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Prepared for the U.S. Department of Energy
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DOUGLAS UNITED NUCLEAR MONTHLY REPORT

SEPTEMBER 1970

[redacted]

DOUGLAS UNITED NUCLEAR, INC.

RICHLAND, WASHINGTON

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October 19, 1970

DOUGLAS UNITED NUCLEAR

MONTHLY REPORT

September 1970

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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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REACTOR PLANT OPERATIONS

Production Statistics

	KE	N
Input Production - Pu (KMWD)	83.0	32.5
Time Operated Efficiency - %	78.6	36.0
Steam Availability to WPPSS - %	-	32.1

KE Reactor

The reactor remained down in continuation of the August extended outage until September 4 when operation was resumed following AEC-RL startup authorization.

Equilibrium power level was restricted to 3,750 MW by brittle fracture tube power limits. There were two outages and one startup interruption. One outage was the continuation of the August outage, and the other was caused by an unexplained Panellit trip. The startup interruption was caused by a faulty test hole Panellit gauge.

N Reactor

N Reactor was in shutdown status at the beginning of the month. Reactor operation resumed, following the extended summer outage, on September 14. There were four unscheduled reactor shutdowns during the month, caused by the following: a low flow monitor trip; a manual shutdown to investigate a suspected leak through the primary loop dump valve, V4-4; a manual shutdown to investigate an unexplained increase in primary loop makeup rate; and a low flow monitor trip which occurred when number six drive turbine tripped to pony motor operation.

In the last shutdown described above, the automatic scram circuit deenergized the rod safety circuit, but no control rods entered the reactor. The back-up control system, the ball safety system, functioned to shut the reactor down. Investigation of the cause of the failure of the rod safety system to function properly was still under investigation at month end.

The reactor remained in shutdown status at month end. Total outage hours for the month were 472.0.

FUEL FABRICATION

Production Statistics (tons)

	For KE	For N
Billets Extruded	-	8.5
Finished Fuel Produced	34.3	37.2

KE Fuel

AlSi fuels operation continued on a partial-line basis for the production of K5E fuel.

N Fuel

Input and output production was below forecast in order to assure delivery of spike fuel to the reactor.

TECHNICAL ACTIVITIES

A program has been designed to establish a basis upon which to determine the in-service life expectancy and maintenance requirements of the V-11 GRAFOIL valve packing material.

System modifications aimed at reduction of bacterial contamination have been completed for the left horizontal control rods on N Reactor, and a surveillance program has been implemented. Testing has been initiated on the horizontal control rod mockup to check materials compatibility, fluid stability, and bactericide effectiveness.

Fabrication of the prototype subcritical drive system for the Nuclear Flux Monitor System at N Reactor is in progress and preparation for environmental testing of the drive system has been initiated.

Results of ex-reactor impact testing have determined that N Reactor fuel has sufficient strength to prevent excessive deformation in the event of an inlet connector failure accident.

A KE Reactor process tube treated by the electrolytic hydride removal process in February 1970 was found to be leaking at the rate of 10 to 12 gallons per day. Laboratory examination revealed that the leak was caused by a circumferential crack at the Van Stone flange. The flange was masked from treatment during the hydride removal operation last February. No connection between the failure and the cleaning operation is apparent. A complete analysis of the tube is in progress.

Increased emphasis has been focused on developing a recirculating decontamination process for application at N Reactor. DUN-7265, "Action Plan - N Reactor Recirculating Decontamination," was issued and outlines a program that will permit consideration of using the technique during next year's summer outage.

N Reactor graphite cooling heat exchanger No. 2 has been rebuilt, reinstalled, and returned to service.

The N Reactor fog spray system has been functionally tested for pressure surges and certain cast iron components have been replaced.

Steam collection drums which are part of the steam generators at N Reactor have been examined and defects from original construction repaired.

Intermittent once-through cooling for a shut-down N Reactor was established to effect repairs.

The current status of the N Reactor Regulatory Program was presented to the Hanford Subcommittee of the Advisory Committee on Reactor Safeguards.

The new emergency coolant supply line and emergency dump tank at N Reactor were tested following installation and accepted for reactor use.

Further testing to identify the phenomena causing circulating raw water instability at N Reactor and resulting interdependence of the two electrical power sources, was not performed during the recently concluded summer outage because of other priority outage work. Plans, procedures, and equipment are available for implementation of the test program as outage time is available.

Preliminary results have been evaluated on a confinement leak rate test at N Reactor. The leak rate was about 2.5 percent as compared to 1.4 percent the previous year.

Work continues on the contaminated coolant pilot plant. Startup of the facility is targeted for October 1, 1970.

DUN waste management practices regarding reactor effluent streams were presented to the ACRS Subcommittee on August 21, 1970.

Irradiations completed during the report period included 42 quickie activation samples for BNW and WADCO, tantalum for a DUN program, and three tensile specimen capsules for BNW. Customer irradiations were charged to produce polonium for Mound and xenon-128 for ANL.

GENERAL

Three minority employees were added to the work force during the month. In the clerical and related job category, 100 percent of the new-hire additions in the quarter ending September 30 were minorities.

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

Charles D. Harrington

Charles D. Harrington
President

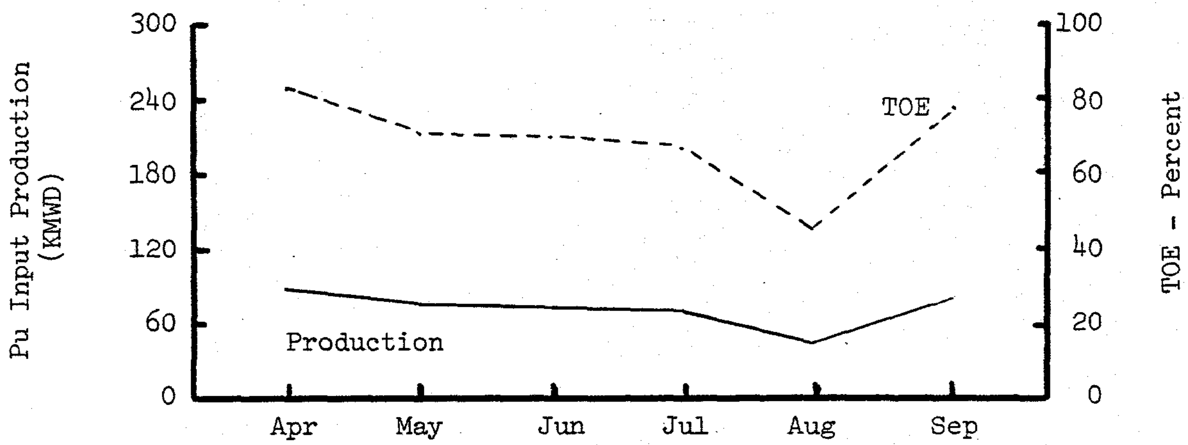
REACTOR PLANT OPERATIONS - KE

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)	83.0
	- U-233 (Equiv. MWD)	0
Power Level (MW)	- Maximum	3,750
	- Average	3,519
Time Operated Efficiency - %		78.6
Number of Outages		2
Number of Startup Interruptions		1
Operating Coolant Flow - 1000 gpm		201.7
Fuel Charge (Tons)	- 94 Metal	27
	- Natural U	215
Fuel Element Failures		0
Helium Losses - 1000 cu.ft.		187.1

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 on the preceding page.

Power Level

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

Reactor Outages

Date Down	Outage Hours	Remarks
September 1	86.0	August outage continued.
September 14	67.6	Unexplained Panellit trip.
September 17	0.5	Faulty Panellit gauge 3A test hole.

The August outage was continued until September 4 when AEC-RL startup authorization was received at 0800 hours and operation was resumed at 1400 hours.

