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**DOUGLAS UNITED NUCLEAR
MONTHLY REPORT**

OCTOBER 1969

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DOUGLAS UNITED NUCLEAR

MONTHLY REPORT

OCTOBER 1969

DOUGLAS UNITED NUCLEAR, INC.

Richland, Washington.

Work performed under Contract No. AT(45-1)-1857 between the
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SUMMARYREACTOR PLANT OPERATIONSProduction Statistics

	<u>KE</u>	<u>KW</u>	<u>N</u>
Input Production - Pu (KMWD)	64.0	74.9	103.7
- U-233 (Equiv. KMWD)	9.4	9.1	-
Time Operated Efficiency - %	57.1	64.3	87.0
Steam Availability to WPPSS - %	-	-	86.0

K Reactors

Maximum power levels of 3920 MW at KE and 3845 MW at KW were achieved, being restricted at KE by the bulk outlet water temperature limit of 95 C and at KW by inlet water piping brittle fracture considerations. Low production resulted from a long scheduled outage at each reactor and two unscheduled outages at KE. The plutonium made was largely fuel grade.

The thorium and supporting 94 Metal fuel were discharged completely from KW and partially from KE, with replacement by natural uranium. Discharge of the remaining thorium loading from KE, scheduled for late November and early December, will complete the DNR U-233 program.

N Reactor

Reactor input production and WPPSS electrical generation (501,610 MWH) were new one-month record highs. On October 20 total generation to date by the N Reactor-WPPSS complex reached 10 billion KWH, making this nuclear facility the first in the world to achieve this milestone.

Reactor power level was administratively limited at 4000 MW. Rod scram tests, primary loop leak repairs, and mandatory standards checks were completed during the one scheduled outage. The two short unscheduled outages both resulted from automatic flow monitor trips.

Steam generators 1A and 1B were decontaminated successfully and their retubing was started.

FUEL AND TARGET FABRICATIONProduction Statistics (Tons)

	<u>For KE & KW</u>	<u>For N</u>
Billets Extruded	-	0
Finished Fuel Produced	226.2	19.2

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K Fuels

AlSi canning operations continued on the basis of two lines per day, five days per week. Construction continued on the hot-die-sizing facilities under project DCE-524.

N Fuels

Fabrication work concentrated on reducing the in-process inventory of fuel assemblies. There were no production fuel extrusions.

TECHNICAL ACTIVITIESK Reactors

Technical bases for tube power limits to be applied for a natural uranium loading have been documented and forwarded to AEC-RL. The bases presented have been used for the development of Process Standards. Satisfactory progress is being made in developing a computer code for automating calculations for tube power limits.

Static code calculations have been made to determine vertical safety rod system worths for cases with one or more safety rods failing to enter.

Tests were conducted at KE Reactor on six pairs of process tubes (one aluminum and one Zircaloy) to determine the most effective chemical formulation, concentration, and temperature for a through-reactor decontamination. The tests showed which agent was most effective, disclosed a problem with rear nozzle concentration of radioactivity, and indicated that the decontaminant would have to be preheated to be effective (reactor decay heat being inadequate).

Laboratory data on the ratios of Pu-236 to Pu-238 in neptunium-graphite wafer samples from the four-tube test block reinforce earlier conclusions that Hanford is capable of producing medical grade Pu-238.

Physics and hydraulics analyses have been completed on proposed oralloy fuel models. Calculations are proceeding on vertical positioning of oralloy columns in the test loading for compatibility with speed-of-control criteria. Temperature transients were analyzed for proposed models to determine the required column lengths.

Three coupon holders, each containing four different types of lithium-bearing spline samples, have been charged into KE Reactor to be irradiated for a full fuel-charge cycle to determine irradiation embrittlement.

N Reactor

A reactor systems study is being conducted to identify problems and quantify solutions for improved operability. A summary of equipment-induced causes of forced shutdowns and other historical data is being prepared. A systems analysis of the flow monitor is being made to develop a detailed approach to

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solution of the problem of the flow monitor system as a source of forced shut-downs other than legitimate high and low flow experiences.

A study report on a plant process surveillance system for N Reactor is in preparation. One of the justifications for the system is replacement of the intended functions of the ISI-609 process computer now located in the reactor building.

Four sets of digital simulations of the High Pressure Injection System were conducted to determine the manual reaction times required to preclude excessive loss in coolant inventory after single equipment failures.

Sixteen proprietary decontaminants produced by four companies are being evaluated in an attempt to find an improved decontaminant for carbon steel piping.

Two additional metal-water reaction tests were performed with irradiated outer fuel elements in the continuing program of fission product release studies.

GENERAL

A post-installation review of the Univac 9300 procurement and startup was made, in accordance with AEC Manual Chapter 1801, to provide useful information to Federal agencies involved in computer procurement and operation. Currently, at a point thirteen months after installation, the Univac 9300 has fulfilled all objectives of production computing and the additional function of data transmission from 100-N Area to the Univac 1108 in the Federal Building.

The structure of the basic education portion of the TOP Program was changed substantially for the current class of trainees. Individual progress appears to be satisfactory under the revised program. Formal individual evaluation will be made in the near future.

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

Charles D. Harrington

Charles D. Harrington
President

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REACTOR PLANT OPERATIONS - KE & KWPRODUCTIONGeneral

Production, power levels, efficiencies and related statistics for the two K reactors are tabulated below. Combined input production and time operated efficiency (TOE) for the past six months are shown on the following chart:

Statistical Summary

	<u>KE</u>	<u>KW</u>	<u>Total</u>
Input Production - Pu (KMWD)	64.0	74.9	138.9
- U-233 (Equiv. MWD)	9.4	9.1	18.5
Power Level (MW) - Maximum	3,920	3,845	7,765
- Average	3,615	3,757	7,372
Time Operated Efficiency - %	57.1	64.3	60.7
Number of Outages	3	1	4
Number of Startup Interruptions	0	0	0
Operating Coolant Flow - M gpm	190.6	193.6	384.2
Fuel Charge (Tons) - 94 Metal	319.1	86.9	406.0
- Natural U	57.9	354.0	411.9
Fuel Element Failures	0	1	1
Helium Losses - M cu. ft.	98.5	76.2	174.7

OPERATING EXPERIENCEReactor Loadings

Front face maps showing the loadings in KE and KW are reproduced on the two pages which follow page B-6. The tonnages listed are approximate; actual fuel charge totals are tabulated above.

The side blanket thorium load and the supporting 94 Metal fuel were discharged from KE Reactor. At KW Reactor, thorium from the core and from the buckled zone, along with support metal, was discharged. Both reactors were reloaded with natural uranium. Discharge of the remaining thorium from KE Reactor, scheduled for November and December, will complete the DNR U-233 program.

Power Levels

The maximum power level at KE Reactor was restricted by the bulk outlet water temperature limit of 95 C. Precautions taken because of inlet water piping brittle fracture considerations restricted the KW level.

Reactor Outages

<u>Date Down</u>	<u>Reactor</u>	<u>Outage Hours</u>	<u>Remarks</u>
October 5	KE	210.0	Scheduled charge-discharge.
October 18	KE	38.7	Correction of a process tube water leak. Aluminum tube 4188 was replaced.
October 19	KW	265.9	Scheduled charge-discharge. Fuel inspection confirmed a suspected 94 Metal element failure in tube 2285.
October 29	KE	70.8	Panellit trip caused by surge in water pressure from the diesel back-up facility. (See Equipment Experience.)

EQUIPMENT EXPERIENCEEmergency Diesel to Process Water Check Valves - KE Reactor

The 24-inch "A" emergency diesel to process water check valve failed to open during a routine functional test on October 11. The valve was removed from the reactor, inspected, and returned to service after evaluating the forces required to open the valve. Reliability tests were run to demonstrate that the valve opened when closed under controlled conditions. These tests were successfully demonstrated.

On October 29, the 182 diesel last-ditch water pumps automatically started during the draining of A-B risers at KW Reactor. Pressure transients on the last-ditch water system created a pressure instability at KE Reactor that

resulted in a scram. The "A" emergency diesel water to process check valve was stuck and required 81 psig delta pressure to open. The "B" emergency diesel water to process check valve required approximately 20 psig delta pressure to open. Both check valves were removed and mechanical stops were installed. The pressure fluctuations which caused the reactor to scram are being investigated.

