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**Pacific Northwest Laboratory
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**DOUGLAS UNITED NUCLEAR
MONTHLY REPORT**

OCTOBER 1971

DOUGLAS UNITED NUCLEAR, INC.

RICHLAND, WASHINGTON

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DOUGLAS UNITED NUCLEAR

MONTHLY REPORT

OCTOBER 1971

DOUGLAS UNITED NUCLEAR, INC.
Richland, Washington

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By Authority of CG-NMP-1
RM Oten, 9-28-94.
By Jerry Moley, 9-30-94.
Verified By J E Savely 10-4-94

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SUMMARY

N Reactor operated during 22 days of the month with a time operated efficiency of 69.5 percent and electrical production of 427.54×10^6 kwh. New daily and weekly generation records were established.

A small seismic shock was felt at N Reactor on October 25. No effects on plant operation or facilities were observed.

C. D. Harrington
President

PLANT OPERATIONN REACTORProduction and Operating Experience

The reactor operated on 22 days of the month with a TOE of 69.5 percent and electrical production totaling 427.54×10^6 kwh - 10.7 percent of the 4 billion kwh goal. FY 1972 production through October totals $1,305.33 \times 10^6$ - 32.8 percent of the goal. Total actual and credited electrical generation for the fiscal year is $1,796.686 \times 10^6$ kwh which is 45 percent of the goal. A one week generation record of 144,260,000 kwh was set October 31 and a single day generation record of 20,620,000 kwh was achieved on October 28.

The reactor power level was limited to that required to meet WPPSS steam requirements. The maximum level reached was 3765 MWt with WPPSS generating 855 MWe for a thermal efficiency of 22.7 percent.

The October 8 scheduled shutdown terminated a 30 day and two hour period of uninterrupted dual plant operation which surpassed the previous record of 25-plus days achieved in 1967. There were 227 outage hours: 222.7 hours were scheduled and 4.3 were unscheduled as detailed below. The outage plan was executed and steam availability to WPPSS was established only 11 hours after the originally scheduled time. The eleven hour delay included the two scrams occurring during startup.

<u>Shutdown Date/Time</u>	<u>Startup Date/Time</u>	<u>Duration Hours</u>	<u>Outage Type</u>	<u>Cause</u>
10-8/2101	10-18/0345	222.7	Scheduled	Refueling, Maintenance
10-18/0700	10-18/0800	1.0	Scram	Flow Monitor - Valve Leak
10-18/0850	10-18/1207	3.3	Scram	Flow Monitor - Valve Leak

Statistical Summary

Input Production - Pu (KMWD)	78.4
Pu Production - % of Forecast	114.7
Electrical Generation (KMWH - WPPSS)	427.54
Total Actual and Credited Electrical Generation FY 1972 to date	1796.68
Power Level (MWt) - Maximum	3,765
Time Operated Efficiency - %	69.5
*Reactor Availability (RA)	69.5
*Plant Factor (PF)	71.7
*Electrical Response Factor (ERF)	122.8
Fuel Failures	0
Helium Consumption (1000 cu.ft.)	349.8
Fuel Oil Consumption (bbl.)	8,903

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*Definitions

RA is that fraction of the month during which the reactor could make steam for electrical power generation whether or not it is actually in service.

PF is the ratio of actual power produced to the total design rated power capability.

ERF is the ratio of power produced to that scheduled.

Tritium Shipment Equipment

The problems with the Quality Assurance records for isotope cask No. 1 were finally resolved by the fabricator, J. A. Jones. These records have now been reviewed and accepted by N Plant Facility Design leaving only the Super Tiger to trailer tie-down quality records yet to be approved.

The vendor's calculation pertaining to these tie-down devices are currently being reviewed for compliance to mandatory regulations. It appears that a favorable determination on these records will be made some time early in November.

Operator Recertification

Recertification testing for N Control Operators is in progress and the written tests have been completed during the month except for those not available due to personal absences. Included was a requalification test for one Control Operator returning after extended absence. Oral and demonstration tests are currently underway. An extension of the November 1 completion date has been received from AEC-RL.

