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MONTHLY REPORT

FEBRUARY 1969

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March 18, 1969

MONTHLY REPORT

FEBRUARY 1969

DOUGLAS UNITED NUCLEAR, INC.

Richland, Washington

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

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SUMMARY

REACTOR PLANT OPERATIONS

Production Statistics

	<u>C</u>	<u>KE</u>	<u>KW</u>	<u>N</u>
Input Production - Pu (KMWD)	42.5	89.9	73.8	76.1
- U-233 (Equiv. KMWD)	10.5	13.9	10.5	-
Time Operated Efficiency (%)	65.7	82.7	70.5	70.3
Steam Availability to WPPSS (%)	-	-	-	69.6

C & K Reactors

Maximum power levels (4050-4060 MW) at the K reactors continued to be limited by inlet water piping brittle fracture considerations.

The nondefense plutonium program to produce Pu-240 in the 8-12 percent range is continuing and thorium loads are being irradiated at KE and KW. Weapons grade plutonium is being produced from natural uranium at C Reactor.

N Reactor

Reactor power level was administratively limited at 4000 MW.

Two reactor outages occurred, one unscheduled and one scheduled. The former resulted from a tube failure in the Foster Wheeler boiler. The scheduled outage was for reactor maintenance, charge-discharge, and the removal of a process tube for destructive testing. The V-11 valve program for replacing defective original actuators, and repacking the leaking units, was continued. Another leak in the FW boiler was found and repaired during the scheduled outage.

The 48-hour operational testing of the two CE package boilers was conducted. A number of problems still exist, mainly at full load. The vendor is making corrective modifications.

FUEL AND TARGET FABRICATION

Production Statistics (tons)

	<u>For C & Ks</u>	<u>For N</u>
Billets Extruded	-	69.0
Finished Fuel Produced	163.6	34.3
Thorium Canned	13.3	-

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C & K Fuels

AlSi canning operations continued on the basis of three lines on the day shift, five days per week. Bumpers or self-supports were attached to 81 percent of the fuel elements produced. The current thorium canning program was completed.

N Fuels

Production was increased to make up for production lost in January due to difficulty with unbonding in extrusion.

TECHNICAL ACTIVITIES

C & K Reactors

Fracture toughness characteristics of three samples each of base metal and weld metal from the D riser at KE Reactor were obtained from the Lawrence Radiation Laboratory at the University of California. At -76 F, the minimum critical stress intensity factor (K_{Ic}) was 45,000 to 50,000 psi $\sqrt{\text{in}}$ for the base metal and 60,000 to 65,000 psi $\sqrt{\text{in}}$ for the weld metal. Initial results for the growth of cracks by fatigue appear comparable to those reported in the literature for other mild carbon steels. Charpy test results for samples from the dome of the D riser indicate that this metal has acceptable fracture toughness.

Calculational techniques and computer programs for determining K tube power limits to avoid the release of fission products following a dual riser failure have been completed for the E-Q loads. Initial calculations indicate that the current tube power limit of 1960 kw is conservative.

Analysis of the three KE Zircaloy tubes which were sandblasted in March 1967 to remove case hydride layers, and which operated until December 1968 with anodized spacers, is essentially complete. Examination indicates that anodized spacers will protect the tube from excessive hydriding if no surface hydride is present. The sandblasting was only partially successful, however, and those areas where the hydride was not completely removed continued to hydride. An electrochemical process has been developed which will remove all surface hydride but will not attack the base Zircaloy metal. Parametric studies are in progress to determine if this process would be practical for in-reactor use.

The 10 kg plutonium irradiation test was charged into KE Reactor on February 20. Test objectives are to establish the basic neutronic interactions which govern the transmutation of plutonium in a K reactor, and to investigate plutonium fuel characteristics associated with such irradiations.

Reactor effluent water diffusion studies being conducted at Washington State University under Contract DDR-1114 have been temporarily delayed by the extremely cold weather. Temperatures down to -50 F caused extensive damage to the river model recirculation system.

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N Reactor

The irradiation testing of Mark IV fuel elements continues to give favorable results. Visual observation of discharged test fuels indicates that clad thickness has little or no effect on the swelling rate. A significant reduction in swelling rate, especially for inners, is shown for the alloy 503 fuel.

In the Zircaloy process tube monitoring program, the third precracked tube section has now been cycled 290,000 times between 1,250 and 25,000 psi hoop stress without initiation of cracking. The cyclic stress will be raised to 37,500 psi to expedite crack initiation so that critical crack length determinations can be completed in a reasonable time. A tube with an additional year of exposure was removed from the reactor for examination and testing.

FEATURE REPORT

The appended summary report describes the secondary and tertiary coolant systems for B and C Reactors following deactivation of the D, F, and H plants, and for C Reactor after the shutdown of B Reactor in February 1968.

GENERAL

Plans for increasing the number of minority group employees on the Company's payroll are still being held in abeyance pending a final decision concerning the future level of production reactor operations, and the lifting of the present freeze on employment.

Company personnel had no disabling injuries in February, and no radiation exposures exceeded operational control.

Carl W. Kuhlman
for Charles D. Harrington
President

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REACTOR PLANT OPERATIONS - C & Ks

PRODUCTION

Reactor production, power levels, efficiencies and related statistics are tabulated on the next page.

Overall reactor input production and time operated efficiency for the past six months are charted below:

The production of nondefense plutonium in the 8-12 percent Pu-240 range is continuing with both natural and enriched uranium at the K reactors. The C Reactor natural uranium load is now programmed for weapons grade material.

OPERATING EXPERIENCE

Reactor Loadings

Front face maps showing the loadings at C, KE and KW Reactors are reproduced on the three pages which follow page B-6. The tonnages listed are approximate; actual fuel totals are given on page B-2.

The K reactors remain charged with loads of thorium and ⁹⁴ Metal, as required for the assured delivery of 360 kg of U-233.

Power Levels

The maximum power levels of 4050-4060 MW at the K reactors represented continued operation under limits resulting from the inlet piping brittle fracture potential. The KE power level was further limited during the last half of the month by the February 17 charging of material for PTA-150, "Pu-Al Irradiation" (see Mission 3 in Section D). To month end, this test area of the reactor

PRODUCTION REACTOR STATISTICS — FEBRUARY 1969

REACTOR	C	KE	KW	TOTAL
INPUT PRODUCTION — PU — KMWD	42.5	89.9	73.8	206.2
— U-233 — EQUIV. MWD	1,046	13,918	10,509	25,473
POWER LEVEL (MW) — MAXIMUM	2,440	4,060	4,050	10,550
— AVERAGE	2,311	3,879	3,738	9,928
TIME OPERATED EFFICIENCY — %	65.7	82.7	70.5	73.0
OUTAGE TIME ALLOCATION — %				
CHARGE — DISCHARGE	18.8	4.8	15.3	13.0
FAILED FUEL REMOVAL	0	0	0	0
WATER LEAKS	3.6	0	0	1.2
TUBE REPLACEMENT	0	0	0	0
OTHER MAINTENANCE	1.8	2.2	9.7	4.5
STANDARDS CHECK	3.8	2.3	0.2	2.1
PRODUCTION TESTS	6.3	8.0	2.2	5.5
PROJECT WORK	0	0	0	0
OTHER	0	0	2.1	0.7
TOTAL	34.3	17.3	29.5	27.0
NUMBER OF OUTAGES	2	1	4	7
NUMBER OF STARTUP INTERRUPTIONS	0	0	0	0
WATER LEAKS — TUBE	1	0	0	1
— VAN STONE	2	0	0	2
NEW TUBES INSTALLED	3	0	0	3
FUEL CHARGE — (TONS) — NATURAL URANIUM	137.1	3.5	2.9	143.5
— ENRICHED URANIUM	64.0	369.2	360.0	793.2
FUEL ELEMENT FAILURES	1	0	0	1
HELIUM CONSUMED — M CU. FT.	310.3	82.6	99.8	492.7
WATER TO REACTOR				
NORMAL OPERATING FLOW — GPM	98,700	189,600	190,300	478,600
pH	6.56	6.67	6.69	—
DICHROMATE — PPM	0.49	0.50	0.90	—

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B-2

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operated at only about 67 percent of the desired level. Corrective loading changes will be made during the next outage.

The C Reactor, restricted by the bulk outlet water temperature limit of 95 C, and nine-pump operation, achieved a maximum power level of 2440 MW.

Reactor Outages

Seven reactor outages were experienced as summarized below:

<u>Date Down</u>	<u>Reactor</u>	<u>Outage Hours</u>	<u>Remarks</u>
February 4	KW	99.6	Scheduled outage for charge-discharge, repair of HCR No. 9, and miscellaneous maintenance work.
February 8	C	59.8	Process tube water leak. One front and two rear Van Stone flange leaks were repaired. Also accomplished some EMS and miscellaneous maintenance work.
February 17	KE	116.0	Technical high limit trip on high speed scanner caused by a faulty strap-on RTD. The outage was continued for charge-discharge and miscellaneous maintenance work.
February 19	KW	41.4	Unexplained Panellit trip. 168 tubes were charge-discharged and EMS work was accomplished.
February 21	C	170.7	Removal of a failed overbore fuel element (PT-103). The leaking ribbed aluminum tube was replaced with a smooth Zircaloy tube. Also completed scheduled charge-discharge.
February 24	KW	41.3	Unexplained Panellit trip. Some wide mesh venturi-orifice screens were installed and EMS work was accomplished.
February 28	KW	15.7	A transient voltage condition caused a trip on the high speed scanner during testing of electric exhaust fan No. 2. Wide mesh venturi-orifice screens were installed and EMS work was performed. Reactor continued down at month end.

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CONF-55

Boiler Operations - K Reactors

Successful burner changing was made on No. 3 boiler in 165-KW on February 6 by use of the "buddy burners". Acceleration tests were made at all possible intervals during the burner changing, and each time the boiler performed satisfactorily. Since February 10, the burner changing by this method has been done daily.

On February 17 successful "buddy burner" changing and related acceleration tests were made on No. 3 boiler at KE. Since February 21, this daily burner changing method has been in progress on this boiler also.

Both of these boilers with modified controls (DCE-505, Boiler Control Modifications Project) performed satisfactorily during the entire month.

EQUIPMENT EXPERIENCE

VSR Channel Sleeves - KE Reactor

The top eight sleeves were examined in VSR channel No. 49 at KE Reactor. The sleeves were cracked their full length and pieces were broken out of five of the sleeves. The borescope inspection could not be made past the eighth sleeve because of distortion. This channel was sleeved in September 1965, and is the most distorted.