EQUIPMENT EXPERIENCE

Vertical and Horizontal Rods

All HCRs continue to be operable with no major binding problems. All VSRs are serviceable with an average hot drop time of 1.77 seconds.

Resistance Temperature Detectors

There were no RTD failures during the month.

Panellit Scrams

Two Panellit scrams were experienced during the month. The first scram occurred on September 14 on Panellit gauge 5571. No abnormality could be found with the charge makeup, charge position, or hardware integrity. The toggle valve was found to be leaking when depressed and is believed to be the cause of the scram. Eight rows of toggle valves were replaced because of aged O-rings. The second Panellit scram was experienced on September 17 when the drive arm became disengaged from the gauge in 3A test hole.

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184-DA Boiler

The work of retubing the 184-DA boiler by the J. A. Jones Company is essentially complete and the boiler is scheduled to be placed into service during the week of October 5.

Emergency Electrical Power Backup

The program for an emergency power source for 100-K is nearing completion. The installation of equipment (750 kw diesel generator and No. 2 fuel supply for a KW boiler) was completed September 18. Acceptance tests have been completed on the diesel generator, but the boiler cold start system has not been demonstrated. It is planned during the next few weeks to complete the demonstration tests of both the diesel generator and boiler cold start. Rough-draft emergency procedures have been prepared and will be finalized following the testing program.

PROCESS ASSISTANCE AND CONTROL

Operational Physics

Operation was relatively stable during September, with one unscheduled outage and a startup interruption around the middle of the month. The samarium dip following the startup on September 4 after the prolonged August outage was made with about 570 cmk in rods and splines.

The spike inventory is high and poison requirements for total control are higher than in the recent past, due to the low average exposure of the fuel. On discharge outages about 140 splines are required for total control. In all calculations the strength of the ball system has been reduced to account for the four ball channels which are out of service.

Minimal physics assistance was required with the stable reactor condition. The scram recovery on September 17 was made without onsite physics assistance, using approved procedures.

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

Effective Central Tubes*	2,126
Flattening Efficiency** - September	.676
- 12-Month Average	.702
Maximum Operating Time Permitting Scram Recovery - Hours***	10
Average Critical Prediction Error - September (cmk)	114
- 12-Month Average	131

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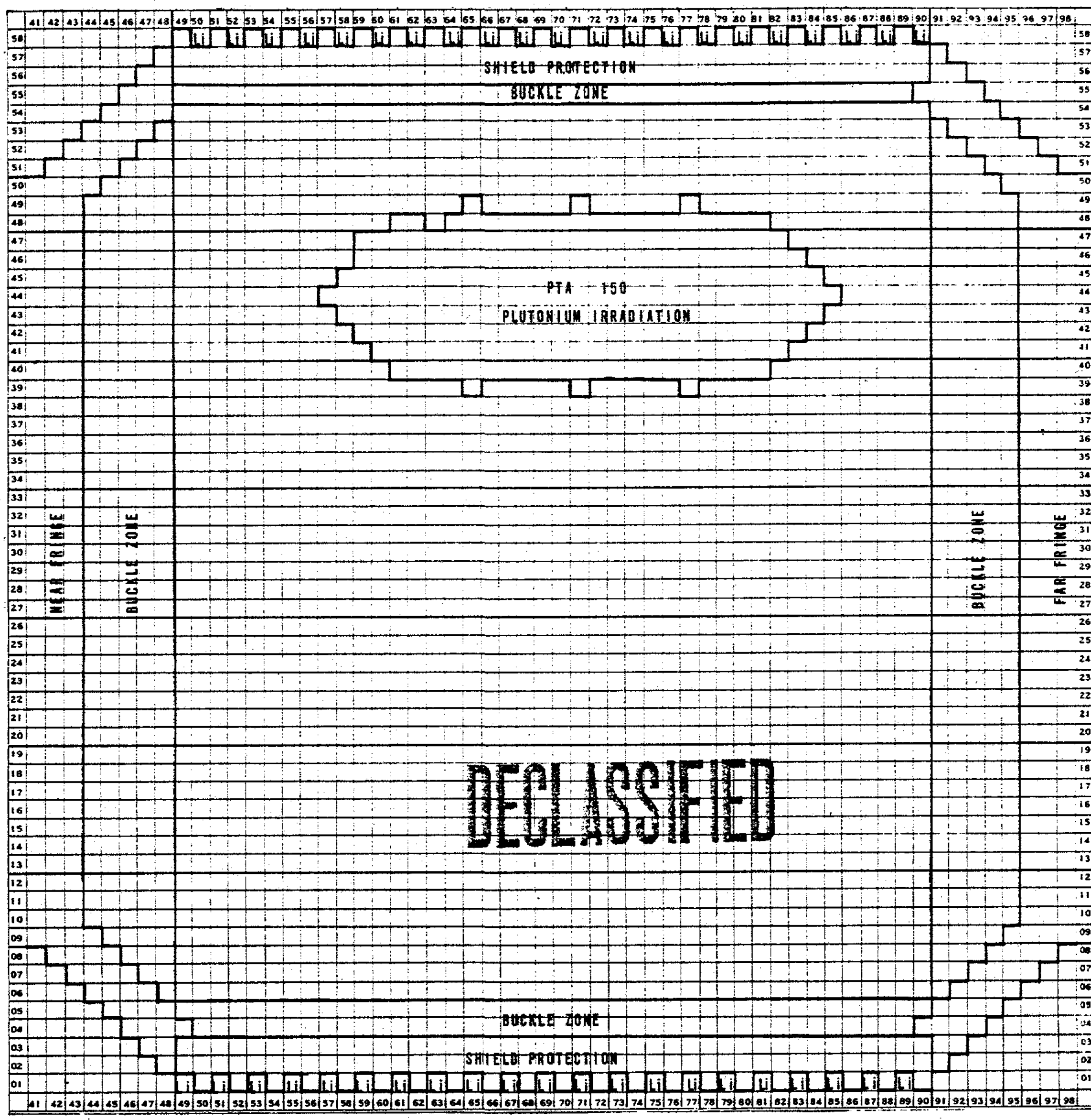
*Reactor power level divided by the average power of the ten most productive tubes which are representative of the reactor loading.

**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.



Zone	Tons	Material
Central	226 61 .23	Natural Uranium 94 Metal Bismuth
Buckled	82	Natural Uranium
Fringe	54	Natural Uranium
Shield Protection	23 .6	94 Metal - Thoria Support Lithium

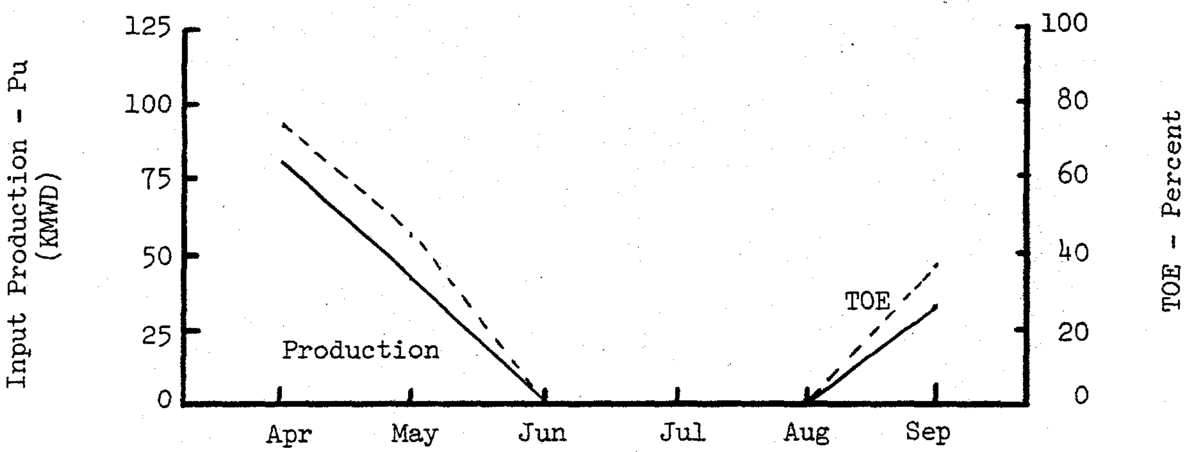
Loading Pattern - KE Reactor

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:



