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DOUGLAS UNITED NUCLEAR MONTHLY REPORT

JANUARY 1970

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RICHLAND, WASHINGTON

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February 18, 1970

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JANUARY 1970

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SUMMARY

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REACTOR PLANT OPERATIONS

Production Statistics

	KE	KW	N
Input Production - Pu (KMWD)	109.4	94.6	56.8
- U-233 (Equiv. KMWD)	2.4	1.1	-
Time Operated Efficiency - %	89.5	77.4	50.9
Steam Availability to WPPSS - %	-	-	49.0

K Reactors

A maximum power level of 4000 MW was achieved at both K reactors, this power being the administrative limit associated with inlet water piping brittle fracture considerations. The single outage at each reactor was as scheduled for charge-discharge.

On January 12, AEC-RL confirmed that plans for placing one K reactor in standby status should be initiated. By month end arrangements had been made for deactivating KW on February 1.

N Reactor

Equilibrium operation was at the 4000 MW maximum authorized power level before and following the 12-day scheduled outage for refueling and maintenance. There were four unscheduled outages, including two initiated by automatic scrams. One such scram occurred when a third Zone Temperature Monitor controller tripped while one ZTM was in a tripped condition purposely due to ZTM system coverage limitations and a second ZTM had already tripped spuriously. The other automatic scram was due to a spurious Flow Monitor trip. The third unscheduled shutdown was manual, due to a 125-volt DC ground, and the fourth resulted from a manual scram initiated upon failure of the control rods to respond to normal operating control switch actuation.

Steam generator retubing work continued in Cell 1.

FUEL AND TARGET FABRICATION

Production Statistics (tons)

	For KE & KW	For N
Billets Extruded	-	19.5
Finished Fuel Produced	221.0	18.8

K Fuels

AlSi canning operations were increased from two lines to four on January 19 to better utilize existing manpower and funds in FY-1970. Project DCE-524 construction was discontinued on the Line 2 hot-die-sizing facilities in Bldg. 313.

N Fuels

Input and output production exceeded forecast.

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TECHNICAL ACTIVITIES

K Reactors

Calculations for expansion of the process standards system for K reactor tube power limits have been completed for the natural uranium loading and six-pump operation, and restricted top-of-riser pressure, which will provide a satisfactory degree of flexibility in reactor operation. Additionally, a sensitivity analysis of tube power limits to key assumptions and parameters is underway.

The possibility of increasing the K reactor advantage in producing plutonium with a low Pu-241 content by means of segmental discharge has been confirmed by current studies. An additional benefit resulting is a reduced throughput for a constant Pu-240 content.

NpO_2-graphite elements for the wafer growth analysis test were charged in early January for 25 days of irradiation. Designs for thin-annulus tubular elements (graphite core and water core) have been improved to allow the use of identical parts and fabrication methods for the tubular sections.

Calculations on yield and quality predictions for Pu-238 produced in thin-annulus elements indicate that the graphite-core element is the more efficient producer, but the water-core element will produce Pu-238 at 90 percent quality more rapidly. In-reactor testing to confirm these calculations is planned for March.

Technical bases for the oralloy irradiation have been completed, and the initial 14 columns of K7 elements are scheduled to be charged into KE Reactor during the wye-joint outage.

The investigation to explain pressure surges that have occurred in the emergency KE-KW coolant crosstie during diesel startups continued. It has been concluded that the presence of air in the crosstie would contribute to generation of pressure surges in any event, but that reduction of the diesel acceleration rate would significantly reduce the magnitude of the surges.

A full-crossheader, through-reactor decontamination test is scheduled for KE Reactor for the third week in February. Liquid samples have been removed from the trench, wells, and ponds and will be compared to post-decontamination samples to provide waste disposal data for a full-reactor decontamination.

The Zircaloy coupon samples in KW Reactor, which are being irradiated to investigate hydride mechanisms and the efficiency of case layer removal by the electrolytic process, will be discharged in February because of that reactor's deactivation. A ten-tube test to demonstrate the anodic stripping and electro-chemical etch processes in KE Reactor is being considered for March. The test will be used to demonstrate that case-removal arrests base metal hydrogen buildup and to determine if the final etch step is required.

There are indications that the increase experienced in rear-face radiation dose rates may be caused by an increase in Sc-46 concentration. Rear-face hardware films will be analyzed to determine if this radioisotope is a significant contributor, and other analyses of the inlet water and process chemicals are planned to determine the source of the parent scandium.

N Reactor

Engineering Specifications defining finished fuel quality as required at the reactor have been drafted as an integral part of the Quality Assurance Program for N Reactor fuel.

A presentation of the Process Surveillance Systems Study was made to AEC-RL on January 9, emphasizing the safety and operability benefits of the system.

An important breakthrough in the program to produce a high-expansion, long-lived foam for the noble gas removal study appears to have been realized. The highly effective formulation has qualities which are far superior to those obtained with the 150 others tested.

Production Test PT-NR-116, "Hot Unfired Standby Test," was completed. The final test included 15 successive light-offs on each Combustion Engineering boiler without failure. It was concluded that these boilers can safely be placed on the hot, unfired standby condition as a routine mode of operation while the Foster Wheeler boiler is in service provided certain maintenance steps are taken.

IRRADIATION SERVICES

Work continued on the development phase of the program for producing pure Xe-128 from iodine as requested by the Argonne National Laboratory. The target capsule containing 450 grams of elemental iodine, being irradiated in KE Reactor, is to yield about 150 ml of Xe-128 for the development separations work.

FEATURE REPORT

The 2750-ton extrusion press in Bldg. 333, which serves primarily for the coextrusion of Zircaloy-clad N fuel, has been demonstrated to be very effective also for the precision extrusion of various other products such as Zircaloy cladding components, Zircaloy process tubing, and uranium fuel cores. These extrusions and the excellent dimensional control obtained are described in the appended summary report.

GENERAL

Reductions in force resulting from the KW Reactor deactivation are scheduled to begin in mid-February. It is expected that these reductions, affecting about 385 DUN employees, will be virtually completed by June 30. As in the past, every effort will be made to assist employees made surplus by the shutdown to locate other employment, either on or off the project.

These force reductions and the associated hiring freeze will have a negative effect on the Company's Affirmative Action Compliance Plan.

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

for Charles D. Harrington
President

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REACTOR PLANT OPERATIONS - KE & KW

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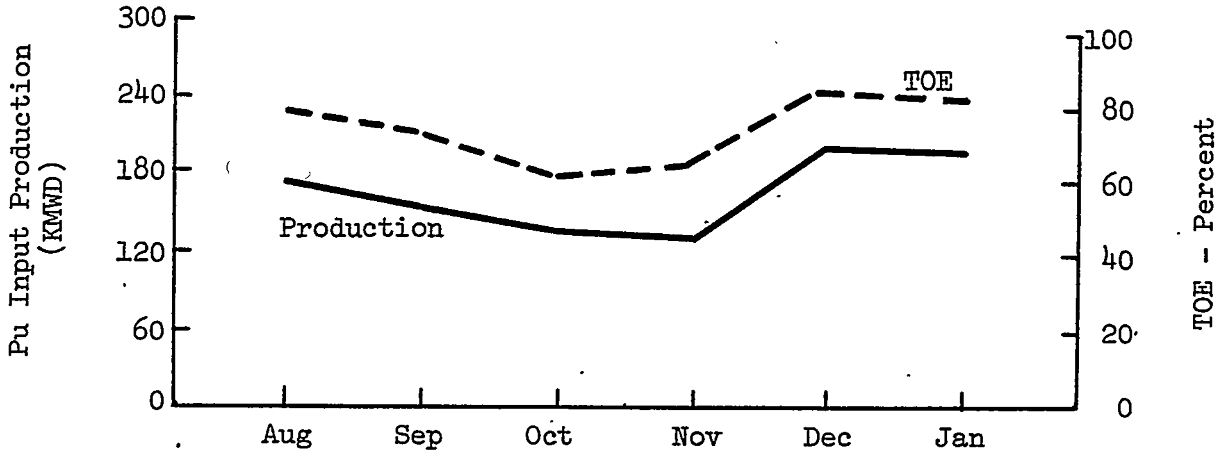
PRODUCTION

General

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



Word was received from AEC-RL on January 12 that the FY-1971 budget provides for the operation of only one of the K reactors, and that plans for placing KW in standby should be initiated. By month end, all arrangements for its shutdown early on February 1 had been completed.

Statistical Summary

	KE	KW	Total
Input Production - Pu (KMWD)	109.4	94.6	204.1
- U-233 (Equiv. KMWD)	2.4	1.1	3.5
Power Level (MW) - Maximum	4,000	4,000	8,000
- Average	3,944	3,946	7,890
Time Operated Efficiency - %	89.5	77.4	83.4
Number of Outages	1	1	2
Number of Startup Interruptions	0	0	0
Operating Coolant Flow - 1000 gpm	199.3	200.9	400.2
Fuel Charged (Tons) - 94 Metal	89.3	83.0	172.3
- Natural U	344.9	357.4	702.3

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	KE	KW	Total
Fuel Element Failures	0	0	0
Helium Losses - 1000 cu. ft.	96.1	108.6	204.7

OPERATING EXPERIENCE

Reactor Loadings

The front face map showing the very similar loadings in KE and KW is reproduced on the page following B-4. The tonnages listed are approximate; actual fuel charge totals are tabulated above.

Power Levels

The power levels at both K reactors were restricted by the 4000 MW administrative limit associated with inlet piping brittle fracture considerations.

Reactor Outages

Date Down	Reactor	Outage Hours	Remarks
January 1	KW	168.4	Scheduled charge-discharge.
January 14	KE	78.1	Scheduled charge-discharge.

EQUIPMENT EXPERIENCE

Wye-Joint Outage - KE Reactor

The KE reactor is scheduled to shut down on February 1 in preparation for the wye-joint replacement outage. J. A. Jones Co. construction forces are scheduled to start work February 2.

Emergency Diesel Water Lines - KE Reactor

Tests were conducted on the emergency diesel system to measure pressures generated in the system when the diesel pumps are started. It was concluded that no significant pressure differential is generated when the diesel starts.

Resistance Temperature Detectors - KE Reactors

An analysis of the scram tape obtained during the January 14 shutdown showed approximately 400 RTDs with response times greater than 3.0 seconds. It is now necessary to test these RTDs with the self-heat method to determine the exact response time. This work will be performed during the wye-joint outage and RTDs with response time greater than 3.0 seconds will be replaced.

[redacted]

Vertical Safety Rods

At KE Reactor the VSR drop times remain at 1.80 seconds and all VSRs are serviceable. At KW Reactor a knuckle rod was installed in channel 68 to eliminate the rod binding problem. VSR No. 49 was declared unserviceable because of binding experienced during attempted withdrawal of the rod for startup on January 8.

Aluminum Tubes

At KE Reactor 49 central zone aluminum process tubes were probologged. Eleven of these tubes were found to have thin walls and will require replacement. All central zone aluminum process tubes have been probologged within the last three months; 17 of these tubes have thin walls and will be replaced during the wye-joint outage.

10,000 HP High-Lift Motor - 190-KW

An inspection of the motor and windings was made on No. 4 high-lift pump at 190-KW. This unit had operated 19 months beyond the scheduled five-year inspection. No evidence of appreciable wear or damage was found to justify disassembly at five-year intervals. Based on this and previous inspections, consideration is being given to extending the disassembly frequency to seven-year intervals.

