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DOUGLAS UNITED NUCLEAR MONTHLY REPORT

AUGUST 1971

DOUGLAS UNITED NUCLEAR, INC.

RICHLAND, WASHINGTON

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DOUGLAS UNITED NUCLEAR

MONTHLY REPORT

AUGUST 1971

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SUMMARY

N Reactor

Steam was supplied to WPPSS on August 2 and 3, as requested, for prestartup testing of the electrical generating equipment. Steam was again supplied to the electrical generating plant on schedule August 10 and the reactor power was held at approximately 2000 MWt to meet the Supply System's 410 MWe requirement. Reactor power was increased on August 17 in response to a request from WPPSS for increased steam generation to meet an increased electrical power demand of 810 MWe. Electrical power generation of 828 MWe at a reactor power level of 3675 MWt was demonstrated at month end through effective trimming of auxiliary steam losses and consumption.

Achievement of the production and availability goals associated with the three-year operating mode has required some major adjustment and redirection of resource application. A systematic review is currently under way aimed at the identification, scheduling, and implementation of corrective and preventative maintenance activities and the development of engineered solutions needed to achieve the necessary reactor performance requirements.

Charles D. Harrington
President

OPERATIONSN REACTOROperation

The reactor was started up on August 2 and steam delivered on schedule to WPPSS for operational testing of the turbine generators. Reactor power level reached during this testing was 2400 MW. On August 9, the reactor was again started up to deliver steam to WPPSS for the production of a nominal 400 MWe. During the ensuing seven days, the power level was limited to 2000 MW. On August 16, the Supply System requested additional steam to meet an 810 MWe electrical requirement. Power level was raised to 3650 MW and steam equivalent to 810 MWe was delivered.

The reactor was scrammed on August 18 as a result of a 125 volt dc ground and again on August 19 and 21 as a result of two fuel failures. Equilibrium power level was subsequently reached on August 23. Power level was again raised at WPPSS' request reaching a maximum of 3680 MWt and an electrical generation rate of 835 MWe on August 27. Shortly after reaching 3680 MW, the level was lowered to 3675 with a corresponding drop in electrical generation rate to a level of 828 MWe.

Reactor operation continued through month end at 3675 MWt with electrical generation rate of 828 MWe.

Maximum thermal efficiency reached was 22.3 percent as compared to a previous high of 21.2 percent.

The ground in the 125 volt dc system which caused the August 18 outage was located and corrected within the minimum outage time. Similarly, fuel failures occurring in tubes 2360 and 1051 resulted in minimum outages. The suspect fuel charges from 2360 and 1051 were transferred to the fuel examination facilities for confirmation and determination of the cause of failure.

Since WPPSS had indicated that steam for electrical production would not be required until August 9 (later rescheduled to August 10), the interim period following WPPSS operational testing until August 9 was used to perform various maintenance activities and to operate the reactor at low power level (maximum 1700 MW).

During the first nine days of August, the reactor maintained an "available status" and was credited with 151.2 million kwh. During the next seven days, while reactor power level was restricted to about 2000 MWt and a nominal electrical generating rate of 400 MWe, credit for 130.7 million kwh was received. These latter credits were based upon reactor availability at an electrical generation rate of 810 MWe.

Over the entire month, credit was received for 514,976 million kwh. Potentially, credit for 578.9 million kwh was achievable. Thus actual performance was at the 89 percent level.

A total of 298.370 million kwh was actually produced and, based upon a plant capacity of 810 MWe, represents 83 percent of the power requested by WPPSS. It should be noted that achieving an electrical generation rate of 835 MWe is a first for N Reactor-WPPSS complex and is the highest achieved nuclear generation rate for a single reactor in the world.

Statistical Summary

Input production - Pu (KMWD)	61.9
Electrical generation (KMWH) - WPPSS	298.37
*Equivalent electrical generation - (KMWH)	514.97
- FY 72 to date	789.726
Power level (MWt) - maximum	3,680
Fuel failures	2
Fuel charge (tons) - 94 Metal	331.09
-125 Metal	64.68
- Natural Uranium	<u>0.37</u>
Total	396.14
Helium consumption (1000 cu. ft.)	**(606.1) credit
Fuel oil consumption (bbl.)	17,986
Time Operated Efficiency - %	80.0

* Reflects availability status.

** Error in July reported consumption figure corrected.

The loading pattern and fuel failure data are shown in Appendix B.

Equipment Experience

During preparation for delivery of steam to WPPSS for their operational testing, a trip was experienced on the No. 4 drive turbine. The cause of the trip was identified as a drift in the set point in the oil system pressure regulator. The regulator was repaired and subsequent operation has been satisfactory. The period following operational testing of WPPSS turbine generators was used advantageously to perform a number of maintenance tasks. Significant items are described in following paragraphs.

A fault in the circuit of the controller for channel 2146 in the Zone Temperature Monitor System was located as a broken connection of a resistor. The connection was repaired and the controller returned to service. The remaining channel controllers were inspected and found in satisfactory condition.

