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**Prepared for the U.S. Department of Energy
under Contract DE-AC06-76RLO 1830**

**Pacific Northwest Laboratory
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MONTHLY REPORT

NOVEMBER, 1966

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This document consists of
67 pages.

December 15, 1966

MONTHLY REPORT
NOVEMBER, 1966

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By Authority of RM. Hen
CG-NMP-1 8-3-94
By DK Hanson 8-15-94
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SUMMARYREACTOR PLANT OPERATIONS

Reactor input production (Pu) was 333.0 KMWD, 223.4 at the two K reactors and 109.6 at the three smaller units. U-233 input production was 7622 equivalent MWD. Power levels at the smaller reactors were restricted by the bulk outlet water temperature limit of 95 C, and at the K reactors by the administrative limit of 4400 MW.

Overall time operated efficiency (TOE) of the reactors was 70.1 percent. At the three small reactors, TOE averaged 59.9 percent reflecting a 21-day outage at B reactor for the modification of rear face hardware and for the other necessary maintenance work. TOE at the K reactors averaged 85.2 percent. Full statistics for all five reactors are tabulated on page B-2.

Results of the three-month half-plant test at C reactor, to demonstrate the effect of alum feed rate control of zeta potential, have indicated beneficial effects with respect to phosphorus-32 concentration in the reactor effluent. The arsenic-76 has not been affected. To confirm these initial results, it is planned to continue the half-plant test, but with the test and control halves of the plant reversed.

FUEL AND TARGET FABRICATION

Fuel production totaled 194.4 tons of natural uranium elements and 185 tons of 94 Metal elements, representing 78.1 percent of the September forecast. Diversion of operators to the N Fuels Operation of GE-HAPD continued throughout November to provide assistance on coproduct work. This reduced canning line operations by 17 percent. Production was lowered an additional 5 percent due to delayed receipt of fuel cores from NLO. The canning of thoria target elements was resumed early in the month.

Fuel core inventory at month end was 1053 tons, a 2.3 month's supply. Finished fuel inventory was 1256 tons, a 2.4 month's supply.

Installation of the facility for anodizing aluminum dummies was completed on November 1, and single-shift operation was begun on November 9. These coated dummies are for use in Zircaloy tubes at the K reactors to inhibit hydriding. Their production rate was increased steadily to about 85 percent of design capacity by month end. Two-shift operation was started on November 28.

TECHNICAL ACTIVITIES

The scheduled demonstration of the Hanford-developed safing seal by Douglas United Nuclear was conducted at F reactor on November 16. It was attended by representatives of 43 of the United Nations General Assembly delegations, and by 18 representatives of various news media. AEC representatives present included Commissioner S. M. Nabrit. One safing seal assembly was installed on the front face of the reactor, and previously applied seals were removed to illustrate the detection of attempted

seal circumvention. Tours of the reactor building also were conducted. This was the first public opening of a DUN-operated production reactor to foreign nationals and the press.

Streamlining KE reactor downcomer channel entrances resulted in definite improvement in downcomer flow characteristics, including: a 20-25 percent increase in flow in four channels, a decrease in turbulence in the downcomer, and a decrease in water level in the approach box. In view of this flow capability improvement, these streamlining changes will be made also at KW reactor.

An analysis of drop times for vertical safety rods at the K reactors indicates that nuclear safety requirements for the E-D load will be met.

ADVANCED PLANNING

An MFC proposal to evaluate applications for Co-60 produced at Hanford with cobalt-coated process tubes has been issued for approval. A proposal to use a deactivated reactor for spatially dependent physics transient tests is being readied for submission to the AEC's Division of Reactor Development and Technology.

The technical portion of the upgraded CAGE program has been completed. Debugging of the complete program, including CAGE - Financial, is about 90 percent complete.

Study continued on determining the value that orolloy would have if used as a sweetener to re-enrich Hanford reactor tails for recycle. Preliminary results indicate a marginal cost of \$80 per gram for incremental neptunium produced in all fuel streams of assay greater than 0.947 percent U-235.

Basic input elements directed toward the evaluation of a long range plan have been completed. These include environment (competition and market), strengths and weaknesses, and capital requirements.

IRRADIATION SERVICES

The preparation of strontium-85 for radiation sources for use in the NASA biosatellite program continued with the shipment of one capsule to Cape Kennedy in November. Seven of these radiation sources have been prepared and shipped since June; current commitments call for shipment of six additional capsules by mid-December to suit the NASA schedule to place a biosatellite in orbit this year. The target strontium is irradiated by Douglas United Nuclear, and the radiostrontium is encapsulated by Battelle-Northwest.