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Statistical Summary

Input Production - Pu (KMWD)	32.50
Electrical Generation (KMWH) - WPPSS	155.39
- 184-N	4.69
Power Level (MWt) - Maximum	4,000
- Average	3,017
Time Operated Efficiency - %	36.00
Steam Availability - %	32.10
Number of Shutdowns - Scheduled	0
- Unscheduled	4
Fuel Failures	0
Fuel Charge (Tons) - 94 Metal	327.58
- 125 Metal	65.28
- Natural Uranium	.37
Total	393.23

Helium Losses (1000 cu.ft.)	332.17
Fuel Oil Usage (bbl.)	24,596

OPERATING EXPERIENCE

Reactor Loading

The reactor loading at month end is shown on the front face map that follows on page BN-7. No fuel charging or discharging was undertaken during the month.

Reactor Power Level

The maximum authorized reactor power level of 4000 MWt was achieved during equilibrium operation. The main steam header pressure during reactor operation at power levels greater than 3600 MW was maintained above 123 psig, thus permitting an electrical generation by WPPSS of 800 MWe.

Reactor Outages

The reactor was in shutdown status at the beginning of the month as a result of the scheduled summer outage extending from the previous month. Four unscheduled outages were experienced.

Startup from the scheduled summer outage was made at 0733 hours on September 14. The reactor was automatically shut down by a low flow monitor trip at 1303 hours on September 14, and operation was resumed at 2240 hours on September 15. The reactor was manually shut down at 1534 hours on September 19 to investigate a V-4 valve leak-through problem, with startup at 1918 hours on September 20. At 1250 hours on September 27, the reactor was again manually shut down, this time to investigate an unexplained increase in primary loop makeup flow. Startup was effected at 0222 hours on September 30. An automatic low flow monitor trip scrammed the reactor at 0525 hours on September 30. The reactor remained in shutdown status through month end.

Unusual Condition

The reactor automatic scram on September 30 tripped the rod safety circuit, but no rods entered the reactor. The backup control system, the ball safety system, functioned properly and the reactor was safely shut down. Investigation of the cause of the failure of the rod safety system and the ensuing corrective action was not complete at month end.

EQUIPMENT EXPERIENCE

N Confinement System

During performance of the annual confinement system test, four significant deficiencies were detected:

- The backup closure bag on steam vent CVV-206-4 failed to inflate due to over-tightened ball detect screws on the bag canister cover.

- Visual inspection revealed that seven steam vent valves were not seating properly because the cable clamps on the valve shaft drive pulleys had slipped. As a preventive measure, all clamps were removed, dressed, and reclamped to assure a positive connection.

- The Zone I exhaust backup closure gates leaked due to a missing rubber seal on the north gate.

- The No. 2 exhaust fan valve opened automatically when the "closed" solenoid valves were deenergized. All four solenoid valves were replaced and retested.

The system was successfully tested and placed into full service.

N Rupture Monitor

The Equipment Maintenance Standard test of the sample delay time was successfully performed using a newly developed 60-channel electronic data system. This new method eliminates the hand manipulation of each sampling channel's valves, resulting in a large saving of operator and monitor time and about 24 R per year in radiation exposure to personnel.

Emergency Cooling System

The entire emergency cooling system valve gas actuating system piping, solenoids, filters, and dip legs were blown down, cleaned, and repaired to improve the operational reliability of the system prior to the startup from the scheduled summer outage. All the inlet (V-3) and outlet (V-4) valves were functionally tested and demonstrated to be within all Standard requirements.

Dump tank drain down tests were scheduled following the performance of the dump tank acceptance test procedure to provide information for analyzing the flow capacity of the new effluent control system. Following startup, there were indications of leakage of hot water into the dump tank, possibly from the V-4 or RV-2 primary loop pressure relief valve leak-through. Tests were performed to determine the source of the hot water and to provide procedural methods for control of the dump tank temperature. Conditions at the dump tank (flow and temperature) were satisfactory for full power level operation by month end.

Primary Loop

The RV-2 primary loop pressure relief valves were functionally tested satisfactorily. V-4 valve leak-through was determined to be nominal, as designed. PCSV-203-5, cell 5 isolation valve, failed to open when the valve stem parted from the valve gate. It was necessary to place the left half of the reactor on once-through cooling in order to isolate the valve for repairs. A procedure was developed which provided for intermittent once-through flow with

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demineralized water, thereby reducing the risk of using less than primary loop quality water in the system. The valve gate was removed from PCSV-203-5 and the valve bonnet replaced. The system is operating with no isolation valve on the PCS header in cell 5.

A total of seventy (70) V-11 valves was worked on by Construction forces during September. Total valves worked on during the extended summer outage are as follows:

1.	Manual operators installed only - valve not repacked.	142
2.	Valves repacked only - operators previously installed.	47
3.	Valves repacked and manual operators installed	492
	Total	681

Valves yet to be repaired:

1.	Install manual operator only	3
2.	Install manual operator and repack	22
3.	Repack only	52
	Total	77

Two V-12 valves were replaced during this report period; the total number replaced during the summer outage is now 84. However, two additional Grayloc hub leaks were repaired for a total of 36 hubs repaired on 30 valves.

Raw Water System

Development Test No. 233, Fog Spray Surge Tests, was performed on September 4 to determine the automatic start mode for the diesel driven fog spray pumps that will limit surge pressure to a maximum of not more than 267 psi, and to determine the effectiveness of the accumulator tank in performing the function of a surge suppressor. Higher rated flanges were installed to improve the system integrity.

184-N Turbine Generator

A General Electric turbine engineer consultant was called in to help analyze turbine governor problems. A leaking hydraulic flange in the governor system was located and repaired. Control linkage was found to be binding. The binding problem was eliminated, the linkage adjusted, and the turbine returned to service.

Equipment Modifications

The following equipment modifications were completed:

EMM-26, "Relocation of Low Vacuum Trip and Alarm Switches for Drive Turbine - 109-N," provided for relocating, for convenience and ease of monitoring, the low vacuum trip and alarm switches from near the turbine exhaust plenum to the drive turbine control panel.

EMP-453, "Emergency Signals - 1310-N Building," provided for installation of an evacuation alarm system in the 1310-N Building to provide added assurance of personnel awareness to emergency conditions.

MDC-N-70-56, "W Platform Control Power Rewiring," provided for circuit changes in the W platform control console to permit one person to transfer control power from the W console station to the charge machine console without leaving the control power in "off" while changing stations.

MDC-N-70-62, "Rod Low Flow Automatic Reset," provided for installation of a variable time delay reset in the annunciator indicators for low rod flow to permit identification of the annunciator before it is obscured by the automatic reset feature of the system.

MDC-N-70-66, "Emergency Telephone Circuit Revision," provided for documentation of changes made in the emergency telephone communications system that eliminated excess circuits, reconnected deactivated stations, and corrected an erroneous intertie.

MANUFACTURING ENGINEERING ACTIVITIES

Equipment Maintenance Standards

One new N Reactor Standard was issued; of the 42 N Reactor Standards scheduled for preparation, 17 have been issued, five are being routed for final approval, 16 are being prepared, and four remain to be started.

V-12 Valve Linear Actuator

An integral linear actuator has been designed and a prototype has been fabricated. This actuator combines the valve stem packing and diverting mechanism for the V-12 valves. Actuation is effected by primary loop water acting on the underside of a piston to "divert", and higher pressure (about 2100 psi) makeup water acting on the upper side of the piston to "return to normal". The entire mechanism has only one static seal.