Resistance Temperature Detectors (RTDs) - K Reactors

Forty faulty immersion RTDs were located at KE Reactor during the month of January. Eighteen of these were due to water migration through stem pitting. The High Speed Scanner Temperature Analysis Program (HSSTAP) has been structured so that it can be applied by maintenance personnel. Increased emphasis on RTD resistance-to-ground status at KE Reactor has resulted in increased application of the HSSTAP.

PROCESS ASSISTANCE & CONTROL

Operational Physics

Flattening was relatively high for all reactors with the exception of KE after the February 17-22 outage, during which 225 process tubes were committed to the PuAl test block installed under PTA-150. This loading reduced total reactor power by approximately 350 Mw, and flattening by about 8 percent.

Since the K reactors are still tube power limited, some small power loss was expected initially with the decrease in flattening due to the PuAl block. This situation will be improved at the first outage of opportunity during which several thoria columns in the buffer zone will be displaced with 94 Metal.

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Some operational physics parameters of interest were:

	Reactor		
	C	KE	KW
Effective Central Tubes (ECT)*	1633	2227	2270
Flattening Efficiency** - February	0.87	0.71	0.73
- 12-mo. average	0.83	0.72	0.70
Maximum Operating Time Permitting Scram Recovery - Hours***	10	10	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 currently approved startup procedures.

Production Fuel Performance

There were no production fuel failures during the report period.

The following table shows production fuel failure frequencies as number per million elements discharged, for the 12- and 24-month periods ending January 20:

	<u>12 Months</u>	<u>24 Months</u>
C Reactor - Natural Uranium	9	33
- 94 Metal	41	31
K Reactors - Natural Uranium	4	6
- 94 Metal	7	8

Use of C Reactor Fuels in K Aluminum Tubes

Analytical results have indicated that, from the thermal hydraulic standpoint, it would be feasible to utilize C Reactor fuel in ribbed tubes in the fringes of the K reactors. To gain further insight into potential utilization of C fuel elsewhere in KE and KW, machinery has been set in motion to obtain flow tests in the BNW Hydraulics Laboratory. These tests will provide better information with respect to flow splits and other hydraulic characteristics of the fuel.

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The physics aspects of utilizing C2N fuel elements in place of K4N, and C3E in place of K4E, would not introduce significant reactivity, control or specific heat generation differences. Water loss and temperature coefficient differences would be of the order of a milli-k or less, and differences in specific heat generation rates in an equivalent flux would be on the order of 0.5 percent.

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<u>Zone</u>	<u>Tons</u>	<u>Material</u>
Central	98 41 (Includes 14 in overbore block)	Natural Uranium 94 Metal
Ring	12 20	Natural Uranium 94 Metal
Fringe	25	Natural Uranium
Thoria Support	19	94 Metal
Thoria	4	Thoria

Loading Pattern - C Reactor

<u>Zone</u>	<u>Tons</u>	<u>Material</u>
Central	242	94 Metal (for Thoria Support)
	2	Natural Uranium
	16	Thoria ("X" Designates Tubes)
	1	Special Depleted Uranium (PITA-048)
Buckled	65	94 Metal (for Thoria Support)
	6	Thoria ("X" Designates Tubes)
Blanket	55	94 Metal (for Thoria Support)
	11	Thoria

Loading Pattern - KE Reactor

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<u>Zone</u>	<u>Tons</u>	<u>Material</u>
Central	244	94 Metal (for Thoria Support)
	1	Natural Uranium
	16	Thoria ("X" Designates Tubes)
	1	Special Depleted Uranium (PITA-048)
Buckled	66	94 Metal (for Thoria Support)
	5	Thoria ("X" Designates Tubes)
Blanket	53	94 Metal (for Thoria Support)
	11	Thoria

Loading Pattern - KW Reactor

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REACTOR PLANT OPERATIONS - NPRODUCTIONGeneral

Reactor production, power level, and related statistics are tabulated below. Input production and time operated efficiency (TOE) for the past six months are shown on the following chart:

Statistical Summary

Input Production - Pu	- KMWD	76.14
Electrical Generation (KMWH)	- Total	337.4 (per WPPSS)
Power Level (MW)	- Maximum	4,000
	- Average	3,868
Time Operated Efficiency	- %	70.3
Steam Availability	- %	69.6
Number of Shutdowns	- Scheduled	1
	- Unscheduled	1
Fuel Failures		0
Fuel Charge (Tons)	- 94 Metal	301.3
	- 125 Metal	67.3
	- Natural U	0.4
	Total	369.0

Helium Losses	- M cu.ft.	557.7
Fuel Oil Usage	- bbl	20,721

OPERATING EXPERIENCE

Reactor Loading

The reactor loading at month end is shown on the front face map which follows page BN-6. Ten process tubes were charge-discharged during the scheduled outage starting February 16. This included the discharge of 8 columns of Mark IV fuel at high exposure.

Power Level

Power level was administratively limited at 4000 MW throughout the month.

The main steam header pressure (MSHP) was limited to some degree throughout February because of a shortage of reactor reactivity. The MSHP of about 115 psig which was maintained during most of the month enabled electrical generation by WPPSS at 745 to 768 MW.

Reactor Outages

The two N Reactor outages and their principal causes are summarized below:

<u>Date Down</u>	<u>Outage Type</u>	<u>Outage Hours</u>	<u>Cause</u>
Feb. 7	Unscheduled	40.3	Leak in Foster Wheeler boiler.
Feb. 16	Scheduled	159.6	Charge-discharge and maintenance.

EQUIPMENT EXPERIENCE

Package Boilers

The 48-hour operational testing of the two Combustion Engineering package boilers was conducted on February 14-16. A number of flameouts were experienced (12 on the No. 1 boiler and 3 on the No. 2 boiler). A review of the data collected during the testing has identified some problem areas, mainly at full load. Combustion Engineering has recommended and is making corrective modifications.

Foster Wheeler Boiler

A leak in the Foster Wheeler boiler resulted in an unscheduled outage on February 7. The No. 5 tube on the west end front buckstay had failed when the inner edge of the pedestal pulled loose from the tube. A second leak was detected and repaired during the scheduled outage starting February 16. This leak, as was the first one, was in a floor tube.

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Equipment Modifications

The following Equipment Modifications were completed:

- EMP-202 - Installation of a burner light-off monitor in the 184-N control room (completed except for running of the Acceptance Test Procedure).
- EMP-262 - Installation of fuses on each of the undervoltage relays servicing the motor starting breakers in the 480V main switchgear in Room 144 of the 105-N Building.
- EMP-309 - Installation of indicating lamps in the 105-N control room to indicate the open or closed position of the CONDV-243 valves.

Coolant Systems

Primary Loop Equipment

Process tube 0758 was removed during the scheduled outage to fulfill tube inspection requirements. A new tube was installed, and the channel returned to service.

Process tube 0864 experienced an inadvertent coolant reduction to a below-normal value during the scheduled outage. The tube had been "stingered" to permit inlet connector valve repair, and the stinger hose became crimped due to elevator movement. The tube was emptied, borescoped, determined to be in good condition, and then recharged.

Process tube connector valves (V-11 and V-12) continued to be problems and required considerable maintenance radiation exposure to make them all operable. Three V-11's were repacked and new manual actuators installed. One V-12 (1564) that would not operate was replaced. A functional test of nine V-11's with new actuators showed that only three would operate with 35 psi air supply. Supplement 2 to PT-NR-105 is being circulated for approvals. This supplement authorizes an increase in the number of manual V-11 valve actuators to 100.

Modified cap tools were tested on the ten tubes charge-discharged. Improved performance was noted, and additional modification needs identified.

Steam generator retubing by Combustion Engineering in Cell 2 continued on schedule. Some tube sheet cladding defects have been found and corrected as the work progresses. The tube sheet of steam generator 1A was inspected to determine if it, too, has such cladding defects. The inspection was not conclusive due to moisture in the dome area. Additional inspections will be made next month. Radiation levels in Cell 2 were significantly reduced by filling the primary piping with demineralized water from the instrument backwash system, and then flushing through drain valves. Dose rate in one area was reduced from 150 mR/hr to 30 mR/hr. No. 2 primary pump was disassembled and blanking flanges installed to prepare for cell hydrostatic testing later on.

An oil leak developed on the No. 4 primary pump lube oil system during operation and an undetermined amount of lube oil (about 3 barrels) was discharged into the Cell 4 sump. The oil and water mixture was collected in about 20 drums and required contaminated disposal. The cause of the leakage was not positively identified. The system was checked during the February 16 outage, and the leakage did not recur following startup.

Inspection of the No. 1 high lift pump revealed that the internal splitting vane in the pump volute is cracked in the same manner as in the No. 2 and 3 units. The pump vendor has been contacted, and engineering observation is continuing.

Secondary System

A number of control failures on the CONDV-243 secondary system makeup valves resulted in the requirement that these valves be operated manually. Corrective measures on the automatic system so far have not been effective in gaining adequate control. Engineering investigation is underway.

Ball Safety System

Forty ball hoppers were functionally tested following the February 16 shutdown as part of the continuing test program. There were no failures. Five hoppers were reworked to correct rust indications. In addition, hoppers 15 and 48 were reworked and returned to service. Hopper 15 had been found about half full of balls on February 7, with the other half in the ball channel. Hopper 48 had tripped during the previous startup preparation and had been left empty during the ensuing operation.

Control Rods

Low coolant flow conditions persisted in the control rods. Efforts to improve flow by cleaning orifices, blowing down sensing lines, and replacing control valves were generally successful. It is not planned to chemically clean any more rods until the cooling tube leak problem can be further explored.

Examination of the seal section tubing from two rods, to determine the extent of pitting and to evaluate the cause of leaks observed in rods 68 and 69 in December 1968, has been postponed until March to permit fabrication of replacement tubing sections. If, as expected, severe pitting is observed on the seal section tubing, examination of a tip section will be considered.

Waste Management

On February 21, a small amount of oil was noticed on the surface of the river below the oil storage tanks. Investigation indicated that a small amount of oil was rising from the river bottom at a level below the oil trap provided during the previous leak. All oil lines were hydrostatically tested and found tight. No. 2 oil storage tanks are being tested one-by-one at month end. It may be that the leakage is from a residual pocket of oil that formed during the previous leak, and which has only now broken through to the river. At month end the leakage had diminished to almost none.

