five-year extension of the DUN prime contract with the Atomic Energy Commission was signed. Under the terms of the contract the Company will continue to operate Hanford's fuel fabrication facilities and production reactor complex for the Commission.

CAPITAL APPROPRIATIONS

The following project proposals were transmitted to AEC-RL Budget Division during June:

- DAE-540 - Smoke Density Monitors, 100-300 Areas
- DAP-541 - Boiler Feedwater Deaerator
- DCP-542 - High Pressure Injection System Improvements
- DCE-544 - Critical Instrument Power Supply

Project directive DCP-537 - Fencing Waste Retention Ponds and Leaching Trenches, was received from AEC-RL June 12, 1970 in the amount of \$25,000. The directive specified the use of chain link fence from 100 area shutdown facilities rather than hog wire fence and wood posts which had been recommended at an estimated cost of \$23,150.

APPROVAL LETTERS

At the close of the reporting period, final action was pending on the following requests:

<u>Number</u>	<u>Subject</u>	<u>Date of Transmittal to AEC-RL</u>
ATD-148 Add. #2 Sup. #1	Merit Salary Budget for Exempt Salaried Personnel - CY 1970	June 15, 1970
ATD-175 Add. #2	Overhead Charges Applicable to Diversi- fication Work to be Used in Determining Credits to the AEC	June 12, 1970
ATD-193 Add. #1	Computer Program Leasing	June 17, 1970

DOCUMENT INVENTORY

The Document Control Center destroyed an all-time high of 8,503 Secret documents in the paper cleanout campaign this month. For the first time since November 1, 1965, the total Secret document holdings are under 30,000.

EMPLOYMENTEmployment Summary

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

Force Reduction

Planned force reductions resulting from the shutdown of KW Reactor are essentially on target at the end of FY 1970. During the six-month period, January 1 to June 30, 1970, 325 employees were removed from the rolls; 109 were placed with other employers.

Union Relations

The Union has requested arbitration concerning the practice of sending work to the ITT/FSS reproduction facilities in the Federal Building. The Union contends that such actions are in violation of the "contracting-out" clause of the DUN-HAMTC agreement.

SAFETY

No personnel radiation exposures exceeded operational control.

Month-end safety statistics were:

Disabling injuries - June	0
- CY to date	2
Days since last disabling injury	58
Man-hours since last disabling injury	445,600

APPENDIX A

PROJECT STATUS SUMMARY - REACTOR FACILITIES

AUTHORIZED PROJECTS

<u>Number & Title</u>	<u>Authorized Funds - \$</u>	<u>Percent Complete</u>		<u>Remarks</u>
		<u>Design</u>	<u>Construction</u>	
<u>Single-Pass Reactors</u>				
DAP-526, Deactivation of Hanford Production Reactor C	105,000	100	44	Design for the 105-C fire alarm system completed and procurement funds released to J. A. Jones.
DCP-535, Sodium Sulfite System - 182-K Building	25,300	100	100	The ATP was completed on May 28. Several minor exceptions remain to be cleaned up. The project has been completed.
<u>N Reactor</u>				
GCP-406, Improved Safety Platforms and Accesses - 100-N Area	300,000	100	93	Cell 3 snubbers completed June 19.
GCP-411, Effluent Control Program - 100-N Area	1,830,000	100	77	Section II - All fabrication on ERW piping completed June 1. Transfer area encasements installed June 12. Section IV - Held in abeyance pending approval of increased funding requested in Rev. 2 to the project proposal. Section V - Field activity is complete. Resolution of the contractor's claims for extras is in progress. Tie-in of new dump tank to existing ECS in progress.

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DUN-6850

G-1

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Number & Title	Authorized Funds - \$	Percent Complete		Remarks
		Design	Construction	
DCE-519, Replacement of Bridge Crane and Hoist System with New Crane System - 105-N Storage Basin Area	465,000	99	5	Installation of crane rails started on June 22.
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)	9/23/69
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)	3/31/70
DCP-529, Gravity Drainage System and Disposal Basin - 100-N	200,000	7/9/69
DAP-530, Upgraded Electrical Services and Lighting - 1100-N and 1101-N Buildings	78,000	8/12/69
DAP-531, Establish 1102-N Office Building - 100-N Area	45,000	8/12/69**

* Changes have been made in these dates to make them conform to the plan to use the actual date appearing on the proposal itself.

** DUN has requested that this proposal be returned.

<u>Number & Title</u>	<u>Funds Requested</u>	<u>Date to AEC-RL*</u>
DCE-532, Isolation of Process Coolant Risers - K Reactors	270,000 ⁽¹⁾	8/25/69***
DCP-538, Heat & Ventilation System Improvements - 105-N & 109-N	330,000	5/18/70
DCE-539, Upgrade Flux Monitor System - N Reactor	335,000	5/15/70
DAE-540, Smoke Density Monitors for DUN-Operated Facilities	55,000	6/11/70
DCP-542, High Pressure Injection System Improvements - N Plant	138,000	6/17/70
DCE-544, Reactor Instrumentation Critical Power Supplies - N Reactor	65,000	6/15/70

* Changes have been made in these dates to make them conform to the plan to use the actual date appearing on the proposal itself.

*** AEC-RL has advised that this proposal is to be returned to DUN.

(1) Additionally, this project requests the use of \$240,000 of operating funds.

PROJECT PROPOSAL PREPARATION

<u>Number & Title</u>	<u>Design Criteria</u>	<u>Project Proposal</u>
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

APPENDIX BEMPLOYMENT SUMMARY
(with employee allocations)

	<u>5/31/70</u>	<u>6/30/70</u>
<u>CONTRACT PERSONNEL</u>		
<u>02 Programs</u>		
Douglas United Nuclear	1297	1301
Assisting Other Contractors	<u>18</u>	<u>16</u>
Total - 02	1315	1317
<u>Other Programs Under AEC Contract</u>		
Assisting Other Contractors and WPPSS	35	35
Special Irradiations	8	5
Other Programs - Standards	<u>2</u>	<u>2</u>
Total - Other Programs	45	42
Total Contract Personnel	<u>1360</u>	<u>1359</u>
<u>COMMERCIAL ACTIVITIES PERSONNEL</u>		
	<u>24</u>	<u>25</u>
TOTAL FORCE	1384	1384