Boiler Flameouts - 165-KW

Two boiler flameouts were experienced at 165-KW. The first occurred on October 18 during a routine change of the forced-draft fan from turbine to motor drive. Continuity of operation was not affected since normal conditions were restored in nine minutes. The second flameout was experienced on October 30 during a routine burner change. High fuel oil pressure was experienced during the change and subsequent boiler tests could not reproduce a flameout. Consequently, the original problem is believed to have been in the fuel oil regulators. The reactor was down at the time of the second flameout.

Vertical Safety Rods

At KW the VSR pin inspection and replacement program was completed. Four VSR channels were inspected for sleeve damage and all were found to be in good condition.

Horizontal Control Rods

At KE the overall operation of the HCRs continues to be good. Numbers 8 and 12 rods will have to be replaced in the next 1-1/2 to 2 years because of boron burnout.

At KW a slight hump in the graphite track of No. 9 HCR was removed. This relieved a binding problem previously associated with this rod. HCR No. 8 was removed and the channel cleaned. Borescoping indicated the channel to be clear and tests confirmed the rod's operability throughout its full range of travel.

Through-Reactor Decontamination - KE Reactor

In accordance with Production Test Authorization 158, six sets of two tubes each were decontaminated at KE Reactor during the outage begun on October 5. This first-time trial of through-reactor decontamination at a K reactor is discussed under Technical Activities, page D-3.

Resistance Temperature Detectors - KE Reactor

There were nine RTD failures at KE during the month. One of these failures would have resulted in a reactor scram if the technical high limit feature of the High Speed Scanner had not been bypassed. Three of the failures were in tubes subjected to through-reactor decontamination.

During normal random testing at KE Reactor, it was discovered that several immersion-type RTDs indicated slow responses. The reason for this response

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deficiency has not yet been explained, but the slow response units have been identified for maintenance action. The response analysis computer program was modified to accept data from immersion units, and was used to analyze a temperature data tape generated during an imposed scram.

Venturi Changeout - KW Reactor

Approximately 300 venturis were changed. The majority of these were required to adjust flows for a change in process tube loading.

ENGINEERING ASSISTANCE

Thermoluminescent Dosimetry (TLD)

Calibration of the TLD system has been completed. A standard deviation of less than $\pm 7\%$ of the average reported dose from the true dose received is considered to be accurate over a wide range of gamma energies and environmental conditions. Extremity dose studies are presently underway at N Reactor (for workers' feet in the segregation pit) and at the 300 Area (for operators' hands on the hot-die-size line in the 313 Building).

Information meetings were held with all Radiation Monitoring Supervisors, and with Operations Division and Section management, to acquaint these people with the TLD system and its capabilities.

Emergency Planning

Information material for the emergency manual has been revised and widely distributed for comments. Final issuance will be made following receipt and evaluation of the comments. Requests for suggested changes in the emergency procedures were made at the same time so that they can also be revised as soon as practical.

A study has been implemented to determine the severity of the personnel exposure problem that could be associated with an accidental dumping of a cask of irradiated uranium in a transfer area at the K or N reactors. The results of this study will be used to determine if changes in emergency actions and/or procedures, as well as additional safeguards, are indicated.

PROCESS ASSISTANCE AND CONTROL

Operational Physics

The 94 Metal E-Q loading has been completely displaced with a natural uranium loading (except for the top and bottom shield protection material) in the KW Reactor, and the initial startup following the loading transition occurred on October 30. This startup was the first utilizing the brittle fracture tube power limits for the natural uranium loading. A combination of the new limits and excess reactivity of 2 mk lower than anticipated restricted the reactor power level to 3620 MW, about 200 MW lower than initially estimated. The power

level and excess reactivity are related such that if more excess reactivity were available to flatten the power distribution, a higher power could be achieved; and conversely, if a higher power level could be achieved, the excess reactivity would increase, thus providing for better flattening. The excess reactivity in the samarium dip was $0.4\% \frac{\Delta k}{k}$, but since the excess reactivity will increase as long term gains are realized, better flattening will be achieved, making possible a higher power level than at present. Since the fuel exposure is very low following the recent 2000-tube discharge, the graphite temperature coefficient of reactivity is low, and therefore xenon instability is not a problem which requires a higher excess reactivity for control flexibility in the samarium dip.

The KE Reactor has converted only the near and far side blankets to natural uranium, and therefore, the reactor loading is still primarily 94 Metal E-Q material. The spike inventory to be employed for the startup following the transition to a natural uranium loading will be increased based on observed requirements for the KW Reactor startup. Approximately 250 central zone spike columns were used for the KW Reactor, 25 more than have been in any past natural uranium loading.

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

	Reactor	
	KE	KW
Effective Central Tubes (ECT)*	2265	2220
Flattening Efficiency** - October	0.72	0.71
- 12-Month Average	0.72	0.72
Maximum Operating Time Permitting Scram Recovery - Hours***	11	10

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

**ECT divided by the number of power generating tubes.

***The maximum operating time subsequent to a cold startup following which a scram recovery could be made using the currently approved startup procedures.

Production Fuel Performance

The 94 Metal element which failed in tube 2285 of KW Reactor was charged on June 13 and had an exposure of 1267 MWD/T. Failure is attributed to groove corrosion. Examination showed essentially complete loss of two sets of self-supports.

The following table shows production fuel failure frequencies in the K reactors,

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as number per million elements discharged, for the 12- and 24-month periods ended September 20:-

	<u>12 Months</u>	<u>24 Months</u>
Natural uranium	24	5
94 Metal	7	8

Product Accountability Studies

Computer runs have been made to determine conversion ratio and heat generation rates in C2N and O3N fuel elements if utilized in aluminum fringe tubes of the K reactors. Calculated conversion ratio percentage variations relative to K4N are small - of the order of 0-1 percent. For current purposes, such data are being reported as percentage differences rather than on an absolute basis.

Control Rod Strength

Calculations of the boron-10 burnup in the horizontal control rods, coupled with an experimental measurement relating boron-10 density to rod strength, have shown that a total of eleven rods in the two K reactors will reach the 10 percent worth loss point over a one-year period beginning in 1.5 years. The calculational method is the same as that used to evaluate the cadmium burnup in the C Reactor rods during the first few years of operation. The method utilizes the control strength of a rod to determine the fraction of the neutrons captured within the local control region of the rod. The measurement relating control strength to boron-10 density was performed in the Hanford Test Reactor by measuring the worth of a small rod section containing known quantities of boron-10.

Methods under consideration for verifying the calculations and measurements are (1) analyze the B₄C grit extracted from a rod and/or (2) measure the neutron attenuation through a burned-down portion and a full-strength portion of the rod. The former is destructive while the latter is not. The neutron attenuation method would include a comparison to the attenuation through a rod section with a known quantity of boron-10. Verification is planned for early in 1970.

Tube Probologging

Probologging of 20 aluminum process tubes was recently completed at KW Reactor. The results would indicate no immediate concern for tube life. The minimum thickness measurement obtained was 48 mils on tubes 1967 and 4361. Both of these tubes were previously probologged, 1967-KW in October 1966 and 4361-KW in October 1968, and measured 55 mils. Each tube has been in the reactor since October 1962. The remaining tubes ranged in mils remaining from 50 to 55, with the average being 53 mils. An intensive tube probologging program was recently conducted at KE Reactor on some 200 process tubes. The results are expected to be available soon.

High Speed Scanner

The High Speed Scanner Temperature Analysis Program (HSSTAP) has been incorporated into the Equipment Maintenance Standards (EMS) Program.

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<u>Zone</u>	<u>Tons</u>	<u>Material</u>
Central	237	94 Metal (for Thoria Support
	10	Natural Uranium
	19	Thoria ("X" Designates Tubes)
	1	Special Depleted Uranium (PITA-048)
Buckled	65	94 Metal (for Thoria Support)
	6	Thoria ("X" Designates Tubes)
*Blanket	47	Natural Uranium (Side)
	23	94 Metal (for Thoria Support)
	4	Thoria (Top and Bottom)

*Side Blanket thoria and thoria support metal have been discharged.

Loading Pattern - KE Reactor

B-A

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<u>Zone</u>	<u>Tons</u>	<u>Material</u>
Central	222	Natural Uranium
	64	94 Metal
	1	Special Depleted Uranium (PITA-048)
Buckled	82	Natural Uranium
Near and Far Fringe	47	Natural Uranium
Shield Protection	22	94 Metal (Thoria Support)
	5	Thoria

Loading Pattern - KW Reactor

B-B

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REACTOR PLANT OPERATIONS - NPRODUCTIONGeneral

Reactor production (all 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:

The input production and electrical generation totals are record one-month highs, both exceeding the previous records established in April 1969. On October 20 the N Reactor-WPPSS complex produced its 10 billionth kilowatt-hour of electrical power, thereby becoming the first reactor in the world to achieve this milestone.