The prototype is currently undergoing off-reactor tests. Twelve additional units are being fabricated.

Pipe Freeze Technique

Prototype tooling has been tested that has the capability of freezing and isolating the 105-N stag-arm connectors in less than six minutes. Liquid nitrogen is used in conjunction with an easily attached self-contained freeze shell. The total weight of a full nitrogen container and attached freeze shell is less than 150 pounds.

Welding Tool Development

An automatic welding head, for welding Grayloc hubs to connector tubing, has been developed and constructed. Several test welds have been made to establish welding procedures. The test welds have passed bend, tensile, and radiograph examinations.

Grayloc Flange - Hub Repair Liners

A new repair technique is being developed for badly cut Grayloc flanges. This technique uses a thin steel liner which is inserted into the conical surface, thereby renewing this surface and the flange normal surface. The liner is welded to the outer surface of the flange.

PROCESS ASSISTANCE AND CONTROL

Reactor power level following the summer outage was restricted until September 22 to 3000 MWt or less while measurement and adjustment of rupture monitor hold-up time was accomplished.

A calculation of total control immediately following the ball drop substantiated that there was adequate protection with 27.5 mk of inserted rod and 56.8 mk of inserted balls for an overall shutdown margin of 46.8 mk.

Traveling wire flux monitor graphs made subsequent to the September 14 startup demonstrated that the residual balls in the middle of the reactor are still present, though additional burnout is indicated.

Physics assistance was given for three startups during the month, and the fourth was handled by Operations personnel using established procedures without onsite coverage by the assigned physicist.

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

Effective Central Tubes	830
Flattening Efficiency - September	0.83
- 12-Month Average	0.81
Maximum Operating Time Permitting Scram Recovery - Hours	∞
Average Critical Prediction Error - September	125 cmk
- 12-Month Average	125 cmk

*For definitions see page B-4.

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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	
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21	E	E	X	X	X	G	G	G	F	F	F	G	F	F	G	G	G	F	G	F	G	G	G	G	F	G	G	G	G	X	X	X	E	E	21
20	E	E	X	X	X	G	F	G	G	G	G	F	F	G	G	G	F	36	F	G	G	F	F	G	G	G	G	G	G	X	X	X	E	E	20
19	E	E	X	X	X	F	F	F	F	F	F	G	G	G	G	G	G	G	F	G	F	F	G	F	F	G	G	F	F	X	X	X	E	E	19
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	G	X	X	X	E	E	18
17	E	E	X	X	X	G	F	G	G	G	G	F	G	G	G	G	G	G	G	G	G	G	G	G	F	G	G	G	G	X	X	X	E	E	17
16	E	E	X	X	X	G	G	[redacted]	N	F	G	G	G	G	G	G	G	G	G	G	F	F	G	F	G	F	G	G	G	X	X	X	E	E	16
15	E	E	X	X	X	G	G	G	G	G	F	G	G	G	G	G	G	F	G	G	G	F	G	G	F	G	G	G	G	X	X	X	E	E	15
14	E	E	94	X	X	G	G	G	G	F	F	F	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	X	X	X	E	E	14
13	E	E	X	X	X	G	F	G	G	G	G	F	G	F	G	G	G	G	G	G	G	F	G	G	G	F	G	G	F	X	X	X	E	E	13
12	E	E	X	X	X	G	F	F	G	F	G	G	G	G	G	G	G	G	G	G	G	F	G	G	G	G	G	G	G	X	X	X	E	E	12
11	E	E	X	X	X	G	G	G	36	G	G	G	G	G	G	36	G	G	F	G	G	G	G	F	G	36	G	G	G	X	X	X	E	E	11
10	E	E	X	X	X	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	F	G	G	G	G	G	G	G	G	X	X	X	E	E	10
09	E	E	X	X	X	X	G	G	G	G	G	G	G	G	G	G	F	G	G	G	G	G	G	G	G	G	G	G	X	X	X	X	E	E	09
08	E	E	E	X	X	X	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	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	G	G	G	G	G	G	G	G	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	G	F	X	X	X	X	E	E	E			06
05				E	E	E	X	X	X	X	X	X	F	X	X	X	F	X	X	X	X	X	X	X	X	X	X	X	E	E	E				05
04					E	E	E	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	E	E	E					04
03						E	E	E	E	E	E	E	E	E	E	E	36	E	E	E	E	E	E	E	E	E	E	E	E						03
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	220	Mk-IC (94 Metal - Fringe)
F	89	Mk-IV (94 Metal - High U-236)
G	444	Mk-IV (94 Metal - Central)
N	1	Mk-IB (Natural U)
X	231	Mk-IA & Mk-IV (125 & 94 Metal)
	985	Total
	19	Total PTs
	1,004	Grand Total

PT-NR No.	No. Tubes	Description
94*	7	Mk-IV Demonstration
113	5	Evaluation of the End Spider Inner Support System
136	6	Mk-IV - Nonbonded End Closures
	1	Graphite Sample Channel
	19	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

FUEL FABRICATION - KE REACTOR

PRODUCTION

General

Production of AlSi-bonded fuel for KE Reactor was 103.9 percent of forecast. One hundred percent of the elements produced had self-supports attached.

Acceptable Fuel Elements Produced

Tons		34.3
Yields:	Current Month - %	92.0
	FY to Date - %	92.4

Month-End Inventories - Tons

Bare Uranium Cores		830*
Finished Fuel:	AlSi-Bonded	1,411*
	Hot-Die-Sized	14
Thoria Elements		14

*These totals include 73 tons of bare cores and 95 tons of finished fuel in sized used in the smaller reactors.

OPERATING EXPERIENCE

The overall operating efficiency for the month was 99.7%. All of the down-time was charged to equipment malfunctions.

The AlSi fuels operation continued on a partial-line (about 3/4) basis for the production of K5E fuel.

EQUIPMENT EXPERIENCE

The #5 Acme Gridley lathe has been tooled to ream the base of K5E cans to remove a fold defect area, and the operation has been successful thus far. Over 10,000 cans have been reamed with one reamer. It was necessary to use a lubricant on the tool to obtain a consistently smooth surface; however, the lubricant caused the chips to stick in the base of the cans and they could not be removed with an air jet. It was necessary to design a special can basket holder so the cans could be degreased (to remove chips) manually before being processed in the cap and can cleaning machine.

PROCESS ASSISTANCE AND CONTROL

Nothing significant to report.

FUEL FABRICATION - N REACTOR

PRODUCTION

Input Production	
Total Billets Extruded	62
Tons Extruded	8.5
Percent of Forecast	24.2
Output Production	
Total Finished Fuel Assemblies	1,440
Tons Output	37.2
Percent of Forecast	79.0
Uranium Utilization - %	81.4
Month-End Inventories - Tons	
Bare Uranium Billets	130
Finished Fuel	194

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OPERATING EXPERIENCE

The September forecast attainment was sacrificed to assure the delivery of about four tons of spike fuel required by the reactor. In addition to this inner spike run, the shop continued processing pre-extruded material and completed 80% of this material.

EQUIPMENT EXPERIENCE

The Campbell recut saw was provided with a set of quick-change stops to accommodate all current recut fuel lengths. The set consists of six basic length stops with incremental spacers to compensate for variations such as those caused by heat treatment, prior machining, or position in the fuel assembly.

PROCESS ASSISTANCE AND CONTROL

Mark I outer support inventory includes 80,000 supports that were purchased when the nominal Mark I height specification was 0.125 inch instead of the present 0.128 inch. The supports are also out of specification with respect to crown radius and crown length. The crown height (measure of crown radius) over a 0.400-inch chord is specified as 0.009 to 0.011 inch. The crown height on these supports ranges from 0.0075 to 0.0085 inch.