FUEL FABRICATION

Production and Operating Experience

The extrusion press, which was down three days near month end due to hydraulic system difficulties, was operating normally at month end. Processing problems at the braze stations lowered throughput for final assembly.

Statistical Summary

Input Production

Billets Extruded - tons	35.5
Percent of Forecast	91.0

Output Production

Finished Assemblies	972
Tons Output	20.8
Percent of Forecast	63.0

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Uranium Utilization - %

A12	80.0
NO9	82.5

Month End Inventories

Bare Uranium Billets	116
Finished Fuel	220

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PROCESS ASSISTANCE

Performance Characterization of Inconel-600 Tubing
From N Reactor Steam Generator 6B

A monitoring program was initiated early in the operation of N Reactor to evaluate the behavior of steam generator heat exchanger tubing subjected to the severe environs of high temperature primary liquid coolant and secondary system steam. Severe leakage in the originally installed steam generators resulted in retubing all units with Inconel-600 except 6A and 6B which were originally installed with Inconel-600 heat exchanger tubing, and cell 5 generators which have the original 304 stainless steel tubing. Replacement of the stainless steel tubing was necessitated by gross stress corrosion cracking and leakage. Table I lists the service dates for the heat exchangers as currently tubed.

TABLE I

INCONEL-600 RETUBED HEAT EXCHANGER SERVICE DATES

<u>Steam Generator</u>	<u>Inconel-600 Service Date</u>
4A	6/64
6A, B	6/66
3A, B	5/68
4B	11/68
2A, B	9/69
1A, B	Not yet in service.
5A, B	304 stainless steel tubing since N Reactor startup.

Since commercial power reactors have experienced some failures of Inconel-600 heat exchanger tubing, the surveillance program is being continued. The program consists of the periodic removal of tubing samples from the generators for destructive examination. Samples were removed for study from steam generator 4A after 2 years' service, 6A after 3.3 years' service, and once again from 4A after 5 years' service.^{1,2} The examinations were performed using standard metallographic techniques in conjunction with ultrasonic and eddy current testing. The results of these analyses gave no evidence of stress corrosion cracking or other deleterious effects, but the analytical techniques

1. DUN-SA-92, "Performance of Inconel-600 Tubing in N Reactor Steam Generators," W. K. Kratzer, May 14, 1969.
2. DUN-SA-149, "Examination of Inconel-600 Tubing from N Reactor Steam Generator 6A," W. K. Kratzer, November 18, 1971.

were limited. The most recent tubing sample was removed from steam generator 6B in April 1971 after four years of service. This sample was examined using scanning electron microscopy and X-ray microprobe techniques to provide much higher magnification and resolution of surface features than possible with the metallographic technique. This examination was concentrated along a section of tubing corresponding to the primary coolant inlet. Of particular interest were the tubing support areas on the outer surface where crevices result in areas of impaired heat transfer. In addition, detailed examination of the inner surface and the characteristics and compositions of corrosion films on both surfaces were made.

The results of the examination can be summarized as follows:

- The outer surface, secondary steam side, of the tubing displayed a thin uniform crud layer approximately 5 microns thick. At the edge of the No. 1 tube support area, the layer increased in thickness to about 50 microns and decreased to 10 microns in the center of the support area. The crud consists mainly of iron oxides as expected. Appreciable concentrations of manganese, zinc, and copper were also detected with the copper concentrated at the crud-tubing interface. The exterior crud does not appear to be tightly attached.
- The outer surface displayed one area of pitting near the center of the No. 1 tube support area. The pits averaged 5 microns in depth. The pitting is not of a nature that would be expected to cause premature tubing failure. The site was not considered to be active at the time of tube removal. The possibility exists that the pits were present in the fabricated tube or resulted from the leaching or selective corrosion of localized surface impurities early in the tube service history.
- The inner primary coolant surface displayed a thin (~ 1 micron) uniform crud layer over the entire tube section examined. Based on normalized X-ray microprobe data, the film appeared to again consist mainly of iron oxide. Localized areas with high silicon content were noted for which no explanation exists, and to which no particular significance is attached. Aluminum in small quantities was also present in the crud layer.
- The inner surface displayed prominent intergranular etch pits. The pitting was most severe adjacent to the tube sheet (primary coolant inlet) but the depth was estimated at only 10 microns. Both the severity and frequency decrease with decreasing primary coolant temperature. The pitting process could not be identified as being active or passive, or intermittent, at time of removal. The pits were at least partially filled with crud.