PROCESS ASSISTANCE AND CONTROL

Operational Physics

The thoria and supporting fuel in the PuAl test block was recharged during the KE Reactor outage. Four thoria and thirty-four 94 Metal columns were replaced with natural uranium, resulting in a tube power decrease of 25 to 30 percent in the PuAl columns.

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

	Reactor KE	Reactor KW
Effective Central Tubes*	2155	2203
Flattening Efficiency** - January	0.70	0.70
- 12-Month Average	0.72	0.71
Maximum Operating Time Permitting Scram Recovery - Hours***	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 the currently approved startup procedures.

K Reactor HCR Poison Considerations

The deactivation of KW Reactor will alleviate the problems of HCR replacement for the near future. Earlier studies had concluded that no KE Reactor rods would be expected to decrease to 90 percent effectiveness in the next two years. In the meantime, selected KW control rods would be made available for exchange if necessary.

Production Fuel Performance

The following table shows production fuel failure frequencies in the K Reactors, as number per million elements discharged, for the 12- and 24-month periods ended December 20, 1969:

	12 Months	24 Months
Natural uranium	120	4
94 Metal	7	7

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Zone	Tons KE	Tons KW	Material
Central	212	229	Natural Uranium
	66	61	94 Metal
Buckled	82	82	Natural Uranium
Near and Far Fringe	48	47	Natural Uranium
Shield Protection	23	22	94 Metal (Thoria Support)
	7	5	Thoria

Loading Pattern - KE & KW Reactors

REACTOR PLANT OPERATIONS - N

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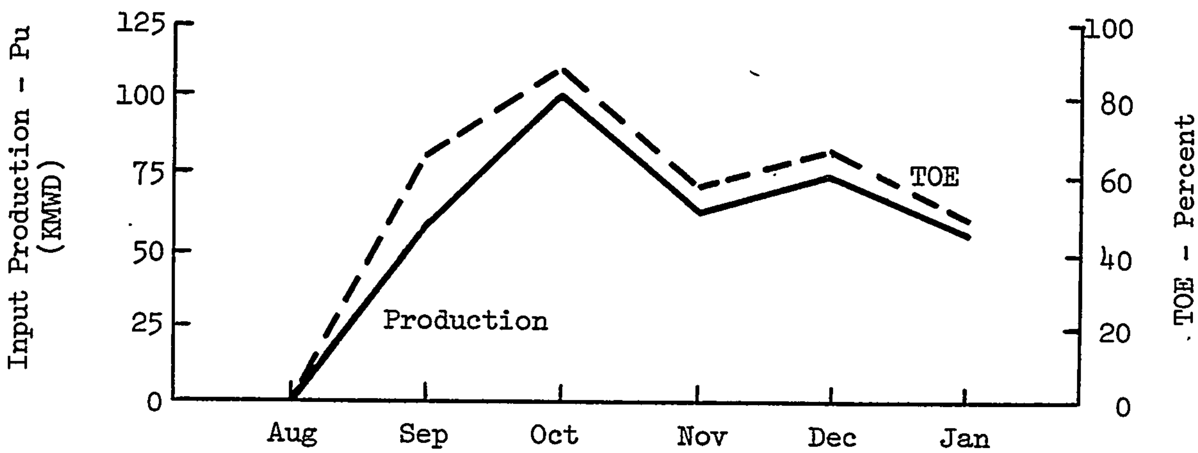
PRODUCTION

General

Reactor production (a combination of fuel grade and blend material), 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)		56.84
Electrical Generation (KMWH)	- WPPSS	271.64
	- 184-N	4.97
	- Total	276.61
Power Level (MW)	- Maximum	4,000
	- Average	3,602
Time Operated Efficiency - %		50.9
Steam Availability - %		49.0
Number of Shutdowns	- Scheduled	1
	- Unscheduled	4
Fuel Failures		0

Fuel Charge (Tons) - 94 Metal	321.96
- 125 Metal	65.35
- Natural U	0.37
- Total	387.68
Helium Losses - 1000 cu. ft.	359.73
Fuel Oil Usage - bbl.	19,984

OPERATING EXPERIENCE

Reactor Loading

The reactor loading at month end is shown on the front face map which follows page BN-6. A total of 211 fuel columns were charge-discharged during the scheduled outage which started on January 14. Eight of the freshly charged tubes were recharged because of high charging pressures observed during their initial charging.

Power Level

Reactor power level was administratively limited at 4000 MW throughout the month. Even though a shortage of reactivity existed at month end, it was possible to maintain main steam header pressure at approximately 123 psig throughout the month, thus enabling electrical generation by WPPSS at their station's design rating of 800 MWe.

Reactor Outages

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

Date Down	Outage Hours	Cause
January 2	26.8	Automatic scram when three Zone Temperature Monitor controllers tripped. One ZTM was in a trip condition purposely due to ZTM system coverage limitations, a second ZTM tripped spuriously, and (before the second trip could be cleared) a third ZTM trip occurred.
January 3	16.1	Manual reactor shutdown because of a 125-volt DC ground on circuit C-11. The ground was traced to a faulty limit switch in the door of filter cell A at 117-N.
January 14	291.9	Scheduled shutdown for charge-discharge and maintenance.
January 28	22.8	Automatic scram due to a flow monitor trip.

Date Down	Outage Hours	Cause
January 29	7.5	Manual scram due to failure of the control rods to respond to operating control switch actuation.

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EQUIPMENT EXPERIENCE

Primary Loop

The major maintenance activity on the primary loop was the continuation of leak repair efforts. Included were the repacking of fifty-four V-11 valves by the J. A. Jones Co., the replacement of three V-11 valves by plant forces, and the replacement of fourteen V-12 valves by J. A. Jones. Other significant leak repairs also were effected. Manual operators were installed on the repacked V-11 valves.

Primary loop leakage rate at month end was about 220 gpm, as compared to about 350 gpm at the beginning of the month.

The flow monitor trip which scrammed the reactor on January 28 occurred during trip adjustment following the startup from the scheduled outage. The flow monitor trip was reset before the trip identification was recorded, so the actual fault was not determined. A complete check-out of the system revealed several small toggle valve leaks, and two controllers were found in need of corrective action; it is not known whether any of these faults were responsible for the trip.

An oil leak noted on No. 2 primary coolant pump was caused by failure of the inboard bearing assembly. As a result, the labyrinth seal was damaged and a new mechanical seal was required. Repairs were made using parts from the No. 1 primary pump which is currently out of service.

Boiler Experience

Tube leaks experienced in the FW boiler on January 8, and in the CE-1 boiler on January 24, were repaired. Those in CE-1 have been concentrated in one small area at the bottom of the lower drum. Both CE boilers are now maintained on hot unfired standby during reactor operation in accordance with a newly issued Process Change Authorization.

The FW boiler now has 26 tubes plugged on the west end of the convection section.

Based on internal inspection and operating experience, the following renovation work is indicated for this boiler:

- Replace sixteen rows of convection section tubes on the left side. (288 tubes).
- Replace entire roof section of tubes (180 tubes).

- Repair expansion folds in the gas seal on the left side in the mud drum area.

- Repair gas leak in the steam drum area.

- Replace refractory of the firebox floor.

- Clean convection and firebox tubes on the fire side.

The estimated cost of this work is $240,000-$250,000.

Horizontal Control Rods

During the scheduled outage, it was confirmed that an out-of-service control rod (No. 34), with no cooling, had been left in the reactor following the November outage, and had been subjected to the subsequent full-power operation. This rod had been taken out of service during the July 1969 extended outage due to a coolant leak in the tip section, and plans had been made to remove the leaking tip and install a new one during the January 1970 outage. In the meantime the rod tip was inserted into the reactor during the November outage, to reduce radiation levels in the inner rod room, but was not withdrawn to the full-out position prior to reactor startup. Since the coolant was valved off, the rod became overheated during reactor operation and some melting occurred. Tip removal was accomplished, during the scheduled outage initiated January 14, by removing the step plug and retrieving the broken sections of the rod. A new tip was installed and the rod returned to service.

Investigation of the rod controls malfunction on January 29 revealed that breaker CX-29 had tripped open due to a failure of a surge suppressor on control relay 8K41. The opening of CX-29 interrupted the "in" - "out" control function from the operating console and de-energized the 3/4-inch limit switch.

Ball Safety System

The reactivity level in the reactor is lower than calculateions indicate it should be, and traveling wire flux monitor traverses indicate a dip in reactivity near the center of the reactor that closely resembles what a full ball channel would cause. However, an extensive ball channel test program during the scheduled outage failed to reveal any balls in the ball channels. It is known that some balls from ball hopper testing operations have leaked through separations in the ball channels and are lying along ledges in the graphite moderator. Ball hopper test drops were limited to selected channels to be inspected for channel separation (see also page DN-2).

Circulating Raw Water System

Vibration measurements made at No. 4 circulating raw water pump indicated a sharp increase in magnitude. During the scheduled outage, this pump unit was taken out of service to permit inspection. The B-bus supply power for No. 4 pump was transferred to the No. 3 pump unit to provide the two B-bus supplied pumps.

Confinement System

The Cell A filters were removed from service on January 2 because of low charcoal filter efficiencies. A laboratory analysis revealed a charcoal efficiency of 92.3% at 100 C and 100% relative humidity. Previous laboratory tests (June 20 and August 7, 1968) had shown an efficiency of 97.5%.

Flux Monitor System

The sub-critical and intermediate flux monitor systems functioned normally during the last reactor startup. During the January 14 outage, the faulty No. 2 intermediate chamber was replaced with an available Westinghouse prototype.

Equipment Modifications

The following equipment modifications were completed during the month:

EMP-256, "V-27 Valve Elimination and Piping Modification," authorized removal of the leaking V-27 valve from the DS (EHP) line in the 105-N Primary Piping System and defined the associated piping and electrical modifications.

EMP-382, "105-N and 109-N, Zone 1 Exhaust Smoke Detectors," provided two Pyr-A-Larm duct-type smoke detectors in the outlet of the 109-N Zone 1 exhaust fans. These new detectors were integrated into the existing 105-N Pyr-A-Larm System.

DC-3016, "Bracing for N Area Battery Racks," provided bracing and anchor bolts to these battery racks to strengthen them sufficiently to withstand a design basis earthquake.

DC-3038, "CRWP Safety Circuit Revision," provided a time delay of 0.4 ± 0.1 seconds in the CRWP two-out-of-three coincident safety circuit trip. This change will minimize the potential for an inadvertent reactor scram from a CRW low-flow trip.

DC-3048, "Revised Input Circuit for Intermediate Range Flux Monitors," provided a maintenance current of 2 x 10 amperes into the input of the intermediate range monitor by the addition of a precision resistor. This change increases the number of decades the system can follow the flux decay down following a reactor scram or shutdown.

MDC-N-69-61, "Helium Gas Meter Compensator," provided a helium gas meter which compensates for changes in gas pressure and temperature.

PROCESS ASSISTANCE AND CONTROL

Operational Physics

There is a significant flux depression in the vicinity of ball column 49, which affects the reactor all the way from top to bottom. It is not unreasonable to expect that an effect of around 2 mk may be caused by the observed flux dip, largely accounting for the difference between calculated loading status and observed reactivity of 2.5 to 3 mk, which has appeared since the November 1969 outage.