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Testing

During an attempt to remove graphite samples from ball hopper test channel No. 60 it was found that the graphite alignment in the channel had altered making it impossible to remove all the samples. Investigation of this problem is underway, and further work will be scheduled for this channel.

The thermocouples installed under Development Test 45 to check the Zone I concrete temperatures at the piping penetrations, with and without cooling water on, was continued. On February 18, four additional thermocouples were installed. Final results are not yet available.

Production Test 102, "Return of N Reactor Auxiliaries Condensate to a Single Quadrant", was initiated twice, but terminated both times due to system instabilities caused by the CONDV-243 valve control problem. The test will be continued.

Development Test 35 is being initiated to obtain information for calibration of flow monitor Barton transducer units and verify the high and low trip points prior to reactor shutdown while the primary loop is still pressurized. Currently a test facility is being installed in the 109-N Module Shop to allow calibration of the Barton units under pressure.

PROCESS ASSISTANCE & CONTROL

Operational Physics

The main steam header pressure was kept down to about 115 psi to maintain enough excess reactivity in rods for reasonable flattening. This excess reactivity dropped by mid-February to a relatively low value of 3.3 mk for the inboard rod configuration, which was used to obtain high axial flux flattening as well as radial flattening. Excellent flattening had been obtained with this rod configuration until the excess reactivity dropped to 3.5 mk, below which value the radial flattening began to decrease gradually. After the mid-February outage, the rod configuration was changed from the inboard to an outboard pattern. These two rod patterns are shown below:

Typical Inboard Pattern

Inlet		29	39	49	59	69	79	89	Outlet	
	18	28	38	48	58	68	78	88	98	
07	17	27	37	47	57	67	77	87	97	107
06	16	26	36	46	56	66	76	86	96	106
05	15	25	35	45	55	65	75	85	95	105
04	14	24	34	44	54	64	74	84	94	104
03	13	23	33	43	53	63	73	83	93	103
	12	22	32	42	52	62	72	82	92	
	21	31	41	51	61	71	81			

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Typical Outboard Pattern

Inlet												Outlet	
		18	29	39	49	59	69	79	89				
07	17	28	37	47	57	67	77	87	97			107	
06	16	26	36	46	56	66	76	86	96			106	
05	15	25	35	45	55	65	75	85	95			105	
04	14	24	34	44	54	64	74	84	94			104	
03	13	23	33	43	53	63	73	83	93			103	
12		22	32	42	52	62	72	82	92				
		21	31	41	51	61	71	81					

1. Rods normally in during operation.
2. Even numbered rods enter from right side;
odd numbered rods enter from left side.

The outboard pattern allows more rods to be physically present in the reactor for the same excess reactivity than does the inboard configuration. The outboard rods see a lower relative neutron density and are, therefore, worth less than inboard rods. The outboard rod configuration will provide better flattening and better distributional control during operation when the excess reactivity is less than about 3.5 mk. After the March charge-discharge outage, during which the excess reactivity will be increased, an inboard rod pattern will again be utilized.

Some operational physics parameters of interest were:

Effective Central Tubes (ECT)	840
Flattening Efficiency - February	0.84
- 6-mo. average	0.83
Maximum operating time permitting scram recovery - hours	24

Fuel Performance - Billet Fabrication Studies

The 25 Mark I-C fuel columns fabricated from upset forged billets and 12 columns fabricated from direct cast billets are performing satisfactorily. The first columns from upset forged billets are due to be discharged in March and from direct cast billets in June or July.

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	41	42	43	44	45	46	47	48	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64	65	66	67	68	69	70	71	72	73	74
34																																		
33							C	C	C													C	C	C			C	C	C					
32						C	C		C				76		76	76			C				76	C	C		C	C	C					
31							X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X		C	C			
30				C	C	C	X	X	X	X	X	X	X	E	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X		C	C		
29			C	C		X	X	X	X	X	E	E	E	A	A	A	A	E	E	E	E	E	E	E	E	E	94	X	X	X	X		C	C
28			C	C		X	X	X	X	E	A	A	A	E	E	E	07	E	E	E	E	E	E	E	E	E	E	E	X	X	X	C	C	C
27	C	C	C	X	X	X	07	E	E	E	E	E	A	E	E	E	A	E	E	E	E	E	94	07	E	07	A	X	X	X	X	C	C	C
26	C	C	X	X	X	X	A	E	A	A	E	A	E	E	E	E	A	E	E	E	E	C	E	A	94	94	94	X	X	X	X	C	C	C
25	C		X	X	X	A	A	E	E	A	A	E	E	A	E	A	A	A	E	E	A	A	E	E	E	E	E	E	X	X	X	C	C	C
24	C	C	X	X	X	07	E	07	E	A	A	07	A	A	E	E	E	A	94	07	E	E	E	07	E	07	96	E	E	X	X	X	C	C
23	C		X	X	X	96	96	96	96	E	C	96	96	96	E	96	C	A	94	A	E	A	A	C	E	E	96	A	E	X	X	X	C	C
22	C		X	X	X	A	C	94	C	96	96	96	E	96	94	A	E	96	96	C	C	E	96	E	E	96	96	96	X	X	X	C	C	C
21	C		X	X	X	07	C	07	C	C	C	07	E	07	E	94	E	E	E	07	94	E	E	07	E	07	94	94	A	X	X	X		C
20	C	C	X	X	X	E	E	E	E	E	E	A	96	A	A	E	96	E	E	96	95	E	96	E	E	E	E	E	E	X	X	X	C	C
19	C		X	X	X	96	96	96	E	E	E	E	E	94	E	E	E	E	96	95	E	96	95	96	E	E	E	E	E	X	X	X		C
18	C	C	X	X	X	07	E	07	C	E	C	E	C	E	C	E	C	E	E	07	C	E	E	07	E	07	E	E	E	X	E	X		C
17	C	C	X	X	X	E	E	A	A	A	E	A	E	E	E	A	94	94	94	94	C	94	A	E	E	E	A	E	E	X	X	X	C	C
16	C	C	X	X	X	E	A	D	E	E	E	E	E	E	E	E	A	07	E	E	E	E	E	E	95	E	E	A	E	X	X	X	C	C
15		C	X	X	X	07	A	A	A	E	E	A	A	07	A	A	E	E	A	07	A	E	A	07	E	07	A	C	A	X	X	X		C
14	C	C	X	X	X	01	A	01	A	E	E	E	01	01	01	A	01	A	94	01	95	01	01	C	01	A	94	01	01	X	X	X		C
13	C	C	X	X	X	E	E	A	A	E	E	E	A	E	A	94	07	94	E	E	E	E	C	E	E	94	E	E	E	X	X	X		C
12	C		X	X	X	07	E	C	E	E	E	A	A	A	A	E	94	E	94	E	E	E	E	07	E	07	E	E	C	X	X	X	C	C
11	C	C	X	E	X	A	E	E	E	E	A	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	X	X	X	X	C	C
10	C	C	X	X	X	A	E	94	E	E	E	E	E	E	E	E	E	E	A	E	E	E	E	E	E	E	E	E	E	X	X	X	C	C
09	C	C	X	X	X	X	A	E	E	A	A	07	E	07	A	A	E	94	94	A	E	E	E	07	94	07	E	E	X	X	X	X	C	C
08	C	C	C	X	X	X	E	A	A	A	E	E	A	A	A	E	A	E	E	A	C	C	C	E	E	E	E	E	X	X	X	X	C	C
07		C	C		X	X	X	X	A	E	A	A	A	E	E	E	07	C	E	A	E	E	E	E	E	E	X	X	X	X	C	C	C	
06			C	C		X	X	X	X	A	E	07	E	07	A	E	A	E	A	07	E	E	E	E	X	X	X	X	X	X	C	C	C	
05				C	C	C	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	C	C	C	
04					C	C	C	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	C	C	C	
03						76	C				C	C				C	C	C	C	76	C													
02							C	C					C	C	C			C	C	C	C	C												
01																																		
	41	42	43	44	45	46	47	48	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64	65	66	67	68	69	70	71	72	73	74

Fuel Code	No. Tubes	Description	PT-NR No.	No. Tubes	Description
	62	Mk-IC (94 Metal Fringe Normal)	76	6	Fuel Meltdown Test
A*	112	Mk-IC (94 Metal Central)	94+	30	Mk-IV Demonstration
C**	185	Mk-IC (94 Metal)	95	5	High U-236 Content Fuel
D	1	Mk-IC (Natural U)	96	31	Mk-I from Upset Forged Billets
E	277	Mk-IC (94 Metal - High U-236)	01	12	Mk-I from Direct Cast Billets
X***	242	Mk-ICA (125-94 Metal)	07	40	Initial Full Length Mk-IV Columns
	879	Total		1	Blank Channel or Empty Tube
	125	Total PTs			
1004		Grand Total		125	Total

*50% NFS **100% NFS ***94 Metal-25% NFS +Includes Mk-IV High U-236 Content Fuel and 2 tubes w/Mk-IA 125-94 Metal and Mk-IV 94 Metal.

LOADING PATTERN - N REACTOR

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FUEL AND TARGET FABRICATION - C & K REACTORS

PRODUCTION

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General

Production of AlSi-bonded and hot-die-sized fuel elements for the C and K reactors was 88.7 percent of forecast. Eighty-one percent of these elements had bumpers or self-supports attached.

Thoria target element production was completed.

Acceptable Elements Produced

	Finished Production (tons)	Yield - Percent	
		Current Mo.	FYTD
AlSi-Bonded Fuel	140.0	96.6	96.3
Hot-Die-Sized Fuel	23.6	91.7	81.8
Thoria	13.3	93.1	95.6

Month-End Inventories

	<u>Tons</u>
Bare Uranium Cores	1053*
Finished Fuel: AlSi-Bonded	1419*
Hot-Die-Sized	22
Thoria Elements	46.8

*These totals include 114 tons of cores and 81 tons of finished fuel made for the smaller reactors.

OPERATING EXPERIENCE

Overall operating efficiency of the AlSi-bonding lines was 99.2 percent. Final assembly of hot-die-sized overbore fuel elements was delayed which accounts for the below-forecast production. Downtime was assigned 50 percent to equipment malfunctions and 50 percent to operations. AlSi canning operations continued on the basis of three lines on the day shift, five days per week.

EQUIPMENT EXPERIENCE

Tests of cast agitator baskets indicated they will last 50-100 percent longer than the existing baskets which are fabricated on-site. The cast baskets also

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cost about one-third as much as the fabricated units. In addition, these new baskets are more uniform in construction, hence require little or no shaping after installation.