Tests have indicated that the use of thicker shoes will compensate for the slight deviations from specifications in the support dimensions. It is, therefore, planned that the subject supports will be used under controlled conditions with specially shimmed shoes to provide acceptable support height on the finished fuel assemblies. Use of these supports will result in a cost savings of about $15,000.

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TECHNICAL ACTIVITIES

N REACTOR OPERABILITY PROGRAM

Primary Coolant System

V-11 Valve Packing Life Test

A program designed to establish a basis upon which to determine the in-service life expectancy and maintenance requirements of the new GRAFOIL V-11 valve packing material has been initiated. Test fixtures have been designed and fabrication started. Procurement of reactor tested packing material for use as standards during testing is in progress.

Horizontal Control Rod (HCR) System

System modifications to reduce the bacterial contamination of the horizontal control rod invert emulsion hydraulic fluid were authorized by Design Change 3098, "Water Oil Invert Emulsion Fire Resistant Hydraulic Fluid for 105-N HCR Drive," and have been completed for the left bank horizontal control rods. A surveillance program authorized in PT-NR-153, DUN-7037, "HCR Invert Emulsion Hydraulic Fluid," has also been implemented. No system problems have occurred. Rod timing is within Equipment Maintenance Standards, and bacterial concentration is diminishing as predicted. Testing has been initiated on the HCR mockup to demonstrate an effective bactericide, and will be continued over the next several months to check materials compatibility, fluid stability, and bactericide effectiveness.

Temperature Monitor Systems

During the extended summer outage, 107 all tube temperature monitor and 61 zone temperature monitor resistance temperature detectors (RTDs) were repaired.

A development test, DUN-7199, "Development Test Authorization 227, Temperature Monitor Instrumentation Inside N Area Thermo Barrier 105 Building Outlet Site," authorizing the monitoring of rear pipe space, process tube, and structural steel temperatures has been issued. The data will provide a basis for the specifications for strap-on RTDs used in the All Tube Temperature Monitor System.

Nuclear Flux Monitor System

In support of Project DCE-539, "Upgrade Flux Monitor System," dated May 15, 1970, a schedule of approximately 115 drawings showing the detailed design of the upgraded nuclear instrumentation system has been made.

Fabrication of the prototype subcritical drive system, using water as the hydraulic fluid, is in progress, and preparation for environmental testing of the drive system has been initiated.

FUEL PERFORMANCE PROGRAM

N Reactor Fuel Strength Studies

A testing program to determine the strength requirements of the fuel under accident conditions has been completed. The limiting accident case identified was the failure of an inlet connector. In this event, reverse flow in the associated process channel would displace the fuel and spacer charge over the gap that exists between the most upstream spacer and inlet nozzle cap, and would exert dynamic loads upon the inlet nozzle and cap assembly, the spacer charge, and the fuel.

Previous experiments have demonstrated that, under conservatively estimated kinetic energy imparted to the fuel by reverse flow, the energy absorbed by the spacer charge upon impact with the inlet nozzle cap would be sufficient to prevent damage to the cap and the roll-joint between the inlet nozzle and process tube. It had not been adequately proved, however, that (1) the fuel would withstand the impact forces without loss of cladding integrity, or (2) the spacers would not deform in such a manner to obstruct coolant flow through the inlet nozzle port and out through the break.

These potential problems have been resolved by the recent tests which experimentally simulated the accident conditions as closely as possible in ex-reactor testing facilities. These experiments consisted of impact testing prototypic fuel-spacer columns in a vertical drop tube and evaluating results by examination of the prototypic fuel and flow testing of the spacers.

A comprehensive series of impact tests was carried out under conditions more severe than those calculated for an inlet connector failure accident with the nominal 28-inch gap between the upstream spacer and inlet nozzle end cap. The results and conclusions were as follows:

- Plastic deformation occurred in dummy fuel elements having cores with yield strengths less than 5,000 psi. With adjustments made to account for lower clad strengths at reactor operating temperatures, a core yield strength of 9,200 psi would be required to prevent excessive deformation. In contrast, the minimum in-reactor expected yield strength of uranium under the same conditions would be greater than 20,000 psi. This indicates a safety factor greater than two with respect to excessive clad deformation under the assumed accident conditions. These results and conclusions are based upon the observation that only the outer fuel cylinders would be deformed by impact. Deformation of the inner fuels did not occur because most of their kinetic energy was transferred to the outer elements by the locking clip mechanism. Excessive displacement of the inner cylinders was prevented by buckling of the adjacent downstream spacer.

- All spacers were bowed and plastically deformed by the impact. The major amount of buckling occurred in the spacer adjacent to the fuel; consequently, there would be minimal potential for flow blockage at the port leading to the inlet connector. Tests of deformed spacer columns in the full scale flow loop indicated a maximum flow reduction of only seven percent, which is considered to be negligible.

- The results of the studies have shown that, with the current spacer design and fuel charge makeup, the mechanical and hydraulic aspects of flow reversal situations would not pose any unresolved safety questions with respect to fission product confinement.

Dichromate Control as a Function of Fuel Surface Temperature

Analysis of test results (DUN-7166, "K Reactor Dichromate Test Results," July 29, 1970, Secret) has identified a relationship between fuel clad corrosion and fuel surface temperature and fuel residence time as a function of dichromate concentration. Based on the data, dichromate concentration should be adjusted upward whenever fuel surface temperatures exceed 120 C and the exposure time is greater than 70 operating days. For fuel surface temperatures greater or less than 120 C, but with exposure time under 70 operating days, no requirement for adjustment of the dichromate concentration is indicated.

In order to facilitate administration of such a requirement, correlations are being developed to enable ready determination of fuel surface temperatures from the pertinent reactor operating parameters.

PLANT LIFE PROGRAM

K Zircaloy Tube Hydriding

A KE Reactor process tube (3058) treated by the combined (including hydrogen fluoride) electrolytic hydride removal process in February 1970 was found to be leaking at the rate of 10-12 gallons/day. The leak was isolated to the rear eight inches of the tube by selective pressure testing. During tube removal the nozzle was stuck to the gunbarrel flange studs and subjected to considerable impact stress before it broke loose. A circumferential crack was observed at the Van Stone flange. The crack surface had sufficient film buildup to indicate that it was most likely not caused by the removal operation but was the source of the operating leak. The flange was not treated during the hydride removal operations last February.

No other penetrations of the tube were found in the last eight inches of the tube in visual examination in the laboratory. A fresh crack (no film) was found at a dent in the tube at about the 15-inch point. All indications are that this damage was caused during tube removal. A complete analysis of the downstream 60 inches of the tube is underway. No connection between the failure and the cleaning operation has been found at this time.

N Ball Graphite Channel Renovation

Physical renovation of the scheduled ball channels was completed on August 27, utilizing the procedures authorized by Design Change 3089. Acceptance testing of the renovated channels was completed successfully in accordance with Equipment Maintenance Standard N-103-12. A detailed report of the complete channel renovation work is in preparation.

DECONTAMINATION PROGRAM

Recirculating Decontamination Process Development

The emphasis on developing a recirculating decontamination process for application to N Reactor was increased during the month. This increased emphasis will permit consideration of the use of the recirculating technique for decontaminating the reactor at the beginning of next year's long maintenance outage.

In the previous four summers, N Reactor piping has been decontaminated utilizing a hot, inhibited phosphoric acid solution at approximately 10,000 gpm single pass. Because of limited capacity for storing hot water, concentrated chemicals, and wastes, maximum chemical contact time with this technique is about thirty minutes.

Although these decontaminations have been considered generally satisfactory, the post-decontamination radiation levels have been successively higher each year. Data show the effectiveness of the process on exposed carbon steel surfaces to be the same as always, i.e., essentially complete removal of deposited activity.