Valve leaks in the primary coolant system were repaired to eliminate water hammer in the low pressure drain system.

Emergency dump tank modifications were made to permit better control of tank temperatures during reactor operation.

Rupture monitor sample flows were corrected.

Steam generator 6B was determined to be leak free.

Additional equipment experience and maintenance activities are detailed in the Systems and Equipment Section of the report.

Tritium Shipment

Coproduct target elements for the first tritium shipment continue to be held in dry storage (NPR Cask) waiting delivery of the shipping containers.

The current estimated delivery date of the first isotope shipping cask is September 10, 1971.

The trailer mounted Super Tiger overpack shipping container arrived in Richland on August 16. On August 27, a test load of approximately 14 tons was made of the trailer-mounted Super Tiger and related equipment. No major problems were encountered, but it was evident that trailer jacks will be required to level the load to overcome the trailer deflection of about two inches incurred when the weight is transferred from over the rear axle of the trailer to the Super Tiger interior.

Present plans call for making the final tests of the Super Tiger and related equipment during the week of September 13.

FUEL FABRICATIONOperation

An extension of the X09 extrusion run delayed the extrusion of matching B09 outer fuels. The delay, along with operating problems at the braze, clad and bond test stations, lowered output production.

Defect rates at the processing station were low for all categories and should result in excellent yields.

Statistical Summary

Tons extruded	49.0
Finished assemblies	725
Tons output	17.1
Uranium utilization - % B09	86.5
X09	--
Month-end inventories - Tons	
Bare Uranium Billet	112
Finished fuel	261

PROCESS ASSISTANCESafety Review - Shipment of Uranium Cement Billets

For many years, metallic uranium and zirconium scrap and chips, produced as a byproduct of fuel manufacture, were burned to an oxide and returned to the feed site. Shipping procedures, containers, and criticality specifications were developed and used for shipment of uranium oxides.

The basic need for reducing the metallic fines to oxide is to avoid the pyrophoric hazard in handling of finely divided metallic uranium and zirconium. A secondary advantage resulted from the lesser criticality hazard associated with uranium oxides as compared with metallic uranium.

The difficulty of meeting the health and environmental requirements associated with burning metallic uranium to an oxide provided a strong incentive to develop an alternate method for preparing and shipping finely divided and scrap uranium. Consequently, development of a replacement shipping method was initiated. A method was conceived which consists of encapsulating the metal fines as aggregate in concrete billets (using a mixture of Portland cement, limestone, and gypsum as the masonry cement). A safety evaluation was performed consisting of criticality review, a pyrophoricity safety review, a structural evaluation of the billets, and permissible shipping configurations.* Procedural limitations and shipment mass limits have been identified which ensure against accidental criticality, spontaneous combustion, or spread of uranium contamination during shipment.

The development program and the safety review have resulted in the conclusion that the program as described in DUN-7654 meets the safety requirements defined in USAEC Manual Chapter 0529, "Transportation of Radioactive and Fissile Material," and that uranium and zirconium metallic fines can be safely shipped as an aggregate in concrete billets.

Based on this conclusion, AEC-RL has been requested⁽¹⁾ to obtain an amendment to Department of Transportation Special Permit No. 5467 to authorize shipment of the cement-uranium billets in NLO containers. Upon approval, it is expected a number of billets will be shipped to NLO for their use in testing the process for recovery of the uranium.

N Reactor Rod Safety System - Response Time Study

For speed-of-control purposes, N Reactor Process Standards require that individual rods enter the reactor from full-out to 75 percent inserted in 1.5 seconds. The rod system is backed up by the ball safety system in that failure of three rods to

*The evaluation is contained in document DUN-7654, "Safety Review - Shipping Masonry Cement Billets Containing Scrap Uranium in NLO Boxes," JT Long and H Toffer (to be issued).

⁽¹⁾ Letter, CW Kuhlman to OJ Elgert, "Safety Review - Shipping Masonry Cement Billets Containing Scrap Uranium in NLO Boxes," August 31, 1971.

meet the response time requirement will cause ball hoppers to empty into the reactor. The number and location of ball hoppers activated is correlated to the number and location of rods failing to enter 75 percent in 1.5 seconds. To provide a margin of safety for possible increase in response time and to avoid inadvertent ball drops, the Equipment Maintenance Standards (EMS) require individual rods to scram, from full-out to 75 percent inserted in 1.2 seconds under test conditions. Rod drive systems are adjusted to be within the EMS time response requirement before reactor startup.