The excess reactivity available for control during the current operating run is very low, but is expected to suffice for operation through February.

Operation with uncooled rod 34 in the reactor caused the melting temperature of the internal aluminum to be exceeded and some boron carbide grit presumably was lost from the rod. However, rod calibration data taken on January 25 showed the shape and magnitude of the calibration curve for the new rod 34 to be consistent with the curves obtained for rods 33 and 37. Therefore, it was concluded that the amount of boron carbide grit lost from the overheated rod had been minimal, and that adjustment for shadowing of residual boron was not necessary for the newly installed rod.

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

Effective Central Tubes*	795
Flattening Efficiency* - January	0.79
- 12-Month Average	0.81
Maximum Operating Time Permitting Scram Recovery - Hours*	24

*For definitions see page B-3.

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

Fuel Code	No. Tubes	Description
CE	5	Mk-IC (94 Metal - Fringe)
D	2	Mk-IC (94 Metal - High U-236)
E	258	Mk-IC (94 Metal - Fringe & Central)
ES	60	Mk-IC (94 Metal - Fringe)
F	250	Mk-IV (94 Metal - High U-236)
G	182	Mk-IV (94 Metal - Central)
N	1	Mk-IB (Natural U)
X	230	Mk-IA (125 - 94 Metal)
	988	Total
	16	Total PTs
	1004	Grand Total

PT-NR No.	No. Tubes	Description
76	1	Fuel Meltdown Test
94*	10	Mk-IV Demonstration
01	4	Mk-I & Mk-IV from Direct-Cast Billets
[redacted]	1	Graphite Samples Channel
	16	Total PTs

*Includes Mk-IV high U-236 content fuel, 1 tube with Mk-IA 125-94 Metal, and 6 tubes with Mk-IV-AA 125 Metal and Mk-IV 94 Metal.

Loading Pattern - N Reactor

[redacted]

FUEL AND TARGET FABRICATION - K REACTORS

PRODUCTION

General

Production of AlSi-bonded fuel for the K reactors was 126.3 percent of forecast, reflecting the change from 2- to 4-line operation initiated on January 19 to better utilize existing manpower and funds in FY-1970. Eighty-four percent of these elements had self-supports or bumpers attached. Due to the scheduled deactivation of KW Reactor, construction was discontinued on the Line 2 hot-die-sizing facilities in Building 313 (Project DCE-524). Work remaining on HDS Line 1 is being completed.

Acceptable Elements Produced

		Yield - Percent	
	Tons	Current Month	FYTD
AlSi-Bonded Fuels	221.0	95.8	95.6

Month-End Inventories

		Tons
Bare Uranium Cores		973*
Finished Fuel:	AlSi-Bonded	1,063*
	Hot-Die-Sized	19
Thoria Elements		14

*These totals include 132 tons of bare cores and 110 tons of finished fuel of the sizes used in the smaller reactors.

OPERATING EXPERIENCE

Canning line efficiency of the AlSi-bonding lines was 98.6 percent. Downtime was assigned 36 percent to equipment malfunctions and 64 percent to operations. Only minor processing problems were encountered with the starting of two additional canning lines.

EQUIPMENT EXPERIENCE

Nothing significant to report.

PROCESS ASSISTANCE AND CONTROL

Nothing significant to report.

FUEL AND TARGET FABRICATION - N REACTOR

PRODUCTION

Input Production

Total billets extruded	125
Tons extruded	19.5
Percent of forecast	114.7

Output Production

Total finished fuel assemblies	756
Tons output	18.8
Percent of forecast	125.3

Uranium Utilization - %

Uranium Utilization - %	89.1

Month-End Inventories - Tons

Bare uranium billets	265
Finished fuel	238

OPERATING EXPERIENCE

Input production was above forecast to make up for production lost in December.

EQUIPMENT EXPERIENCE

Nothing significant to report.

PROCESS ASSISTANCE AND CONTROL

Two hundred and seventeen fuel assemblies which had exhibited white oxide on the surfaces of the fuel supports and end closure welds following autoclaving were irradiated under a Coextrusion Process Specification Waiver.

The oxide layer on these fuels varied in thickness and visual appearance; hence, prior to reactor charging, a detailed characterization of the specific degrees of oxide was made. Post-irradiation examination of these fuels indicated that no severe corrosion occurred. The corrosion which did take place was uniform, and no accelerated break-away corrosion in the form of pits or discontinuities in the Zircaloy surface was observed. Samples of these fuels which were retained in the fabrication shop can now be used as visual standards.

TECHNICAL ACTIVITIES - K REACTORS

RESEARCH AND DEVELOPMENT

DECLASSIFIED

Basic Production

Brittle Fracture Program - K Inlet Piping

Technical Bases for Tube Power Limits

Calculations for expansion of the structure for K reactor tube power limits have been completed for the natural uranium loading. Current efforts involve development of correlations to enable adjustment of limits for a range of void coefficients and reactor flattening efficiencies. The result will be a structure of limits that will provide a satisfactory degree of flexibility in reactor operation. Additionally, a sensitivity analysis of tube power limits to key assumptions and parameters is underway and should be complete by the time the correlations have been developed. At that time, final documentation of results will begin for the dual riser, reduced TORP (top-of-riser pressure), six-pump case for the natural uranium loading.

The computer program for automation of void-reactivity kinetics is being debugged and is scheduled to be completed by the first of February. The program then will be applied for development of tube power limits for the six-pump, increased TORP case. It is expected that, on completion of the next case, the analytical process for tube power limits will be routine.

The document presenting the void coefficient curves for K4 and K5 natural and enriched fuels used in establishment of dual riser limit cases has been prepared and is being issued as DUN-6634.

Segmental Discharge Studies

The possibility of increasing the K reactor advantage in producing plutonium with a low Pu-241 content by means of segmental discharge has been confirmed by current parametric studies. In addition, there is the advantage of reduced throughput for a constant Pu-240 content (about 11 percent decrease in uranium requirements for weapons-grade material). In 6 percent Pu-240 material produced in the segmental discharge mode, the Pu-241 content would be approximately 10 percent less than currently produced. In segmental discharge material irradiated to the same MWD/ton as presently done, the Pu-241 content would be approximately 25 percent below that in current weapons-grade plutonium.

Through-Reactor Decontamination

A Production Test permitting a full-crossheader, through-reactor decontamination at KE employing preheated decontamination solutions is routing for approval. Because of the wye-joint outage postponement, this test is now scheduled during the third week in February.

Predecontamination control liquid samples have been taken from the K Area trench, from wells between the trench and the river, and from seepage ponds. These samples have been analyzed for pH and radioisotope content. A sampling program has been established to obtain samples from the same locations following the decontamination in order to provide waste disposal information for a full-reactor decontamination.

Zircaloy Process Tube Hydriding

Zircaloy coupon samples are being irradiated in the KW Reactor to investigate hydride mechanisms and the efficiency of case layer removal by the electrolytic process in arresting hydrogen buildup in the K process tubes; these coupon samples have logged about 12 months of residence in the reactor. Originally, it was intended that they be irradiated until June 1970 to increase assurance that sufficient hydride buildup would have occurred in the samples to verify trends shown by the six- and nine-month coupons. It now is planned to discharge the coupons in February because of the announced reactor deactivation.

Conditions have been established for adapting a fluoride etch (0.5% HF) into the electrolytic hydride removal process for removing the last trace of zirconium hydride from the tube inside surfaces. A Production Test has been written for a 10-tube reactor demonstration of the case-hydride removal process in March at the KE Reactor. The ten tubes would then be used for (1) a practical demonstration that case removal arrests base metal hydrogen buildup, and (2) determining whether the final etch step is necessary to provide the desired protection.

Rear-Face Dose Rate Study

There are several indications that the increased rear-face radiation dose rate experienced over the past few years at the K reactors may be due to an increase in Sc-46 concentration. The Sc-46 to Zn-65 ratio increased from 0.1 to 1.0 in mid-1967, while the Zn-65 concentrations remained similar to previous years (the ratio is used to eliminate seasonal variations). In addition, recent data from the through-reactor decontamination tests show a three- to five-fold increase in Sc-46 over levels found on fuel elements and process tubes descaled in 1962 and 1966. Because of a high energy gamma emission and an 83-day half life, Sc-46 could be a major contributor to dose rates.

Analyses of rear-face hardware films are planned to ascertain if Sc-46 indeed is a significant contributor, and other analyses of the inlet water and process chemicals are planned to determine the source of the parent scandium. Test results already available indicate there was no increase in parent scandium in the Columbia River between 1962 and 1969. A comparison of effluent water analyses and water treatment changes indicates that the Sc-46 may be added during the flocculation steps.

Half-Plant Dichromate Test

Examination of the fuel elements exposed to coolant containing 0.5 ppm and 1.0 ppm sodium dichromate in a half-plant test in KW Reactor has been completed. Because of the large amount of data involved in this test, the computer analysis program will be modified to permit graphical comparisons and statistical analyses of the data.

Visual inspections verified earlier limited observations that at the higher temperatures corrosion was severe in both environments but was slightly worse at 0.5 ppm dichromate. A very pronounced dependence of localized corrosion on temperature was found. At 90-95 C tube outlet temperature, only uniform corrosion was observed. At 100-105 C, extensive shallow ledge corrosion on the fuel bottom lateral surfaces and slight erosion-corrosion at the bottom supports was observed. At 107-108 C, groove corrosion was commencing on the bottom lateral surfaces and most of the support crowns were severely corroded. Support corrosion increased from very severe at 112 C outlet to complete removal at 115 C outlet, and groove corrosion of the lateral bottom clad surfaces penetrated to the AlSi layer. Fuel exposure ranged from 850 to 1250 MWD/T during a total of 82 days of testing.

Product Flexibility

Pu-238

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Neptunium Irradiation

The eight elements for PTA-194, "NpO_2-Graphite Wafer Growth Analysis," were charged into a central zone column in KW Reactor on January 5 for a 25-day irradiation.

Designs for the graphite core and water core tubular (thin annulus) elements have been improved to allow the use of identical parts and fabrication methods for the tubular sections containing Np-237 target material. Only the end caps will be different to allow sealing of the graphite core element and free passage of water through the water core element.

The two neptunium-graphite samples sent to ARHCO for an independent determination of their Pu-236/Pu-238 weight ratio are scheduled for early count-rate analyses. Arrangements are also being made for independent analyses of NpAl elements to provide additional information on the success of the clustering technique on reducing Pu-236 contamination.

Calculations have been completed on yield and quality predictions for Pu-238 produced in thin-annulus elements. The calculations show the graphite-core element to be the more efficient Pu-238 producer, but the water-core element will produce Pu-238 at 90 percent quality more rapidly. These calculations will be checked by irradiating 800 grams of Np-237, to be equally divided between the two element designs. The in-reactor test is planned for March.

Analyses of target effectiveness with pin elements circumferentially located in a graphite core indicate this concept would also be acceptable from the physics standpoint. Calculated absorption rates relative to homogenized cores varied from 82 percent for a four-pin element to 91 percent for eight pins, with the ultimate limit of either multiple pins or an annular design approaching the homogenous core case in the range of 3 to 5 percent.