PROCESS ASSISTANCE AND CONTROL

Nothing significant to report.

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FUEL AND TARGET FABRICATION - N REACTORPRODUCTIONInput Production

Total billets extruded	325
Tons extruded	69.0
Percent of forecast	135.3

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Output Production

Total finished assemblies	1,344
Tons output	34.3
Percent of forecast	118.3

<u>Uranium Utilization - %</u>	74.8
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Month-End Inventories - Tons

Bare uranium billets	264
Finished fuel	343

OPERATING EXPERIENCE

Because of the low production last month due to unbonding problems at extrusion, both input and output production this month were significantly increased. This accounts for the high percentages when compared to the official forecast.

EQUIPMENT EXPERIENCE

Nothing significant to report.

PROCESS ASSISTANCE AND CONTROLDual-Electrode End Welding

The dual electrode technique was used to end weld 100 Mark IV outer fuels utilizing the operating parameters and limits established earlier for Mark I outer and Mark IV outer fuel geometries. Three welds were rejected out of 400 total welds (two welds per end). Metallographic examination of various samples showed a normal recast weld structure identical to that of the single electrode process. The clad melt and weld penetration were determined to be comparable to that with ends welded by the single electrode process.

This dual electrode technique, which will significantly reduce the time to process fuels through the welding step, will be incorporated as a regular production operation. The established operating parameters will be added to the Quality Control Standards for both the Mark I and the Mark IV outer fuel geometries.

Preheating of Billets for Extrusion

The reduced reject rates due to unbonding which have been obtained from the stepped preheating cycle of billets prior to extrusion are partially the result of the reduced temperature differential between inner and outer copper-silicon sleeves. Raising the furnace temperature incrementally every two hours allows the billet to approach thermal equilibrium throughout the cycle.

The time required at the final interval varies from 9 to 10 hours, for the two older preheat furnaces, to 4 to 5 hours for the two new furnaces. The 16 to 22 hours duration of the total stepped preheating cycle makes it desirable to develop other means of lowering the inner sleeve-outer sleeve temperature differential while reducing the length of the preheat cycle. Under present practice, the lubed billet is placed in a steel preheat can which is then loaded into the furnace. Heat is transferred to the billet through the steel can wall and ends, thereby heating the OD of the billet directly and relying upon conduction in the billet assembly itself to heat the billet ID.

An experimental preheat can is being developed which will allow the furnace air to enter the ID of the billet assembly. It is expected that, by heating the billet ID in this direct manner, temperature differentials will be significantly reduced so that preheating cycle time may also be lowered.

Fuel Recovery Process (Iron Contamination in End Welds)

This recovery program was established to reclaim approximately 350 fuels brazed with rings contaminated with iron particles. The finished autoclaved fuel exhibited rust spots on the end weld. The reprocessing operation developed to remove these iron particles had the following basic steps:

1. Mask off the OD and ID portions of the fuel adjacent to the weld areas.
2. Vacublast the weld areas to remove the oxide formed during autoclaving.
3. Immerse the blasted end of the fuel in a 20% by volume nitric acid solution for 15 minutes.
4. Rinse and dry.
5. Reweld the fuel end.
6. Immerse the rewelded end of the fuel in a final etch solution using the normal final etch procedure.

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7. Rinse and dry.
8. Re-autoclave using the standard autoclave cycle.

The program was completed with remarkable results. Four hundred fifty-eight outer ends were reprocessed, from 345 Mark IV 26-inch outer fuel assemblies; a total of only seven ends from seven different fuels was rejected.

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TECHNICAL ACTIVITIES - C & K REACTORS

RESEARCH AND DEVELOPMENT

Mission 1 - Basic Production

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1-A. Brittle Fracture Program

Fracture Mechanics Testing Program

Three samples of weld metal and three of base-plate metal from the D riser of KE Reactor were subjected to brittle fracture tests at UCLRL to measure critical stress intensity factors at -76 F, 32 F, and 68 F.

The following tentative conclusions are based on the data obtained thus far:

- Fractures at -76 F were sufficiently brittle (no perceptible plastic deformation along the fracture front) to represent a reasonably valid indication of minimum critical stress intensity (K_{Ic}) of 45,000 to 50,000 psi $\sqrt{\text{in}}$ for base metal and 60,000 to 65,000 psi $\sqrt{\text{in}}$ for weld metal.
- Fatigue crack growth rates measured during the fatiguing process prior to fracture testing indicate growth rates comparable to other mild carbon steels. These values will be refined and measured over a broader range in future tests.
- At temperatures below the crack propagation transition temperature, the stress intensity required to propagate a crack is less than that required to initiate the crack; i.e., once a crack begins it would continue to propagate until it reached a zone of sufficiently low stress or sufficiently high toughness that the stress intensity is reduced to below the arrest value. For example, to initiate a crack in the base metal at 0 C would require a stress intensity of $\sim 50,000$ psi $\sqrt{\text{in}}$. As the crack lengthens, the stress intensity factor increases. If the crack reaches a weld before the stress intensity factor increases to $\sim 60,000$ psi $\sqrt{\text{in}}$, it will stop since this is the crack arrest value for the weld. If the stress intensity exceeds $\sim 60,000$ psi $\sqrt{\text{in}}$ at this point, it would propagate through the weld and continue into the adjacent plate.
- Sample size and configuration being used appear adequate for testing the remaining samples.

Three mechanical testing laboratories given three identical steel samples have returned Charpy and tensile test data. All three laboratories--Atlas, Metallurgical Engineers, Inc., and Pittsburgh Testing Laboratory--reported essentially identical results. Results of Charpy tests of samples

from the dome of the D riser of KE Reactor indicate that one heat of dome material is adequately notch ductile at any operating temperature. The Charpy tests yielded an average of 16-1/2 ft. lb. at 33 F and the tensile tests gave an average yield strength of 33,900 psi.

1-B. Technical Bases for K Reactor Tube Power Limits

DUN-5362, "Interim Report: Development of Tube Power Limits for K Reactors", has been issued and copies have been forwarded to AEC-RL. This document summarizes the sequence of accomplishments toward development of revised technical bases, the analytical methods adopted for determination of tube power limits, and the status of the program as of January 13. As related to the status, principal items presented were as follows:

- The development of the analytical method for determining tube power limits for any given reactor operating condition was complete.
- With conservatively chosen assumptions for current operating conditions of the reactor under the 1960 KW tube power limit, peak fuel cladding temperatures following a dual riser failure would not exceed 540 C. Thus, according to the criterion that limits the peak temperature to 580 C, the present tube power limits are adequate for assurance that there would be no nuclear safety implications associated with simultaneous loss of coolant supply to two adjacent inlet risers.

The parametric study of revised tube power limits with consideration of void-reactivity feedback effects is nearly complete and a correlation has been developed for calculation of tube power limits for a conservatively chosen tube power distribution. In its present form, as shown by the equations below, the correlation relates the tube power factor for the limiting process channel to a modified inlet temperature that incorporates effects of true inlet temperature, bulk outlet temperature, and the void-reactivity gain associated with the aggregate exposure of fuel in the region where voiding is probable.

$$T_{FL} = 1.0 + 0.02422 (T' - 5.2)$$

$$T' = (T_{in} + 2.4) \left[\frac{R + 8}{18} \right] + 1.765 (95 - T_B) \left[\frac{R + 2.75}{12.75} \right] + 2.45 \\ (10-R) - 2.4$$

where:

T_{FL} = ratio of limiting tube power to average tube power (average tube power is reactor power divided by the number of heat-producing tubes)

T' = adjusted inlet temperature, °C

T_{in} = true inlet temperature, °C

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T_B = reactor bulk outlet temperature, °C

R = local void reactivity gain, mK

With an assumption of the ratio of limiting to average tube power (T_{FL}), together with inlet temperature and reactivity gain, the two equations can be solved for reactor bulk outlet temperature. This temperature then is used in combination with the normal reactor flow rate to determine reactor power, from which can be determined the limiting tube power. The inlet temperature, bulk outlet temperature and reactivity gain used are also constrictive on reactor operation.

Simplified techniques for expressing tube power limits are being explored. Methods for implementation of the limits are under study. As it now exists, the correlation for tube power limits applies only to the postulated dual riser accident and is subject to the following limitations:

- Maximum normal reactor flow rate = 196,000 gpm
- Minimum tube flow rate = 71.8 gpm
- Maximum bulk outlet temperature = 95 C
- Maximum front to rear peaking factor = 1.15
- Minimum ratio of peak to average tube power = 1.28
- Reactor loading = E-Q, 1000 MWD/T maximum average exposure
- 51-piece columns of K5E fuel

The correlation was developed for a realistic distribution of tube powers that can be shown to be conservative; i.e., the "bottom tilt" case for which the maximum tube powers are assumed to be at the bottom of the central zone, with a uniform gradation of decreased tube powers towards the top. This distribution has been found to produce the most severe void-reactivity feedback effects.

Future studies are to be made to (1) explore refinements such as the feasibility of adequately defining less severe tube power distributions, (2) determine the impact of the tube power limits on achievable reactor powers, and (3) to investigate tube power limits required for less severe coolant loss accidents.

Preliminary indications are that restrictions in reactor power due to tube power limits would be nominal, and would be superseded by the 4400 MW administrative limit or the 95 C bulk outlet temperature limit during significant periods throughout a 1-year operating cycle, during which inlet temperatures would vary according to historical records.

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1-C. Zircaloy Process Tube Hydriding

Process Tube Examination

Analysis of the downstream portion of the three tubes from KE Reactor which were sandblasted in-situ (numbers 1966, 1968 and 2163) is essentially complete. It now appears quite certain that where the case layer was removed completely there has been no further hydriding.

Of the three tubes examined, No. 1966 was very poorly cleaned and there was a residual case layer over most of the downstream 60 inches. Total hydrogen levels up to 1163 ppm were found. Tube 1968 had only a few spots where there was some residual case, and tube 2163 was completely free of case layer hydrogen. Where there was no case layer, the base metal content was about where it should have been at the time they were sandblasted. These findings have been confirmed by metallography. These data indicate quite conclusively that, when the case layer is removed and anodized spacers are used, hydriding is effectively controlled. This experiment also shows that the major deficiency to this method of tube renovation is the difficulty of getting complete and uniform removal of the case layer.