FEATURE REPORT

Included at the back of this issue is a feature report which describes: (1) the safety and control rod systems of the production reactors, (2) the problems which have developed with them, largely due to moderator distortion, and (3) the corrective rod channel renovation and rod modification work which has been accomplished. As noted in that report, all problems encountered with

these critical systems have either been corrected, or appropriate remedial measures have been readied for use when needed.

GENERAL

There were no disabling injuries, and no radiation exposures exceeding operational control limits.

The Company has been advised that its Pension Plan, Salaried Savings Plan, and Wage Savings Plan have been approved by the Internal Revenue Service.

for Charles D. Harrington
President

REACTOR PLANT OPERATIONS

PRODUCTION

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

The average exposure of irradiated natural uranium fuel discharged from the small reactors was 592 MWD/T and, at the K reactors, 660 MWD/T. The average exposure of enriched fuel discharged from the small reactors was 755 MWD/T and, at the K reactors, 943 MWD/T.

OPERATING EXPERIENCE

Reactor Outages

There were nine reactor outages, totaling 987.9 hours, and five startup interruptions resulting in 24.7 outage hours. In addition, one reactor continued down on an outage initiated in October. The outages and their principal causes are listed below:

<u>Date</u>	<u>Reactor</u>	<u>Hours Down</u>	<u>Cause</u>
Nov. 1	KW	65..5	Reactor continued down on a scheduled outage initiated October 29 for charge-discharge and maintenance.
2	D	58..9	Scheduled charge-discharge..
4	B	513.7	Scheduled charge-discharge, replacement of rear face hardware, and other maintenance.
8	KE	42.8	Scheduled charge-discharge.
19	C	84.9	Scheduled charge-discharge and dual area (B-C) trip-out test.
24	C	29.1	Correction of a water leak in process tube 0357.
24	D	140.8	Removal of a failed fuel element from tube 4180 and scheduled charge-discharge.
26	KW	38.7	Unexplained Panellit trip.
28	KE	64.0	Scheduled charge-discharge. The outage continued over month end.
30	C	15.0	Correction of process water leak and scheduled charge-discharge. The outage continued over month end.

Significant Activities

The failed I&E natural uranium fuel element removed from D reactor had reached only 33 percent of goal exposure. Failure was due to mechanical damage.

REF ID: A63770

PRODUCTION REACTOR STATISTICS - NOVEMBER, 1966

REACTOR	B	C	D	KE	KW	TOTAL
INPUT PRODUCTION - PU - KMWD	15.9	53.1	40.5	111.6	111.7	333.0
- U-233 - EQUIV. MWD	485	1538	1193	2092	2314	7622
POWER LEVEL - MW (MAX.)	2025	2295	1975	4400	4400	15095
- MW (AVG.)	1855	2217	1896	4373	4365	14706
TIME OPERATED EFFICIENCY - %	28.7	79.9	71.3	85.1	85.3	70.1
NUMBER OF OUTAGES	1	3	2	2	1	9
NUMBER OF STARTUP INTERRUPTIONS	0	1	2	1	1	5
OUTAGE TIME ALLOCATION - %						
CHARGE - DISCHARGE	4.7	5.9	17.9	10.6	9.3	9.7
FAILED FUEL REMOVAL	0	0	0.9	0	0	0.2
WATER LEAKS	0	3.7	1.8	0	0	1.1
TUBE REPLACEMENT	2.7	1.2	0.2	0	0	0.8
OTHER MAINTENANCE	49.4	5.1	1.1	0.8	1.8	11.6
STANDARDS CHECKS	0.4	4.2	2.9	0.6	0	1.6
PRODUCTION TESTS	5.7	0	2.7	2.2	0.6	2.2
PROJECT WORK	6.1	0	0	0.1	1.1	1.5
OTHER	2.3	0	1.2	0.6	1.9	1.2
TOTAL	71.3	20.1	28.7	14.9	14.7	29.9
WATER LEAKS - TUBE	1	1	0	0	0	2
- VAN STONE	0	0	0	0	0	0
NEW TUBES INSTALLED	13	1	9	1	0	24
FUEL CHARGE - TONS NAT. U.	155.4	152.3	148.6	330.0	310.9	1097.2
- TONS ENR. U.	54.0	61.3	57.9	101.3	111.1	385.6
FUEL ELEMENT FAILURES	0	0	1	0	0	1
HELIUM CONSUMED - M CU. FT.	92.1	253.1	130.2	255.5	166.0	896.9
WATER TO REACTOR						
NORMAL OPERATING FLOW - GPM	88,800	107,000	94,000	216,000	216,000	-
pH	6.61	6.58	6.65	6.69	6.70	-
DICHROMATE - PPM	1.00	1.00	0.93	0.90	0.95	-

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In compliance with Process Standards, the B-C dual area trip-out test of the backup water supply system was conducted November 21. Results of the test are being compiled and analyzed.