High Enrichment Utilization

Plutonium Utilization Studies

Submissions on utilizing fuel-grade plutonium for enrichment spiking on a short-term basis, and for an advanced driver fuel on a longer term basis, were included in a report prepared as ARH-AOP-18, "Plutonium Fueling at Richland - Interim Report."

Oralloy Test Planning

The reference document for Production Test planning has been completed as DUN-6618, "Technical Bases of K Reactor Oralloy Production Test," and preparation of the Production Test document has been started. Current plans call for charging of the initial 14 columns of K7 elements (3.75 wt% uranium in aluminum) into KE Reactor shortly before the end of the extended outage for wye-joint work.

Environmental and Regulatory Technology

Nothing to report.

Waste Management

Nothing to report.

DECLASSIFIED

ENGINEERING AND TECHNOLOGY - REACTORS

Pressure Surges in the K Reactor Emergency Crosstie

Analytical investigations have continued into the cause(s) of pressure surges in the emergency crosstie that have occurred on diesel startup, and to explore ways of alleviating the problem. An improved mathematical model has been developed that permits better simulation of the system. Peak crosstie pressures following diesel startup have been calculated as shown in the following table:

	Peak Pressure - psia		
Reactor	Instantaneous Acceleration	10-Second Acceleration	20-Second Acceleration
KW	654	330	220
KE	666	250	180

These calculations assumed the presence of air in each end of the crosstie and a linear acceleration rate of the diesel-powered pumps. It is concluded that air in the crosstie would contribute to generation of pressure surges in any event, but that reduction of the diesel acceleration rate would significantly reduce the magnitudes of the surges. A peak pressure of approximately 410 psia would be required to lift the A-B riser check valves of an operating reactor.

Thus, the calculations show that severe water hammer caused by opening and sudden slamming of the check valves during diesel startups could be averted by retarding acceleration of the diesels. Experimental verification of the results is in progress, and the feasibility of controlling diesel acceleration is being investigated.

Project Engineering

The status of construction projects relating to K reactor facilities is summarized in Appendix A. The impact of the scheduled KW deactivation on these projects has not been fully evaluated. As the attendant decisions are formalized, appropriate entries will be made in Appendix A.

ENGINEERING AND TECHNOLOGY - FUELS AND TARGETS

Nothing to report.

DECLASSIFIED

TECHNICAL ACTIVITIES - N REACTOR

RESEARCH AND DEVELOPMENT

DECLASSIFIED

Basic Production

Process Surveillance Systems Study

A presentation was made to AEC-RL on January 9. Justification for and desirability of the system were reviewed, with emphasis on the safety and operability benefits. Also presented was a review of reactor applications of digital process control systems in the United States, the United Kingdom, and Canada. Principal conclusions included:

- All large United States power reactors scheduled to go on line in the early 1970s will employ process computers.
- Canada and the United Kingdom both employ active as well as passive digital control of power reactors.
- Of the large production reactor installations in the United States, only Hanford does not apply passive or active digital control.

Primary Coolant Control System Simulations

DUN-6560, "N Reactor Primary Loop Coolant Shrinkage," was issued. Comparisons of shrinkage data from four recent scrams with data reported for Phase I operation are made. Shrinkages utilized in the simulations include the maximum observed during recent scrams as well as the maximum observed during reactor operation.

All-Tube Temperature Monitor System

DUN-6632, "Reduction of Exposure and Costs at N Reactor through Replacement of RTDs with Thermocouples," has been completed and issued. This report shows that, when general performance of resistance temperature detectors (RTDs) and thermocouples is compared in similar environments, the RTD failure rate is 2-1/2 to 8 times as high as that of the thermocouples. This difference is due to the delicate construction of RTDs and the susceptibility of their measuring circuits to grounds.

Graphite Distortion

Channel 1648 was borescopically examined. The channel appears to be in good condition, with no evidence of graphite fractures. However, there is some separation between the trunnion blocks and tube blocks in the upstream and downstrean ends of the channel. Also, the position of the adjacent ball channel liner blocks indicates that the channel has some degree of bowing and/or there has been some contraction of the liner blocks.

Ball channels 22, 36, 49, 50, 59, 61, 62, and 68 were examined to determine whether the horizontal and vertical movement observed at the upper channel block junction ($\sim$12 feet from top of unit) in channel 50 in November was generally present throughout the stack, and whether the distortion of channel 50 had increased. The distortion appeared limited to the nine centrally located channels. No changes from the original positions were noted in channels 36, 62, and 68. Each of the other channels exhibited a vertical separation at graphite layer 76, about 138 inches down from the unit top. Gaps as wide as one inch were observed. A number of safety balls were seen to be resting on ledges within the separations in channels 59 and 61. The only other abnormality found was a slight liner separation at the bottom of channel 22.

A series of tests with a ball channel mockup was initiated to evaluate the amount of ball leakage which could occur through vertical separations of varying severity.

Product Flexibility

Thulium Activation (PT-NR-89)

Preliminary calculations with the computer code YSOGEN yielded results which compare favorably with observed calorimetry measurements. Agreement for activation of the three thulium samples, corrected for the 475-day decay prior to measurement, averages within 3 percent.

Environmental and Regulatory Technology

Metal-Water Reaction and Fission Product Release Studies

The fifteenth metal-water reaction test with irradiated N Reactor fuel was performed. This test (SNH-15) was the third under molten uranium conditions and was the first in which the uranium of an outer element was molten in steam. Hydrogen evolution from the Zircaloy-steam and uranium-steam reactions was measured for 7 minutes. The test was then terminated due to failure of a joint in the quartz furnace housing the element. A large quantity of uranium was extruded upon clad failure, exposing considerable uranium surface area for reaction with the steam. The hydrogen evolution rate was the highest measured to date.

Detailed data from all tests to date are being integrated into an interim report to be issued in the near future.

Cesium Ratio Studies Affecting Meltdown Analyses

Attempts to reconcile the Cs-137 to Cs-134 ratio, reported from irradiated fuel meltdown test observations, with yields calculated with the RIBD code have not been successful to date due to the expected uncertainty of cross sections involved. Cesium ratios from the recently sampled batches of N Reactor fuel will be compared with the meltdown test data following completion of chemical analyses.

Meteorological Studies

The N Reactor meteorology system has been operational for the past month. Some difficulties developed with the tape, but a readjustment of the recorder flux-check amplifier boards and a strict tape head cleaning program alleviated the trouble. Wind sensor icing also occurred recently.

The software program has been reevaluated and a schedule for completion and cost estimate is being made.

Twenty-one months (January 1968 through September 1969) of data from all field sites, including N Reactor, have been analyzed and summarized. From these data, joint frequency tables will be generated for:

- Wind speed versus wind direction at a site.
- Wind direction versus time of day for a site.
- Wind direction versus wind direction for two sites.
- Wind speed versus wind speed for two sites.

Illustrations of streamlines at ground level for three times during the day for 60 days are being prepared.

I-131 Dilution Studies

A report has been issued summarizing field work last summer when I-131 released from a fuel rupture at N Reactor was traced from the 1301-N crib to the river bank springs. The conclusion of the study is that the minimum travel time of I-131 from the crib to the river was approximately 9 ± 1 days, and the maximum concentration appeared 15 ± 1 days after release to the ground.

Noble Gas Removal Study

All phases of this study are progressing approximately on schedule. Full-scale development tests will begin in February on both the cryogenic and foam encapsulation modes of removal. A background and status report on the study, "Progress Report for October-December 1969," DUN-6653, is being prepared.

In the period of a year, many formulations employing some rather unique materials have been tried and have met with limited success. In the quest for a long-lived high expansion foam, some 150 formulations were tested which lasted from a few minutes to several months. These are now available depending upon the needs of the user.

However, recently an important breakthrough in the area of producing a high expansion, long-lived foam appears to have taken place. A high degree of success has been achieved with a formulation consisting of:

Two parts polyethylenimine

One part polyvinylalcohol

Two parts surfactant or foaming agent

Five parts water

This formulation mixes readily with water, and no separation of the solution has been noted after 30 days. Thus, storage of the mixed formulation appears practical over a long period. Generation of foam via the use of a flooded diffusion plate with expansion ratios of up to 300:1 while encapsulating the total air flow has been successful.

Foam degradation appears linear from the top of the container or exposed layer at the rate of about 1 inch per week. The foam recently generated into clear plexiglass containers shows no internal breakdown in the structure of cells such as is common to formal firefighting foams leaving only a skeleton effect.

The foam exhibits a two-thirds solution run-off in a few hours, which does not appear to affect the foam. It is suspected that the good fluid flow properties exhibited by the foam when initially generated are in part due to the excess solution carried with the bubbles during generation.

The solution run-off has been collected and additional foam produced exhibiting the same qualities as the initial product. This would indicate that no specific ingredient has been removed from the formulation and retained by the bubbles. The bubbles, when pricked with a sharp point, do not burst as a normal soap bubble does. Rather, a very slow collapse takes place as the contained gas escapes through an ever-widening hole originating at the point of rupture. The bubbles appear to set into a sticky semi-plastic state as time progresses.

Because of the observed nature of the foam, the encapsulation of iodines and iodine products as methyliodide appears to be quite feasible.

Waste Management

Nothing to report.

ENGINEERING AND TECHNOLOGY - REACTOR

CE Boiler Light-Off Tests

Production Test PT-NR-116, "Hot Unfired Standby Test," conducted to determine and develop the light-off capabilities of the Combustion Engineering boilers from a hot, unfired condition was compelted after the test sequence was repeated twice. The final test included 15 successive light-offs on each boiler without failure. Two previous attempts to qualify the boilers failed but served to identify fundamental problems with the ignitor lighting sequences.

It was concluded that the CE boilers can be safely placed on the hot, unfired standby condition as a routine mode of operation while the Foster Wheeler boiler is in service provided certain precautions are taken: (1) cleaning out the pressure taps and lines to pressure switch PS-14 at least once each month; (2) cleaning the in-service ignitors at least every three days; and (3) keeping the spare ignitor ready for immediate installation and light-off.

Circulating Raw Water System

Results have been received from BNW on the CRW system piping void measurements for Production Test NR-125, Supplement 2, "A-B Electrical Bus Hydraulic Interdependence Test." Since analysis of the results is just underway, only qualitative statements can be made concerning the test.

It appears that air coming out of solution between the condenser and the RWRV-102 valve is not the problem. Some voids were detected downstream of the RWRV-102 valve and are being considered cavitation voids. Although voids were detected at almost all valve positions, they were more abundant when the valve was almost shut. If the voids were caused by air inleakage, the void fraction should not have changed as it did. The instability previously seen during operation and earlier testing could not be reproduced during PT-NR-125, Sup. 2.

Only two conditions are known to have changed since PT-NR-125, Sup. 1: (1) the river temperature dropped about 18 F, and (2) tube leak repair was done on the surface condenser during which a major tube leak was eliminated and other tube plugs were upgraded. Present plans are to test again with the tube leak restored to its former status and with warmer river temperatures.