Statistical Summary

Input Production - Pu (KMWD)	103.67
Electrical Generation (KMW) - WPPSS	501.61
- 184-N	6.61
Power Level (MW) - Maximum	4,000
- Average	3,841
Time Operated Efficiency - %	87.0
Steam Availability - %	86.0
Number of Shutdowns - Scheduled	1
- Unscheduled	2

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Statistical Summary (cont'd)

Fuel Failures	0
Fuel Charge (Tons) - 94 Metal	309.85
- 125 Metal	67.18
- Natural U	<u>0.37</u>
Total	377.40
Helium Losses - M cu. ft.	224.52
Fuel Oil Usage - bbl.	13,365

OPERATING EXPERIENCE

Reactor Loading

The reactor loading at month end is shown on the front face map which follows page BN-6. There was no fuel charge-discharge.

Power Level

Power level continued to be administratively limited at 4000 MW. It was possible to maintain main steam header pressure (MSHP) at approximately 123 psig, thus maximizing the electrical generation capability of WPPSS at the station's design power of 800 MWe.

Reactor Outages

The three reactor outages and their principal causes were as follows:

<u>Date Down</u>	<u>Outage Hours</u>	<u>Cause</u>
October 7	89.9	Scheduled outage for horizontal rod test and maintenance. Primary loop leak repair was included in maintenance effort.
October 11	5.4	Unscheduled automatic scram from a flow monitor trip due to a leaky toggle valve.
October 13	1.9	Unscheduled automatic scram from a flow monitor trip. The cause of the scram was not positively identified, but occurred during an adjustment of the open diode scanner cross-bar voltage and is believed due to an intermittent fault in the cross-bar switch.

EQUIPMENT EXPERIENCE

Primary Loop

Retubing of steam generators 1A and 1B started on October 20, following the successful decontamination of these units.

Twenty-one MK-IIIA manual operators were installed on V-11 valves, one new V-11 valve with manual operator was installed on tube 0866, and three front connector valves were repacked and manual operators installed. Also, thirteen rear connector diversion (V-12) valves were installed and a number of other leaks were repaired within Zone I. As the leaking V-12 valve stem packings cannot be retightened, a new type of metal packing is being developed as a replacement. A prototype has been fabricated and is being tested. The tests are showing that the bellows-spring-piston concept is good, and that a rotary O-ring seal is practical. If these seals prove to be as successful as the tests indicate to date, they should contribute substantially to the reduction of primary loop leakage.

Secondary System

High chloride content continued to be a problem in the secondary system, and greater than normal feed and bleed were required during the month to control the chloride level. Selected surface condensers are scheduled for leak testing during the November outage, with the 184-N turbine-generator surface condenser being the primary suspect.

Boiler Experience

Tube leaks were repaired on the No. 1 Combustion Engineering (CE) boiler and on the Foster Wheeler boiler. The CE-1 and CE-2 boilers carried the steam load requirements during the Foster Wheeler boiler outage, thus avoiding an unscheduled reactor outage.

Production Test NR-116, "Hot Unfired Standby Test," was initiated on October 15 and continued through month end.

Horizontal Control Rods

Testing of the horizontal control rod (HCR) system was continued under PT-NR-121, "Fire Resistant Hydraulic Fluid Surveillance," when the reactor was shut down for rod testing on October 7 (see also Technical Activities, page DN-7). The five "slower than normal" rods continued in this status, but were no worse than on previous tests. Only HCR-17 was slower than specified by the Process Standards. Scram times for the five rods were:

<u>Rod No.</u>	<u>Scram Time</u>
17	2.94 seconds
52	1.25 "
94	1.37 "
96	1.37 "
97	1.11 "

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Rods 34 and 63 were out of service at month end due to a coolant leak in the former, and failure of the latter rod to operate during scram recovery. Rod 63 was manually pulled and removed from service.

Rupture Monitor System

The No. 3 gross gamma rupture monitor turret went out of service on October 23. The problem was traced to a grounded wire on a circuit board for process tube sample 1641. The ground was caused by a combination of moisture and defective insulation on a 700-volt lead. Investigation revealed EMP-411, Addendum 2, had been improperly installed such that there was no warning on loss of the 700-volt power to the turret. Correction is scheduled for the November outage.

Equipment Modifications

The following equipment modifications were completed during the month:

DC-3016, "Bracing for N Area Battery Racks," added sufficient bracing and anchor bolts to enable the racks to withstand a design basis earthquake.

DC-3023, "CRWP Motor Underpower Relay Installation," added underpower relays to each CRWP motor to protect these motors from severe transient fault conditions that could originate outside the AEC transmission system.

MDC-N-69-17, "1310-N Sump Pump/Recirculation Pump Interlock," provided an interlock to prevent both pumps from running at the same time.

MDC-N-69-44, "Irradiated Fuel Cart Control Modification," provided a control circuit at the discharge chutes which will allow limited movement of the fuel carts from the discharge chute station.

MDC-N-69-46, "Process Elevator Bypass Valve Circuit Breaker Isolation," provided power from separate breakers for each elevator's bypass valve.

PROCESS ASSISTANCE AND CONTROL

Operational Physics

The modified inboard rod withdrawal sequence and pattern has been very successful in providing flattening through the power ascension and at equilibrium power. Experience with this distribution has shown that the assumed maximum peak-to-average ratio in the axial flux distribution is currently more conservative than is necessary, and appropriate action will be taken to relax the limits somewhat if the current trend continues.

Calculations show that the void flooding accident for the Mark IV lattice is more limiting than the cold water injection accident, and therefore this effect is now included in the total control calculations. It is anticipated

that total control requirements for the void flooding accident in the Mark IV lattice will be somewhat more limiting during the power ascensions than the current cold water injection requirements, but no significant increase in non-equilibrium losses are anticipated as a result.

Rod insertion time tests run on individual rods following the recent hydraulic fluid change showed that the insertion times of four rods were slightly longer than normal but still within Process Standards requirements. Calculations of total control requirements at equilibrium power showed that these rods could be locked out of service within total control requirements. These rods were locked out to prevent their insertion as a result of a spurious scram so that individual timing tests could be performed following a shutdown.

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

Effective Central Tubes (ECT)	807
Flattening Efficiency - October	0.82
- 12-Month Average	0.82
Maximum Operating Time Permitting Scram Recovery - Hours	24

Failed Fuel Examination

Metallographic examination of fuel failures No. 51 and No. 52 at the Radio-metallurgy Laboratory, BNW, has confirmed that wire fretting caused both. Operating conditions for the fuel columns involved were as follows:

	<u>Failure No. 51</u>	<u>Failure No. 52 (PT-96)</u>
Tube No.	1167	2060
Fuel Identification	CG005 (outer) TZ09-15 (inner)	CB20-10 (outer) TT98-08 (inner)
Position in Column	4	1
Fuel Model; Loading	MK IV; 16 pc. at 26.1"	MK I; 18 pc. at 23.2"
Date Charged	5/8/69	11/20/68
Date Failed	6/30/69	7/7/69
Tube Exposure; Goal	215 MWD/T; 2430 MWD/T	1530 MWD/T; 2200 MWD/T
Tube Power	4460 KW	4410 KW

The point of failure on No. 51 was located approximately four inches from the numbered end on the inside surface of the outer element. This location is adjacent to a solid support on the inner element. Indications of fretting were

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also observed at two other areas adjacent to the point of failure.

The point of failure on No. 52 was located at the center of the inner surface of the outer element, 120 degrees from the locking clip location. Adjacent to the point of failure there were indications that a buggy spring support had been in contact with the clad surface of the outer element. All three of the center buggy spring supports were still intact. A summary document of the failure examinations is being prepared.

Control Rod Coolant Tubing

Metallographic examination of the pits observed in the aluminum seal section tubing removed without chemical cleaning from control rods 34 and 46 showed that the worst pits extended 2/3 of the way through the 65-mil wall. Experience with these tubes clearly indicates that the pitting is unrelated to chemical cleaning.

Reactor System Operability Study

An operability study is being conducted for the purpose of identifying problems and quantifying solutions for improved operability. Currently being prepared is a status report which summarizes equipment-induced causes of forced shutdowns and other historical data. Also included is a rating of 34 pertinent N Reactor systems by "scram potential", considering complexity, maintenance history, access to control poison, and forced shutdown history.