Two significant surges occurred on the electrical distribution system, but no interruption of production resulted. The first surge (BPA System) caused a momentary drop of about 10 psi in coolant pressure at both K reactors; the second surge, which took place during WPPSS generation rejection tests at N reactor, caused a 2 psi dip at the Ks.

The high speed scanning system at KW reactor, which monitors the effluent water temperature of individual process tubes at the rate of 4000 per second, was tied into the No. 1 safety circuit and the Acceptance Tests were completed. The reactor scram feature of the system remained by-passed pending further approvals and revisions to Process Standards.

Three 2 1/2-hour Operating Studies classes were attended by an average of 30 exempt personnel. The subjects of Heat Transfer, Reactor Materials, Process Standards and Nuclear Safety were covered

EQUIPMENT EXPERIENCE

At B reactor, under Design Change 1080, the installation of available K-type rear nozzles and larger connectors was completed on all process tubes. This new hardware replaces equipment that was failing with increasing frequency, causing unscheduled reactor outages. The process water flow increase since these piping changes approximates 4000 gpm.

At 190-C, the No. 12 process water pump motor failed on November 22. During the period of repair, which extended through month end, C reactor coolant was supplied by nine pumps instead of all ten. This resulted in decreases of 3500 gpm in process water flow and 55 MW in reactor power level. The cause of the motor failure is not yet known.

Beneficial use of the relocated B-C Limited Area fence and the new Security Patrol badge house was achieved November 30. This simplification permits both B and C reactors to be within a single Exclusion Area. Substantial savings in personnel, grounds maintenance, and fence maintenance will be realized.

Of the 12 vertical safety rod channels scheduled for enlargement and sleeving at each of the K reactors, ten have been completed at KE and seven at KW. Enlargement of the eleventh channel at KE was in progress at month end. In addition, two biological shield openings and one thermal shield opening have been enlarged at each K reactor.

A new step-plug and rod tip were installed on horizontal control rod No. 12 at KW to alleviate a binding problem.

Ball 3X pressure sensing line modifications, under Design Change 1127, were completed at KW. This work was previously completed at KE. These modifications assure redundancy in the sensing system.

The recent successful rear face jackbolt hole foaming of 69 tubes at D reactor led to the decision to complete this work on all tubes at B, C and D reactors as soon as practicable. The procedure involves piercing the jackbolt hole and injecting a silicone foam which seals the annulus between the gunbarrel and thermal shield. This seal prevents the loss of helium and CO₂ from the reactor. Scheduling of the work is dependent on the procurement of foam kits and the required tools.

In support of the program for horizontal control rod renovation, procurement has been initiated on several types of rods. The types on order are: the K reactor conventional-type rod, the K reactor Inconel water-cooled rod, and the newly designed water-cooled aluminum rods for B, C and D reactors (see Feature Report beginning on page C-1).

Occasionally, the surface of a poison spline (used in process tubes for reactivity control) is roughened when it is being reclaimed. To eliminate this roughness, the use of a special roller to smooth and resize all used splines is being evaluated. Equipment development has been completed, and the first pilot batch of 100 splines is expected to be processed in a few weeks. The overall spline reclamation program is expected to result in an annual saving of about \$35,000.

The present process tube nozzle puller has a maximum capacity of 25,000 pounds, which frequently is insufficient to pull process tube nozzles which have become "frozen" or "cemented" to the studs that bolt them to the gun-barrel flange. A new puller with a pulling capability of 100,000 pounds has been designed, fabricated, and tested on-reactor successfully.

PROCESS ASSISTANCE & CONTROL

Process Physics

Flattening efficiency at all five of the DUN-operated reactors was above the average for the previous 12 months. Startup efficiency was similarly good, with a particularly good series of startups at C reactor in which the average nonequilibrium loss was 0.125 equilibrium day. Minor loading changes and adjustment of fuel exposure distribution have been accomplished at both K reactors in preparation for the E-D loading scheduled for late December.

A 40 percent K reactor type control rod has been placed in the No. 4 position at C reactor. Preliminary measurements indicate that its total reactivity worth is roughly 106 percent that of the rod it replaced.

The roughly six percent higher reactor power resulting from the flow increase accruing from the changed rear-face hardware at B reactor will have no significant physics effects on reactor operation. Similarly, the 9-pump operation at C reactor with lower than normal flow, and consequently lower than normal reactor power, produced no noticeable effects of physics interest.