There was no evidence of any deleterious process capable of endangering heat exchanger tube life. The pitting on the inner surface has not progressed significantly beyond that observed during the examination of tubing from cell 6A. A rapid, unexpected acceleration of the pitting could become significant to heat exchanger tube life, and therefore the monitoring of the pitting on the inner surface to determine if the process has stagnated or is progressing slowly is planned.

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Decontamination Program

Maximum reductions in rear pipe space radiation levels can be achieved by decontaminating both the external and internal piping surfaces. To date, only an internal process has been used at N; therefore, a program was established to develop the procedures for spray application of a decontaminant to the external rear pipe space surfaces during the summer 1972 outage. A major task of this program is to establish compatability of the decontaminant with rear pipe space materials.

Turco 4521, an oxalic acid-citric acid based decontaminant, has been selected for application. The selection was based on avoiding the localized galvanic and pitting attack characteristics of phosphoric acid based solutions. An organic solution was also desired to facilitate the post-rinse thermal degradation of residual decontaminant. Turco 4521 emerged as the choice from preliminary testing of several candidates.

Beaker corrosion tests were conducted at 180 F on materials that would be contacted by the acid reagent during a spray decontamination of the rear pipe space between the V-12 valves and the rear risers. Table II defines these materials and material couples and summarizes test results.* The maximum corrosion rate observed was 3.4 mils/day (about 0.1 mils/decontamination) on A 193-B7 low alloy steel, the Grayloc stud bolt material. No accelerated corrosion attack was noted on coupled materials during tests conducted for up to 20 hours. Crevice samples likewise showed no deleterious effects. All test results were judged to be satisfactory.

Some residual oxalate deposition was observed in the crevices. The potential effects of these deposits are the subject of a current evaluation program. RTD cables repaired with exposed silver soldered copper connections are also being tested.

* DUN-7787, "Corrosion of N Reactor Rear Pipe Space Materials in Turco 4521 Decontaminant," W. K. Kratzer and C. A. Richardson, October 11, 1971.

TABLE II

TURCO 4521 BEAKER CORROSION TEST RESULTS AT 180 F

Material	Coupon Number	Condition	Corrosion Rate, mils/day			Material Use
			1 Hour	Exposure 3 Hours	20 Hours	
A 193-B7	R1		2.17	1.70	2.86	Grayloc bolts
Grayloc Bolt	R2		2.30	1.96	3.35	
A 194 Gr 11	Q1		1.96	0.94	1.78	Grayloc nuts
Grayloc Nut	Q2		1.18	1.30	1.73	
A 105-61T	003		0.52	0.39	0.47	Grayloc hubs welded to connectors
Grayloc Hub	004		0.53	0.39	0.48	
A 106-B	ND8		0.77	0.41	0.54	Connectors
Connector Pipe	ND9		0.97	0.45	0.58	
AISI 4140 (Sheet stock)	H1		2.06	0.92	1.31	Grayloc clamps, seal rings
	H2		2.24	1.02	1.30	
A 105 Gr 11 (sheet stock)	01		0.87	0.36	0.42	V-12 valve body
	02		0.57	0.31	0.38	
	03	coupled, crevice, to IN 3	0.58	0.38	0.33	
	04	coupled, crevice, to IN 4	0.81	0.43	0.33	
Inconel 600 (sheet stock)	IN1		0	0	0.07	Rupture monitor lines
	IN2		0	0	0.07	
	IN3	coupled, crevice, to 03	0	0.02	0.06	
	IN4	coupled, crevice, to 04	0	0.04	0.06	
A 515-Gr 7 (A 212-B) (sheet stock)	G1		0.70	0.38	0.48	Rear riser primary pipe
	G2		0.66	0.38	0.51	
	G3	coupled, crevice, to AC3	0.74	0.40	0.39	
	G4	coupled, crevice, to AC4	0.86	0.37	0.37	
304 stainless steel (sheet stock)	AC1		0.15	0.17	0.27	
	AC2		0.10	0.16	0.29	
	AC3	coupled, crevice, to G3	0.05	0.12	0.17	
	AC4	coupled, crevice, to G4	0.08	0.12	0.19	