Samarium Oxide Burnout

A sample of samarium oxide balls believed to include balls present in the reactor for ten full-power days were measured against the standard NPR ball sample in the Hanford Test Reactor. The measured worths were identical, which is as predicted by DUN-5740, "Reactivity Worth of Irradiated Ball Columns in N Reactor."

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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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Fuel Code	No. Tubes	Description	PT-NR No.	No. Tubes	Description
C	69	Mk-IC (94 Metal - Fringe)	76	4	Fuel Meltdown Test
CE	61	Mk-IC (94 Metal - Fringe)	94*	7	Mk-IV Demonstration
D	162	Mk-IC (94 Metal - High U-236)	96	15	Mk-IC From Upset-Forged Billets
E	242	Mk-IC (94 Metal - Fringe & Central)	01	12	Mk-IC From Direct-Cast Billets
ES	2	Mk-IC (94 Metal - Fringe)	01	2	Mk-IV From Direct-Cast Billets
F	117	Mk-IV (94 Metal - High U-236)	07	47	Initial Full Length Mk-IV Columns
G	21	Mk-IV (94 Metal - Central)		1	Graphite Samples Channel
N	1	Mk-IB (Natural U)			
X	241	Mk-IA (125 - 94 Metal)			
	916	Total		88	Total PTs
	88	Total PTs			
	1004	Grand Total			

*Includes Mk-IV high U-236 content fuel, 1 tube with Mk-IA 125-94 Metal and Mk-IV 94 Metal, and 2 tubes with Mk-IV-AA 125 Metal and Mk-IV 94 Metal.

Loading Pattern - N Reactor

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FUEL AND TARGET FABRICATION - K REACTORS

PRODUCTION

General

Production of AlSi-bonded fuel for the K reactors was 103.3 percent of forecast. Forty-nine percent of these elements had self-supports or bumpers attached. Project DCE-524 construction work continued on the hot-die-sizing facilities in Bldg. 313.

Acceptable Elements Produced

	<u>Tons</u>	<u>Yield - Percent</u>	
		<u>Current Month</u>	<u>FYTD</u>
AlSi-Bonded Fuel	226.2	96.5	95.3

Month-End Inventories

	<u>Tons</u>
Bare Uranium Cores	1003*
Finished Fuel: AlSi-Bonded	1079*
Hot-Die-Sized	39*
Thoria Elements	19.5

*These totals include 124 tons of cores and 129 tons of finished fuel made for the smaller reactors.

OPERATING EXPERIENCE

Canning line efficiency of the AlSi-bonding lines was 98.7 percent. Downtime was assigned 69 percent to equipment malfunctions and 31 percent to operations. AlSi canning operations continued on the basis of two lines on the day shift, five days per week.

EQUIPMENT EXPERIENCE

Hot-Die-Size End Bonding Coil

A new double-layered end bonding coil was designed to increase the inside surface area, to reduce the air gap and to lower the coil resistance in an effort to reduce the heat loss in the coil. Greater length-to-diameter ratio was incorporated into the design resulting in a more effective coupling. The material used was heavy rectangular cross section tubing with a 0.093 inch wall, hence winding these double layer coils was a difficult task; it was accomplished by the development of special winding fixtures and tooling as shown in

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Figure C-1 (on page C-A). The final assembly before insulation was applied is shown in Figure C-2 (page C-A).

Using thermocoupled fuels, the test coil provided faster heating rates by raising the end bond cup from room temperature to 550 C in 30 seconds as compared to the 60 seconds required by the coils currently in use. It was found, however, that the net heat delivered by the test coil to the U-Ni-Al interface was less because the holding or soaking heat was not sufficient to bring the interface to temperature until late in the bonding cycle. This fact prevents the use of this coil for end bonding of hot-die-sized fuels.

PROCESS ASSISTANCE AND CONTROL

Two samples of extruded cores for the long fuel hot-die-size program were examined for metallurgical characteristics according to the established uranium quality control procedures. The micro structure was inspected for carbide and nitride size and distribution. A macro photo was checked for non-metallic stringer orientation and grain size. The only variation observed from what has been the norm in regular production uranium was the grain size. These two cores had slightly larger grain size than regular production uranium but were within the maximum allowable specification.

Figure C-1. Double-Layer End Bonding Coil Winding Fixture.

Figure C-2. Double-Layer End Bonding Test Coil.

FUEL AND TARGET FABRICATION - N REACTORPRODUCTIONInput Production

Total billets extruded	0 (as forecasted)
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Output Production

Total finished fuel assemblies	751
Tons output	19.2
Percent of forecast	120.1

<u>Uranium Utilization - %</u>	-
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Month-End Inventories - Tons

Bare uranium billets	264
Finished Fuel	364

OPERATING EXPERIENCE

Coextrusion production commitments were officially reduced, effective October 1. Due to the inherent inertia of the process, a surplus in in-process inventory resulted. Consequently, to balance output, non-production extrusions (e.g., those for Sandvik Special Metals and Atomics International) were scheduled for the entire month. The in-process inventory of virgin material and material to be recut will be reduced to a level compatible with present production rate.

EQUIPMENT EXPERIENCE

Nothing significant to report.

PROCESS ASSISTANCE AND CONTROLConversion of Uranium Chips and Fines to Oxide

Three processes are being studied in an effort to replace the uranium oxide burner which is used to convert uranium chips and fines to an oxide.

Dissolution of chips and fines in nitric acid is successful for recovering uranium; however, the process is very slow and large amounts of nitrous fumes are given off. Hence this process is not completely desirable because of atmospheric pollution considerations.

Dissolution of chips and fines in hydrochloric acid has an advantage over

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nitric acid in that only hydrogen is released during the process and the production of toxic fumes is eliminated. The only concern with this process is the chloride content in the resulting sludge, since NLO requires that total chloride content in the product not exceed 1/2 w/o. The uranium is readily dissolved and it appears that this process has good potential.

Oxidation of chips and fines under water in a pressure vessel at 200-400 C is also being investigated. Uranium oxidizes in this medium, releasing hydrogen. The primary problem area is whether the process can adequately be controlled when large masses of uranium are being oxidized. An initial test at 200 C with five pounds of uranium was easily controlled with respect to temperature and pressure. Presumably when the hydrogen release has ceased, all metal has been oxidized and the process is complete. If this assumption is correct, the hydrogen release (which can easily be monitored) could serve as a process control.

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TECHNICAL ACTIVITIES - K REACTORS

RESEARCH AND DEVELOPMENT

Basic Production

Brittle Fracture Program - K Inlet Piping

Technical Bases for K Reactor Tube Power Limits

Document DUN-6289 REV 1, "Initial Technical Bases for Tube Power Limits for the K Reactors - Natural Loading," has been issued and information copies forwarded to AEC-RL. The bases presented in the document have been applied for development of Process Standards for operation of the K reactors with natural uranium loadings.

Methods described in Supplement V of the K Reactor Hazards Summary Report were applied to develop bases for power limits for protection against coolant loss from failure of two adjacent supply pipes, with initial six-pump operation and 405 psig top-of-riser pressure.

In their present form, the bases present technical limits for reactor power level over the range of inlet temperatures between 11.8 and 22.6 C, and will apply conservatively for lower inlet temperatures, as illustrated in Figure D-1 (next page). Each power level limit is coordinated with a set of tube power limits; one for each of the basic fuel types in the natural loading (enriched K5E and K4E, and natural K5N and K4N, for Zircaloy and ribbed aluminum tubes, respectively).

Basic tube power limits were calculated for an axial flux reading of 15 percent above theoretical chopped cosine distributions. The 15 percent peaking is typical of values that exist during normal operation but is often exceeded during startup conditions. Consequently, correction factors were developed to adjust bulk reactor and tube power limits for axial peaking up to 37.5 percent, which is well above the maximum expected amount. The effect of peaking on tube power limits is illustrated in Figure D-1.

To insure availability of power limits when required, the initial limits are based on a limited range of reactor operating parameters. Reactor inlet temperature and axial peaking were the only process parameters considered as variables. All others were taken at their limiting (most restrictive) values. Work is in progress to expand the initial bases to encompass the potential full range of process variables for the natural uranium loading. The resulting system of limits will provide for greater flexibility of operation within the constraints associated with minimizing the potential consequences of a dual supply piping failure.

Satisfactory progress is being made in developing a computer code for automating calculations for tube power limits. It is expected that by early November a subroutine will be available to calculate voiding transients for the first reactivity feedback calculation in the iterative

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Figure D-1. Variation of Tube Power Limit with Inlet Temperature for K5E Fuel. Peaking Factors are Percentage Increases above a Normal Chopped Cosine.

procedure used for determination of the post-accident power transient for a given case. The program for determining subsequent voiding patterns is expected to be completed by the first of December.