Tube 3361-KW, removed from KW on December 19, was found to be relatively low in total hydrogen content. This was the tube which had been charged with the 2.1 E-N load and had operated at high temperatures. A companion tube removed earlier had the highest hydrogen level ever encountered in a K tube, but the more recent data indicate that short-term high-temperature operation does not necessarily result in rapid hydriding.

Tubes 1960- and 1961-KE, which were installed March 23, 1967, and were operated until December 10, 1968, with anodized spacers, were found to be completely free of hydrogen in the downstream 60 inches. These data confirm earlier tests of up to 9 months duration in which anodized spacers prevented hydriding in new tubes.

Hydride Removal Techniques

Work is proceeding on an anodic hydride removal process. At present, the process parameters--time, temperature, current density, voltage, and electrolyte strength--are being tested. Electrolyte additives, to prevent precipitation of zirconium compounds and to enhance removal of the last trace of hydride needles, will also be studied.

This work to date indicates the process will work satisfactorily at 0.1 amp/cm² with maximum voltage limited to 50 V. Three wt% HNO₃ solution is an adequate electrolyte and approximately 60 minutes is ample time for removing a reasonably heavy case layer. These four parameters are all feasible from the reactor use standpoint. An outstanding advantage of this process is that it will not attack the base metal and it is possible to tell from the current and voltage relationship when the cleaning is complete.

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1-D. Physics Code Development

The HAMMER program has for some time altered the absorption and nu-sigma fission cross-sections in the thermal group as a result of thermal leakage corrections. The change in absorption cross-sections is considered valid, but the change in nu-sigma fission has yielded poor multiplication values in cases where the fission cross-section is low or zero. The error was corrected and cases were run to check the coding. The change has removed the discrepancies between nonleakage and leakage-corrected numbers in the thermal region. It also has restored the necessary condition that the k-infinity and k-effective be identical with a zero axial input buckling. A calculational case of a PCTR experiment was run to check the changes against a known k value and also to check against an old HAMMER version (before the thermal leakage correction was attempted). The differences in k values and other parameters of the experiment between the HAMMER version and the new version were sufficiently small to add the correction to the PCF file.

During resolution of the nu-sigma error, a discrepancy was found in the method for calculating the fission source and its application which did not provide the anticipated "flat" source. This problem is under study.

In the thermal library, the scattering cross sections are missing for several isotopes. Most of these values are constant or nearly constant and can be added easily.

Solution to the nu-sigma fission problem mentioned above was necessary to check out the new uranium isotopes to be added to the library. Several cases have been run using U-238 and U-235 separately and in a 1 percent enriched KV-type lattice.

1-E. Corrosion Studies

Four inhibitor systems were investigated for their ability to control pitting corrosion and corrosion product tuberculation in filtered Columbia River water.

A long-chain organic polymer film dispersing material, Betz DE-287B, failed to control either pitting or tuberculation and is being dropped from further consideration. Hydrazine at a concentration of 10 ppm is also being dropped from further testing because it did not completely control pitting corrosion, although it did reduce uniform corrosion.

Two other inhibitor systems, Betz Dianodic 187, a zinc-phosphate-chromate formulation (at a concentration of 10 ppm) and Calgon TG-10, a zinc-phosphate formulation (at a concentration of 3 ppm), were investigated for their ability to control tuberculation in stagnant water simulating KE-KW Reactor cross-tie line conditions. Both materials significantly reduced tuberculation, film weights, and uniform corrosion, but did not completely control tuberculation. The catalyzed sodium sulfite presently being considered for this application thus appears to offer slightly better tuberculation control.

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Additional tests are being considered employing the Betz Dianodic 187 and Calgon TG-10 at higher concentrations. The concentrations could be raised considerably before the cost would equal that of using catalyzed sodium sulfite.

1-F. Graphite Studies

Eighteen specimens of graphite, removed from the moderator of KW Reactor, have been in the ETR to evaluate the effect of continued irradiation. Ten of these samples have been removed and will be replaced with samples of N Reactor graphite and silicon carbide coated graphite. Irradiation of the eight remaining K reactor samples will continue. Most of the K samples which have been irradiated at 800 C or above have gone through turnaround and are expanding. Samples being irradiated at present K reactor moderator temperatures have received the equivalent of 6 or 7 years additional exposure, but have not yet reached turnaround; their contraction rate appears to be decreasing.

Mission 2 - Coproduct

2-A. Lithium-Aluminum Splines

A requisition has been issued for the fabrication of spline samples containing dispersions of LiAl , LiMg , and Li_2CO_3 . These compounds would be in an aluminum matrix, in the proper proportions to give an effective lithium concentration of 3.2 wt% - the same as in the alloy splines which have already been tested. These additional spline samples will be irradiated to determine if they are less susceptible to embrittlement.

Mission 3 - Transplutonium Technology

3-A. Ten-Kilogram Plutonium Irradiation

Production Test PTA-150, authorizing the irradiation of 10 kg of plutonium in KE Reactor, was approved and documented as DUN-4747. Supporting studies and documentation have also been completed: DUN-5676, "Shutdown Neutron Flux Traverse - PT-4561 - 105-KE - January 20, 1969", and DUN-5479, "Total Control Calculations for the PuAl Production Test (PTA-150)".

The objectives of this Production Test are to establish the basic neutronic interactions which govern the transmutation of plutonium in a K reactor, and to investigate plutonium fuel characteristics associated with such irradiations. The plutonium-aluminum elements contain 6.0 wt% plutonium dispersed in a matrix of 1100 alloy aluminum. The test loading consists of a slab of 13 PuAl columns interspersed with eight thoria columns, surrounded by 48 thoria columns, surrounded by 156 columns of 94 Metal.

The test loading was charged on February 20. The reactivity and heat generation of the PuAl columns are somewhat below desired levels.

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3-B. Irradiation of 250 Kg of Plutonium

Calculations were completed for a survey study of Hanford capabilities for irradiating up to 250 kg of plutonium. Four cases were considered for varying charge and discharge dates.

The Am-243 and Cm-244 yields were approximated from the plutonium burnout data obtained from the B Reactor test. For the case of initial charging October 1, 1969, and final discharge March 1, 1972, the approximate Am-243 would be 1.3 kg and the Cm-244, 0.2 kg. The Am-243/Am-241 ratio would be approximately 0.9, and that of Cm-244/Cm-242 approximately 3.0

Criticality parameters for these elements have been determined by HAMMER calculations with several different fractions of Pu-240. The results are:

Percent Pu-240	Critical Mass (lb)	Critical Volume (gallons)	Critical Cyl. Dia. (inches)	Critical Slab (inches)	Critical M/A (lb/ft ²)
0	217	23.3	14.8	6.8	41.1
5	263	28.3	16.0	7.6	45.0
20	446	46.4	19.1	9.6	58.1

The assumed physical characteristics of the element are:

Element diameter (core)	1.30 inches
Element diameter (clad)	1.48 inches
Element length	8.5 inches
Core volume	184.88 cm ³
Core density	2.76 gm/cm ³
Plutonium concentration	2% weight
Pu per core	10.205 gm

The HAMMER calculations have not been checked experimentally, but efforts are underway to do so. Some experimental work has been done on both 5 w/o and 1.8 w/o Pu in PuAl elements in light water.

The minimum exposure yields and related assay were determined for their effects on critical mass considerations in the ARHCO dissolvers. Inserts would be required if ARHCO were to base limits on unirradiated PuAl. Thus, it is desirable that ARHCO be assured a given burnup level or minimum exposure level.

Mission 7 - Target Space Enhancement

7-A. Large Channel Facility - C Reactor

The fuel failures from positions 3 (incipient) and 4 of one-inch overbore tube 2868 were examined in Radiometallurgy, confirming that failure resulted from corrosion and erosion of the cladding adjacent to the upstream end of a self-support. In general, non-uniform corrosion of the remaining portions of the cladding was minimal. The rod of element 3 revealed little non-uniform corrosion.

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Four columns at present have exposures in excess of 1550 MWD/T, six additional columns have more than 1100 MWD/T, and the remaining 33 columns in the test block have accumulated 650 MWD/T or less. Present plans are to irradiate the entire block without discharge pending further resolution of the outlook for C Reactor deactivation.

Investigation of the Zircaloy process tube failure associated with the fuel rupture in overbore tube 2868-C indicates that the fuel rupture blister pushed a hole through the tube wall. There was no evidence of overheating of the Zircaloy. It showed good ductility and elongated approximately 30 percent before failure. There was no hydrogen pickup. Apparently, this rupture occurred at a point where there was no graphite around the tube since normal clearance between tube and graphite is about 0.030 inch and it appears the protrusion on this tube was at least 0.2 inch. It may be that the combination of a high OD/wall ratio and annealed material make this tube significantly more susceptible to this type of failure than are the K tubes.

On February 21, a fuel failure was experienced in ribbed aluminum overbore process tube 3364-C at an exposure of approximately 750 MWD/T. Removal was accomplished with 6000 pounds of pushing force. Fuel examination is scheduled.

7-B. K Reactor Overbore Safety Analysis

An evaluation of possible reactivity limitations versus channel and fuel designs for K reactor overbore is being performed by use of the program MOFDA. Rod-in-tube fuels have been designed by the program for 0.6, 0.8, and 1.0 inch overbore, including Zircaloy and aluminum clad fuels in Zircaloy and aluminum process tubes. Aluminum clad fuels in aluminum process tubes, utilizing an inner aluminum rod for flow balance, have also been designed for the three overbore sizes. Both operating and accident conditions have been incorporated for the various fuel designs.

Due to the large volume of data produced by these calculations, a computer program to analyze the data is being written. This program will summarize the data output by MOFDA as well as calculate the control system strengths for the various cases. These summarized results will provide the basis for significantly reducing the required number of overbore calculational models.

Mission 8 - Nuclear Safety

8-A. Uranium-Aluminum Reaction Studies

Consultation with Dr. P. Chiotti of Ames Laboratory has continued during the past month to evaluate more fully his measurements of the heats of formation of uranium-aluminum compounds. The temperature-dependence of the heat capacities of these uranium-aluminum compounds is being measured.

Another area of interest in the uranium-aluminum reaction is the kinetics of reaction. Chiotti has given some indication of rate and has agreed to forward some of his experimental results to DUN so that a more thorough analysis of reaction kinetics for the formation of UAl_4 can be performed here.