On August 17, following 196 hours of operation, a rod scram occurred as a result of a grounded electrical circuit. Rod scram speeds were measured and analyzed, and 19 of 86 rods were found to have scram speeds of more than 1.2 seconds. Two of the 19 had a response time greater than 1.5 seconds, and the response time of another was measured just at 1.5. Sixteen of the 19 rods were left side rods. The left side rod hydraulic system contains invert-emulsion fluid, while the right side contains the original type turbine oil. The average speed for the 19 rods was 1.36 seconds while the average response time for all left and right side rods was 1.168 and 1.127 seconds, respectively. The average response time for the rod system was well under the 1.5 second response time specification and therefore no reactor safety problem existed. However, two rods exceeded the 1.5 second response limit, while a third equaled the response limit. Thus, had the response time of the third rod been a few hundredths of a second slower, a part of the ball safety backup system would have been actuated.

Because of the safety and reactor availability implications of slow rods, the program to identify the cause of slowed rod response time has been expanded.

Previous analysis of rod response times under scram conditions indicate a preponderance of slow rods to be on the left rod bank. Evaluation of the system condition for rods with response time greater than 1.5 seconds shows the floating bearing bushing skews in its housing causing binding between interior of the bushing and the piston rod. The basic reasons for these observations are not currently understood.

An empirical evaluation of the trend in individual rod response times is being firmied so that preventive corrective action on slow rods can be appropriately taken.

Development Test 279, "Effect of Rod Exercise on Scram Speed," is being implemented to obtain additional data on a controlled basis. In this test, approximately half the safety rods are to be moved inward and outward 10 percent once a day. The remaining half of the safety rods will not be moved. Following a scram, a comparison of response times of the two sets of rods will be made. Results from this test should provide additional insight into the effect of the invert-emulsion fluid used in the rod drives on the response time.

Reactor Startup and Power Ascension

The production gains that are potentially available by attaining the fastest reactor startup and ascension to equilibrium power level permitted by Process Standards has been under continuing evaluation. Under the new AEC-RL-WPPSS

agreement, the potential increased electrical production provides an added incentive to bring N Reactor to equilibrium power level as rapidly as safety requirements will permit.

Specific startup events along with the time requirements associated with the event for both "normal" and "fastest" startups are shown in Table I. Also shown in the table are the appropriate Process Standards which limit the rise to equilibrium power level. These power level-time requirements are shown graphically in Figure 1. At a nine percent Pu-240 assay, the difference between these two startup-power ascension rates amounts to about 1/2 kg of plutonium per startup. The incremental electrical production amounts to approximately \$23,000 per startup.

One of the principal deterrents to approaching the "fastest" startup mode of operation is the ability to rapidly calculate and apply process tube limits. Through the cooperation of Computer Science Corporation (CSC), arrangements for calculating reactor temperature limits for the hours between 10 p.m. and 7 a.m. via teletype hookup with CSC are being firmed. Additional training of reactor personnel in the use of these techniques is proceeding. In addition, some refinements in calculation techniques are evolving.

Fuel Element End Filming

The fuels manufacturing process includes self radiographing of each fuel piece to ensure that the outer .002-inch of the end weld surface averages less than one percent uranium. Uranium contamination in the weld bead is of concern since it would increase in-reactor corrosion rates, increase the activity of the coolant water, and possibly be a source of potential reactor fuel failure. "End filming" of the finished elements has provided assurance that no uranium is contained in the fuel ends; however, only seven end film rejects have been found in a group of 50,000 fuels, and the presence of uranium in the end welds has never been confirmed by analytical techniques. This experience supported consideration of eliminating the end filming operation from the process and using other process inspection steps to detect uranium contamination as a means of effecting some cost savings.

A test designed to determine if uranium contamination in the end welds could be detected at end testing, weld inspection, or at post-autoclave inspection, has been completed. Uranium strips .10-inch thick were brazed and welded into the end closure between the end cap and fuel cladding of a selected number of fuel pieces in an effort to obtain uranium contamination between three and ten percent.

The results of this test indicate that uranium weld contamination cannot be reliably detected at weld inspection, bond testing, or at post autoclave inspection. The current fuels process contains no other operation where uranium contaminated welds can be reliably detected.

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TABLE I

Startup Event	Time from One Event to the Next		Basis for Fastest Times (Process Standards Limits)
	Normal	Fastest	
1) Start rod pulls	--	--	A-105 Rod Withdrawal Rate
2) First indication of critical (with two subcritical insts.)	25 Min.	15 Min.	A-105 Rod Withdrawal Rate
3) 150 MWt	25-30 Min.	15 Min.	A-105 Power Rate of Rise and Primary Coolant Temp. Transient Limits B-200 Pressurizer Temp. Transient Limits
4) Hold at 150 MWt to allow primary coolant to reach 300 F	60-90 Min.	60 Min.	A-105 Advice B-200 Primary Cooling System Temp. Transient Limits
5) 500 MWt	15 Min.	12 Min.	A-105 Power Rate of Rise B-200 Primary Cooling System Temp. Transient Limits B-245 Flow Map Required
6) Hold 500 MWt to check power meters and in- crease turbine speeds	2-1/2 to 3 Hrs.	1-1/2 Hrs.	B-200 & A-105 Primary Cooling System Limits and Power Rate of Rise Limits B-265 Power Level Check
7) 2000 MWt	1 Hr.	20 Min.	A-105 Power Rate of Rise Limits B-200 Primary Cooling System Temp. Transient Limits
8) Hold 2000 MWt for power check	1-1/2 Hrs.	15 Min.	B-250 Temp. Map B-265 Power Check During Scram Recovery
9) 3200 MWt	2 Hrs.	30 Min.	A-105 Power Rate of Rise Limits B-200 Primary Cooling System Temp. Transient Limits B-265 Tube Power Limits
10) Hold 3200 MWt for tube power check and TWFM run	24 Hrs.	13 Hrs.*	B-265 Thermal Power Level Checks
11) 3650 MWt	4 Hrs.	20 Min.	A-105 Power Rate of Rise Limits B-200 Primary Cooling System Temp. B-265 Tube Power Limits