BUCK static code calculations have been performed to determine VSR system worths for cases with one or more safety rods missing. Due to central shadowing in both the side-to-side and front-to-rear directions, the strongest effect is calculated with a corner rod missing, and the system strength is calculated to change from 32 to 26 mk in that case. The effect of an adjacent VSR in the next bank is greater than that of a second nonadjacent rod in the outer bank. The effective system strength was calculated to be 24 mk with the second missing rod to be the adjacent one in the next bank. The case of the second missing rod in the outer bank adjacent to the missing corner rod is yet to be calculated. The most limiting two-missing-rod cases calculated to date have indicated a difference of about 50 C in the fuel heating transient subsequent to voiding.

Fatigue Tests

One additional fatigue test was completed on KE Reactor A riser base metal with a mean stress level of 4000 psi to determine the effects of residual stress on fatigue crack growth rates. Results are not yet available. An additional test is being run at a mean stress level of 10,000 psi.

Through-Reactor Decontamination

A series of tests was conducted (under PTA-158) at KE Reactor on October 7 to determine the most effective decontamination formulation, concentration and temperature for conducting a through-reactor decontamination. Six pairs of tubes were decontaminated by injecting preheated decontamination solutions (inhibited organic acids) into the front nozzle at a flow rate of 1.5 gpm. Each tube pair consisted of an aluminum tube and a Zircaloy-2 tube. Measurements of the decontamination effectiveness were obtained from radiation levels of the downstream spacers before and after the decontamination, continuous radiation measurements obtained from a shielded detector attached to each decontaminated tube outlet nozzle, liquid samples periodically taken from the outlet nozzle during the decontamination, and visual inspection of the spacers and discharged fuel elements at the completion of the tests.

The tests showed that Turco 4306-C at a concentration of 6 oz/gal and a temperature of 90 C provides the most effective decontamination. Turco 4528-F and Wyandotte 5061 under the same conditions provide almost as effective decontamination. Tests with Turco 4528-F at 6 oz/gal and 50 C and at 3 oz/gal and 90 C resulted in poor decontamination. A test at 75 C with 3 oz/gal each Turco 4306-C and Turco 4528-F, a mixture which appeared in laboratory tests to be more effective than either formulation above, resulted in a marginal decontamination.

One potential problem area was found during the single-tube tests which will require further evaluation. During all six tests, the downstream nozzle area of each tube became very radioactive during the water flush following

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the decontamination. This radioactivity was rapidly flushed away from the nozzles on the aluminum tubes, but persisted in the nozzles of the Zircaloy tubes until the nozzle cap was removed. It is possible that the flow pattern in the Zircaloy tube nozzles is slightly different than in the aluminum tube nozzles, and this could cause holdup of particulate matter removed during the post-decontamination flush. The nozzle cap used on the Zr-2 tubes during the test, however, had a slight leak, and it is more likely that particulate matter collected at the leak area. Further evaluation is required to determine positively why the Zircaloy tube nozzle radiation levels remained high following the flush.

If reactor decay heat is used as the source of heat to raise the decontamination solution from 15 C to 90 C, it is calculated that a temperature of 75 C will not be reached until the solution has passed over 26 elements in a column of 38 elements. Thus, only the downstream 12 fuel elements would be decontaminated. For other column lengths, the point at which the solution reaches 75 C will be approximately proportional to the above figures. It should be remembered that this calculation is for a tube with a 90 C outlet temperature and very few tubes will be at this temperature.

In addition to the poor front-to-rear thermal distribution, tests conducted on a complete crossheader (DUN-6245) show that a large side-to-side tube outlet temperature variation can be expected. For example, these tests showed that, with several tubes on the crossheader operating at a maximum ΔT of 21 C utilizing reactor decay heat, a full 27 percent of the remaining tubes achieved only 75-85 percent of this ΔT and another 14 percent achieved a ΔT of even lower percentage of maximum. Assuming the same distribution occurs for a full-reactor decontamination employing decay heat and a maximum outlet temperature of 90 C, then 27 percent of the tubes can be expected to reach a temperature of 71 to 79 C while another 14 percent of the tubes will be at less than 71 C. Thus, a total of 41 percent of the tubes would be expected to exhibit either a poor or a marginal decontamination. These results will be checked in a second flow test as soon as it can be scheduled.

The above examples indicate there may be too large a fraction of the total reactor fuel columns not obtaining an adequate decontamination to justify additional work on the concept of utilizing reactor decay heat to heat the decontaminating solutions. Studies have been started to investigate the feasibility of using preheated solutions for through-reactor decontamination.

Evaluation of Process Tube Flattening

PTA-191, "Examination of K Reactor Tube Flattening," was written and approved during the month. The intent of this test is to determine whether the observed process tube flattening, caused by shrinkage of some trunnion blocks onto the tube, is continuing and at what rate. The test specifies measurements of Zircaloy and aluminum tube diameters, dimensions of graphite blocks in tube channels, and a test hole charging of test fuel elements to assess current fuel support damage. To date measurements have been completed on: (1) Al tubes 4564 and 4570-KE, (2) test hole 3B-KE, and (3) empty tube channel 3066-KE.

Measurement equipment functioned perfectly during the test, and data are presently being analyzed.

Product Flexibility

Pu-238

Neptunium Irradiation (PTA-163)

The ratios of Pu-236 to Pu-238 reported by the BNW Analytical Laboratory on two neptunium-graphite wafer samples from the four-tube block are 0.28 ppm $\pm$ 0.07 and 0.28 ppm $\pm$ 0.05 for the aluminum-clad and zirconium-clad elements, respectively. Counting was continuing on two additional wafers. Long counting times are required to establish statistical accuracy due to the very low level of detection desired relative to background activity. These data reinforce earlier conclusions regarding Hanford capabilities for producing medical grade Pu-238 from neptunium irradiation, and indicate some additional potential from irradiation in blocks. The two columns containing neptunium-aluminum cores were discharged late in the report period. Pu-236 data on the latter material will permit more precise assessment of block-irradiation benefits.

U-236 and Np-237 Buildup Studies

The americium target column in tube 1678-KE, charged in May 1969 under PTA-171, is continuing to operate satisfactorily, with discharge expected after approximately one year of irradiation. Preliminary discussions on post-irradiation analyses indicate the required separation of americium from curium to be more complex than for natural uranium fuels, thus involving higher sampling costs.

Additional work on the GRAPH code, to be used in plotting earlier high U-236 test data, has achieved sufficient debugging of the error-bar subroutine to permit its use in analyzing the comprehensive data in that study.

Plutonium Burning Irradiation (PTA-150)

A total of 54 columns in the buffer region of the 225-tube block containing the 10 kg PuAl test have been replaced with natural uranium. Sixteen columns contained thorium, and 38 contained supporting 94 Metal. These changes have resulted in the desired power increase in the PuAl itself of approximately 20 percent to an average of about 1200 kw column power.

Oralloy Utilization

Physics and hydraulics analyses have been completed on oralloy fuel having C4T and K6 dimensions (C Reactor solid thorium elements and K Reactor 210 Metal I&E elements, respectively). With 3.75 wt% uranium used in the solid elements and 4.25 wt% in the I&E elements, approximately 15 columns would be available from the 10 kg expected to be used for the solid design and seven columns from the five kg expected to be used for the I&E design.

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Calculations are proceeding on vertical positioning of oralloy columns in the test loading for compatibility with speed-of-control criteria.

Temperature transients were analyzed for fuel models proposed for the initial oralloy Production Test to determine the required column lengths. To avoid complications that might be involved in relation to tube power limits for the base loading, the criterion chosen for column length was that process tubes charged with oralloy elements would not void during the transient that would result from loss of cooling to dual inlet risers.

Temperature transients were calculated for both the I&E and solid model oralloy fuel columns. The study utilized a revised HEAT program to calculate the transient expected for the oralloy fuel, using the flow decay and void-reactivity power transient developed for the base loading under dual riser failure conditions. The oralloy columns had been designed to operate at the same power as an equivalent length K5N column. Thus, pre-accident K5N operating column powers were used as the starting conditions for the oralloy columns. A series of computer runs were made to find the column length such that the non-voiding tube power limit was no less than the expected maximum column power. This non-voiding column was found to consist of 41 pieces for the K6 model and 30 pieces for the C4T model. These column designs are being utilized in preparation of the Production Test.