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Mission 10 - Columbia River

10-A. Corrosion and Water Quality Studies

Results from the neutron activation analyses of a Columbia River water sample collected November 14, 1968, above the Hanford reactors, in parts per billion, are: Na(2220), K(1190), Rb(<3.5), Cs(0.35), Sc(0.002), Cr(<0.4), Mn(2.8), Fe(<13), Co(0.02), Cu(<0.13), Zn(<7.0), Hg(<2.5), As(2.5), Sb(0.24), Se(0.25), Br(5.6), Ag(<0.5), Au(<0.03) and U(<0.7). The most significant changes since August are a reduction in Zn and Fe concentrations from 100 to <7 ppb and 300 to <13 ppb, respectively.

Analyses of a duplicate of the August sample show that the high values for Cu, Ag, Sb and Se in the original sample were due to contamination introduced when new analytical procedures were employed. Corrected values are: Cu(<0.01), Ag(<0.5), Sb(0.3), and Se(<0.2).

10-B. High Turbidity Coolant

The influence of higher process water turbidity on effluent water radionuclide concentration is being investigated in a half-plant test at KE Reactor. Phase II of the testing (turbidity of 0.2 JTU) was completed on January 27, and Phase III testing (0.3 JTU) was initiated. To the end of February, there continued to be no significant differences in the effluent radionuclide concentrations between the two sides of the reactor. The higher turbidities have been successfully obtained by feeding lower amounts of filter conditioner and it has not been necessary to extend filter runs.

ENGINEERING & TECHNOLOGY - REACTORS

MFC-12 - New Capabilities of Production Reactors

A summary briefing on the MFC-12 activities was presented to DUN management on January 24, and to AEC-RL management on February 10.

The final summary report on MFC-12 (DUN-AOP-180) requires only final management approval before issuance.

Reactor Effluent Disposal

Reactor effluent water diffusion studies being conducted at Washington State University under Contract DDR-114 have been temporarily delayed due to extremely cold weather. With temperatures down to 50° below zero, all activity in the Columbia River model building was halted. Failure of the heating system and the subsequently frozen and broken water pipes resulted in extensive damage to the model recirculation system.

Smoke Density Monitoring Prototype

Detailed design for the prototype installation of a smoke density monitor has been completed. It is planned to install the prototype on the No. 1 boiler

stack at 165-KW. The work will be incorporated with the boiler control system modifications on that boiler which are currently in progress.

Confinement Studies

The dynamic seismic analysis of the K Area facilities to be made by John A. Blume and Associates will include analysis of the reactor block. Security clearances on their key personnel as well as their facility have been extended to include this work for DUN. An initial package of engineering drawings and documents has been prepared and transmitted for their use.

Project Engineering - Reactor Facilities

The status of approved construction projects relating to single-pass reactor facilities is summarized in Appendix A.

ENGINEERING & TECHNOLOGY - FUELS & TARGETS

Correction of Nitrogen Oxide Mist Problems in the 300 Area

A program has been initiated to reduce gaseous effluents from fuel fabrication processes utilizing nitric acid. This program contemplates an extensive study of problems with nitric acid mists and mixed nitrogen oxides (NO_x). Adverse weather conditions this winter, with frequent and prolonged periods of atmospheric downwash, have aggravated the problem.

The first phase of this program will minimize nitric acid mists originating from the hot-die-size delubrication acid tanks in the 313 Building. Specific action includes the following: install demister pads in the inlet to the exhaust stack, raise the stack to 1.5 times building height for better dispersion, design tank covers to reduce formation of mists, repair scrubbers, and improve efficiency of existing scrubber systems. Following completion of these steps, a similar program will be implemented for the hot-die-size pilot plant in Bldg. 306. An in-depth investigation of NO_x effluents from N Fuels processes is continuing and steps will be taken to reduce NO_x vapors there.

Effectiveness of all of the above actions will be measured by Hanford Environmental Health Foundation personnel who have moved their portable laboratory into the 300 Area and are currently collecting stack and ambient air samples.

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TECHNICAL ACTIVITIES - N REACTORRESEARCH AND DEVELOPMENTMission 1 - Basic Production1-A. Mark IV Fuel DevelopmentIrradiation Testing

The Mark IV demonstration fuel columns and the short-charged Mark IV production model columns are performing satisfactorily. The fuel swelling data from the discharged columns are shown below:

<u>Fuel Designation</u>	<u>No. of Columns Discharged</u>	<u>Avg Max Swelling Rate % V/1000 MWD/T</u>	
		<u>Outers</u>	<u>Inners</u>
Thick clad (94 Metal)	3	0.62	0.72
Base clad (94 Metal)	3	0.68	0.69
Base clad alloy 503 (94 Metal)*	3	0.54	0.29
Base clad (125 Metal)	9	0.52	0.42
Standard design	1	0.84	--

*Alloy 503 contains 250 ppm Fe and 250 ppm Al whereas the standard Mark IV fuel alloy (601) contains 400 ppm Fe and 800 ppm Al.

These data are based on exposures from 600 to 2500 MWD/T. The clad thickness appears to have little or no effect on the swelling rate. A significant reduction in swelling rate, especially for inners, is shown for the fuel consisting of uranium alloy 503. Further studies will define how this particular change in uranium composition affects fuel swelling.

The full length Mark IV columns charged last month are operating satisfactorily. The first discharge of one of the monitor columns is scheduled for September 1969.

Flow Loop Testing

Life testing of Mark IV fuel support systems is continuing in the flow loop:

<u>Fuel Type</u>	<u>Support System</u>	<u>No. of Elements</u>	<u>Loop Time, Hours</u>
26" Mark IV	End-spider	9	3500
26" Mark IV	End-spiders with one or two legs removed	5	2250 (465 prior to leg removal)
12" Mark IV	Spring-stop	6	300

The nine end-spider supported fuels have endured a total of 7.6×10^8 hydraulically induced cycles without any signs of fatigue cracking (a total of 1×10^9 cycles is the goal). Two of the five end-spider supported fuels have two legs removed on both ends, and inspection shows the inner elements to have become displaced about 30 mils in the downstream direction. Thus, end-spiders with two failed legs appear to have sufficient fatigue resistance but inadequate creep resistance.

Mark IV Heat Generation and Reactivity Effects

Calculations of the potential range in specific heat generation rates in a given flux and in reactivity versus variations in Mark IV fuel fabrication tolerances continue to indicate that the 40-column loading of Mark IV fuel currently in the N Reactor has little, if any, effect on the current relatively low reactivity excess.

Mark IV Physics Studies

Comment review of the physics section for the Mark IV Safety Analysis Report has been completed. In place of the 75 MW trip of the intermediate system, whose geometry would permit it to be affected by severe flux distortions, the limiting case is now assumed to be a high level trip due to an excursion from the 400 MW level during a hot startup. The necessary CLUMSY cases have been run to show that maximum fuel and water temperatures would not be excessive in event of such an excursion.

A calculational model is being developed to describe more precisely the N Reactor flux distribution as a function of enrichment level and spike ring loading. Apparently because of differences in the N horizontal and vertical lattice, and inadequately defined reflector effects, analyses using cell-averaged cross sections from the FLEX code have not been successful. The revised approach will utilize a two-part lattice to more adequately describe the graphite and void distribution and DCODE four-group cross sections. The flux distribution description is of particular importance for predicting the reactor flattening effectiveness with revised spike configurations such as the Mark IV-A design.

1-B. Zircaloy Process Tube Monitoring

Evaluation of process tube 2455 is continuing. The third section of this tube, which came from near the center of the reactor, has been cycled 290,000 times between 1,250 and 25,000 psi hoop stress without initiation of cracking. It may be that the 0.35-inch slot, 80 percent through the wall, is not sufficient to cause cracking under these stress conditions. The cyclic stress will be raised to 37,500 psi to expedite crack initiation so that critical crack length determinations can be completed in a reasonable time. In future tests with N tubing, milled slots of greater than 80 percent of the wall thickness may be used.

Hydrogen distribution in the tube wall of various sections of 2455 is also being determined to gain more information relative to the hydriding

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potential of N pressure tubing with continuing reactor operation. An interim report of the evaluation of tube 2455 has been written and is now being reviewed. It is expected that the evaluation of tube 2455 will be completed by May 1, 1969.

Process tube 0758, which was fabricated by Chase Brass & Copper Company (rather than Harvey Aluminum Company), was removed from the reactor on February 18 as part of a continuing program to determine the effects of irradiation on the mechanical and physical properties of N Reactor tubes.

The stress rupture testing equipment has been modified to permit evaluation of N tubing at 475 C. Initial test samples have been prepared so that tests can be started during the next report period. The first two tests will be run with as-received and hydrided (275 ppm) tube sections. Duration of the tests will be 10 to 100 hours.

Four N tubing sections have been hydrided and are being slotted by electro-jetting, rather than milling, for fatigue testing. Two of the sections will have slots 0.35-inch long by 0.010-inch wide through the complete wall thickness. These will be cycled to failure at maximum hoop stresses of 13,000 and 11,000 psi (reactor operating stress is approximately 10,000 psi). The balance of the samples will have slots electrojetted only partly through the wall thickness prior to testing. Tests will also be performed with these samples to determine if there is detectable leakage before the cracks grow to the critical size for unstable propagation.

1-C. Graphite Studies

Graphite samples for burnout monitoring, discharged recently from N Reactor channel 1648, are now being measured.

1-D. Corrosion Studies

Examination of Inconel 600 Steam Generator Tube

On January 23, tube 4AS R5L1 was removed from steam generator 4A for examination as part of the continuing evaluation of the performance of Inconel 600 tubes. The tube section, 27.8 feet long, extended from the tube sheet to two-thirds of the way around the end bend.

The evaluation program for the tube includes external and internal non-destructive examination to locate cracks or localized attack, measurements of film and crud deposits, and metallographic examination of critical regions. These regions include the tube-tube sheet crevice and high stress areas such as the tube-tube sheet weld, the tubing section just outside the tube sheet, tube support areas and bends. External and internal tube surfaces will also be checked for intergranular corrosion.

Initial eddy current measurements on the tube have shown no abnormalities. The entire examination should be completed by April 30.

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Inconel 600 tubed steam generator experience as of January 31 is summarized below:

<u>Steam Generator(s)</u>	<u>Operating Months</u>	<u>Calendar Months</u>
4A	27.4	56
6A, 6B	18.7	31
3A, 3B	5.6	8
4B	1.4	2

Evaluation of Decontaminants

A program for the corrosion testing of proprietary decontaminants has been started in the 1706-KER facilities in an attempt to identify materials with corrosion and activity redeposition characteristics better than those exhibited by Turco 4512-A, the inhibited phosphoric acid used for the N Reactor carbon steel piping decontaminations in 1967 and 1968.