*This number is based on holding power until there is less than 15 mk of rod in the reactor before running Traveling Wire Flux Monitors (TWFM). With very good radial flattening, it may be feasible to reach 3650 MWt on startup tube power limits, thereby reducing the time from 13 hours to about three hours.

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FIGURE 1

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SYSTEMS AND EQUIPMENT

Critical Systems Action Plans

The maintenance and improvement programs at N Reactor which were in progress prior to the January 28 shutdown were based upon an anticipated extended AEC production role over the long-term future. The current 3-year operating mode emphasizes short-term performance requirements. To assure achieving the production and availability goals associated with the 3-year operation, DUN has initiated a management tool which will assure the identification, scheduling and implementation of corrective and preventive maintenance activities and development of engineered solutions to achieve critical system performance requirements.

The N Reactor facility has been divided into 44 major systems which will be reviewed to:

1. Establish system performance requirements to insure that the combined plutonium and electrical production commitments are achieved for the next three years of operation.
2. Identify maintenance and technical support necessary to achieve the established performance requirements.
3. Establish priorities for the identified tasks considering manpower, budget and scheduled outage-nonoutage constraints.

These critical system reviews, called system action plans, will provide the basis for detailed outage planning and scheduling and the performance of those activities required to achieve an annual production of four billion kwh within the constraints of the 3-year operating mode, nuclear safety requirements, and stringent budgets.

Primary Coolant System

Steam water hammers of varying intensity were experienced in the low pressure drain system in early August. Investigation narrowed the cause to one or more of several valves that could be leaking through from the primary loop into the low pressure flush line. During the outage of August 8, several corrections were made including manual seating of V-18-1, repair of broken packing followers on V-4-5, V-4-6, and V-4-8, repair of V-7 operator and seating of V-7. In addition, thermocouples were installed on a number of primary loop lines in order to measure temperature transients downstream of valves during plant startup. Although the water hammer has essentially disappeared, the temperatures indicate that there may still be some minor leakage through several valves. Planning has been initiated to effect a permanent repair at the next scheduled reactor outage.

Valve leak-through along with the high atmospheric temperatures experienced early in the month, resulted in high dump tank temperatures. On August 8, tank modifications were made to permit better control of the tank temperature. The normal overflow capacity of the tank was increased and an additional cold water supply to the tank was installed for use during periods of high temperature. These

modifications were successful and no further dump tank temperature problems have been experienced.

On August 17, during a power level increase, a faulty low trip of unknown origin on a process tube flow monitor prevented resetting the flow monitor low en masse low trip to the setting required for power ascension above 2000 MWt. A procedure was developed to determine the identity of the faulty low trip without shutting down the reactor. This procedure was applied successfully and delayed the reactor power ascension schedule only about 45 minutes. It is estimated that an 18 to 20 hour delay would have occurred had an outage been required.

A pressurizer cold spray control system failure on August 19 during reactor startup was traced to a stuck check valve, the valve was freed to correct the problem without a reactor shutdown.

Primary loop leakage during the month has remained steady and low as reflected in the average injection rate of 1150 gpm. By comparison, this injection rate during the previous operating period in January was 1800 gpm. Unaccounted for leakage at this time is estimated to be between 150 and 200 gpm.

125 Volt DC System Ground and Correction

Following reactor startup in mid-July, several experiences of grounds occurred on the 125 volt dc system. Since the normally ungrounded 125 volt dc source is used to power all critical control functions in the plant, considerable effort was expended to isolate and correct the problems. Although undesirable, the occurrence of these grounds was not surprising considering the extent and magnitude of the reactor work that occurred and the length of the preceding outage. However, through the employment of sophisticated techniques, developed at N Reactor for detecting and isolating grounded circuits, little liability was incurred on plant availability and no compromise was made to nuclear safety.

The techniques for troubleshooting the grounds require skilled engineers and technicians and involve a specially designed ground detector which detects simultaneously occurring positive and negative grounds, a simple but comprehensive procedure for determining when corrective action is required following the advent of a ground, a portable isolated dc power supply, and comprehensive troubleshooting procedures.