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Disposition of High Exposure Mark-IV AA Fuel

The disposition of five Mark-IV AA columns charged under PT-N-94 is summarized in the following table. As directed by the AEC-RL, these five columns were discharged before reaching their authorized exposure during the scheduled October outage. No effort was made to segregate these fuel pieces from the normal goal discharged fuel. Therefore, no post-irradiation examination is possible.

TABLE III

DISPOSITION OF HIGH EXPOSURE MARK-IV AA FUEL

<u>Tube Number</u>	<u>Approximate Exposure At Discharge - MWD/T</u>	<u>Authorized Exposure MWD/T (by PT)</u>
1443	2725	6000
1843	2748	2700
2644	2130	2500
2670	2200	2500
2948	2300	2500

Discharge of these five columns completes removal of all fuel from N Reactor irradiated to exposures higher than that corresponding to the 9 percent Pu-240 production operating mode.

Braze Development - Coextruded Fuels

Historically, one of the highest reject rates in coextruded fuel manufacturing is associated with end closure brazing as measured by nondestructive tests for the cap to core and cap to clad bonds. Part of the quality of the braze and resulting bond is associated with preparation of the parts for brazing, induction coil alignment and crimping techniques. Currently, the detection of inadequate cleaning of parts, improper coil alignment and crimping techniques, and deficiencies in braze station equipment operation may go undetected for some time due to the time lag between brazing and nondestructive testing.

The "as brazed" surface is rough, affects crystal life adversely, and does not provide an adequate signal for a positive test. To prepare the brazes for testing, the fuels are beta heat-treated, vacublasted and end machined to provide smooth ends for testing. Thus, the necessary processing steps after brazing can result in considerable amounts of fuel backlog between brazing and nondestructive testing.

A potential method to speed up quality information feedback from the testers to the braze operation is to braze a smooth, thin Zircaloy disc onto the end of the fuel, thus providing a smooth surface which would permit immediate testing after brazing. The disc would be placed on the braze ring before the braze cycle is started. During induction heating the disc heats up faster than the braze ring and appears to promote better braze ring melting and flow due to the added heat.

[REDACTED]

into the top of the braze. After melting and argon pressurization the disc is forced into the end closure cavity and provides a smooth end amenable to testing. It may also tend to force more metal around the base of the braze.

This method of brazing has been tried on five end closures and appears promising. A much larger number of fuels (reject pieces) will be used to further test the revised brazing method to develop additional information and data on improvement to the reject rate.

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EQUIPMENT AND SYSTEMS PERFORMANCE

Primary Loop Leakage

Primary loop leakage increased during the month as indicated by the high pressure injection flow rate. At month end, the injection flow was 1400 gpm as compared to 1260 gpm at the end of the previous month. Several significant leaks were repaired during the scheduled outage, but at least two have appeared since. One is from the V-7 valve, the diversion system flush valve, which is believed to be unseated and the cause of a steam-water hammer in the low pressure flush header [FL (PL)] in the outlet pipe space. The other leakage is believed to be from the RV-2-1, primary loop pressure relief valve, which may be causing a slight steam-water hammer in the high pressure flush header [FL (HP)] in the outlet pipe space, and the V4-8 primary loop dump valve. Neither leak is judged to be severe enough to require a plant shutdown, and repairs will be attempted during the next outage.