Further investigation of the cause of the circulating water low pressure trip on December 4 showed:

- Air in the sensing lines to the differential pressure gauges across the Duo-Chek valves changed the effective trip setting from the specified 13.25" H_2O to 16.0" H_2O.
- The actual differential pressure of the operating pumps (Nos. 1, 2, and 4) were 30", 27", and 26". These low pressures resulted from low river level (378.5') and system throttling (PCA 195-69).
- CWR system cycling was observed on December 26 which showed dump condenser No. 12 to be particularly sensitive to system upsets.

PT-NR-125, Sup. 3, will be conducted during the next scheduled outage to reproduce this cycling with instrumentation on the CRW system to monitor the critical parameters.

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V-11 Valve Studies

A summary of the various proposals for eliminating V-11 valve leaks was documented in DUN-6500, "Engineering Study, V-11 Valves." This report also includes comments relative to costs, exposure requirements, and manpower involvement. Analysis of this information indicates two chief alternates for consideration; namely, replacement of the valve with a spool piece, or a repacking program for the in-place valves.

Replacement of the V-11 valves with spool pieces entails either the fabrication of a special spool piece, or modification of the in-place valve to enable it to perform functionally as a spool piece. A rather high cost would be involved in fabricating and replacing the valve with a special spool piece. On the other hand, modification of the in-place V-11 valve for use as a spool piece, while less costly, would require considerable personnel exposure. Also, either approach would result in abandonment of the design function of the valve. Accordingly, it was decided to undertake the valve repacking program.

Power System Transient Stability Computer Program

In the past, periodic stduies have been made of the reactor electrical power system with regard to stability during transient fault conditions.

There is a need to update these power transient studies in view of the changes to the in-plant system loading due to reactor deactivations. Previous studies were accomplished by renting the use of a network analyzer. The current method for making this analysis is by digital computer means. Since a commercial computer program is not available for this analysis, a digital computer program has been written for use in analyzing a complex power system where reactor coolant pumping is involved. This new program is called "POWER". Final testing is now being scheduled.

Process Water Conservation During N Reactor Charge-Discharge

Document DUN-6516 with the subject title has been issued. The report summarizes the process reasons which cause the water loss, the costs involved, and the occasional delays in charge-discharge work due to shortages of make-up water. The document reviews a number of alternates aimed at conserving water and provides recommendations. It concludes, however, that under current operating conditions, no changes are justified.

Project Engineering

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

ENGINEERING AND TECHNOLOGY - FUELS AND TARGETS

Fuel Engineering Specifications

Engineering Specifications have been drafted as an integral part of the Quality Assurance Program for N Reactor fuel. These specifications define finished fuel quality as required at the reactor; if they are violated, the fuel is rejected. If fuel fabrication process specifications are violated, the fuel may be accepted, provided the finished fuel quality meets the requirements of the Engineering Specifications. As a safeguard to fuel quality, this system of specifications also provides for review of all process changes to determine whether or not these changes would violate the Engineering Specifications.

Uranium Billet Fabrication

The upset forge and direct cast processes have been under evaluation as alternate billet fabrication routes in an effort to increase uranium utilization and decrease costs. Development efforts have shown that upset forging is now to the point that a percentage of the billet deliveries are being manufactured by this process and direct casting is now ready for further increased billet deliveries. A review of the work performed on both processes is discussed below, together with a projection for future work on these programs.

Upset Forging

In January 1969, 325 Mark IV billets were requested for delivery during the first three quarters of CY-1969. After initial slow deliveries, RMI completed delivery on time. Although a number of billets created canning problems that were caused by a billet ID bow, recent work has shown that the material can be canned and coextruded the same as normal production material. To date, approximately 160 of these billets have been successfully coextruded and no abnormal manufacturing problems have been associated with the material.

A total of over 900 inner and outer upset forge elements have been successfully irradiated with no failures. Data collected from pre- and post-irradiation measured columns of upset forge material also confirmed that swelling rates were comparable to those of standard material.

Based on favorable coextrusion and irradiation data, it has been decided that upset forging is now a process comparable to the standard process. To enable further work on upset forge material, 250 such billets have been requested for delivery during the January-July 1970 period.

Direct Casting

During the past year, 18 outer and 22 inner direct cast billets were delivered to Hanford for evaluation. Due to a high incidence of bleeding pits found in the billets during pickle inspection, this material has not been coextruded. These billets will require additional work at Hanford to remove the bleeding areas before they can be processed.

Twelve columns of outer fuels extruded from direct cast billets received in 1968 have been under irradiation. Ten of these columns have just been discharged, and the remaining two columns will be irradiated to approximately 4000 MWD/T. Of the 10 columns discharged, one was pre-irradiation measured and will be measured for swelling and dimensional changes. This work is scheduled for completion by the end of February.

Recent discussions with NLO indicate that the direct cast process is now ready to produce acceptable inner and outer billets by a consistent manufacturing process. In order to further evaluate direct cast material, about 64 outer and 44 inner direct cast billets will be ordered for delivery by the end of FY-1971.

IRRADIATION SERVICES

PRODUCTION OF Xe-128

The vacuum system, including the separation and purification equipment, has been assembled for the Xe-128 development and production work requested by Argonne National Laboratory. Final preparations are being made to start trial separation runs using natural xenon and elemental iodine.

The target capsule containing 450 grams of iodine is still undergoing irradiation to obtain approximately 150 ml of Xe-128 for the development separations work.

HIGH TEMPERATURE STRUCTURAL STEEL IRRADIATION

The Naval Research Laboratory, Washington, D. C., has authorized $17,100 to design, fabricate, and irradiate capsules containing structural steel charpy impact specimens. The capsules will have flux tailoring shrouds and a controlled temperature environment. The data obtained from this test are aimed at evaluating the spectral effect on the neutron damage to structural steel used in civilian power reactor pressure vessels. The capsules will be charged into KE Reactor this fiscal year.

ROUTINE IRRADIATIONS

One $(U\text{-}Pu)O_2$ fuel development capsule was irradiated in the KW Reactor Snout facility for BNW in support of the LMFBR program.

Sixty-three activation analysis capsules were irradiated in the K reactor Quickie facilities for BNW.

Six cold tensile specimen capsules were irradiated in the 4C Snout facility at KW Reactor for BNW.

DECLASSIFIED

ADMINISTRATION - GENERAL

EMPLOYMENT

KW Reactor Deactivation

As required for implementation of the AEC's decision in mid-January to deactivate KW Reactor, and the plans for its shutdown on February 1, reductions in force have been scheduled to begin in mid-February. It is expected that these reductions, affecting about 385 DUN employees, will be virtually completed by June 30.

As in the past, every effort will be made to assist employees made surplus by the shutdown to locate other employment, either on or off the project. First consideration will be given to excess DUN employees by other Hanford contractors. Other firms, including DUN's parent companies, that may have openings which laid-off employees might appropriately fill are being contacted so that employees can be referred to them. In addition, opportunity will be given to employees in excess categories to elect voluntary layoff.

Affirmative Action Compliance Plan

The hiring freeze and layoffs resulting from the KW deactivation will have a negative effect on the Company's Affirmative Action Compliance Plan.

The only two new-hire additions to payroll this month were American Indians from the Yakima Indian Agency who were added prior to the KW shutdown announcement. Contacts with the Agency in an effort to obtain candidates for the DUN Cooperative Education Program had been unproductive.

One of the two new Cooperative Education students this school year dropped out of the Program this month, leaving two students remaining in this Program of the four who started.

Statistical Summary

DUN personnel totals and employee allocations as of December 31, 1969 and January 31 are shown in Appendix B.

APPROVAL LETTERS

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

Number	Subject	Date of Transmittal to AEC-RL
AP-39	Pension Plan portion of letter entitled "Pension Plan, Salaried Savings Plan, and Wage Savings Plan"	January 12, 1966

Number	Subject	Date of Transmittal to AEC-RL
ATD-123 Add, #3	Subscriptions - Fiscal Year 1970	December 9, 1969
ATD-180	Lay Off for Lack of Work Appendix "B" Modification MO-23	February 24, 1969

NEW PAYROLL SYSTEM

A detailed review is in progress on converting the existing DUN 9 PAC Payroll System to a COBOL Payroll System. It is estimated that a cost savings of $60,000 per year in computer and related costs may be realized. Economies in clerical effort and system maintenance also appear likely. An additional advantage would be the new system's compatibility with computers available offsite in the event of a local computer failure.

ADEQUACY OF OPERATING CONDITIONS

The response to a request from AEC-RL for information on alterations to the production reactors and supporting facilities, as needed to maintain safe and reliable operating conditions, was submitted on January 16. The document contains data for FY-1970 through FY-1972, and is designed to accompany the five-year capital plan for FY-1972 through FY-1976.

CONFERENCE PAPER

A paper entitled "Descriptions and Uses of a Critical Systems Data File for Nuclear Plants" was accepted for the March 16-18 Plant Engineering and Maintenance Conference, sponsored by the American Society of Mechanical Engineers. The paper describes the computerized recording, auditing, and retrieval of test and inspection data for equipment critical to the safety and performance of Hanford production reactors. The system, which facilitates reliability improvements of critical systems, has been in use for six months and can be applied to any other nuclear of similar plant complex.

SAFETY

No personnel radiation exposures exceeded operational control.

Month-end safety statistics were:

Disabling injuries - January	0
- CY to date	0
Days since last disabling injury	178
Man-hours since last disabling injury	1,550,800

APPENDIX A

PROJECT STATUS SUMMARY - REACTOR FACILITIES

AUTHORIZED PROJECTS

Number & Title	Authorized Funds - $	Percent Complete Design	Percent Complete Construction	Remarks
Single-Pass Reactors				
DCE-505, Boiler Control Improvements, 165-KE & KW	410,000	100	97	Project proposal Revision 3 has been prepared and is being routed for internal approvals. It requests completion date extension to April 1 to allow time to finish exception items.
DAP-510, Discharge Chute Clearing Equipment - K Reactors	220,000	100	92	Design revisions have been submitted to AEC-RL Project Engineering for approval. Installation at KE is scheduled to start on February 10.
DAP-516, Storage Building Addition - 105-KE & KW	142,000	100	0	Scheduling of construction has been delayed by the freeze on funds.
DAE-518, Effluent Radio-iodine Monitor - KE & KW Reactors	100,000	100	99	Acceptance testing is in progress.
DAP-526, Deactivation of Hanford Production Reactor C	75,000	97	44	Fire alarm and protection drawings have been reviewed and comments returned to Vitro/HES.

Number & Title	Authorized Funds - $	Percent Complete Design	Percent Complete Construction	Remarks
N Reactor				
GCP-406, Improved Safety Platforms and Accesses - 100-N Area	300,000	100	75	Access ladders in the inlet and outlet pipe spaces at the minus-16' and minus-24' levels were installed January 20. Work is continuing on the 109-N roof.
GCP-411, Effluent Control Program - 100-N Area	1,830,000	100	62	Section II - The 30" Lasker pipe was received January 7, completing the order. Section IV - Held in abeyance. Section V - Welding of the tank structure is essentially complete. Installation of distributor piping, hydro and leak tests, pour retaining wall, paint interior, and insulate exterior are major items to be completed.
DCE-519, Replacement of Bridge Crane and Hoist System with New Crane System - 105-N Storage Basin Area	269,000	94	0	The Record of Purchase for cranes was transmitted for AEC-RL approval January 8.
DCP-528, Fire Protection System Improvements - 100-N	40,000	0	0	An interim authorization for design in the amount of $40,000 was received December 30.