By month end, all known dc grounds were either cleared or their circuits isolated from interference with critical control functions. The isolated circuits are temporarily energized from a portable dc power supply and will be checked, cleared, and restored to normal during the next scheduled reactor outage. Meanwhile refinement work is continuing on improving the techniques for troubleshooting grounds.

Secondary Coolant System Water Quality

A surface condenser leak on No. 3 turbine was isolated as the cause of high chloride concentrations in the secondary system. Steam generator 6B, which was found leaking in July, was thought to be leaking again, but recent operation at higher power levels showed this generator to be leak free. Secondary system water quality has been within good control since August 18.

Cell No. 1 Prefilming

Cell No. 1 prefilming began on August 13. The prefilming process forms a protective oxide coating on the steel and inconel surfaces of the steam generators by recirculating water within the cell at 300-310 F for about 30 days while the cell is isolated from the primary loop. Corrosion products generated during the process are removed by a feed and bleed arrangement and a non-radioactive oxide film forms on the surfaces of the steam generators. This protective film reduces the subsequent corrosion rate and results in a lower rate of radioactivity build-up in the tubes as well as in the primary loop. The operation is proceeding very well and will be complete in time for the cell to be returned to service following the October outage.

Graphite Cooling System (GCS)

A graphite cooling system dump occurred at the time of the scram on August 18 when a system surge suppressor failed in the 125 volt dc control circuit. On previous dumps plugging of 10 to 15 percent of the 640 tubes with debris from the raw water has been experienced. Cleanup of the system on these occasions has required several days of outage time. During the CY 1970 summer outage, a flush system was installed in the GCS emergency raw water supply line and periodic flushes are conducted in order to keep the line as free of debris accumulation as possible. Following the dump on August 18, no GCS tubes were plugged and only four tubes showed any flow change. These four tubes were checked, but no debris was found and their flows subsequently returned to normal. The incident afforded an opportunity for the first full test of the effectiveness of the raw water flush system.

Resistance Temperature Detector (RTD) Repair Program

The short-term results of the present RTD repair program described below indicate a marked improvement over previous repair experience. Currently, the number of RTD channels needing repair are the lowest since January 1969. The table below shows the trends during this period. In the table, ATTM stands for all tube temperature monitor and ZTM stands for zone temperature monitor.

<u>Date</u>	<u>ATTM Unreadables</u>	<u>Readable only on Manual Scanner*</u>	<u>ZTM RTDs Failed & Bypassed</u>
January 1969	0	33	1
July 1969	12	61	13
After Summer Outage-1970	25	158	30
January 1971	18	143	21
After Summer Outage-1971	0	175	0
August 1971	11	49	7

*The 1003 ATTM RTDs are normally read on an automatic data logger.

After the summer outage this year, the number of RTD channels on the manual scanner were relatively high (175) because it was planned to use reactor heat to dry out the RTD cables whose cable ends had been treated with a resin which allowed moisture to leave the cable but retarded moisture entering the cable. The result of utilizing reactor heat is shown by the fact that only 49 RTD channels remain on the manual scanner as of this report.

The favorable results mentioned above are attributable to a new repair program. The program includes the specialized application of a compound of several resins (invention disclosure is in process), extensive craft training in making repairs, audit of failures during operation and after reactor shutdown with detailed planning and selection of which repairs would be made and improved repair procedures. The overall effort received an important assist from the reduction of rear face water leaks accomplished by the V-12 repair program.

It is believed that the cable end treatment was one of the more significant improvement items. The end treatment consists of attaching a reservoir to the open cable end, filling the reservoir with the special resin, heating the cable to initiate drying and cause expansion of gasses in the cable such that when the cable cools the resin will be drawn into the magnesium oxide. The cables are then spliced, leaving the connection open to the environment. Approximately 135-150 C is needed to achieve cable drying. This is reached during reactor operation with the results tabulated above.

Equipment Maintenance Standards (EMS)

The program for developing EMSs is progressing well. Although no new or revised standards were approved this month, two new and two revised standards were issued for approval and one for comments.

The overall status of the EMS program is currently as follows:

	<u>July</u>	<u>August</u>
Standards scheduled	46	46
Standards issued	31	31
Standards routing for approval	3	5
Standards in preparation	11	10
Preparation not started	1	0

Standby Boilers

Two procedure changes were accomplished during the month to reduce standby boiler operating costs. During the operation at 2000 MWt from August 10 to 17, a Process Change Authorization (PCA) was authorized to permit reducing boiler load from 120,000 to 60,000 lb/hr at a reactor level of 2000 MWt instead of 2400 MWt as previously required. The change resulted in a savings of \$1,500 per day when operating at the 2000 MWt level. Later in the month, the minimum standby boiler load was reduced from 60,000 lb/hr to 30,000 lb/hr. This change is estimated to save about \$100,000 per year in fuel oil costs. Further boiler load reductions are being considered.