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Summary of Operational Physics Data

Reactor	B	C	D	KE	KW
Effective Central Tubes (ECT) (1)	1510	1605	1505	2335	2335
ECT - 12-Month Average	1495	1595	1495	2265	2235
Flattening Efficiency (2)	0.828	0.882	0.825	0.775	0.775
Equilibrium Scram Recovery Time - Minutes (3)	2	20	None	None	None

- (1) Effective Central Tubes---Reactor power level divided by the average power of the ten most productive tubes which are representative of the reactor loading.
- (2) Flattening Efficiency---ECT divided by the number of power generating tubes.
- (3) Equilibrium Scram Recovery Time---The maximum time which could elapse between scram and first indication and still permit a successful scram recovery using currently acceptable startup procedures.

Production Fuel Performance

The production fuel element failure (K5N uranium) which occurred at KW reactor on October 21 resulted from a manufacturing defect. The parent lot was declared failure prone and discharged from the reactor.

Fuel performance control charts indicate no change in performance from expected levels. The following summary shows failure frequencies for the past 24 months:

Failures Per Million Fuel Elements Discharged
(Through October 31, 1966)

	Past 3 Months	Past 12 Months	Past 24 Months
Small Reactors - Natural U	0	10	32
Small Reactors - Enriched U	0	14	25
K Reactors - Natural U	32 ⁽¹⁾	14	9
K Reactors - Enriched U	0	6	13
C Reactor - Natural U	0	470	1,463

- (1) High by a factor of approximately two as the result of all reactors being down during the work stoppage of July and August.

Thoria Target Performance

One confirmed and one suspected failed thoria target elements were discharged from KW reactor in October. The confirmed failure was stuck, requiring removal of the process tube and target charge together. Failure mechanism will be determined after the target element is removed from the tube.

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The suspected failure had appeared to cause a 10 psi increase in inlet pressure on the process tube. Subsequent examination of the target charge at the 105-C Fuel Examination Facility has failed to reveal a failure.

Reactor Effluent - Concentration of Phosphorus-32 and Arsenic-76

B and D Reactors

The alum feed rate at both B and D reactors averaged 16.5 ppm during October. The coolant pH was maintained at 6.6, and the sodium dichromate feed was held at 0.9-1.0 ppm. The average concentrations of phosphorus-32 and arsenic-76 during October, together with some historical data, are given in the following table:

	<u>Phosphorus-32</u> <u>(pCi/ml)</u>		<u>Arsenic-76</u> <u>(pCi/ml)</u>	
	<u>B</u>	<u>D</u>	<u>B</u>	<u>D</u>
October 1966	9.3	3.7	80	65
September 1966	10.3	4.4	80	80
October 1965	12.0	5.1	310	65
October 1964	3.6	8.2	140	90

The October 1966 concentrations did not differ significantly from the previous month.

C Reactor

In mid-1961, alum feed rates at all water plants were increased to a nominal 18 ppm. This change did reduce the amount of radionuclides in the reactor effluent, but it has been recognized that 18 ppm is not necessarily optimum and may be in excess of requirements during some periods of the year. Small-scale tests had demonstrated that sufficient alum to bring the zeta potential of the floc to neutral resulted in the most efficient removal of parent material.

To demonstrate the effect of alum feed rate control by zeta potential, a half-reactor test was initiated at C late in August. The alum feed rate on half the reactor was fixed at 18 ppm. The alum feed rate on the other half was adjusted so that the zeta potential of the floc would be neutral. During October this required 12 ppm alum. A coolant pH of 6.6 and a dichromate feed rate of 1.0 ppm were common to both sides of the reactor. The phosphorus-32 and arsenic-76 concentration in the reactor effluent during October were:

	<u>Phosphorus-32</u> <u>(pCi/ml)</u>		<u>Arsenic-76</u> <u>(pCi/ml)</u>	
	<u>Test</u>	<u>Control</u>	<u>Test</u>	<u>Control</u>
October 1966	8.1	11.8	125	120
September 1966	9.4	13.2	95	85
October 1965		8.5		135
October 1964		8.0		130

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The October results are similar to those of September and prior years.

It does appear that beneficial effects are resulting with respect to the phosphorus-32 concentration. The arsenic-76 concentration is unaffected. This half-plant test at C is being continued, and on December 1 it is planned to reverse the test and control halves of the plant.

K Reactors

Coolant pH at these reactors remained at 6.6-6.7 during October, and the sodium dichromate feed rate was maintained at 0.9 ppm. The alum feed rates were 13.6 and 16.9 ppm at KE and KW, respectively. The phosphorus-32 and arsenic-76 concentrations in the effluent of these reactors during October were:

	<u>Phosphorus-32</u> (pCi/ml)		<u>Arsenic-76</u> (pCi/ml)	
	<u>KE</u>	<u>KW</u>	<u>KE</u>	<u>KW</u>
October 1966	4.8	7.2	65	80
September 1966	6.1	7.4	45	60
October 1965	7.0	16.4	50	370
October 1964	6.3	4.0	65	45

The phosphorus-32 concentrations are similar to those of September, but the arsenic-76 concentrations are somewhat higher. The high results from KE reactor during October 1965 were due to operation with 3 ppm alum feed.