Inspection of ECS Flow Monitor Sensing Lines

The September report noted that the Emergency Cooling System flow monitor sensing lines would be inspected during the October outage for stress corrosion cracks or other leaks that would affect their performance. The visual inspection was accomplished at outage flow conditions and no leaks or indications of stress corrosion cracking were found. A follow-up inspection is planned on these lines during the December outage.

V-11 and V-12 Repair Program

Early in the scheduled October outage, a leak inspection was made of the V-11 and V-12 valve pipe spaces to determine the status of valve leakage and to establish the need and priority for valve repairs to be made during the outage. The leak inspection revealed four V-12 valve leaks and one ZTM V-12 valve leak. These five valves were scheduled for repair during the three days allotted for valve repair, and all five leaks were repaired during the outage time scheduled. A five percent improvement in work efficiency was achieved over the 1971 summer outage performance; however, the exposure per valve increased due to the higher radiation field in which the work was done.

The leak inspection of the V-11 valves revealed no leaks in the approximately 800 valves that have been repacked with Grafoil. Minor leaks were detected in the non-Grafoil packed valves. These findings provide good support for the continuing use of Grafoil in this repair program. The V-11 leaks that are currently being detected are occurring in non-Grafoil packed valves. The rate of new leak occurrence and the severity of these leaks indicate that repacking the remaining V-11 valves with Grafoil can wait until the summer 1972 outage.

The leak inspection of V-12 valves also supports the repair program using Grafoil. In the October outage inspection, no leaks were detected from valves repaired with Grafoil during the 1971 summer outage. The nine hub leaks which were detected were new, non-Grafoil leaks. If this trend continues, it would appear that packing leaks will now be the major area of concern.

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Rupture Monitor Heat Exchangers

The rupture monitor system continuously samples a flow sample from each process tube for the purpose of detecting fuel element ruptures. Cooling of the sample is required and for this purpose individual tube-in-tube heat exchangers are used with Columbia River raw water circulated on the shell side for single pass, counter flow cooling. For some time the system has experienced plugging problems due to both chemical buildup and silt collection. Despite considerable maintenance effort, the heat dissipation capability of the system continues to decrease. Proper heat dissipation is necessary to maintain standardized measuring conditions in both the sample and the system components. Therefore, a program has been developed directed at recirculating a chemical cleaning solution through the partially open exchangers and the replacement of all plugged exchangers.

The original system was fabricated in closely packed banks of exchangers having a shell of 1/2-inch OD Monel tubing and a sample tube of 1/4-inch OD Inconel. Planned replacement exchangers will be built slightly larger in dimension to provide increased heat transfer characteristics and improved secondary flow to retard plugging.

To develop a cleaning agent, samples of the deposit from the shell side of the heat exchangers were tested in a variety of cleaning solutions which resulted in selection of a two percent solution of Nalclean 68 capable of dissolving over 90 percent of the plugging material. Tests on an actual exchanger revealed that the two percent solution would clean the exchanger in approximately four hours using simulated reactor parameters. Corrosion tests on Inconel, Monel, and carbon steel - the three main materials in the system - resulted in corrosion rates of less than one mil/hr in all cases.

Development of a method for in-place cleaning of the plugged rupture monitor heat exchangers has involved a significant effort. Methods investigated included ultrasonic excitation, mechanical vibration, backflushing, and high temperature baking. None of the tested methods accomplished the desired amount of deposit removal.

During the December outage, it is planned to complete installation and test equipment for in-place cleaning of partially plugged exchangers and, additionally, initiate the replacement of plugged exchangers.

Cell 1 Prefilming

PCS V203-1, isolation valve between Cell 1 and the primary loop, could not be unseated during the October outage and prevented returning the cell to service. Following startup from the outage, the valve was unseated hot and Cell 1 will be returned to service at the next convenient outage so that another cell can be removed from service for EMS and other work.

Primary Coolant Pump Mechanical Seals

The 100 N primary coolant flow pumps were originally equipped with mechanical seal rotating faces fabricated of Type 304 stainless steel. The stainless steel face was supplied with a Stellite overlay wearing surface to improve seal life.