Promising materials passing the screening corrosion tests will be tested for effectiveness of activity removal and retention. The final phase of testing, if any decontaminants justify it, will include corrosion testing of the N Reactor primary system materials, full size component decontamination tests, and multitube reactor decontaminations. Completion of the certification program for alternate decontaminating solutions is scheduled for mid-1970.

Mission 7 - Target Space Enhancement

7-A. End-Cap Flux Suppression Calculations

Data generated to date indicate that adding hafnium to the fuel end-caps to suppress the flux rise due to the Wilkins effect would probably be impractical. An end-cap containing 2 percent hafnium would significantly reduce power peaking at the fuel and end-cap interface; however, it would also reduce k-infinity by about 7.5 mk.

Mission 8 - Nuclear Safety

8-A. N Meteorological System

Acquisition and processing of meteorological data are continuing. "Flux Checks" errors introduced by the tape unit have been solved by programming techniques which compare data bits to those recorded 1 minute earlier; data bits which deviate by only a small margin from the earlier data are accepted despite "flux check" error indications.

8-B. Soil Studies

On February 7, a solution of 11 kg of sodium dichromate was injected into the 12-inch line to the N crib at the valve building. This injection was performed to enable timing of the flow between the crib and river bank springs, to determine whether any channeling is taking place. Samples have been taken of the spring and wells around the crib to determine ground water flow rates.

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ENGINEERING & TECHNOLOGY - N REACTOR

Decontamination of Steam Generators

Experience at N Reactor now includes three successful decontaminations of large heat exchangers. The method consists of the application of a hot alkaline permanganate treating solution for 3 to 4 hours, a thorough rinsing, then application of a special acid solution to remove the treated corrosion film. Final rinsing completes the process.

The process can now be considered proven, and future decontaminations of equipment contaminated by similar films can be planned with a high degree of confidence of success. Pre-decontamination exposure rates above 2000 mR/hr can be reduced to less than 10 mR/hr in heat exchangers. Slight modifications to the basic procedure should permit good decontamination of virtually any equipment contaminated by pressurized water reactor primary coolant, as long as proper access of the chemicals to the contaminated surfaces can be assured.

N Meteorological System Readout

Study work was completed on a digital display for the N Meteorological system which can be used at the central and at the remote stations to simplify calibration. A digital display unit has been designed and materials ordered for fabrication.

Ball Safety System

Recent failures of a number of ball hoppers at N Reactor to open fully instigated a series of tests to determine the flow rate of balls out of the hopper versus percent of gate opening. Ball drop tests were conducted on the test hopper in the 1705-N development laboratory. It was found that the optimum gate opening for maximum flow rate of ball out of the hopper was not the full open position, but instead about a 60 percent (1 3/4 inches) open position. There is a ten second reduction in drop time at the 60 percent open position versus the full open position. Following are the comparative drop times:

<u>Gate Opening</u>		<u>Drop Time, seconds</u>
<u>%</u>	<u>Inches</u>	
100	2 7/8	37.1 (avg. 10 drops)
60	1 3/4	26.7 (avg. 10 drops)
50	1 1/2	31.6 (avg. 10 drops)
35	1	Balls did not drop

Evaluation of the ball trip mechanism to determine the requirements for permanent modifications is continuing. A study is also in progress to determine if the existing trip mechanism can be eliminated. This possibility appears promising inasmuch as the ball drop times reported above were obtained by using the locking cylinder for the tripping function. (The trip mechanism was deactivated.)

Confinement System Reliability Analysis

Fault-tree modeling of the N Reactor Confinement System has been started. To date, modeling of the 105-N and 109-N fog spray systems is complete. The fog spray fault-trees for both buildings are broken into two sections: (1) failure of fog spray station to come on upon demand, and (2) failure of fog spray station to turn off upon manual command. Since the fog spray systems for the 105-N and 109-N Buildings are independent, with the exception of the control circuitry power supplies, two separate fault-trees are required. The fault-trees start at the pressure switch level and end at the station control butterfly valve. Four hundred and three "X" components have been used in the model.

Project Engineering - Reactor Facilities

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

ENGINEERING & TECHNOLOGY - FUELS & TARGETS

Precision Extrusion - Uranium Sleeves

After the conversion from Mark I to Mark IV as the standard fuel for N Reactor, a large number of Mark I outer uranium billets remained in inventory. It was found that these billets could be converted into Mark IV billets, and used to make Mark IV outer fuels, by inserting a thin-walled uranium sleeve in the billet ID to reduce the ID to the proper size.

Test coextrusions utilizing the sleeved billet design demonstrated that this type of two-piece billet yielded completely acceptable fuel with a metallurgical bond occurring at the resultant uranium-uranium interface within the fuel cross section.

The uranium sleeve components for this program were manufactured by extrusion directly to size utilizing a 6-inch OD by 2 3/4-inch ID by 20-inch long billet encased in 0.065-inch copper-silicon.

As-extruded surfaces were very good, and OD control was ± 0.007 -inch with 95 percent of all pieces falling within a ± 0.005 -inch tolerance. ID's were 100 percent controlled to within ± 0.005 -inch. Wall thickness variation averaged 0.007-inch, or 5 percent of the wall thickness, with a maximum of 0.028-inch. Ninety-seven percent of the pieces measured less than 0.015-inch variation. The above data were gathered from five extrusions (85 components).

Twenty-five extrusions have been made to date and approximately 300 of the sleeve components have been used to produce Mark IV outer tube coextrusions with no difficulties reported. An additional 400 sleeves will be required to convert the remaining Mark I billets to Mark IV.

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Zircaloy Component Manufacture

Significant progress was made in the development of an acceptable process for the onsite manufacture of outer Zircaloy cladding components. Four billets were pierced to enlarge the ID from 2-inch to approximately 5 1/2-inch, and one billet (pierced at an earlier date) was extruded to produce outer shells for the Mark I inner billet.

A most significant advancement in the area of Zircaloy piercing was the development of a method of recovering the piercing tip between extrusions so that the same tip can be used to pierce several billets in succession. The tooling differs from piercing tooling used earlier in two respects:

1. A traveling mandrel is used to guide the piercing tip.
2. An insert in the shear die is used to facilitate removal of the piercing tip from the billet ID following the pierce.

The billet is inserted into the 7.580-inch liner. The piercing tip and centering block assembly is placed at the rear of the billet. The 5-inch stem is then brought forward and the traveling mandrel inserted through the tip and billet and into the ID of the die insert. The piercing tip is pushed through the billet to within 1/4 to 1/8-inch of the die. The mandrel is then retracted into the stem and the main ram is backed off about 12 inches. The container is opened slightly and the die insert is removed (a length of pipe is shoved through the die backer to knock the insert out) and the container is reclosed. The main ram is then brought forward and the tip is pushed the remainder of the way through the billet and into the space vacated by the die insert. The pierced billet is removed from the liner on a sled affixed to the die carriage. A solid follower block is used to push the billet out of the container and onto the sled.

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IRRADIATION SERVICES

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FUEL TECHNOLOGYMixed-Carbide Fuel Development - United Nuclear Corporation

The two mixed-carbide test irradiation capsules were charged into test holes 0065 and 0074 at KW Reactor. Seven of the subcapsules are operating as expected except for three gas leaks in the in-reactor portion of the capsules. Two of the subcapsules are operating at a lower temperature than predicted.

MATERIALS DEVELOPMENTIrradiation Damage to Reactor Metals - Battelle-Northwest

A controlled-temperature capsule containing reactor structural material tensile specimens was discharged from a side General Purpose facility at KE Reactor.

An instrumented graphite sample boat containing several dozen reactor structural material tensile specimens was discharged from the E test hole at C Reactor. The materials had been irradiated for one year, and the tensile specimens operated at 1200 F during reactor operation.

One tensile specimen capsule was charged and one discharged from the Snout facility spider at KW Reactor.

Nuclear Graphite - Battelle-Northwest

Nine boats containing graphite samples were charged into the rod room end of the 4B test hole channel at KE Reactor.

Graphite Coating Studies - Douglas United Nuclear

Two capsules containing various grades of graphite were irradiated in the Snout facility at KW. This is a program to evaluate a process for deposition of siliceous coatings on graphite in-reactor.

ISOTOPE PRODUCTIONProduction of Potassium-40 - Battelle-Northwest

Sixteen capsules containing 100 grams of potassium carbonate were discharged from the magazine facility at KE Reactor. These samples, with more than one year of exposure, were irradiated to produce K-40 at about one percent abundance.

ROUTINE IRRADIATIONS

Seventy-one Quickie activation analysis capsules were irradiated in the K and C facilities for Battelle-Northwest.

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ADMINISTRATION - GENERALFULL COST RECOVERY

The letter of December 19, 1968 to the Atomic Energy Commission regarding overhead charges applicable to diversification work to be used in determining credits to the AEC was reviewed by various AEC-RL divisions, and an agreement to relatively minor changes was reached. Hopefully, formal AEC approval will be obtained early in March.

APPROVAL LETTERS

At the close of the reporting period, approval by AEC-RL was pending for 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. #1	Salary Structure for Exempt Salaried Personnel	February 12, 1969
ATD-130	Exempt Overtime, Appendix "B" Modification MO-13	December 11, 1968
ATD-156 Add. #4	DUN Education and Training Program - Tuition Refund Plan	November 13, 1968
ATD-175	Overhead Charges Applicable to Diver- sification to be used in Determining Credits to the AEC	December 19, 1968 (to be resubmitted)
ATD-176	Conduct of Employees and Consultants of AEC Cost-Type Contractors and Certain Other Contractors	December 20, 1968
ATD-178	Increase in Per Diem Rate, Appendix "B" Modification MO-22	February 11, 1969
ATD-180	Lay-Off for Lack of Work	February 24, 1969

FLUX RELIEF FOR SPECIAL IRRADIATIONS

The AEC's Division of Classification accepted the DUN flux tailoring proposal with a limitation to one percent of reactor volume. Topics implementing this action are included in the new Section 300 of the Hanford Classification Guide

and DUN-4700 (the Local Guide for Hanford Special Irradiations) is being revised accordingly.

FILE IMPROVEMENT WORKSHOP

Representatives of the National Archives and Records Services conducted the File Improvement Workshop in the Federal Building from January 27 through February 14. Sixty-five DUN employees completed the course and were awarded Certificates. General comments from the attendees indicated the course would be beneficial to them in handling and controlling paper work.