PLANT MODIFICATION & IMPROVEMENT

Project Progress

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

SPECIAL STUDIESN Reactor Pneumatic Shuttle Irradiation Facility

Several Hanford single pass reactors, notably KE Reactor, contained facilities wherein a wide spectrum of specialty or developmental irradiations could be conducted. While these facilities accommodated many of the testing needs of various AEC contractors, certain of these facilities were highly instrumental to the continuation of high priority programs for the AEC Division of Biology and Medicine under the direction of Battelle Northwest. With the placement of KW Reactor into standby status on February 1, 1970, and the potential for similar action regarding KE Reactor, an informal study was initiated to assess the costs of relocating or installing a newly designed facility(ies) in N Reactor.

Basic design features and costs for several facilities were scoped in the study and reviewed with AEC-RL. Balancing of cost factors against the relative priority or status of various AEC contractor programs identified a facility, called the N Reactor Pneumatic Shuttle Irradiation Facility, as the only design for which funding appeared feasible. On August 10, 1971, official authorization and funding was received from AEC-RL to proceed with the fabrication and installation of this facility.

The facility will cost \$45,000 including fabrication and installation. Achievement of operable status is scheduled for mid-November, 1971, but could be influenced somewhat by unanticipated variability in the N Reactor outage schedule. The facility will handle only encapsulated nonfissionable samples unless circumstances dictate the desirability of additional flexibility and an expanded safety analysis. Design of the installation necessitates limitations of sample geometry and heat generation rates on test samples. The shuttle facility will be installed under a design change and operated under a production test authorization, both of which are currently routing for approval.

The in-reactor portion of the facility will be installed in right-side horizontal rod channel No. 23 following removal of the resident isotope rod and shield penetration modifications. Three cylindrical aluminum sample (shuttle) tubes will be housed within an outer titanium sheath. An inner stainless steel sheath which also surrounds the shuttle tubes, forms the boundary for entering and exiting coolant flow. Coolant supply will be obtained from the existing horizontal rod coolant supply source. Existing reactor safety circuitry will also be utilized although wider and less restrictive coolant trip settings will be applicable.

The three shuttle tubes will penetrate to various locations in the flux spectrum to accommodate variations in test requirements. Capsules will be charged from the outer rod room through tubing runs to the desired shuttle tube under the impetus of pressurized helium. Gas flow is reversed for sample discharge. Ex-reactor switching equipment will permit discharge to a decay cave or directly to a transportable cask depending on sample activity.

Controls for shuttle tube operation will be located in the right side outer rod room to permit direct physical control of personnel to the area during capsule discharge. The safety review now in progress is expected to show that loss of coolant during reactor operation would necessitate a reactor shutdown. With this probable exception, the facility is designed to provide minimum disruption to reactor operation and outages.

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The safety review is scheduled for completion by September 3, 1971. Authorizations for installation and operation of the facility are currently being prepared and will be routed for approvals during September 1971. Procurement and fabrication activities have also been initiated. Outage work has been tentatively scheduled for the removal of rod No. 23 during the September outage. All aspects of the project are on schedule with respect to the mid-November 1971 target date for achieving operational status.

Zircaloy-2 Corrosion

Zircaloy-2 clad fuel for N Reactor is chemically cleaned in an aqueous solution containing 2.5 to 3.5 percent by volume hydrofluoric acid and 40 percent nitric acid at 26 C prior to subjecting the fuels to irradiation. The purpose of chemically etching the Zircaloy-2 cladding is to remove approximately .0007 to .0013 inch of metal from the fuel surface, thus assuring complete removal of the worked surface layer and any surface contaminants.

The etchant solution is normally prepared by using liquid hydrofluoric acid or by additions of crystalline ammonium bifluoride to nitric acid. With use of either source of fluoride ion it is important to maintain stringent control of the etching and rinsing processes in order to ensure that good corrosion resistant Zircaloy-2 surfaces are prepared. Incomplete rinsing following etching can cause the etchant to continue to react on the fuel surfaces with the reaction products adhering or drying to the fuel surfaces.

Subsequent exposure in high temperature water or steam results in the formation of cloudy or white Zircaloy-2 surfaces that indicate accelerated corrosion of the areas having residual etchant. Zircaloy-2 clad N Reactor fuels are given an autoclave prefilm treatment in 750 F, 1500 psig steam prior to irradiation. When white oxide or accelerated corrosion occurs during prefilming, expensive fuels are often scrapped due to lack of information with which to predict in-reactor corrosion rates and oxide thicknesses.

A corrosion in-reactor test was completed to determine the effects from residual surface etchant on Zircaloy-2. The test included comparison of hydrofluoric acid versus ammonium bifluoride as etchant, comparison of the effects of various amounts of etchant staining as determined by delay in transfer time to rinse, and comparison of the performance of low (100 psig) and high pressure (1500 psig) steam prefilming and as-etched coupon preparation.

The results indicate that corrosion rates of Zircaloy-2 in-reactor do not become catastrophic due to etchant contamination. The total corrosion occurring even on samples dried 2.5 hours after etching does not appear catastrophic or detrimental to fuel element performance; however, post-irradiation examinations are needed of oxide thicknesses and uniformity to substantiate this conclusion.