Lithium-Bearing Splines

Three coupon holders, each containing four different types of spline samples, have been charged into tube 2071-KE. They will be irradiated for the full fuel-charge cycle and then removed and examined for irradiation embrittlement. The four different types of splines contain LiAl alloy, LiMg alloy dispersed in aluminum matrix, Li_2CO_3 crystals dispersed in aluminum matrix, and LiF crystals dispersed in aluminum matrix. The holders are in the first, third, and fifth positions so that flux levels will vary.

Environmental and Regulatory Technology

Metal-Metal Reactions

One of the necessary quantities for the determination of heats of reaction of uranium and aluminum at elevated temperatures is the heat capacity of UAl_4 . Since there are no published values of this, an expression has been developed from uranium-aluminum heatup data received from Dr. P. Chiotti of the Ames Laboratory, Ames, Iowa. The expression, assumed to be linear, is:

$$C_p = 24.90 + 0.0111 T \quad (T \text{ in } ^\circ\text{C})$$

where C_p is heat capacity in cal/mole/ $^\circ\text{C}$. This is most accurate over the temperature range 100 C - 600 C, but may be assumed to extend down to the base temperature of 25 C and up to the vicinity of the UAl_4 melting point of 730 C. Figure D-2 (next page) shows the comparative values of the heat capacity of the compound UAl_4 , and the sum of the heat capacity of uranium and four times the heat capacity of aluminum. It is actually the difference between the heat capacity of the compound and the additive heat capacity

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Figure D-2. Comparison of Heat Capacity of UAl_4 and Additive Heat Capacities of Uranium and Aluminum which Form UAl_4 .

that is important in calculating the heat of formation of UAl_4 at higher temperatures.

Waste Management

Nothing to report.

ENGINEERING AND TECHNOLOGY - REACTORS

Improved Hand and Shoe Monitor

The current model hand and shoe monitor, used at Hanford the last 10 years for personnel monitoring, is becoming obsolete from a function, circuit, and hardware standpoint. There is a need for equipment that is more adaptable to performing within the rather broad fluctuations that occur in the radiation background level in which the monitors are located. This need led to a preliminary investigation of a personnel monitor developed and used at the National Reactor Testing Station. This monitor was designed to operate in locations where the background changes are frequently quite similar to those encountered with the environmental conditions existing at Hanford; accordingly, it was decided to further investigate this method of background compensation.

A complete prototype monitor system was then built for field experience. This instrumentation was installed in the 105-N Building on October 22. The prototype instrument contains equipment to indicate the contamination status of the person being monitored. There are lights over each readout section; right hand, left hand, and shoes. If the test shows no contamination, all lights come on at the end of the test showing normal conditions. If a person is contaminated, the normal lights stay off and a light indicating position of contamination comes on. In addition, an audible warbling tone is generated that notifies the monitored person of the contaminated condition.

Development of a Computerized Program for Plant Data Analyses

The analysis of critical equipment reliability and performance on a continuing basis is essential to assuring safe nuclear reactor plant operation. The basic reliability and performance data for critical plant protective and safeguard systems are provided by the Equipment Maintenance Standards (EMS) program. This, to be useful, must be analyzed by a statistical approach.

In order to efficiently retrieve and analyze the EMS data routinely, automatic data processing (ADP) methods are used. By using the ADP method, it is possible to expand the program flexibility and usefulness and thereby provide the desired information on equipment performance and performance trends for necessary operations action.

The data analysis program has been operational on a manual data input basis for approximately two years awaiting the completion of an automated EMS Data File. The Data File, which contains all of the historical equipment data generated by the EMS program over the past three years is now essentially complete. This Data File is being updated at monthly intervals to include the

results of the latest equipment tests and inspection. With the completion of this EMS Data File, the data analysis program is being modified to remove the manual data input limitations. These modifications are designed to permit computer retrieval of the historical data on any equipment item in the EMS Data File, thus providing an automated surveillance of critical equipment reliability and performance levels.

Project Engineering

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

ENGINEERING AND TECHNOLOGY - FUELS & TARGETS

Removal of Nickel from Hot-Die-Size Cores

Extensive process development work continues in the effort to optimize a process of nickel removal from hot-die-size cores, and to minimize the associated air pollution problem. A new weak-acid system has been tested and found to be highly efficient in removing the nickel from the uranium without increasing uranium loss and without causing serious air pollution. The process reaction requires about 20 minutes to remove one-half mil of nickel from uranium and removes less than one mil of uranium per hour. The solution is effective at 40 C and can be handled in stainless steel or PVC tanks. Large-scale tests are being conducted to establish firm reaction rates and solution life.

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TECHNICAL ACTIVITIES - N REACTORRESEARCH AND DEVELOPMENTBasic ProductionCalculation of Control Rod Effects on Flux Distribution

The full N Reactor case has been set up on the FLARE code which is operating sufficiently for running trial cases. The first practical use of the code, following adequate debugging and normalization, is expected to be the generation of rod-worth data and sequential calculations.

Process Surveillance Systems Study

The systems study report, "Plant Process Surveillance System for N Reactor," is being prepared for internal review by November 1. One of the justifications for the system is replacement of the intended functions of the ISI-609 process computer now located in the N Reactor building. Table DN-1 (next page) shows a comparison of performance characteristics. Also, the accompanying Figure DN-1 gives a comparison of dependabilities in the form of availability and reliability. The improvements shown are made possible by the advances which have been made in process computer design and application technologies since the ISI-609 process computer was installed.

Primary Coolant Control System Simulations

Four sets of digital simulations of the High Pressure Injection System were conducted during this report period. The purpose of these simulations is to determine the manual reaction times required to preclude excessive loss in coolant inventory after single equipment failures. A summary showing the status of these simulations is given in Table DN-2, page DN-4.

Flow Monitor System

A systems analysis of the flow monitor is being conducted. The objective of the study is to develop a detailed approach to solution of the problem of the flow monitor system as a source of forced shutdowns other than those due to high and low flows. Table DN-3 (page DN-5) gives a summary of forced shutdowns due to the system for calendar years 1966 through 1968 and for calendar year 1969 to October 17. Definite approaches to solutions have not yet been developed. However, candidate solutions include simplification of reactor startup procedures and automated flow checks and trip span checks.

Proprietary Decontaminant Evaluation

In an attempt to find an improved (increased effectiveness, reduced corrosion) decontaminant for periodic N Reactor carbon steel piping decontaminations, a program was started early in 1969 to screen proprietary decontaminants proposed by companies prominent in the field. The screening process consists of a laboratory beaker corrosion test and an out-of-reactor loop effectiveness

Table DN-1. ISI-609 vs. Replacement System Performance Characteristics

<u>CHARACTERISTIC</u>	<u>ISI-609 SYSTEM</u>	<u>REPLACEMENT SYSTEM</u>	<u>FACTOR OF IMPROVEMENT</u>
Instruction Repertoire	64	100	1.56
Addition Time	720 micro-sec.	3.5 micro-sec.	200
Multiplying Time	28.8 milli-sec.	0.0105 milli-sec.	2700
Word Length	39 bits + parity	24 bits + parity + M.P.	M.P.
Core Memory	8,192 x 39 = 319,488 bits	32,728 x 24 = 985,472 bits	2.45
Bulk Memory	None	600,000 words	
Cycle Time	720 micro-sec.	1.75 micro-sec.	400
Input Rate	6 pts/sec.	150-200 pts/sec.	25-33.3
High Speed Direct Memory Access	No	Yes	Yes
No. of Priority Interrupts	15 max.	58 max.	3.87

Figure DN-1. Dependability Comparison - ISI-609 vs. Replacement System

Table DN-2. Summary of Primary Coolant Control Simulations
Response to a Full Scram.

SIMULATION	CONTROL MODE SWITCHING DELAY (SEC. AFTER SCRAM)	FAILURE CONDITION	INITIAL CONDITIONS	POST-SCRAM HPIP CONTROL MODE	STATUS
All Runs			Pressurizer Level = 21 ft. Pressurizer Press. = 1600 psi		
1.		None	2 HPIPs on Auto	4 on Auto	Complete (DUN-5954)
2.		Loss of Injection Pump Control	2 HPIPs on Standby	3 Auto, 1 Failed	
3.			Injection Flows of 700, 1200, 1700, & 2200 gpm	2 Auto, 2 Failed	
4.			for each simulation	1 Auto, 3 Failed	
	Hand to Auto	Loss of One Power Source	1 HPIP on Auto	1 on Auto	Simulation Complete Not Documented
5.	20		1 HPIP on Hand (set at 800 gpm)	1 on Hand until switched by Operator	
6.	40		2 HPIP on Standby	2 Failed	
7.	60		Injection Flow = 1800 gpm		
8.	80				
	Auto to Hand	Loss of Automatic HPIP Control. Reactor Scram at Pressurizer Level = 14 ft.	Same as 5 thru 8 Above	1 on Hand	Simulation Complete Not Documented
9.	10			3 Failed until switched to Hand by Operator	
10.	20				
11.	30				
12.	40				

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Table DN-3. Flow Monitor Outage Cause Summary.