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Rapid wear of the stainless steel Stellite faces was experienced during the first few months of reactor operation. Seal failures were frequent during this period. Upon inquiry, the pump manufacturer recommended a replacement rotating seal face of 304 stainless steel with an aluminum titanium overlay wearing surface. This type seal was installed in all operating pumps by July of 1964; however, operating experience was not encouraging.

The pump manufacturer was again contacted to discuss potential methods for improving seal life, and in July 1965, the manufacturer presented a newly developed tungsten carbide three-piece rotating seal assembly as a potential replacement for the sapphire rotating face and advised that it had been tested on a limited basis with satisfactory results. Two such seal assemblies were ordered and were placed in service in August of 1966; performance has been satisfactory. They were visually inspected in October of 1969 when the pump was removed from service for Cell 1 steam generator retubing. At that point, the estimated service for the seals was 23,000 hours compared to roughly 10,000 hours for the previous seals. Based on the performance of the tungsten carbide test installation, additional tungsten carbide seals were ordered and placed in service. To date, these have operated successfully. Accordingly, routine use of these seals is planned as a replacement for failed sapphire seals.

Control Rods

Rod 25 was removed from service during the month to provide an access hole for the pneumatic shuttle irradiation facility. The titanium sheath of the rod tip has been salvaged for reuse as a spare tip assembly.

Nineteen rod drive hydraulic seal assemblies were replaced on rods that had slow scram times during the October outage. Three of the seals were of a new design to be tested as part of the program to correct the slow rod problem.

One rod water leak was repaired and four other rods were found to be leak free by pressure testing.

Charge-Discharge Equipment

Several component breakdowns occurred during the refueling operation which prevented completing this work within the time scheduled. The trampoline trapped some short fuel elements and extra time was required to clear off the elements and adjust the trampoline profile. The major breakdown occurred when the mine cart cable broke as a result of damaged underwater cable sheaves. Temporary repairs were made to complete the charge-discharge, but new sheaves of improved design will be installed.

Temperature Detectors (RTD)

Extensive efforts were made during the October outage to eliminate all unreadable temperature detectors. Thirty-four were restored to service, but at startup four detectors remained unserviceable, and 15 were out of service at month end. Failed detectors are generally original installations. Repaired detectors of improved design are giving reliable service, but repair rates are limited by the availability of personnel radiation exposure.

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Seismic Event

A small seismic shock was felt at N Reactor at 11:35 a.m. on October 25, 1971. The intensity of the shock was insufficient to affect the seismoscopes at N, and the reactor continued in operation.

An immediate check of all instruments and recorders showed no effect at all of the shock except for the following:

- The recorder of deareator storage tank level at 184 Building showed a "wobble" corresponding to an approximate 4-inch change at the time of the event.
- The thermal shield bulk flow recorder showed a very minor, almost indiscernible bobble at the time of the event.

The WPPSS plant indicated no effect on their turbine-generation operation. One annunciator, "Potable Water Storage Tank Level Change" was received at 184-N. This tank is their day tank and is located atop 185 N. No other perturbations were observed.

An immediate assessment of the effect on the N Plant was made by the Operations Division manager and the managers of Engineering and Process Technology Sections. The results of the instrument responses, checks of all control systems, and preliminary physical checks were reviewed. Following this review, it was determined that all control and safety systems were operable and the decision was made to continue plant operation; however, a more detailed inspection was ordered. There is no evidence that N Reactor systems did not perform exactly as expected under a shock of the intensity indicated by the seismic network that recorded the event.

Engineering personnel and Operations representatives subsequently visually inspected all building structures and process equipment for signs of cracking, displacement, or other damage. Included were all motor control centers and large switchgear, large pipe hangers and snubbers, pump and motor mountings, shielding walls, shielding door fit, tank mountings, and tank leakage possibilities.

Boiler structures in the 184 Building were inspected and observations taken inside fire boxes for possible tube leaks.

Diesel pumps were functionally tested and gave normal responses.