PROJECT PROPOSALS AWAITING AUTHORIZATION

Number & Title	Funds Requested	Date to AEC-RL
GCP-411, Rev. 2 - Effluent Control Program - 100-N	$2,010,000 (new total)	9/24/69
DCE-519, Rev. 2 - Replacement of Bridge Crane and Hoist System with New Crane System - 105-N Storage Basin Area	400,000 (new total)	7/10/69
DCP-525, Fire Protection Improvements - KE	225,000	5/2/69
DCP-527, Graphite Cooling & Fog Spray - N	97,000	5/9/69
DCP-528, Fire Protection System Improvements - 100-N	290,000	6/4/69
DCP-529, Gravity Drainage System and Disposal Basin - 100-N	200,000	7/10/69
DAP-530, Upgraded Electrical Services and Lighting - 1100-N and 1101-N Buildings	78,000	10/2/69
DAP-531, Establish 1102-N Office Building - 100-N Area	45,000	8/19/69
DCE-532, Isolation of Process Coolant Risers - K Reactors	270,000*	9/25/69
DCP-535, Sodium Sulphite System - 182-K Building	25,300	1/15/70

*Additionally, this project requests the use of $240,000 of operating funds.

PROJECT PROPOSAL PREPARATION

Number & Title	Design Criteria	Project Proposal
DCE-519, Rev. 3 - Replacement of Bridge Crane and Hoist System with New Crane System - 105-N Storage Basin Area	Completed	In preparation. Cost is now estimated at $465,000 on the basis of a recent crane bid and more complete design.
DCP-528, Rev. 1 - Fire Protection System Improvements - 100-N	Completed	Completed - in approval status (addition to scope).
Export Water System Backup - 181-D (for 200 Areas)	Engineering Study is essentially complete	In preparation.
Stack Monitoring Improvements - 100-N	Revision complete - in approval status	Awaiting approval of revised design criteria.
High Pressure Injection and Seal Water Improvements - 109-N	In progress	

APPENDIX B

EMPLOYMENT SUMMARY
(with employee allocations)

	12/31/69	1/31/70
CONTRACT PERSONNEL		
02 Programs		
Douglas United Nuclear	1602	1594
Assisting Other Contractors	14	13
Total - 02	1616	1607
Other Programs Under AEC Contract		
Assisting Other Contractors and WPPSS	44	44
Special Irradiations	11	8
Total - Other Programs	55	52
Total Contract Personnel	1671	1659
COMMERCIAL ACTIVITIES PERSONNEL	15	16
TOTAL FORCE	1686	1675

FEATURE REPORT

PRECISION EXTRUSION OF URANIUM AND ZIRCALOY

INTRODUCTION

The 2750-ton extrusion press in the 333 Building, which serves primarily for the coextrusion of Zircaloy-2 clad tubular fuel elements for N Reactor, has been employed also for the extrusion of various other products for onsite use (e.g., Zircaloy fuel components, uranium cores, and Zircaloy process tubing). Most of these products have required very precise dimensional control and excellent as-extruded surface quality, as little or no secondary metal working operations (machining, drawing, grinding, etc.) are included in these processes. In addition, the press is being utilized for an increasing volume of non-AEC extrusion work conducted under the Use Permit granted Douglas United Nuclear for this purpose.

This report describes the precision extrusion of Zircaloy cladding components for N fuels, Zircaloy process tubing and uranium cores for the single-pass reactors, and miscellaneous other products. The emphasis is on the fabrication procedures and the dimensional and surface quality results obtained. Some of the programs have been completed, others are just starting and still others are of continuing nature. All are designed to demonstrate onsite extrusion capabilities with various geometries and types of materials.

ZIRCALOY CLADDING COMPONENTS

General

The onsite fabrication of Zircaloy components for the N fuel coextrusion billet assemblies was one of the earliest precision extrusion programs. Each tube-in-tube N fuel model requires four Zircaloy components, each of a distinctly different geometry, two for the outer fuel and two for the inner fuel. Table I below lists the approximate finished dimensions for the four components used in the production of Mark IV fuels. The extruded blanks for each part are shown in Figure 1, appended. Mark I components are very similar in size.

TABLE I

Mark IV Zircaloy-2 Components

	OD	ID	Wall Thickness
Outer Billets:			
Outer Clad	7.340"	7.000"	0.170"
Inner Clad	2.500"	1.880"	0.310"
Inner Billets:			
Outer Clad	5.850"	5.490"	0.180"
Inner Clad	1.250"	0.650"	0.300"

Until recently, only inner components had been manufactured onsite in sizable quantities, these being primarily inner cladding for the Mark I and Mark II outer billets. Now, however, all four sizes have and are being made for both the Mark I and Mark IV models. Moreover, although large quantities of outer cladding have not been made as yet, some dimensional data are available on 12 extrusions of Mark IV outer billet, outer cladding made recently.

Outer Clad for Outer Billets

The process for the manufacture of outer clad components for the outer billets involves a double extrusion operation in which the initial billet is first pierced or cupped to enlarge the ID and then forward extruded to make the component blanks. Figure 2a, appended, shows the initial billet, cupped billet, and extruded part for the Mark IV outer billet, outer component case. The cupped billet is machined both OD and ID prior to making the final extrusion.

The machined billet is extruded from a container of 9.60-inch ID through a 7.39-inch ID die. The mandrel is a free or floating type with an OD of approximately 6.92 inches; it is tapered from front to rear, small to large, to facilitate removal after the extrusion. This taper is about 1.5 mils per inch.

The extruded tubes are machined both OD and ID to produce the finished parts. Because of the tapered mandrel, wall thickness of the as-extruded piece ranges from a nominal 0.215 inch at the front of the tube to 0.195 inch at the rear. These are typical measurements for an extrusion 10 feet in length.

Wall thickness measurements taken on 25 pieces from 12 extrusions were as follows:

Average wall thickness variation	0.009" (4-1/2% of wall)
Maximum wall thickness variation	0.021" (10% of wall)

The wall thickness variation tolerance for Zircaloy clad components is 0.016 inch, total. Only one of the 25 pieces measured exceeded this limit. However, extrusion of this part directly to finished size is precluded by two factors:

1. Tube ovality, in some cases in excess of 0.035 inch.

2. Use of the tapered mandrel, which produces a change in ID of approximately 0.030 inch from the front to the rear of each extrusion.

The ovality noted in the extruded product results from the fact that the ratio of diameter to wall thickness is such that the tube at extrusion temperature (1200 F) does not have sufficient rigidity to maintain its shape and therefore "sags" as it lies horizontally during and immediately after extrusion. If this problem can be overcome, it might prove practical to extrude the OD to size with the ID slightly undersized as at present.

Outer Clad for Inner Billets

Outer cladding for the Mark IV inner billet is manufactured in much the same manner as for outer billets with the exception of billet and tooling sizes. This particular billet is pierced and extruded from the 7.58-inch press liner. Figure 2b, appended, depicts the steps for production of this part. To date few of these parts have been produced and significant data for discussion are not available.

Inner Clad for Outer Billets

The inner clad component for the outer coextrusion assembly is produced by a single extrusion step involving again the 7.58-inch liner. As stated before, this part has been made onsite in large quantities for only the Mark I and Mark II fuels (about 500 and 2,000 respectively). About 200 of the Mark IV inner clads shown in Figure 1 have been produced onsite.

In all cases this part has been extruded directly to finished ID size with about 0.025 inch per surface of OD machining overstock. Typical dimensional results are shown in the following data compilation involving 14 recent extrusions of inner clad components (200) for Mark I outer billets:

Average OD	2.831" (Range ±0.003")
Average ID	1.963" (Range ±0.006")
Average Wall Thickness Variation	0.013" (3% of Wall)
Maximum Wall Thickness Variation	0.036" (9% of Wall)

The use of tapered mandrels accounts for the 12-mil range of ID dimensions recorded. Here a mandrel taper of 0.5 mil per inch has been found to more than double mandrel life by minimizing the contact area between the mandrel and the hot extrusion. In the case of this particular clad component, only the OD and wall thickness dimensions are critical and a slight variation in ID can be tolerated.

Surface quality and OD and ID dimensional control with Zircaloy extrusions of this approximate geometry have been such that extrusion directly to size would seem possible. However, wall thickness variations in excess of the 0.016 inch specification have precluded this from a yield standpoint. Although a maximum variation of 9% of the total wall thickness is excellent for an extruded piece, it is far in excess of the specified tolerance for the finished part. Of the 200 pieces from which the above data were taken, approximately 75% were within this tolerance prior to OD machining.

Inner Clad for Inner Billets

Inner clad components for the inner fuel geometries are also produced by a single extrusion step, but utilizing a 6.07-inch press liner. The parts are extruded directly to final ID size with about 0.015 inch per surface of machining overstock. Most of the work in this area has been quite recent, with 20 extrusions (representing about 250 Mark I and Mark IV components) having been made within the past year. The Mark I part differs from the Mark IV part listed in Table I only in OD size (1.32 inches).

The following data compiled from 11 extrusions (130 pieces) of Mark I inner cladding and five extrusions (75 pieces) of Mark IV components show the excellent results obtained:

	Mark I	Mark IV
Average OD	1.365"	1.378"
Range of Values	±0.003"	±0.004"
Average ID	0.647"	0.646"
Range of Values	±0.004"	±0.005"
Average Wall Thickness Variation	0.010"	0.009"
% of Wall	3	3
Maximum Wall Thickness Variation	0.026"	0.025"
% of Wall	7	8

Tapered mandrels again accounted for much of the variation in ID dimension. A look at 30 pieces of the Mark IV parts produced in three extrusions made with straight mandrels revealed an ID control of ±0.001 inch.

Of significance in the extrusion of this particular part is the fact that until recently it was thought impractical to make this part by extrusion of a tube, and all Zircaloy components of this geometry were made by drilling the ID of a solid, extruded rod. Experience (both here and offsite) had shown that the small diameter mandrels required would not stand up under the required pressures and temperatures, and often would not last through even a single extrusion. Now, however, the use of tapered mandrels and shorter extrusion billets (12" maximum length) has increased mandrel life to an average of four extrusions each, making the extruded-tube route far superior economically to the drilled-rod method.

THIN-WALLED ZIRCALOY PROCESS TUBING

A development program was conducted in 1967 to determine the feasibility of producing thin-walled Zircaloy reactor process tubing by extrusion directly to finished dimension. The following dimensions and tolerances were targeted for the extruded tubing:

OD	2.741" ± 0.005"
ID	2.651" ± 0.005"
Wall Thickness	0.045" Nominal
Wall Thickness Variation	0.0035" (7 1/2% of wall)

The tolerances shown conform to ASTM standards for "Extra Close Tolerance Tubing" and are much more precise than normal as-extruded tolerances for tubing of this size. Also, part of this specification is a surface finish requirement of 0.005-inch maximum pit or dent and 0.002-inch maximum scratch or gall (extrusion mark).

Several extrusions were made using various billet designs and extrusion conditions before a workable process was found. All extrusions were made from a 6.07-inch press liner using a Zircaloy billet jacketed in copper silicon. A 0.065 inch wall copper-silicon sleeve was used in the ID, with various billet ODs and heavy-walled Cu-Si outer sleeves used to keep the extrusion reduction ratio to within the capacity of the press.