The use of either hydrofluoric acid or ammonium bifluoride as etchant does not appear to cause any corrosion rate difference. Hydrogen pickup, being an important parameter on Zircaloy clad performance, was determined only on the out-of-reactor comparison samples and was found to be normal.

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ADMINISTRATION - GENERALUNION RELATIONS

The Hanford Atomic Metal Trades Council and the various Hanford contractors involved were unable to reach agreement regarding seniority of employees transferred between contractors in conjunction with the transfer of some functional responsibilities. Therefore, in accordance with previously agreed upon procedures the matter was referred to arbitration. The arbitration hearing was held on August 25. The arbitrator's decision is expected about September 7, 1971.

SAFETY

No personnel radiation exposures exceeded operational control.

Month-end safety statistics were:

Disabling injuries	- August	0
	- CY to date	0
Days since last disabling injury		419
Man-hours since last disabling injury		2,485,084

APPENDIX A

PROJECT STATUS SUMMARY - REACTOR FACILITIES

AUTHORIZED PROJECTS

<u>Number & Title</u>	<u>Authorized Funds - \$</u>	<u>Percent Complete Design</u>	<u>Construction</u>	<u>Remarks</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	54	0	Drawings have been approved and materials are on order. A project proposal revision has been prepared to incorporate further work in 100-D Area due to abandonment.
DAP-549, Rev. 1, Deactivation of KE Reactor and 100-K Area	160,000	60	46	All of the special-order material for No. 3 boiler modification has been ordered by J.A. Jones and delivery is scheduled for early September. Vitro has issued the heat and ventilation drawing for comments.
<u>N Reactor</u>				
GCP-411, Rev. 3, Effluent Control Program, 100-N Area	1,960,000	95	97	Installation of 182-N sprinkler system in progress. Deluge system design for 182-181-N is 80% complete. Dump tank instrumentation ready for completion next outage.
DCE-519, Rev. 4, Replacement of Bridge Crane and Hoist System with New Crane System, 105-N Storage Basin Area	465,000	100	99	Bids for two-ton hoist received 8/25/71.
DCP-528, Rev. 1, Fire Protection System Improvements, 100-N Area	150,000	50	0	Purchase specification for cable coating being prepared. Design of fire protection for 117-N in progress. By directive the work authorized is based on the scope of work covered in Rev. 3, below.

AUTHORIZED PROJECTS (continued)

<u>Number & Title</u>	<u>Authorized Funds - \$</u>	<u>Percent Complete</u>	<u>Design</u>	<u>Construction</u>	
DCE-539, Upgrade Flux Monitor System, N Reactor	335,000	90		0	Fabrication of the new control room instrument panels is nearing completion. Bids for the new ion chambers have been received, reviewed, and a split order between the two responsive bidders has been recommended. The bid package has been submitted for AEC approval.
DAE-540, Rev. 1, Smoke Density Monitors for DUN-operated Facilities 100-Areas	38,000	100		10	DUN Purchasing has placed all orders for engineered equipment with a mid-October delivery date promised. J. A. Jones Company has continued installation work at 184-N.
DCP-542, High-Pressure Injection System Improvements, N Plant	138,000	0		0	All work stopped January 28, pending decision on N Reactor operation. Alternate approaches are being studied.
DCE-551, Training Mock-Up for V-12 Valve Replacements, 185-D Building	49,500	100		1	Five sheets of working drawings have been approved. Material lists have been prepared and some orders have been placed.
DCE-552, Control Rod Coolant Recirculation Facilities, N Reactor	235,000	20		0	Design progressing. Specifications for heat exchangers and pumps are being prepared.

PROJECT PROPOSALS AWAITING AUTHORIZATION

<u>Number and Title</u>	<u>Funds Requested</u>	<u>Date to AEC-RL</u>
DCP-528, Rev. 3, Fire Protection System Improvements, 100-N Area	350,000	7/8/71 (Replaces previous unauthorized Rev. 2)
DCP-529, Gravity Drainage System and Disposal Basin for 100-N Area	200,000	7/9/69
DCP-553, Protection of Equipment, 109-N Building	85,000	7/12/71 (Replaces previous unauthorized Project DCP-538, Heat and Ventilation System Improvements, 105-N and 109-N Buildings, which was returned 7/29/71.)

DUN-7741

PROJECT PROPOSAL PREPARATION

<u>Number and Title</u>	<u>Design Criteria</u>	<u>Project Proposal</u>
DAE-540, Rev. 2, Smoke Density Monitors for DUN-Operated Facilities, 100 Areas	Activity Started	Activity Started. Incorporates work at one active boiler in KE Area.
DAP-547, Rev. 1, Permanent Markers for Buried Wastes in Terminated 100-D, 100-F, and 100-H Sites	Completed	Prepared for approvals.