Reactivity Effluent - Activity Data

The following table shows the average curie/day discharge for six radio-nuclides in the reactor cooling water effluent during October 1966:

<u>Reactor</u>	<u>As-76</u>	<u>P-32</u>	<u>Zn-65</u>	<u>I-131</u>	<u>Cr-51</u>	<u>Np-239</u>
B	36	4.1	3.6	0.30	137	30
C	53	4.7	14.0	0.36	291	56
D	33	1.9	7.9	1.19	156	33
KE	34	2.5	3.6	0.04	66	21
KW	<u>76</u>	<u>6.8</u>	<u>16.1</u>	<u>0.36</u>	<u>126</u>	<u>45</u>
Total	232	20.0	45.2	1.25	776	185

FUEL AND TARGET FABRICATIONPRODUCTIONAcceptable Elements Produced

Type	Tons Input	Elements to Storage--Tons			Yields--Percent	
		Unrestricted Use	Upstream Use	Total	Current Month	FYTD
<u>AlSi-Natural U</u>						
8" Regular	59.7	89.6	18.7	108.3	93.6	92.6
8" Bumper	58.6	50.7	-	50.7	95.3	92.9
8" Self-Support	37.9	8.8	4.1	12.9	96.9	93.0
<u>AlSi-94 Metal</u>						
6" Regular	-	15.3	4.8	20.1	88.9	93.2
6" Bumper	52.2	44.5	-	44.5	95.3	94.1
6" Self-Support	142.3	115.7	1.8	117.5	94.6	95.5
<u>Hot-Die-</u>						
<u>Sized U</u>	19.1	25.4	-	25.4	-	-
Thoria	24.7	17.2	-	17.2	-	-

Procurement and Inventories (in Tons)

Item	Received During Mo.	Placed in Process During Mo.	Uncanned Stock at End of Mo.	Uncanned Mo's. Supply on Hand
Natural U	155.2	175.3	855.0	3.1
94 Metal	219.3	194.5	198.0	1.1
Thoria	18.5	24.7	21.6	1.3

AlSi-Bonded Fuel

Production totaled 78.1 percent of the September forecast. Diversion of operators to the N Fuels Operation of GE-HAPD for assistance in coproduct work continued throughout the month. As a result, 20 canning line shifts, or 17 percent of the total scheduled, were not worked. Production was additionally 5 percent below forecast due to problems with the timing of core arrivals resulting principally from the strike at National Lead in October.

Core for support welding were not available for five days early in the month, and the achievable welding rate after these cores became available could not offset the non-productive period. Production was lowered about 3 percent for this reason. An additional 2 percent lowering resulted from late receipt of certain eight-inch natural uranium cores, and consequent fabrication of the lighter six-inch 94 Metal elements which reduced output tonnage.

The total manufacturing yield decreased from 94.9 percent to 94.1 percent. Reject categories having the most effect on yield were assembly, marred surface, bond and inclusions.

Reclamation of total rejects achieved by category and percent was: closure weld-89 percent, AlSi slopover-98 percent, and rail weld-100 percent.

Hot-Die-Sized Fuel

The plating station input totaled 3300 cores, 407 of which were for development purposes. C reactor overbore fuel (model C6ANS) was processed nearly the entire month for both production and process development. Capability for increasing the throughput rate of this larger model (two to three times heavier than regular) was demonstrated.

OPERATING EXPERIENCE

Four-Inch Fuel Elements

Ten percent of the available canning line time was devoted to the production of small reactor four-inch fuel elements. This model is produced infrequently, the last having been made in April, 1965. Although the usual operating problems associated with production of a fuel model at infrequent intervals were encountered, and lowered the time operated efficiency during this period, the yield will be higher by several percent than in any previous run. About half of the material fabricated will have bumper rails attached in December.

Thoria Canning

Thoria target element fabrication was resumed on November 2 and the production rate build-up has been quite satisfactory. A considerable decrease in elements requiring reworking of weld defects has resulted from the use of a ceramic thoria wafer placed between the powder and the cap. This wafer provides a weld heat absorption media, thereby improving the initial welding success as well as materially aiding the subsequent reworking of any weld defects.