Part of the increase in post-decontamination radiation levels is attributed to incomplete cleaning of crevices and semi-stagnant areas at locations such as Grayloc couplings and ends of process tube nozzles. By recirculating the decontamination solution, the time of application can be extended for at least several hours and, if necessary, for as long as a few days. This longer application period may produce significantly better results at hard-to-clean areas. Further, by utilizing a neutral pH citric acid--EDTA or DTPA reagent--corrosion of reactor materials during the longer period should still be well within safe ranges.

An action plan for developing a recirculating process that can be considered for use in a full-plant application in May 1971 was issued as DUN-7265, "Action Plan - N Reactor Recirculating Decontamination," September 18, 1970.

OPERATIONAL SUPPORT PROGRAM

Intermittent Once-Through Cooling Procedure for Isolation and Repair of PCSV-203-5 - N Reactor

In August, a steam generator cell isolation valve (PCSV-203-5) failed upon routine testing late in the extended summer outage. Because the valve (at the cell outlet) is in direct contact with the inlet risers, the major problem was in isolating it from the reactor core during repair. The solution adopted was to sectionalize the left and right sides of the reactor, then maintain circulation on the right side and intermittently introduce single-pass cooling to the left side with the high-lift diesel pumps through the emergency cooling system piping. To assure that the procedure was fully consistent with nuclear safety requirements, it was carried out as a production test (DUN-7240, PT-NR-232, "Isolation of PCSV-230-5 Using Intermittent Once-Through Cooling").

The intermittent cooling procedure was considered to be required in order to enable efficient utilization of the capacity of the demineralized water facilities. It involved establishing the correct proportions between the period of time during which cooling was supplied and the period of flow interruption. This required analysis prior to the test to assure that:

- A "ratcheting" temperature effect would not occur, in which event the temperature decrease during the cooling period would be less than the temperature increase that had occurred during the prior period of flow interruption;

- The pressure differential after flow was restored would be adequate to reestablish cooling if boiling and tube voiding occurred during a nonflow period;

- Heat-up rate of the fuel during the flow interruptions would not be excessive.

Because of the low heat generation of the fuel that existed after the long shutdown period prior to the test and the low temperature of the graphite, it was possible to develop test procedures that would be practical from the operational standpoint and still be conservative with respect to the foregoing considerations. The test was carried out successfully. Based upon temperature measurements with in-core thermocouple probes, the following observations have been made:

- Natural convection was not as effective as originally expected in removing heat from the active zone.

- The predicted fuel heat-up rate was very conservative. Sufficient heat transfer to the moderator (which was at about 85 F during the test) prevented the stagnant water in the active zone from exceeding 104 F in the high flux region (center of column) with water shutoff times up to one hour.

High Speed Scanner - KE Reactor

The high speed scanner installed at KE Reactor measures, displays, and acts upon signals received from individual process tube outlet water temperatures. Data accumulated since installation of the high speed scanner indicate that certain trimming potentiometers originally installed by the vendor have failed after three to six months of operation. The failure mode is usually loss of setability and failure of the trimpot to retain its setting.

The new state-of-the-art potentiometers (trimpots) installed in the KE high speed scanner by production test have met all requirements. A number of new potentiometers are being procured on a required design basis for additional testing in more critical circuits. Monitoring of potentiometer performance is continuing.

Demineralized Water Storage Tank Membrane Removal

High purity demineralized and degasified water for N Reactor is stored in a one million gallon demineralized water storage tank (DWST). The interior of the carbon steel tank is painted with a three-coat phenolic resin application to prevent iron pickup by the water. The tank has also contained a rubberized fabric bag (usually called the membrane) fastened to the top of the tank wall to prevent oxygen from the air dissolving in the water.

In recent years, cemented seams in the membrane have become weakened by exposure to the warm water. Several repairs have been made but the leakage has persisted such that continual pumping has been required to keep the membrane from sinking.

Engineering studies have determined that the best alternate method of keeping oxygen out of the stored water is to apply a nitrogen blanket to the tank. Work has begun on installation of the equipment, but procurement delays have postponed beneficial use.

During the recent summer outage, the leakage rate of the membrane exceeded the pumping rate and the membrane sank. It was decided to remove the membrane, doing without an oxygen barrier until the nitrogen blanket can be put in service

Rebuilding of Graphite Cooling System Heat Exchangers

The graphite moderator of N Reactor requires additional cooling to control graphite temperatures. A low pressure, low temperature recirculating cooling system was provided for this function. The four U-tube heat exchangers originally installed in this system have a history of tube failures. A retubing program was initiated and is currently underway.

The first unit (Graphite Cooling Heat Exchanger No. 2) has been removed, disassembled, redesigned, rebuilt, reinstalled, tested, and returned to service.

Disassembly confirmed that the tube leaks were due to a wearing through of the tube walls at the tube supports (baffle plates). It was determined that the rigidity of the tube support system was inadequate. The support system was redesigned and the unit was rebuilt with new tubing of Inconel 600.

When the unit was reinstalled and tested, no tubing vibration could be detected at any flow. The unit has been returned to service and a second unit is in the process of being rebuilt.

105-109-N Buildings Fog Spray System

The fog spray system supplies the gamma monitor system and Zone V air conditioning cooling during normal operation and on emergency demand supplies graphite cooling backup, 109 fog spray, 105 fog spray, partial 105 fire protection, and backup to the control rod and thermal shield cooling systems. Normal demand is supplied by the electric-driven "high pressure raw water" pumps while the emergency demands are diesel-driven "fog spray" pumps.

During routine testing of the diesel-driven fog spray pumps, in the latter part of the extended summer outage, high speed cycling of one of the diesel drives occurred. Concurrently, a cast iron reducing flange at a strainer in the 105 fire protection system failed. A review of the system revealed that on one occasion in the past a cast iron reducing flange in a similar position had also failed and that there were six additional cast iron reducing flanges still in the system. These flanges were replaced with higher rated steel flanges prior to startup.

A series of surge tests was run to evaluate the effects of various starting modes of the diesel drives on the system. The tests simulated loss of BPA power, and operation of the system without the fog spray accumulator. Loss of power results in a shock wave from rapid closure of the high pressure raw water pump check valves, and a measurable pressure surge results from the full speed startup of the fog spray diesel drives. The fog spray accumulator was shown to act as a surge tank on the system during the diesel starts.

Although a number of components are underrated in the fog spray system and the full intent of the piping code is not met, examination of the system design, materials, and operating conditions provides assurance that an adequate margin of safety exists to permit continued operation. Code Deviation Record No. 27 was prepared and approved on this basis to authorize continued operation of the system.

Steam Collection Drums

At N Reactor, there are two steam collection drums attached to each steam generator. There are two steam generators per cell and a total of six steam generation cells. Concurrent with the steam generator retubing program, the steam collection drums were found to have a number of internal surface defects at the 42-inch steam header connection where blanks had been welded for the original construction hydrostatic leak tests. It was found that all steam collection drums in the original cells (1-5) had such defects and required repair. Cells 1 through 4 were previously repaired during the steam generator tubing task. The defects in Cell 5 were repaired, together with the additional defects found in the Cell 3 units during the summer outage. The remainder of the cells were found to be in satisfactory condition.

Terminal Marking of Contaminated Burial Sites

Design criteria have been prepared to provide terminal markers for underground contaminated waste burial sites no longer considered active. The burial sites under consideration are located in 100-D, 100-F, and 100-H Areas.

N Reactivity

Initial critical verification and excess in rods as equilibrium was approached both indicated agreement with the anticipated excess reactivity of 4-5 mk at equilibrium. Projections with the TRUE-K code based on tentative charge-discharge plans into June 1971 indicate ample reactivity barring unforeseen ball drop losses in the interim.