No ground cracks were observed and no evidence has yet appeared of any underground pipe breaks or unexplained inventory losses from fluid systems in the plant.

A radiation survey was conducted of all plant areas to find any breach in shielding or release of radioactive fluids.

All inspections and surveys yielded negative results and subsequent operation confirms that no discernible process effects occurred.

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N Reactor Pneumatic Shuttle Irradiation Facility Status

Design Change 3183 (to be issued) authorizes the installation of a pneumatic shuttle irradiation facility in N Reactor horizontal safety rod channel 25. During the October outage, site preparation activities critical to meeting the December 1971 objective for operational status of the facility were completed. Horizontal safety rod 25 and the corresponding shield plug were removed. Enlargement of the thermal shield penetration was partially completed. A new shield plug, designed to accept the shuttle facility thimble, was installed and sealed with a temporary fitting. In addition, supports for shuttle tubing runs were installed in the inner rod room and a penetration of the inner-outer rod room barrier wall was completed. Remaining outage controlling work requirements are within the anticipated time availability for the December outage.

Fabrication of the in-reactor thimble is also on schedule and offsite instrumentation equipment orders have been received. The facility will be mocked up ex-reactor prior to the December outage to verify the design and to establish capsule travel time. Production Test 280, which will authorize routine operation of the facility and which includes the safety analysis, is routing for approvals.

Equipment Maintenance Standards (EMS)

The status of the development of the new EMS system is currently as follows:

	<u>September</u>	<u>October</u>
Standards scheduled	46	46
Standards issued	31	32
Standards routing for approval	8	7
Standards in preparation	7	7

PLANT MODIFICATIONS AND IMPROVEMENTS

Project Progress

The status of proposals and authorized projects is provided in Appendix A.

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OTHER ACTIVITIES

Quality Verification of Maintenance Work

A Work Completion Record (WCR) procedure has been developed which incorporates measures for quality verification of maintenance work on critical N Reactor systems. The procedure was approved for application on a trial basis during the October outage, and was applied to ten pre-selected jobs. The results of the trial application have been reviewed, and some improvements in the WCR procedure are being made for further application.

N Plant Quality Control Program Manual

A manual, "N Plant Quality Control Program," DUN-M-39, that specifies the quality control program used by the Operations Division for the operation and maintenance of N Plant has been issued. The manual describes the general quality control measures to be applied and responsibilities for their application.

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ADMINISTRATION - GENERAL

Union Relations

Twenty separate Appendix "A" meetings were held with HAMTC local unions during the month. These meetings provided all locals represented by the HAMTC with the opportunity to present their views concerning rates of pay and working conditions.

In each case the local union was advised that the Company would make no proposals or counter-proposals until 1972, when the DUN-HAMTC Agreement is open for negotiation.

Safety

No personnel radiation exposures exceeded operational control.

Month-end safety statistics were:

Disabling injuries - October	0
- CY to date	0
Days since last disabling injury	480
Man-hours since last disabling injury	2,700,500

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-547, Permanent Markers for Buried Wastes in terminated 100-D, 100-F, and 100-H Sites.	28,000	90	40	Tracts in 100-H Area have also been staked and setting of posts is under way in 100-F Area.
DAP-549, Rev. 1, De-activation of KE Reactor and 100-K Area.	160,000	100	60	Modification work and acceptance testing on the No. 3 KE boiler was completed 10/5/71.
<u>N Reactor</u>				
GCP-411, Rev. 4, Effluent Control Program, 100-N Area	1,960,000	99	98	Revision No. 4 approved, extending the project to 4/1/72.
DCP-528, Rev. 3, Fire Protection System Improvements, 100-N Area	330,000	70	0	Revision No. 3 was approved and the interim authorization was increased to \$330,000. Order placed for fire retardant material to coat cables in trays in 105-109 Buildings.
DCE-539, Rev. 1, Upgrade Nuclear Flux Monitoring System, N Reactor	335,000	96	8	Revision No. 1 has been approved, extending the project completion date to 10/31/72. Purchase order has been placed for the last major piece of engineered equipment; the startup range chamber positioners. The Certified Project Schedule-Construction has been prepared. System assembly continues at 105-D Instrument Shop.