Only token production was made during one week of the month when wafers were unavailable because of shipments having been changed from air freight to rail freight. Meeting of original reactor recharging schedule required air freight shipment of wafers from the vendor at Cincinnati. However, relaxation of the end product (U-233) specification allowed extension of reactor charging times and similar adjustments in thoria canning rates.

Anodizing of Dummies

Anodizing of aluminum dummies on a production basis began on November 9, after a concentrated shakedown of equipment and process during the period November 1 through 8. These treated dummies are for downstream use in Zircaloy tubes at the K reactors to inhibit hydriding. Initial production was 132 pieces on one shift. This rate was increased steadily, and by month end about 85 percent of design capacity was being achieved. An output of 720 pieces has been obtained in one shift on a peak basis. Two-shift operation was started on November 28. Full design capacity should be realized when the permanent heat exchanger arrives and is installed in January.

Autoclave Failures

There were no AlSi process autoclave failures this report period. The autoclave failure rate for the past 12 months is 5.0 failures per million elements autoclaved.

EQUIPMENT EXPERIENCE

Canning line time operated efficiency averaged 97.4 percent. Of the 2.6 percent downtime, 1.4 percent was chargeable to equipment failure.

About one-fourth to one-half of the equipment downtime usually results from core loader and agitator mechanical problems. A prototype of a modified loader-agitator has been developed and installed on one canning line. The performance of this equipment to date has been highly satisfactory, and it appears that the number of rejects in the "no-can" category may be significantly reduced. Continued on-line operation on an extended basis will be made to assure solid reliability, and to verify that cost savings will accrue from improved time operated efficiency and fewer rejects.

The new thoria target element welder with a water-cooled chuck was installed in mid-November and by month end was in routine production use. The chuck of this welder absorbs heat during the welding operation, thereby reducing the heat transmission to the thoria below the cap which causes blowouts through the weld from expanded gases. Development work on this new welder since the last thoria target element campaign had shown that the cooled chuck virtually eliminates the blowout problem.

The Hanford Test Reactor was out of service for 22 operating hours due to failure of the control rod selsyn. A satisfactory replacement was obtained from one of the deactivated reactors.

PROCESS ASSISTANCE AND CONTROL

Outgoing fuel quality, as indicated by quality certification reinspection, is shown graphically below:

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The "inclusion" reject rate at final inspection has shown a significant increase. Two types of inclusions have been detected; both contain copper and are in the components from one vendor. One type has been identified as chips of a 2000 Series aluminum alloy (containing copper) extruded into the can surface. The other type is also copper, but in random patterns of small spots. It has not yet been determined whether these copper inclusions originate on-site or are attributable to the vendor's process.

Over 200 fuel elements of the C6ANS model (C reactor overbore) were rejected for inclusions appearing as longitudinal black streaks on the outer surface of the element. Some of these streaks were up to three inches long. Spectrochemical analysis revealed that greater than 1 percent copper was present in the black areas. Metallographic examination showed the black streaks to be caused by a copper bearing aluminum alloy imbedded in the surface of the cans during impact extrusion at the vendor's plant. Copper bearing alloy (2000 Series) parts are impact extruded by this vendor, who recognizes two possible causes for contamination of the cans: (1) the tumbling equipment used to apply lubricant to the extrusion blanks may not have been cleaned out properly before processing 8001 alloy blanks for DUN; and/or (2) the lubricated 8001 alloy extrusion blanks which are handled in metal trays may not have been covered properly, and 2000 Series alloy chips and particles may have fallen into the trays.

Investigation into causes of the AlSi slopover rejects from the cans of one vendor is continuing. It has been determined that the AlSi does not flow past the top of the can until sometime after the uranium core is inserted in the can. It has also been determined that annealing new sleeves reduces the AlSi slopover rate by about 27 percent.

TECHNICAL ACTIVITIES

RESEARCH AND DEVELOPMENT

Beginning with this issue, the progress on research and development work conducted by Douglas United Nuclear will be reported by Mission number and title. Unless otherwise defined, the discussion will pertain to the five production reactors operated by DUN. No coverage will be given the two R&D Missions (numbers 4 and 15) in which DUN does not participate.

Mission 1 -- Basic Production

This Mission consists of four areas of work. The first is directed toward the development of new technology and novel processes which are applicable to reducing unit cost or extending the useful life of the production reactors. Examples are the efforts on (1) the hot-die-sized fuel fabrication process, (2) the improvement of fuel stability and fuel cladding materials, and (3) the resolution of problems associated with graphite distortion. Also included is the identification and solution of future problems as they arise from time to time. The second area of work consists of a continuing nuclear safety program which has the prime objectives of upgrading the nuclear safety status of the reactors and of providing development as necessary to assure that deterioration of the reactor components, and proposed operating techniques and conditions, do not reduce the safety of the reactor. The third area of work has to do with the development of technology to produce plutonium having a high Pu-240 content. The fourth area involves the studies and tests made to increase the incidental production of neptunium-237.