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REGULATORY PROGRAM

August Regulatory Meeting

The Hanford Subcommittee of the ACRS met at Hanford on August 20 and 21 to review reactor safety experience, the N Reactor Regulatory Program, and Hanford waste management practices. Regulatory Program presentations to these regulatory agency representatives included: a review of the plant status following installation of the Effluent Control Project; a status report and tour of Effluent Control Project installations; a listing of the resolved issues, outstanding questions, and program content to resolve these outstanding questions; and a status report of selected technical investigations. The purpose of this presentation was to give the ACRS a basis for evaluating program progress to show comparability between AEC-Division of Production and licensed reactors.

Informal supplementary discussions were held between AEC-DRL and DUN personnel on August 19. These discussions provided valuable guidance on Regulatory Program scope and content. In addition, it was evident that a misunderstanding exists on the required depth and breadth of Regulatory Program studies.

GCP-411 - Effluent Control Project

The results of the acceptance tests for the new emergency coolant supply line and emergency dump tank were reviewed and these systems were accepted for reactor use.

The gravity drains and disposal basin portion of the project was not funded by the Atomic Energy Commission for budgetary reasons. Consequently, the emergency dump tank outflow is currently routed to the existing N Reactor crib and drainage canal through the existing 36-inch low pressure flush [FL(LP)] drain line. This permits beneficial use of the dump tank and provides control of noble gases which would otherwise be released directly to the environment through the emergency coolant system liquid effluent path. Tests have shown that the existing 36-inch FL(LP) line and the crib will suitably accommodate flows of approximately 25,000 gpm with the hydraulic head available from this installation. Since this outflow capability is less than full discharge capacity of the emergency coolant system, procedural actions have been established to obtain beneficial use in the interim period until the disposal basin and gravity drain projects can be completed.

IEEE-279 Evaluation

The designs of N Reactor protection systems are being evaluated for compliance with the IEEE-279 criteria prepared for nuclear power plant protection systems. These evaluations show that the designs of the N Reactor protection systems generally comply with the provisions of the IEEE-279 criteria. Apparent noncompliance with the criteria is generally limited to three principal design features. These are:

1. System designs where failure of a single component can cause failure of system action. (This does not apply to automatic shutdown protective systems.)

2. Lack of physical separation of system channels such that system function could be lost due to common environmental consequences.

3. System bypass features which do not automatically remove the bypass function whenever permissive conditions are not met.

With these evaluations of individual system compliance with criteria, it is now possible to evaluate the impact of system noncompliance.

The N Reactor protection system design emphasizes redundancy of systems rather than redundancy and independence of components within each system. Consequently, loss of system function does not necessarily mean loss of protection function. In those instances where this relationship cannot be established, plant modifications will be necessary to provide full compliance with the intent of the IEEE-279 criteria.

NUCLEAR SAFETY ASSURANCE PROGRAM

A-B Electrical Bus Hydraulic Interdependence

Nuclear safety requires that the two electrical power sources at N Reactor be independent. However, a dependent condition can exist under certain conditions because the turbine generator (TG) condenser is cooled by river water from both A Bus powered and B Bus powered pumps. Thus, a loss of A Bus could, under these certain conditions, cause inadequate cooling of the TG condenser and subsequent loss of B Bus. The test series of August 1969 defined conditions under which the dependency would exist, and appropriate Process Standards limits were established to permit safe operation. However, the phenomena that cause the sudden TG condenser flow reductions that amplify the dependency were not precisely defined by the test series. Further testing to define these phenomena were scheduled for the long summer outage but were cancelled because of conflict with other priority outage work such as tie-in of the new emergency cooling system, turbo-generator Class A overhaul, and inspection of the safety control leg for debris. All plans, procedures, and equipment are available for implementation of the test program as outage time is available.

N Reactor Raw Water Systems Debris

Debris Development Test

Development Test 218,""Investigation of Large Particulate Matter in N Reactor Circulating Raw Water System," was completed and a final report was issued. This test was designed to collect and characterize debris which passes through the traveling screens into the circulating raw water system. The main conclusions were that the concentrations of debris entering the system during the various test periods were lower than the maximum permissible concentrations contained in the proposed raw water cleanliness criteria, but that numerous individual particles were observed which were in excess of the maximum volume and surface area specified in the criteria.

Specific actions required to correct the deficiencies of the system are now being developed.

<u>Graphite Cooling System</u>

Final design reviews are in progress concerning the elimination of debris accumulation in the N Reactor graphite cooling system backup supply by installing a flush line and screen.

<u>N Reactor Confinement Leak Rate Test</u>

A confinement leak rate test was completed at N Reactor. The leak rate was about 2.5 percent as compared to 1.4 percent the previous year and 5 percent as a limit.

A test on a selected steam vent inflatable backup closure showed it would not inflate. The butterfly valve was then closed and was found to leak since it was improperly seated. Subsequently, all steam vent backup closures and butterfly valves were inspected and all were reset and cycled at least three times to ensure proper closing. In addition, the release pressure on the inflatable backup closures was reset to ensure proper release.

<u>N Reactor High Level Protection Studies</u>

Studies are in progress to evaluate the transient behavior of N Reactor for various reactor events to more precisely determine the margins of safety provided by the flux and termperature monitoring systems. These studies are expected to result in modification of the system requirements. The studies are being done by application of computer coded reactor transient models.

<u>Other N Reactor Control Studies</u>

Current nuclear safety requirements specify that the entire safety rod system shall be recalibrated periodically to assure that loading or other changes have not affected safety rod system strength in an unpredicted manner. The safety rod calibration tests require a large amount of outage time. A study is in progress to evaluate calculation techniques which may be utilized to reduce the testing requirements.

Checking of the DUNCAP-3D code against an N Reactor model has been largely completed; calculational results have shown good agreement with the D-pattern calibration. Runs are currently being made to compare DUNCAP results with various rod configurations evaluated during the original startup tests.

WASTE MANAGEMENT PROGRAM

Contaminated Coolant Treatment Pilot Plant

Work continues on the pilot plant facility which is being installed in N Area to develop data for the design of a Contaminated Coolant Treatment Facility (CCTF) for N Reactor. The building is complete and the process equipment is installed. The data acquisition equipment is about 85 percent complete. Target date for startup of the pilot plant is October 1, 1970. The process visualized consists of filtering the N Reactor bleed streams to remove radioactive solids with final cleanup by ion exchange. Various types of ion exchange resins will be investigated. Both cartridge and Vacco multisegmented filters which have reduced waste handling requirements will be tested. A sampling system activated by a timing device will allow measurements for total activity, individual radioisotopic identification and quantity, pH, and conductivity for the inlet and outlet streams of each filter and ion exchange column.

Presentation to ACRS Subcommittee

The DUN Waste Management practices were described to the Hanford Subcommittee of the Advisory Committee on Reactor Safeguards during their recent visit in August. The presentations included description of the source, identity, quantity, and relationship to control limits of the principal radionuclides generated by operation of the N and K reactor and fuel fabrication facilities.

SPECIAL IRRADIATIONS PROGRAM

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Isotope Production

One hundred fifty bismuth pieces were charged into five KE Reactor process tubes to produce Po-210 for Mound Research Corporation.

One iodine-containing capsule of the revised stainless steel design was charged into a side test channel to produce Xe-128 for Argonne National Laboratory.

Routine Irradiations

Forty-three quickie activation analysis capsules were irradiated in the KE Quickie Facilities for BNW and WADCO. A downstream snout capsule containing high specific activity tantalum was discharged from the KE Snout Facility for DUN. Three BNW tensile specimen capsules were discharged from the cooled magazine facility at KE.

TECHNICAL ACTIVITIES - FUEL FABRICATION

New Design Mark IV Outer Billets

A study has been completed to determine the incentives and feasibility of converting N fuel production to a longer billet. The study resulted in a new design outer billet having a 21.9-inch length, yielding 12 full-length fuels per extrusion. Conversion plans include:

- A ten billet demonstration of the new billet length during the second quarter of Fiscal Year 1971.
- Ten percent of production converted to the new length by the third quarter of Fiscal Year 1971.