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Number & Title	Authorized Funds - \$	Percent Complete		Remarks
		Design	Construction	
DAE-540, Rev. 1, Smoke Density Monitors for DUN-operated Facilities, 100-Areas	38,000	100	29	Installation of the gas sample ports was completed on both N standby boilers during the October plant outage. Most of the engineered equipment has been received. The remainder is due on site before month end. Final installation is scheduled for the December outage.
DCP-542, High-Pressure Injection System Improvements, N Plant	138,000	0	0	Project activity is at a standstill, pending study and review of alternate approaches.
DCE-551, Training Mock- Up for V-12 Valve Replacements, 185-D Building	49,500	100	50	Construction is slightly behind a 60 percent scheduled completion due to shut down work necessitated by the October outage, but com- pletion by 12/31/71 is expected as scheduled.
DCE-552, Control Rod Coolant Recirculation Facilities, N Reactor	235,000	60	0	Specification for heat exchangers has been approved.
DCP-553, Protection of Equipment, 109-N Building	12,000	13	0	Title I comment drawings are essentially complete.

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PROJECT PROPOSALS AWAITING AUTHORIZATION

<u>Number & Title</u>	<u>Funds Requested</u>	<u>Date to AEC-RL</u>	<u>Remarks</u>
DCP-528, Rev. 3, Fire Protection System Improvements, 100-N Area	\$ 350,000	7/8/71	Revision 3 changes the scope of work on the basis of Factory Mutual Research suggestions and as requested by AEC-RL. To date, \$330,000 have been authorized.
DCP-529, Gravity Drainage System and Disposal Basin for 100-N Area	200,000	7/9/69	Internal review of the proposal is in progress.
DAE-540, Rev. 2, Smoke Density Monitors for DUN-operated Facilities, 100 Areas	44,000	9/27/71	Requests addition to scope of one monitor for the one active boiler in 100-K Area. Letter in preparation to recall the addition because of planned shortening of K metal shipping schedule.
DAP-547, Rev. 1, Permanent Markers for Buried Wastes in Terminated 100-D, 100-F, and 100-H Sites	49,000	9/21/71	Requests coverage of additionally abandoned zones in 100-D Area. By letter, DUN concurs with subsequent oral AEC-RL proposal that the scope be modified. On this basis, the total work is estimated to cost \$42,500.
DCP-553, Protection of Equipment, 109-N Building	85,000	7/12/71	To date, \$12,000 have been authorized.

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APPENDIX B

Loading Pattern - N Reactor

	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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Tube Loading Status

Fuel Code	No. Tubes	Description	PT-NR No.	No. Tubes	Description
94G	420	Mk-IV 0.95% Fuel; Core & Fringe	113	4	Mk-IV Spider Inner Support
94GA	85	Mk-IV 0.95% Fuel; Core & Fringe	136	8	Mk-IV Non Bonded End Closure.
94GB	10	Mk-IV 0.95% Fuel; Core & Fringe	118	25	Mk-IV Performance Test
94GC	60	Mk-IV 0.95% Fuel; Core & Fringe	230	3	Mk-IA Fuel with "W" Springs
94GEV	28	Mk-IV 0.95% Class II Fuel; Fringe		1	Blank Channel
94GAS	7	Mk-IV 0.95% Class II Fuel; Fringe		41	Total
94GBC	7	Mk-IV 0.95% Class II Fuel; Fringe			
94E	112	Mk-1C 0.95% Fuel; Fringe			
71A	2	Mk-1B 0.71% Natural Uranium			
	731				
XA	1	Mk-1A 1.25%/0.95% Spike Enrich.			
XB	195	Mk-1A 1.25%/0.95% Spike Enrich.			
XC	31	Mk-1A 1.25%/0.95% Spike Enrich.			
XD	5	Mk-1A 1.25%/0.95% Spike Enrich.			
	963				
	41	Special			
	1004	Total			

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