1-A. Hot-Die-Sized Fuel

Fuel clad in HDA-4 cladding alloy (10 weight percent silicon, one weight percent nickel) was irradiated under Production Test Authorization (PTA)-010 to evaluate corrosion behavior. Two fuel failures were sustained during the irradiation, both in the HDA-4 clad fuel, and both were caused by groove corrosion. The test was discharged at an exposure of approximately 620 MWD/T after the second failure. During examination of the fuel, two more HDA-4 clad elements were found on which the cladding was completely penetrated. The test included standard AlSi-bonded and hot-die-sized fuel elements as control pieces, and the frequency and severity of ledge and groove corrosion on all three fuel types are summarized below:

	Type of Fuel		
	HDS-HDA-4 Alloy	HDS-8001 Alloy	AlSi-8001 Alloy
Number Examined	141	106	106
Percent with Primary Groove Corrosion	46	36	43
Percent with Primary Groove Corrosion >10 mils	20	18	15
Percent with Primary Ledge Corrosion	8	9	10
Percent with Primary Ledge Corrosion >10 mils	2	2	0

The examination of two columns of each fuel type, hot-die-sized and AlSi-bonded, from the second cycle of monitor columns charged in KE reactor under the authorization of PTA-011 has been completed. No primary ledge or groove corrosion was found on the AlSi-bonded fuel elements; 16 hot-die-sized fuel elements showed primary ledge corrosion, for an over-all frequency of 24 percent. The groove corrosion on three of the pieces was between 10 and 20 mils deep. Examination of the first cycle of monitor columns from KW reactor discharged September 5, 1966, has also been completed. Out of five columns of AlSi-bonded fuel, 17 pieces showed primary groove and nine showed primary ledge corrosion, all of the attack being less than 10 mils deep. The over-all frequency was 14 percent. A total of 43 pieces from the columns of hot-die-sized fuel showed primary groove corrosion and two showed primary ledge corrosion, for an over-all frequency of 24 percent. Three of the hot-die-sized pieces showing primary groove corrosion were attacked to a depth of less than 10 mils, 24 to a depth between 10 and 20 mils, five to a depth between 20 and 30 mils, and on one the cladding was completely penetrated. This failed piece was located in position 5 of the column (near the downstream end) and the cladding was penetrated near the end of a support.

Since the above data from KE and KW reactors provided the required comparison of corrosion behavior of fuels from the two processes, the remaining PTA-011 monitor columns in both reactors were discharged at about 70 percent of goal exposure.

Evaluation of the corrosion behavior of the AlSi braze layer is continuing in an attempt to confirm preliminary data on corrosion rates. The buildup of corrosion products has been found to have a marked effect on these rates. Ultrasonic vibration of the corrosion medium minimizes such buildup on the test pieces. Preliminary data indicate that the rate of penetration of the aluminum cladding of autoclaved and unautoclaved fuel elements is about the same. Penetration of the autoclaved fuel takes place in pits and grooves, while unautoclaved pieces experience over-all corrosion. In both cases, the rate of penetration is reduced as the AlSi braze layer is reached and penetrated. A series of tests is now being planned whereby incremental corrosion rates can be measured at 0.005-inch increments through the cladding and braze layer.

Mission 2 - Coproduct

The objective of this program is to develop the capability for producing large quantities of tritium. The current demonstrated capability of the five reactors, utilizing 94 Metal, is 10 kilograms per year. With 210 Metal, the annual production capability would be about 28 kilograms of tritium. Utilizing fully enriched fuel, an annual production rate of over 50 kilograms per year could be realized. The current R&D program is primarily aimed at developing the capability of producing tritium utilizing 210 Metal fuel at the five reactors. This would require an average specific power increase of about 30 percent over that of natural uranium in plutonium-only production modes.

There were no new developments in this program during the month.

Mission 3 - Integrated Site Operation

The declining uranium processing program, reduced fuel requirements for a defense-oriented nuclear material production, and the advantages of a closely-coupled fuel cycle in producing non-defense isotopes, all tend to indicate a potential incentive for closing the fuel cycle at the reactor sites. The alternate processes to accomplish this at Richland include: the uranium metal-magnesium reduction process, the uranium metal-electrolytic reduction process, and the UO_2 cycle. The present program is limited to engineering and business studies of the alternate fuel cycles and scouting-type experiments.

There were no new developments during the month.