EQUAL EMPLOYMENT OPPORTUNITY

Plans for addition of minority group employees to the Company's payroll are still being held in abeyance pending a final decision concerning the future level of production reactor operations, and the lifting of the present freeze on employment.

SAFETY

No personnel radiation exposures exceeded operational control.

Month-end safety statistics were:

Disabling injuries - February	0
- Year to date	0
Days since last disabling injury	120
Man-hours since last disabling injury	1,220,000

EMPLOYMENT SUMMARY

DUN personnel totals and employee allocations as of January 31 and February 28 are shown in Appendix B.

APPENDIX A

PROJECT STATUS SUMMARY - REACTOR FACILITIES

<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	55	Equipment installation work on No. 1 KW boiler is 80 percent complete.
DAP-510, Discharge Chute Clearing Equipment - K Reactors	190,000	100	70	J. A. Jones work on fabrication is 95 percent complete.
DAP-516, Storage Building Addition - 105-KE & KW	142,000	5	0	No progress since January 17 when design was halted on account of production cutback possibility.
DAE-518, Effluent Radioiodine Monitor - C, KE & KW Reactors	100,000	94	0	AEC-RL has directed Vitro-HES to incorporate comments on the K Plant construction drawings, but do nothing further with C Plant drawings because of possible deactivation of C Reactor.
DCP-522, Modification of Reactor Coolant Crosstie Piping - 105-KE & KW Buildings	35,000 (interim)	75	0	Interim project authorization received-- fabrication and installation withheld. Removal of pipe, fittings, and valves from 190-H has been authorized to J. A. Jones. Piping design is complete.
<u>N Reactor</u>				
GCP-406, Improved Safety Platforms and Accesses - 100-N Area	300,000	100	93	Installation of reactor platforms in progress as outage time permits. Construction schedule approved January 16. Project revision to increase scope in progress.

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PROJECT STATUS SUMMARY - REACTOR FACILITIES (contd.)

<u>Number & Title</u>	<u>Authorized Funds - \$</u>	<u>Percent Complete</u>		<u>Status</u>
		<u>Design</u>	<u>Construction</u>	
GCE-408, W, C & D Elevator Safety Improvements - 105-N	90,000	100	30	Vitro-HES completed revised drawings for latest design concept. Drawings have been issued for comment.
GCP-411, Effluent Control Program - 100-N Area	1,780,000	100	2	Although bids have been received for Diesel Fire Protection, proposed revi- sions to the criteria and design may dictate cancellation and rebid probably about July 1. Bid opening for the Emergency Dump Tank is March 25.
DCE-519, Replacement of Bridge Crane and Hoist System with New Crane System - 105-N Storage Basin	269,000	40	0	Order-of-magnitude preliminary quota- tions were received by telephone, indicating that crane costs will be \$50,000 to \$90,000 higher than estimated. No written quotations have been received.

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APPENDIX BEMPLOYMENT SUMMARY
(with employee allocations)

	<u>1/31/69</u>	<u>2/28/69</u>
<u>CONTRACT PERSONNEL</u>		
<u>02 Programs</u>		
Douglas United Nuclear	1841	1814
Assisting other Contractors	<u>14</u>	<u>15</u>
Total - 02	1855	1829
<u>Other Programs under AEC Contract</u>		
Assisting other Contractors and WPPSS	54	60
Special Irradiations	<u>11</u>	<u>11</u>
Total - Other Programs	65	71
Total Contract Personnel	<u>1920</u>	<u>1900</u>
<u>COMMERCIAL ACTIVITIES PERSONNEL</u>	5	8
TOTAL FORCE	<u>1925</u>	<u>1908</u>

FEATURE REPORTBACKUP COOLANT SUPPLY SYSTEMS - B & C REACTORSINTRODUCTION

A fundamental concept in the safe operation of the Hanford production reactors is the maintenance of sufficient cooling water to the fuel channels such that the integrity of the fuel element cladding is always maintained. Major emphasis therefore has been placed on multiple cooling systems, high-level maintenance, and high reliability of primary, secondary and tertiary coolant systems for the reactors.

As-built, the primary coolant systems for the B and C Reactors consist of electrically powered pumps capable of conservatively supplying sufficient coolant for full-power operation. These pumps are equipped with massive flywheels capable of imparting sufficient energy to supply coolant during the shutdown transient following a reactor "scram" from electrical power loss.

The secondary coolant system of the reactors consists of steam turbine driven pumps designed to start and reach full power automatically in a time sequence such that heat removal capability significantly exceeds the heat generation rate. The secondary system steam turbines at C Reactor are in-line with the electric pumps, and are maintained hot and rotating at full speed with the primary system electric drive. The secondary coolant pumps at B Reactor are parallel with the primary electric driven pumps, and are maintained hot and rotating at about 15 percent of rated rpm.

The tertiary system consists of two 300,000 gallon elevated tanks at each reactor and a raw water system interconnecting all reactor plants. This third system was designed such that, in the remote event of loss of all power, electric and steam, adequate cooling water could still be supplied to the stricken reactor through automatically controlled valves and pumps from the elevated tanks and from other reactor plant sites. The raw water system originally encompassed four reactor plant areas, 100-B, D, F and H. With deactivation of the latter three, a new concept for the third coolant system was desirable for B and C Reactors.

During the resultant studies of B-C coolant backup system modification, it was determined that not only could increased reliability be obtained in the tertiary system, but the reliability of the secondary system could also be improved while achieving significant operating economies. This report discusses the B and C secondary and tertiary coolant systems following deactivation of the D, F and H plants, and the secondary and tertiary systems for C Reactor following the deactivation of B Reactor in February 1968.

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BACKUP COOLANT SYSTEMS - B & C OPERATING

Secondary System

An important feature of the secondary coolant system is the reliability and adequacy of the steam supply to the turbines which drive the pumps. The 100-B Area steam plant (Bldg. 184) and steam distribution system for the B and C plants are shown on Figure 1, appended. Building 184 contains four coal-fired, two-drum type, superheat boilers, each rated at 130,000 lbs. of steam per hour. With both B and C Reactors operating, three boilers were kept on the line and three were necessary to supply all required steam loads. The distribution system was arranged in the form of a loop. On sensing a reduction in steam loop pressure, all boilers accelerated from base steaming rate to full demand load.

Tertiary System

The raw water export system is diagrammed on Figure 2, appended. With its discontinuance, which occurred with the deactivation of D, F, and H plants, the required backup for the elevated tanks at B and C Reactors was attained by the installation of four diesel-driven pumps at the 182-B Building. These pumps were designed to start on failure of the primary electrical system, and to deliver coolant to the reactors through a check-valve system as the coolant pressure dropped below the pressure supplied by the Secondary System (if the latter failed concurrently). This diesel system and associated piping are shown on appended Figures 3 and 4. With two reactors operating, three diesels were required and all four were maintained in service.

The four diesel pumps drew their coolant supply from the 25,000,000 gallon reservoir at the 182-B Building, in which a 24-hour supply was maintained. In addition to this reservoir supply, a portion of the D to B export system was retained so that (within 24 hours) coolant could be pumped from D to B to replenish the reservoir. This part of the export system was kept in service until an independent diesel pumping system was installed at the 181-B river pump house in addition to the steam and electric driven pumps there.

BACKUP COOLANT SYSTEMS - C REACTOR ONLY

Secondary System

When B Reactor was deactivated, the major changes in the secondary coolant supply system at C were in the steam distribution system and the mode of operation. When both reactors were operating, the steam distribution system functioned as one large system with all three boilers supplying a common steam header. After B Reactor was shut down, two of the three boilers could supply the emergency steam demand. Three boilers were continued in operation, however, to improve the reliability of the secondary system. That is, boiler failure could be sustained without system failure. A study revealed that, if the steam system was split and operated as two separate and independent systems, one of the boilers could be deactivated and improved system reliability could still be attained. This would cause no loss of steam system adequacy and would

enable a considerable operating cost savings. Thus splitting the steam system into two independent loops required the closing of six main valves, as shown on appended Figure 5.

It was still considered credible that one boiler could fail during an electrical power outage. Therefore, it was necessary to assign the steam-powered emergency equipment to each loop such that either loop alone would maintain the reactor in a safe condition. These assignments were made as follows:

Loop A - The steam plant boiler feed water pump in the 183-B Building, the emergency filtered water pump in the 183-C Building, a steam-powered exhaust fan in the C Reactor Building, and four of the eight reactor process coolant steam turbines set for automatic operation in the 190-C Building.

Loop B - The turbogenerator in the 184 Building, a steam-powered exhaust fan in the B Reactor Building, a second steam-powered exhaust fan in the C Reactor Building, and the other four reactor process coolant steam turbines in the 190-C Building.

With the two loops set in this manner, if one should fail during a BPA power outage, the remaining loop is adequate for reactor safety. The elimination of one boiler from the secondary system resulted in a savings of about \$60,000 per year.

Tertiary System

With the deactivation of B Reactor, no significant change was made in the third coolant system other than reducing the number of diesel pumps at the 182 Building from four for the B-C case to three for C alone.

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BACKUP SYSTEM ADEQUACY

The adequacy of the C Reactor backup coolant systems is measured frequently. This is done by conducting tests simulating loss of primary pumps alone and loss of primary and secondary pumps simultaneously. These tests have demonstrated the present backup coolant secondary system flow is 200 percent of the flow required to meet the limiting reactor bulk outlet temperature, if both steam loops respond properly. The available coolant flow is 170 percent of the required flow should only one loop respond properly. The elevated tanks and diesel pumps, which comprise the tertiary system, have been demonstrated to make available 123 percent and 130 percent, respectively, of the flow required by the bulk outlet temperature limit.

Revision of the C Reactor coolant backup system has permitted elimination of approximately 20 miles of raw water export piping formerly required to supply coolant for the tertiary system. Splitting of the steam system has essentially resulted in providing two secondary coolant systems, either of which can provide adequate cooling. The addition of diesel pumps to the steam and electric facilities has further enhanced the fundamental concept of three independent coolant systems.

Figure.1. 100-B Steam Distribution System - As Built

Figure 2.
Raw Water Export System

Figure 3. Raw Water Diesel System

Figure 4. Elevated Tank & Raw Water Piping at B & C Reactors

Figure 5. 100-B Steam Distribution System - Split Loop
(all six valves shown are closed)