The billet geometry shown below was used in five extrusions of tubing to the above dimensions with a 20:1 reduction ratio:

	OD	ID	Wall
Outer Copper-Silicon	6.00"	4.32"	0.840"
Zircaloy Billet	4.31"	2.85"	0.730"
Inner Copper-Silicon	2.83"	2.70"	0.065"

These extrusions, which varied in extruded lengths from 13 to 26 feet, were cut into a total of 26 pieces about three feet in length. The following is a summary of the careful measurements taken on each piece:

OD	2.745" ± 0.006"
ID	2.646" ± 0.006"
Average Wall Thickness Variation	0.003" (6% of wall)
Maximum Wall Thickness Variation	0.005" (10% of wall)

The nominal OD and ID size was chosen to provide for 0.002 inch per surface of metal removal during an acid pickling operation. The variation in these dimensions was the result of tube ovality, not differences in the mean OD of any one tube or from tube to tube. As-extruded surfaces were excellent in all cases.

Although it appears that further working, such as a final drawing operation, would be necessary to consistently meet the specifications for this particular process tube, the results of the few extrusions made were encouraging in that high quality tubing, exceeding all normal requirements for extruded tubing, was produced. A tube 47 feet in length recently was extruded; its dimensional data are not yet available.

THIN-WALLED URANIUM SLEEVES

In excess of 800 thin-walled uranium sleeves, each approximately 20 inches in length, were made by extrusion directly to finished size (see Figure 3, appended). These sleeves were used as shims in the IDs of excess Mark I outer uranium billets to convert these billets into a size suitable for the extrusion of Mark IV outer fuels. The Mark I and Mark IV outer billets have similar ODs, but their IDs are 2.80 inches and 2.51 inches, respectively. The sleeve required was one having the following nominal dimensions:

OD	2.775"
ID	2.510"
Wall Thickness	0.132"

The sleeves were extruded from the 6.07-inch press liner. The billets, made from Mark I outer billets machined on the OD from 6.95 to 5.85 inches, were jacketed in 0.065 inch copper-silicon and extruded at a billet reduction of 18.8:1. Each extrusion yielded 17 of the sleeve components for a 91% metal utilization. Fifty extrusions were required to produce approximately 600 Alloy 501 and 200 Alloy 601 components. These were made in two campaigns of 25 extrusions each, separated by a period of four months.

Dimensional results obtained on the more than 800 sleeves were excellent. OD and ID dimensions were 100% controlled to within ± 0.005 inch. Wall thickness variation averaged 0.008 inch, only 6% of the total wall thickness.

Eight hundred coextrusions of Mark IV outer fuel using the thin-wall sleeves have been made with no rejects attributable to their use. Although this program has now been completed, this method of converting obsolete billets into billets of a usable geometry may again be required in the future.

URANIUM CORES FOR THE SINGLE-PASS REACTORS

During 1967 a program to develop capability for the extrusion of I&E uranium cores for the single-pass reactors was conducted, in which more than 1,000 Model 03N cores were extruded. Of these, approximately 850 were extruded oversize and machined, and 150 were extruded to size on the ID and 0.020 inch oversize on the OD.

The uranium billets (7.33" OD x 0.65" ID) were jacketed in 0.065 inch copper-silicon and extruded from the 7.58-inch liner with an area reduction of 28:1 for the cores extruded oversize and 30:1 for the others. The billets were heated to a temperature of 1170 F and extruded at a ram speed of 15 inches per minute.

During the five extrusions in the first campaign, 60 sample pieces were taken for dimensional study. During a second campaign of nine extrusions, 27 sample pieces were taken. These two sample lots yielded the following dimensional data:

	Campaign No. 1	Campaign No. 2
OD	1.434" ± 0.004"	1.436" ± 0.005"
ID	0.400" ± 0.004"	0.400" ± 0.003"
Average Wall Thickness Variation	0.014"	0.014"
% of Wall	3	3
Maximum Wall Thickness Variation	0.029"	0.032"
% of Wall	6	6

As can be seen, a high degree of dimensional accuracy was obtained. Surface finish on the extruded pieces was excellent in all cases. A total of 26 extrusions were processed during this program.

The success of this first attempt at precision extrusion of uranium fuel cores was largely responsible for a decision to extrude some (500) of the longer cores intended for K reactor testing under PTA-181. This core model, called the K11E is 94 Metal, 501 alloy, 10.75 inches in length. These cores are being extruded directly to size, both OD and ID, working to the following dimensional criteria:

OD	1.420" ± 0.005"
ID	0.545" ± 0.006"
Wall Thickness	0.437"
Maximum Wall Thickness Variation	0.020" (<5% of wall)

Time in which to order extrusion billets of the preferred geometry was not available, so billets procured for an earlier extruded core program were used following either a primary extrusion step or an upsetting operating to reduce the inside diameter.

Extrusions made to date using the converted billets have shown varied results. ODs and IDs have been controlled to ±0.002 inch. Wall thickness variation has varied from extrusion to extrusion, and has been found to relate directly to the variation in wall thickness of the parent billet. Billets with variations under 0.030 inch have produced cores consistently within the 0.020 inch specification maximum, while attempts to extrude billets with as much as 0.050 inch wall variation have resulted in cores with as high as 0.050 inch variation. All remaining billets with wall thickness variations in excess of 0.030 inch will be corrected prior to extrusion. Figure 4, appended, shows a six-inch billet and a precision extruded K11E core.

OTHER PRODUCTS

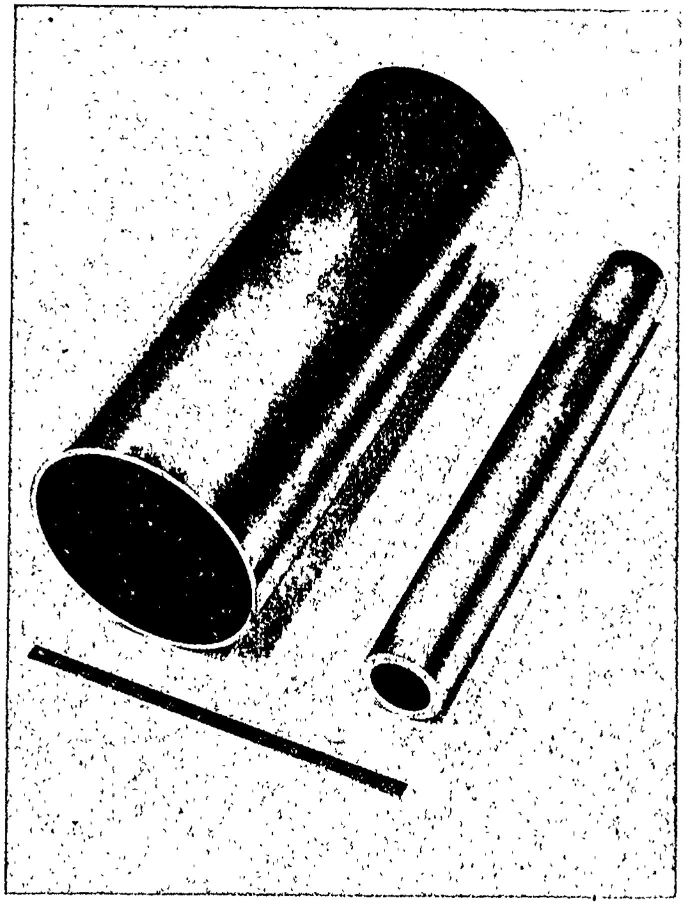
In addition to the Zircaloy and uranium extrusion programs discussed above, success has been achieved with the precision extrusion of various copper, copper-silicon, and aluminum parts. The appended Figures 5 and 6 show some of these other products. Copper and copper-silicon thin wall tubing and copper spacer rings for N fuel coextrusion billet assemblies have all been extruded directly to size. Aluminum spacers and target tubing have also been extruded to close tolerances.

Other materials extruded have included aluminum-uranium and zirconium-uranium alloys.

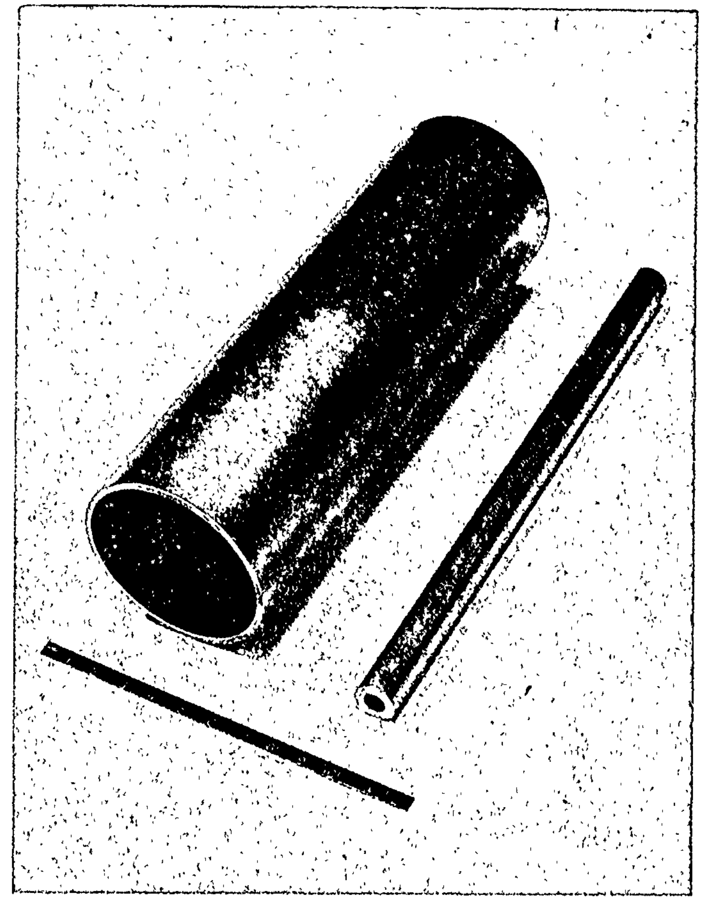
FUTURE EXTRUSION PROGRAMS

It is anticipated that an ever-increasing volume and variety of products will be fabricated onsite by extrusion in the foreseeable future. The extrusion of Zircaloy cladding components for N fuel is expected to represent one-third to one-half of the total requirements by the end of this year. It is hoped that the current development program for precision extruded uranium cores will lead to the routine onsite fabrication of some percentage of the K reactor cores required.

To provide additional onsite flexibility in the extrusion field, and as an aid in developing and improving extrusion techniques, a 500-ton vertical extrusion press has recently been installed in the 300 Area. Although only operational for a few weeks, this press has already demonstrated the capability to handle extrusion work not practical with the 2750-ton press.