Period Covered and Approximate Scram Power Level

	CY-1969 to 10-17-69						CY-1968						CY-1967						CY-1966					
	100	300	500	2000	3000	4000	100	300	500	2000	3000	4000	100	300	500	2000	3000	4000	100	300	500	2000	3000	4000
<u>Flow Monitor Outage Causes</u>																								
I. Equipment Failures																								
a) Toggle valve leaks	2				2		2											1	1					
b) Sensing line breaks						1												1					1	
c) Self-check circuit																							2	
d) Power Supply											1													
e) By-pass jack											1													
f) Transducer							1																	
II. Operating Errors																								
a) Wrong set point					1																			
b) Wrong valving						1								1										
c) Bumped toggle valve																								
d) Ref. Pt. leads to board disconnected	1																							
e) Lost discriminator bias					1																			
III. Actual High & Low Flows																								
a) V-11 trouble									1		1			1										
b) High flow trip	1																							
c) Low flow trip			1		2	1																		
d) Cocked dummy									1		1				1			1						
e) Unknown																								
IV. Miscalibrations												1								1				

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test of each of the proposed decontaminants. Any material appearing promising, compared to the present decontaminants, is then evaluated more fully by laboratory, component, and in-reactor testing. Sixteen decontaminants from four companies are now being evaluated.

Product Flexibility

Pu-238

Arrangements have been made with ARHCO personnel for individual batch dissolution and sampling of PT-NR-95 material which contained higher than normal initial U-236. The data from this test will provide buildup and burnup data of direct production import, as well as neptunium and U-236 data relevant to fuel recycling and Pu-238 production potential.

Environmental and Regulatory Technology

Metal-Water Reaction and Fission Product Release Studies

Two additional metal-water reaction tests were performed with irradiated N Reactor outer fuel elements. These tests, numbers SNH-10 and SNH-11, utilized fuels irradiated to about 2200 MWD/T and cooled about 1.5 years. The fuels were inductively heated in a flowing steam atmosphere inside a 2.71-inch diameter quartz tube (simulating the reactor process tube), and split Zircaloy tubes were used to restrain the end caps as would adjacent fuel pieces.

Fuel clad failure occurred at 1035 C and 1040 C, respectively, in the range experienced on previous tests.

In the SNH-10 test, fuel temperatures were held above the clad failure temperature but below the uranium melting temperature for 80 minutes for metal-water reaction measurements under that condition. The test was then concluded by increasing temperature into the molten uranium range (1235 C) to explore problems that might be encountered in future tests of fuel in the molten state. However, little was gained due to the obstructions to steam flow from the uranium and oxide debris from the original clad failure.

Although hydrogen evolution data has not been fully analyzed for these tests, the hydrogen volume for SNH-10 was about three times that produced during SNH-11 for a comparable time interval. This difference could be accounted for by visual inspection of the fuel which showed considerably more uranium surface area available for reaction.

Waste Management

Nothing to report.

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ENGINEERING AND TECHNOLOGY - REACTOR

Evaluation of Protective Coatings

A zinc-base paint has provided excellent protection to several sections of piping in the raw water system over a seven-year period. To determine if protection could be expected at higher temperatures and in deionized water, potential measurements have been made on bare iron and iron covered with a zinc-base paint in filtered Columbia River water and deionized deoxygenated water at temperatures between 23 C (73 F) and 59 C (138 F). These tests showed that the zinc-base paint would provide adequate cathodic protection to bare iron under all conditions.

Fire Resistant Hydraulic Fluid - Rod Drives

A study of test data involving rod insertion rates following conversion to a new fire resistant coolant in the left HCR rod drives has indicated a minor increase in the average rod insertion times when compared to data taken previous to the fluid changeout. Although the new invert-emulsion hydraulic oil was naturally suspect, other factors such as mechanical deterioration within the rod system were also investigated.

Water collection in the bottom of the oil reservoir in both the reactor installation and in the test mockup, coupled with fouling of filters, caused the initiation of a program which has disclosed bacterial contamination of the new fluid. Constituents of the fluid apparently provide an environment which sustains the growth of organisms. Work is in progress to establish whether a material such as silver nitrate might function effectively as a bactericide. A test program has been initiated with the Biology Laboratory of BNW to check the bacteria count of periodic samples taken from the fluid reservoir at N Reactor. The Laboratory will also attempt to determine the optimum amount of silver nitrate required for complete inhibition of organisms in the fluid.

A test program using the invert-emulsion fluid has been completed on the HCR mockup in an effort to determine the cause for the slow rods on the reactor. Reactor conditions were duplicated as nearly as possible and no slow-down of the test rod was observed. Included were tests attempting to duplicate the slowing down noted on the reactor. The only test which did slow the rod was attained by inserting nitrogen gas in the scram line; with this condition, the rod starts slowly and has a slowed insertion time. However, and somewhat conversely, a troubled on-reactor rod (No. 17) started fast but slowed down during insertion. Changing-out deteriorated rod seals and bushings on this rod significantly improved insert times. Similar work is planned for several other rods which have insert times approaching Process Standard limits.

Project Engineering

The month-end status of construction projects relating to N Reactor facilities is summarized in Appendix A.

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ENGINEERING AND TECHNOLOGY - FUELS AND TARGETS

Optimized NPR Fabrication Route

In an effort to increase uranium utilization and decrease NPR billet fabrication costs, a number of process changes have been suggested by NLO and DUN personnel. These process changes would include the items noted below, all of which are presently being evaluated by DUN, RMI and NLO.

Use of Ingot Top Crop Material Through Primary Extrusion

In normal operation the ingot top crop is sectioned from the NPR ingot prior to primary extrusion by RMI at Ashtabula. By leaving the top crop on the ingot it is hoped that usable uranium normally discarded with the primary extrusion end defects can be minimized by having the top crop evolve as part of the extrusion end defect. NLO is currently processing 50 billets produced in this manner and DUN should have an evaluation of the billet material within the next two months.

Utilization of the Primary Extrusion Nose Crop

It has been standard practice to discard two to three inches of usable primary extrusion with the nose end defect. Historically it was believed that this nose crop material was not usable as it contained only a small amount of alpha working and hence would exhibit a large amount of large grain from the cast ingot. This was probably true when the NPR stream specified Alloy 501; however, since the change to Alloy 601 (with its better grain refinement characteristics) it is felt that this nose defect metal may be usable.

To evaluate this possibility, eight outer and four inner billets containing nose defect uranium will be extruded in the next month and data will be recorded to assess whether the metal has properties which adversely affect coextrusion properties. If no adverse effects are encountered in coextrusion, the Feedsite will be given the option to use two to three inches of additional metal per primary extrusion. This change could result in an increased Feedsite uranium utilization in that it would allow a more flexible cut-up pattern, and in some instances permit another billet to be sawed from each primary extrusion.

Billet Reversal Option

It has been standard practice in the past that the ingot primary extrusion and the coextrusion at Hanford be accomplished in the same direction. When the NPR billet was unpreshaped, it made little difference whether the extrusion direction orientation be maintained as there was no possible means of changing uranium utilization. However, since the change to a preshaped NPR billet design, uranium utilization can be increased if the Feedsite has the option to cut the front or rear preshape on either billet end.

Recent evaluation at Hanford on billets that were reversed (primary extrusion direction and coextrusion direction opposed) showed no adverse effects on

coextruded fuel quality. Consequently, the Feedsite will be given the option to preshape on either end of the raw billet. This process change has been estimated by NLO to increase NPR uranium utilization by as much as 3 percent.

Long NPR Billets

Since the conversion of N fuel fabrication from Mark I to Mark IV, the regular billet lengths (19 3/4 inch outer, 20 inch inner) are no longer optimum; i.e., in routine production at least one short fuel is produced per coextrusion in addition to a number of regular lengths. Since a change in billet length is necessary for continued Mark IV production, it was decided to make the billets as long as possible for maximum uranium utilization and piece yield.