The design includes also a secondary Class III (20.3" long) billet which would yield 11 full-length pieces.

The current length of the Mark IV outer preshaped billet is 19.75 inches, and its extrusion results in ten 26-inch length and one 17-inch length fuels.

New billet designs considered in the study were based on assuming the same amount of defective end material as is now being generated on current extrusions and adding the necessary billet material to effect additional fuel pieces. Three billet lengths were evaluated, all of which were within the feedsite casting capability and the extrusion press capacity. Selection of the 21.9-inch length billet was based both on the resulting savings and the production of full-length fuels in each extrusion. Further, the availability of a Class III billet yielding only full-length fuels is advantageous.

The cost savings from converting N fuels production to a longer billet derive from increased uranium utilization and reduced number of components due to the reduced number of extrusions. These costs have been calculated based on scheduled productions for Fiscal Years 1972-75 and are shown below:

Three Billets @ 21.9 Inches

Fiscal Year	1972	1973	1974	1975
Savings Due to Uranium Utilization	$41,432	$43,842	$45,022	$46,868
Savings Due to Components	23,239	24,449	24,874	26,314
	$64,671	$68,291	$69,896	$73,182

Increasing billet length increases uranium design utilization and thus reduces the amount of scrap returned to NLO for reprocessing. A cost figure of $4.185 per pound of uranium for reclaiming extruded uranium was applied to the difference in scrap generation resulting from the longer billets.

The cost of billet components for longer billets was based upon the reduction in the number of extrusions resulting from the increased billet size as well as a compensating slight increase in the cost of the longer cladding components. The costs do not reflect the approximate ten additional days of extrusion press time per year made available by the reduction in the number of extrusions.

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[redacted]

The design of the billet assembly process allows for various billet lengths, and no problems are expected with the 21.9-inch billet. Billet preheat cans have already been modified to accommodate billets up to 24 inches in length. The 2750 ton extrusion press can be loaded with this length billet, and billets of this size have been successfully extruded well within the press capabilities. An assembled 21.9-inch billet is calculated to be approximately 22.7 inches in length. Approximately 3.9 inches of the press stem will extend into the liner prior to billet upset. The extruded tube using the butt shear technique should be about 30 feet 6 inches long, which is well within the 42 foot runout table length.

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ADMINISTRATION - GENERAL

PROPERTY ACCOUNTING

A physical inventory of catalogued movable equipment assigned to DUN is scheduled to begin on October 5, 1970 in the 100 Areas and on November 9, 1970 in the 300 Area. DUN has approximately 4,800 units with an original acquisition value of $5,705,000. Copies of the inventory procedures and schedule were provided to AEC-RL.

APPROVAL LETTERS

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

Number	Subject	Date of Transmittal to AEC-RL
ATD-78 Add. #4	Personnel Changes	September 23, 1970
ATD-156 Add. #1 Sup. #1	DUN Education and Training Program	September 23, 1970
ATD-196	Feasibility Systems Study	September 10, 1970
	Demineralized Water Storage Tank, 182-N	September 16, 1970

DECLASSIFICATION REVIEW OF COLUMBIA RIVER DOCUMENTS

The AEC-RL Classification Officer has been supplied a list of all classified documents relating to Columbia River studies generated by DUN employees and Washington State University working under contract with DUN. The list will be used to determine the feasibility of conducting a declassification review sponsored by AEC-HQ.

EMPLOYMENT SUMMARY

DUN personnel totals and employee allocation as of August 31 and September 30 are shown in Appendix B.

AFFIRMATIVE ACTION PLAN

Three minority employees were added to the work force during the month. This was offset by the return of two Co-op Trainees to Washington State University and one resignation. In the clerical and related job category, good results were obtained; 100 percent of the new-hire additions in the quarter ending September 30 were minorities. Conversely, no minority was represented in ten additions to the professional staff during the same three-month period.

SAFETY

No personnel radiation exposures exceeded operational control.

Month-end safety statistics were:

Disabling injuries - September	0
- CY to date	3
Days since last disabling injury	84
Man-hours since last disabling injury	565,519

APPENDIX A

PROJECT STATUS SUMMARY - REACTOR FACILITIES

AUTHORIZED PROJECTS

Number & Title	Authorized Funds - $	Percent Complete Design	Percent Complete Construction	Remarks
Single-Pass Reactors				
DAP-526, Deactivation of Hanford Production Reactor C	105,000	100	67	Outside line subcontract work for the fire protection system has been completed.
N Reactor				
GCP-506, Improved Safety Platforms and Accesses - 100-N Area	300,000	100	99	Miscellaneous cleanup work in progress.
GCP-411, Effluent Control Program - 100-N Area	1,830,000	100	96	Section V - In service. ATP results being reviewed for compliance with Design Criteria.
DCE-519, Replacement of Bridge Crane and Hoist System with New Crane System - 105-N Storage Basin Area	465,000	100	75	The two east cranes have been placed on the runway beams and the communication and power rails are being installed. The west cranes are being de-bugged at the vendor's plant.
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.

Number & Title	Authorized Funds - $	Percent Complete Design	Percent Complete Construction	Remarks
DCE-539, Upgrade Flux Monitor System - N Reactor	335,000	10	0	Detail design continues on schedule.
DAE-540, Smoke Density Monitors for DUN-Operated Facilities - 100 Areas	55,000	20	0	Design has continued by HES-Vitro. The Title I drawing depicting equipment layout has been issued for comment.

PROJECT PROPOSALS AWAITING AUTHORIZATION

Number & Title	Funds Requested	Date to AEC-RL
GCP-411, Rev. 2, Effluent Control Program - 100-N	$ 2,010,000	9/23/69
DCP-525, Fire Protection Improvements - KE	225,000	5/2/69
DCP-527, Graphite Cooling & Fog Spray Improvements - 100-N[1]	97,000	5/9/69
DCP-528, Rev. 1, Fire Protection System Improvements - 100-N	394,000	3/31/70
DCP-529, Gravity Drainage System and Disposal Basin for 100-N Area[1]	200,000	7/9/69
DAP-530, Upgraded Electrical Services & Lighting 1100-N & 1101-N Buildings	78,000	8/12/69
DCP-538, Heat and Ventilation System Improvements 105-N & 109-N Buildings[1]	330,000	5/18/70
DCP-542, High Pressure Injection System Improvements - N Plant	138,000	6/17/70

[1]To be returned to DUN by AEC-RL because of long holding periods.

Number & Title	Funds Requested	Date to AEC-RL
DCE-544, Critical Instrument Power Supplies - N Reactor	$ 65,000	6/15/70

PROJECT PROPOSAL PREPARATION

Number & Title	Design Criteria	Project Proposal
Export Water System Backup - 182-D (for 200 Areas)	Completed	Completed. Held by sponsor (ARHCO).
Stack Monitoring Improvements - 100-N Plant	Completed	Prepared. Held for later submission.
Permanent Markers for Buried Wastes in Terminated 100-D, 100-F and 100-H Sites	Prepared	In preparation.

APPENDIX B

EMPLOYMENT SUMMARY
(with employee allocations)

	8/31/70	9/30/70
CONTRACT PERSONNEL		
02 Programs		
Douglas United Nuclear	1273	1267
Assisting Other Contractors	17	17
Total - 02	1290	1284
Other Programs Under AEC Contract		
Assisting Other Contractors and WPPSS	37	37
Special Irradiations	9	6
Other Programs - Standards	4	2
Total - Other Programs	50	45
Total Contract Personnel	1340	1329
COMMERCIAL ACTIVITIES PERSONNEL	22	21
TOTAL FORCE	1362	1350