Loading Pattern - N Reactor

APPENDIX B

AUGUST, 1971

	41	42	43	44	45	46	47	48	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64	65	66	67	68	69	70	71	72	73	74		
34																																			34	
33							EV	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	33	
32							E	E	E	E	E	E	E	E	E	E	E	E	E	13	13	G	E	E	E	E	E	E	E	E	E	E	E	E	32	
31							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	E	E	31	
30							E	E	E	X	X	X	X	X	X	X	X	X	X	18	X	G	X	X	X	X	X	X	X	E	E	E	E	E	30	
29							E	E	E	X	X	94	X	18	G	18	G	G	G	G	G	G	G	G	G	G	G	X	X	X	X	E	E	E	29	
28							E	GA	E	X	X	X	G	GA	G	G	GA	GA	G	G	G	G	18	GA	G	GA	GA	GA	GA	X	X	X	X	E	E	28
27							E	E	E	X	X	X	X	G	G	GA	GA	G	G	G	GA	GA	GA	G	G	G	G	G	X	X	X	X	E	E	27	
26							E	E	X	94	X	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	GA	G	X	X	94	X	X	E	26
25							E	E	X	X	X	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	X	18	X	G	E	25
24							E	E	X	X	X	G	G	36	G	G	G	G	G	G	G	G	18	18	G	G	G	G	G	G	G	X	X	X	E	24
23							E	X	X	X	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	X	X	X	E	23	
22							GA	E	X	X	X	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	X	X	X	E	22	
21							E	X	X	X	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	X	X	X	E	21	
20							E	X	X	X	G	G	G	G	G	G	G	G	G	18	G	G	G	G	G	G	G	G	G	G	X	X	X	E	20	
19							E	X	X	X	G	18	G	G	G	G	G	G	G	G	18	G	G	G	G	G	G	G	G	G	X	X	X	E	19	
18							E	94	X	X	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	X	X	X	E	18	
17							E	X	X	X	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	X	X	X	E	EV	17
16							E	X	X	X	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	X	X	X	E	EV	16
15	EV						E	X	X	X	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	X	X	X	E	EV	15
14							E	E	94	X	X	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	X	X	X	E	14	
13	EV						E	X	X	X	GA	G	G	G	GA	G	G	G	G	G	GA	GA	G	G	GA	G	GA	G	GA	G	X	X	X	E	EV	13
12	G						G	X	X	X	G	G	G	G	G	GA	G	G	GA	G	G	G	G	18	G	G	G	G	G	GA	X	X	X	E	EV	12
11	EV						E	X	X	X	G	G	36	G	G	G	G	G	36	GA	GA	G	G	GA	G	G	G	G	G	X	X	X	E	EV	11	
10		EV					E	X	X	X	G	G	G	G	G	G	G	GA	G	G	GA	GA	G	18	GA	G	G	GA	GA	G	G	X	X	X	EV	10
09	ES						E	X	X	X	X	G	G	G	G	G	G	G	G	G	G	G	G	GA	BA	G	G	G	X	X	X	X	E	09		
08							E	EV	X	X	X	G	G	G	G	GA	AS	G	G	G	AS	AS	G	AS	G	AS	AS	G	G	X	18	X	EV	E	08	
07							ES	EV	EV	X	X	X	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G	X	X	X	X	E	ES	E	07	
06								E	EV	EV	X	X	X	X	G	G	G	G	G	G	18	G	G	G	G	G	18	E	X	X	X	X	EV	ES	E	06
05								E	EV	EV	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	EV	E	05	
04								ES	E	EV	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	EV	ES	E	04	
03								E	E	BC	E	X	E	X	E	X	E	E	EV	E	E	36	36	36	36	EV	E	EV	ES	E				03		
02								E	E	E	E	E	BC	G	BC	BC	BC	BC	E	E	G	E	G	BC	E	E	E	ES	E					02		
01																																			01	

09-01-71

Fuel Code	No. Tubes	Description	PT-NR No.	No. Tubes	Description
E,ES	166	Mk-IC (94 Metal-Fringe)	94	5	Mk-IVAA Demonstration
F	2	Mk-IV (94 Metal-High U-236)	113	4	Evaluation of the End Spider Inner Support System
G	572	Mk-IV (94 Metal-Central & Fringe)			
N	1	Mk-1B (Natural U)			
X	229	Mk-1A & Mk-IV (125 & 94 Metal)	136	8	Mk-IV Non-bonded End Closures
	970	Total	118	16	Mk-IV Performance Test
	34	Total PT's		1	Graphite Sample Channel
	1004	Grand Total		34	Total PT's

Fuel Failure Data:

<u>Tube No.</u>	<u>Charge Date</u>	<u>Exposure (MWD/ton)</u>	<u>% of Goal Exposure</u>	<u>Charge Type</u>	<u>Failure Type</u>
2360	3/17/70	1129	66	94 G	Possible fretter
1051	5/27/50	910	52.7	94 GA	Possible fretter