To evaluate the long billet concept, 30 outer and 30 inner Mark IV billets have been received in new longer lengths of 23.8 and 21.5 inches, respectively. To date, nine long outer billets have been coextruded and results indicate the billet design is adequate to increase piece yield per extrusion from ten full length pieces (plus a 17-inch piece) to 13 full length pieces. No long billet inner coextrusions have been made to date but five such billets are awaiting coextrusion in November. If these coextrusions are successful, partial conversion to long billets may be accomplished within the next two months.

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IRRADIATION SERVICESMATERIALS TESTING"Cold Seeded" Graphite Irradiation - Battelle-Northwest

The first cold seeded graphite irradiation capsule was discharged after reaching goal exposure of 50 days. This instrumented capsule was pulled into the reactor biological shield during operation in order to achieve exactly the desired exposure.

ROUTINE IRRADIATIONS

One hundred six Quickie activation analysis capsules were irradiated in the K reactors for BNW.

The SF foil irradiation capsules were discharged from bottom front-to-rear test hole 0065 at KE Reactor for BNW.

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ADMINISTRATION - GENERALADVANCE FINANCING OF AEC INTEGRATED CONTRACTOR OPERATIONS

On October 7, Douglas United Nuclear submitted to AEC-RL a review of the Company's contract bank accounts and plans for realizing the Commission's objective of reducing average daily bank balances. DUN intends to eliminate the contract bank account at Seattle-First National Bank and to retain those at the Old National Bank and the National Bank of Commerce. As a result of this change, the combined average daily bank balances will be reduced from about \$800,000 to about \$400,000.

AEC MANAGEMENT INFORMATION SYSTEM (MIS)

In view of AEC-MIS plans for earlier data reporting, the overall MIS concept is being reviewed within DUN. A preliminary report of findings will provide the DUN representative to the Atomic Energy Systems, Operations, and Programming Association (AESOP) with data for the first meeting of AESOP.

PRIME CONTRACT NEGOTIATIONS

The first negotiating meeting was held with AEC-RL on October 21.

APPROVAL LETTERS

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

<u>ATD Number</u>	<u>Subject</u>	<u>Date of Transmittal to AEC-RL</u>
AP-39	Pension Plan portion of letter entitled "Pension Plan, Salaried Savings Plan and Wage Savings Plan"	January 12, 1966
AP-59 Add. #2	Salary Structure for Exempt Salaried Personnel	October 24, 1969
ATD-180	Lay-Off for Lack of Work Appendix "B" Modification MO-23	February 24, 1969
ATD-188	Movement of Household Goods Appendix "B" Modification MO-25	October 24, 1969

POST-INSTALLATION REPORT - UNIVAC 9300

A post-installation review of the Univac 9300 procurement and startup was made in accordance with AEC Manual Chapter 1801. The review was made to provide

useful information to Federal agencies involved in computer procurement and operation. Currently at a point thirteen months after installation, the Univac 9300 has fulfilled all objectives of production computing and the additional function of data transmission from Bldg. 1101-N to the Univac 1108 in the Federal Building.

EQUAL EMPLOYMENT OPPORTUNITY

A Negro mechanical engineer was added to the payroll during the month and a Negro instrument specialist was promoted to Engineer.

The structure of the basic education portion of the TOP Program was substantially changed this year. Individual progress appears to be satisfactory under the revised program. Formal individual evaluation will be made in the near future.

EMPLOYMENT SUMMARY

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

SAFETY

No personnel radiation exposures exceeded operational control.

Month-end safety statistics were:

Disabling injuries - October	0
- Year to date	1
Days since last disabling injury	86
Man-hours since last disabling injury	755,600

APPENDIX A

PROJECT STATUS SUMMARY - REACTOR FACILITIES

AUTHORIZED PROJECTS

<u>Number & Title</u>	<u>Authorized Funds - \$</u>	<u>Percent Complete</u>		<u>Status</u>
		<u>Design</u>	<u>Construction</u>	
<u>Single-Pass Reactors</u>				
DCE-505, Boiler Control Improvements - 165 KE & KW	410,000	100	94	Acceptance testing of the fifth boiler modification is under way. Equipment installation on the sixth boiler was started October 7.
DAP-510, Discharge Chute Clearing Equipment - K Reactors	220,000	100	92	Engineering evaluation in progress to resolve design deficiencies.
DAP-516, Storage Building Addition - 105 KE & KW	142,000	100	0	The specification was approved Sept. 30, completing design.
DAE-518, Effluent Radio- iodine Monitor - KE & KW Reactors	100,000	100	95	J. A. Jones continued field work at the 107-KE sample building. DUN fabrica- tion essentially complete.
DAP-526, Deactivation of Hanford Production Reactor C	75,000	84	25	Substation work to provide additional 460-volt capacity for electric heat is under way at 182 and 183 buildings.
<u>N Reactor</u>				
GCP-406, Improved Safety Platforms and Accesses - 100-N Area	300,000	100	100	Revised project proposal, incorporating modified Revision 1 scope change, sent to AEC-RL.

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PROJECT STATUS SUMMARY - REACTOR FACILITIES (contd.)

Number & Title	Authorized Funds - \$	Percent Complete		Status
		Design	Construction	
GCE-408, W, C, and D Elevator Safety Improve- ments - 105-N	90,000	100	95	Physical construction has been completed and ATP performance remains.
GCP-411, Effluent Con- trol Program - 100-N Area	1,830,000	100	43	Section I - Preliminary testing completed October 17 Section II - 182-N header completed in August. Awaiting pipe delivery Section III - Diesel oil day tanks installed and piping completed at tanks October 21. Section IV - No progress. Section V - Emergency dump tank construc- tion underway. Revision 2, requesting a funding increase, submitted to AEC-RL.
DCE-519, Replacement of Bridge Crane and Hoist System with New Crane System - 105-N Storage Basin Area	269,000	76	0	That portion of design required for obtaining bids on crane systems was com- pleted. Bid closing date for cranes is November 17. Revision 2, requesting additional funding, submitted to AEC-RL.

PROJECT PROPOSALS AWAITING AUTHORIZATION

Number & Title	Funds Requested	Date to AEC-RL
GCP-406, Rev. 1 - Improved Safety Platforms and Accesses - 100-N Area	No additional funds.	10/21/69 (resubmitted)
GCP-411, Rev. 2 - Effluent Control Program - 100-N Area	2,010,000 (new total)	9/24/69

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PROJECT STATUS SUMMARY - REACTOR FACILITIES (contd.)

<u>Number & Title</u>	<u>Funds Requested</u>	<u>Date to AEC-RL</u>
DCE-519, Rev. 2 - Replacement of Bridge Crane & Hoist System with New Crane System - 105-N Storage Basin Area	400,000 (new total)	7/10/69
DCP-525 - Fire Protection Improvements - KE	225,000	5/2/69
DCP-527 - Graphite Cooling and Fog Spray - N	97,000	5/9/69 (Held for FY-71 Funds)
DCP-529 - Gravity Drainage System and Disposal Basin - 100-N	200,000	7/10/69 (Held for FY-71 Funds)
DAP-530 - Upgraded Electrical Service and Lighting to 1100-1101-N	78,000	10/2/69
DAP-531 - Establish 1102-N Office Building in N Area	45,000	8/19/69
DCE-532 - Isolation of Process Coolant Risers - K Reactors	270,000	9/25/69

PROJECT PROPOSAL PREPARATION

<u>Number & Title</u>	<u>Design Criteria</u>	<u>Project Proposal</u>
DCE-519, Rev. 3 - Replacement of Crane System, 105-N Storage Basin Area	Completed	In preparation
DCP-528, Rev. 1 - Fire Protection System Improvements, 100-N	Completed	In preparation
Export Water System Backup, 181-D, for 200 Areas	Study started	In preparation
Stack Monitoring Improvements, 100-N Plant	Design criteria prepared but awaiting review of technical basis.	

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APPENDIX BEMPLOYMENT SUMMARY
(with employee allocations)

	<u>9/30/69</u>	<u>10/31/69</u>
<u>CONTRACT PERSONNEL</u>		
<u>02 Programs</u>		
Douglas United Nuclear	1626	1599
Assisting Other Contractors	<u>14</u>	<u>15</u>
Total - 02	1640	1614
<u>Other Programs Under AEC Contract</u>		
Assisting Other Contractors and WPPSS	50	48
Special Irradiations	<u>6</u>	<u>8</u>
Total - Other Programs	56	56
Total Contract Personnel	<u>1696</u>	<u>1670</u>
<u>COMMERCIAL ACTIVITIES PERSONNEL</u>		
	14	17
TOTAL FORCE	<u>1710</u>	<u>1687</u>