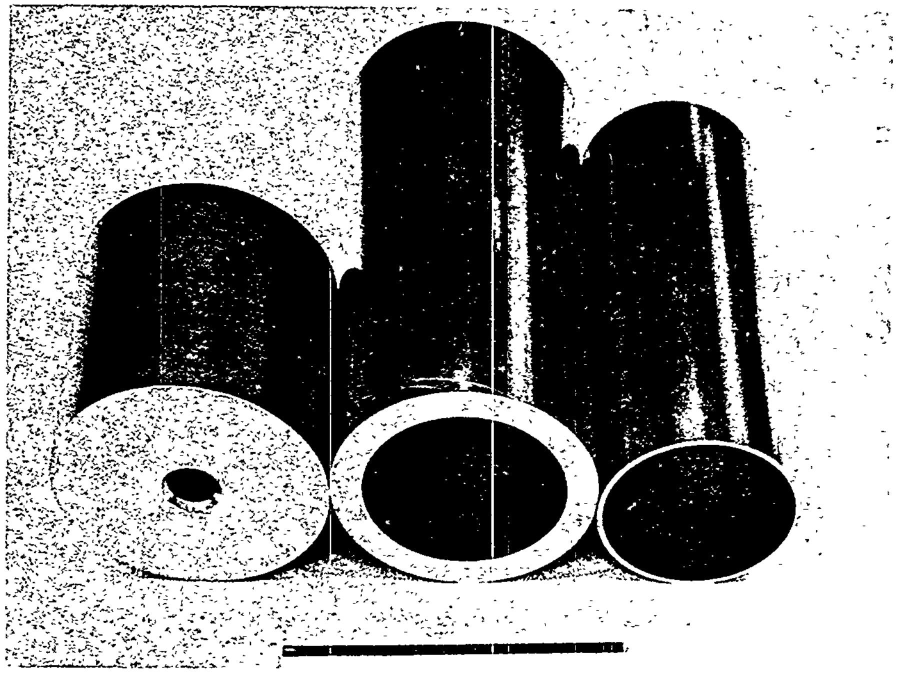
a) Outer Fuel Components



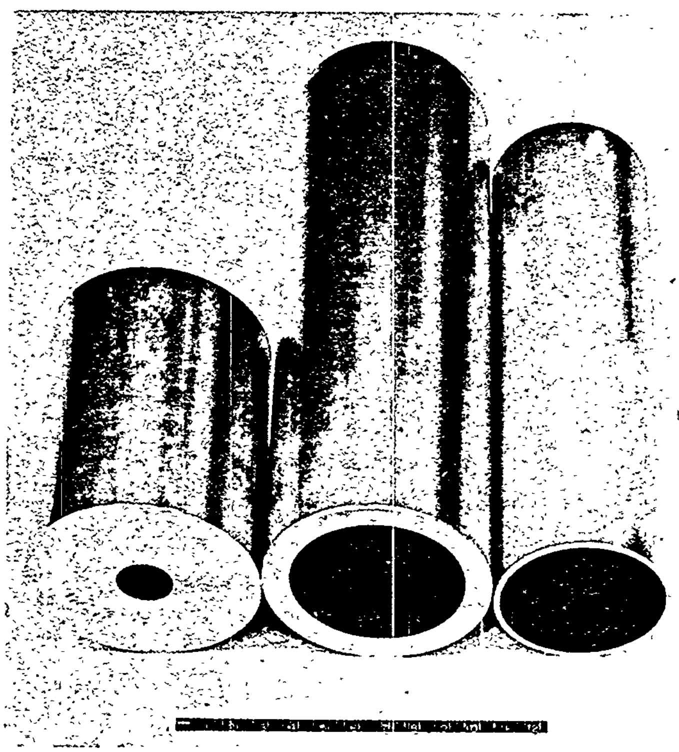
b) Inner Fuel Components

FIGURE 1

Zircaloy-2 Cladding Components Used in the Production of Mark IV Fuels (Unmachined)



a) Outer Cladding for the Outer Fuel



b) Outer Cladding for the Inner Fuel

FIGURE 2

Initial Billets, Pierced Billets and As-Extruded Outer Cladding Components for Mark IV Fuel Production

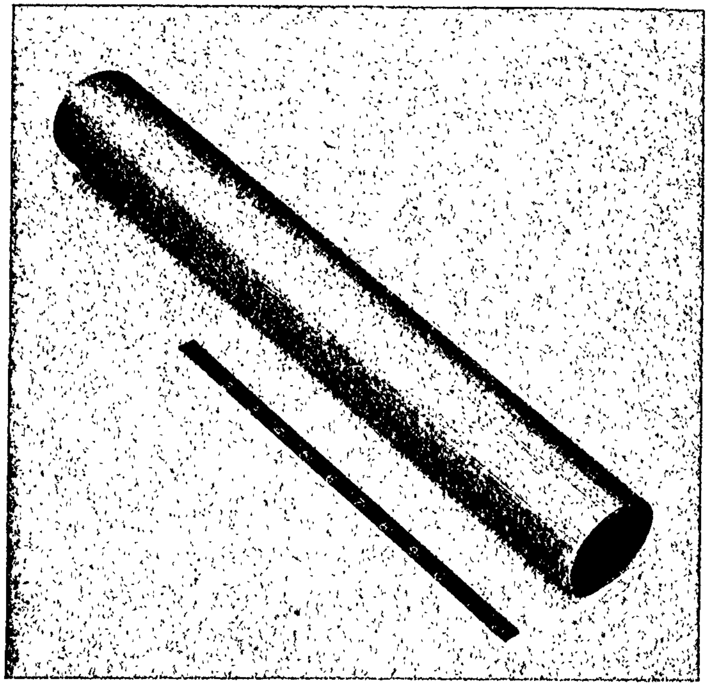


FIGURE 3

Precision Extruded Uranium Sleeve

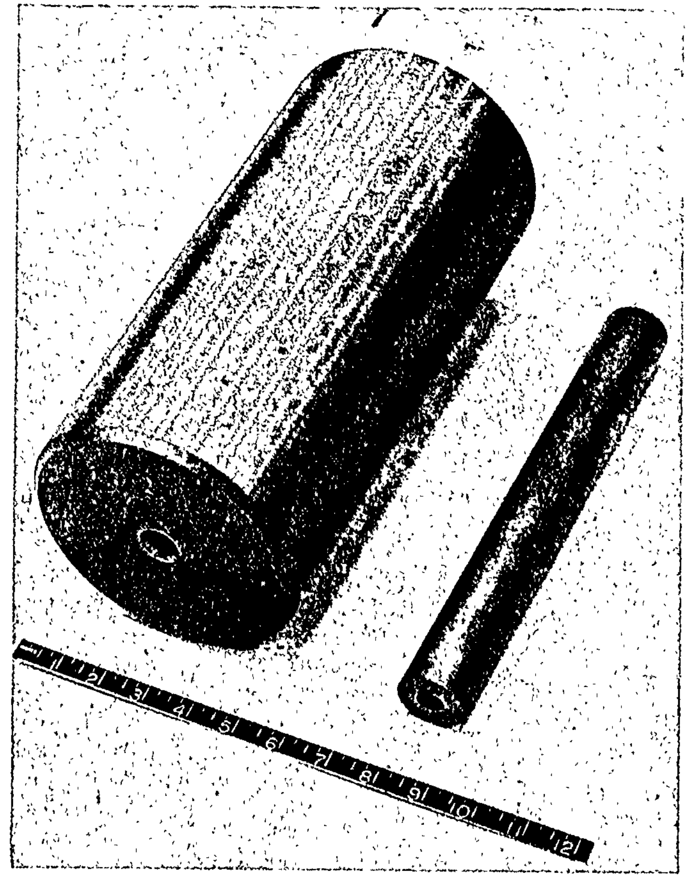


FIGURE 4

K11E Extrusion Billet and Extruded Uranium Core (The Billet Shown has been OD Machined, The Core is As-Extruded)

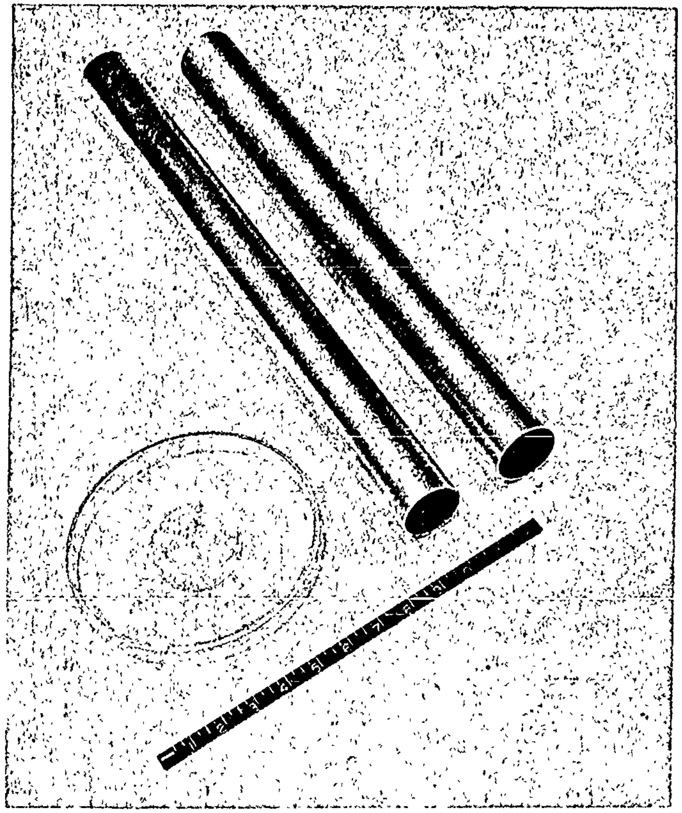


FIGURE 5

Copper and Copper-Silicon Products Fabricated by Extrusion. Shown Here Are Copper Spacer Rings and Inner Sleeves Used in Coextrusion Billet Assemblies

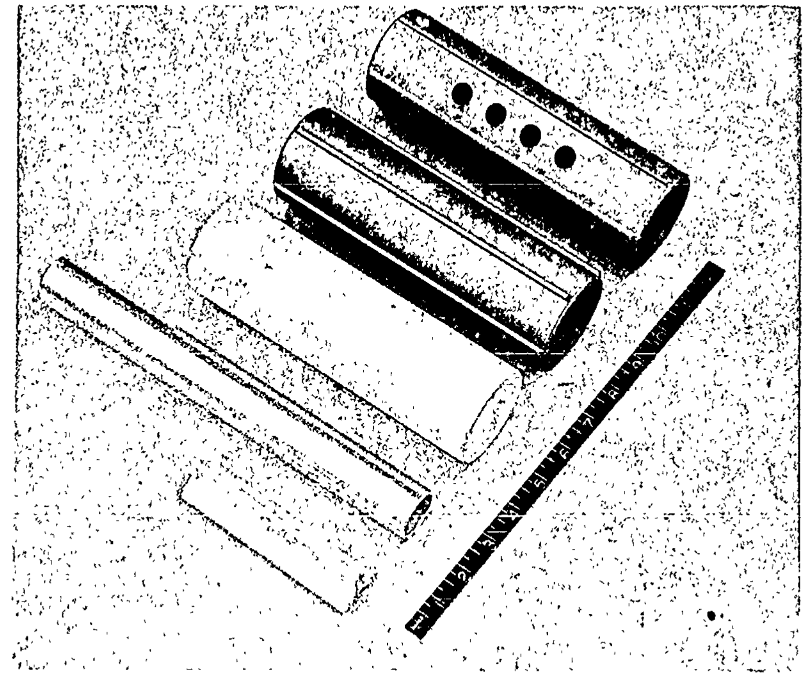


FIGURE 6

Precision Extruded Aluminum Target Tubing and Reactor Spacers and Perfs