Mission 5 - Transplutonium Program

The objective of this program is to identify and provide the capability for producing large quantities of transplutonium isotopes from the irradiation of plutonium in the reactors. The normal graphite moderated reactor neutron spectrum has the significant advantage in such irradiations of minimizing the fission-to-capture ratio for Pu-239, and can easily be tailored to concentrate Pu-240 when product specification would require it.

The study will examine the situation in which a substantial but minor fraction of a reactor is loaded with plutonium. It is visualized that an inventory of up to 500-1000 kilograms of plutonium would be maintained in the reactors. The plutonium would not have a large effect on reactor operability and safety. However, it is necessary to identify the conversion rate parameters, the effects on reactor safety and operation, out-of-reactor criticality limitations, plutonium contamination problems, and the plutonium fuel element fabrication techniques necessary for fabricating and irradiating substantial quantities of plutonium fuel.

Curium-244 is a potential product under this program. The status of DUN work on the multi-site curium-244 production model is described later in this report, under Advanced Concepts & Planning - Operations Research.

Mission 6 - Pu-238 Program

The objective of this program is to increase the national production of Pu-238 in as efficient and economical a manner as possible, by identifying optimum methods and demonstrating the efficient Np-237 utilization capability and relatively rapid conversion to Pu-238 in the DUN-operated reactors. The technology for producing Pu-238 by irradiating Am-241 will be investigated. Assessment of the incentives for developing the latter capability requires detailed and thorough identification.

Two neptunium elements irradiated in the center of a process tube were analyzed for isotopic content. A comparison with predicted values is shown in the following table:

	Sample 1	Sample 2	Predicted From Exposure History
Np Remaining/Np Charged (β)	85.1%	84.8%	83.9%
Pu-238 Produced/Np Charged (α)	12.5%	12.8%	12.6%
Pu-238 Produced/Np Destroyed ($\alpha/1-\beta$)	83.7%	83.6%	77.7%
Pu-238/Total Pu (Q)	89.4%	89.3%	84.3%
Fissions/Np Charged	1.1%	1.1%	--
Pu-236/Pu-238	0.7 ppm	0.5 ppm	--
Weight of Np Initial/Total Target Weight	5.1%	5.2%	--

As seen from the table, excellent agreement was achieved between the two samples. The neptunium destroyed was slightly lower than predicted, but was within the uncertainty of the exposure history. Most significantly, the results indicate that the Pu-238 destroyed by neutron capture was less than predicted. The calculations used the value of 600 barns for the Pu-238 cross-section. Use of 420 barns, as some cross-section data indicate, would have very nearly matched the measured yields. The Pu-236 contaminant in the plutonium was consistent with crude estimates which indicated that the Pu-238 should contain less than 1 ppm of this isotope.

Mission 7 - U-233 Program:

This program has as its objectives the improved definition of thoria powder specifications, the development of economical methods of fabricating high integrity thoria target elements, and the refinement of reactor physics and reactor engineering technology related to the thoria loading of the reactors.

There were no significant developments during the month.

Mission 8 - Co-60 Program

The objective of this program is to define and demonstrate the capability of the reactors to produce massive quantities of cobalt-60.

There were no significant developments during the month.

Mission 9 - Po-210 Program

The effort under this Mission consists primarily of physics studies. The objective of the physics studies is to establish the technology for maximum Po-210 production within safety requirements, and to optimize for minimum cost and minimum interference with other production objectives. In particular, three alternate routes for producing polonium will be investigated: (1) without modifying the reactors; (2) with a reactor modified with large holes for the irradiation of bismuth; and (3) with the reactor modernization program.

There were no significant developments during the month.

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Mission 10 - High Power Density Fuel Program

The objective of this program is to increase the fraction of reactor production which can be allocated to products other than plutonium. The specific goal is to demonstrate the operation of a large number of tubes in a K reactor at tube power levels up to 3800 kilowatts. Attainment of the objective would permit half of the tubes in a K reactor to be removed from heat production service, and to be loaded with target materials, without decreasing reactor power level.

Some of the technical problems to be overcome include: (1) use of alloying and heat treatment to obtain uranium metal with improved irradiation stability and behavior, (2) development of an aluminum-base cladding material and/or coatings with corrosion resistance superior to that of the present 8001-F aluminum alloy, (3) determination of the water voiding reactivity effect and the decreased Doppler effect and necessary associated modifications to the reactor safety control system, (4) analysis of the impact of higher specific power on coolant flow instability or meltdown involved with interruption of the reactor coolant supply, (5) development of the capability to obtain and maintain a more optimum axial flux distribution, and (6) determination of the orificing required to obtain the optimum coolant flow distribution between reactor process tubes.

10-A. Fuel Testing

The initial test of 125 Metal fuel elements was discharged from KW reactor on September 5, 1966, at an average exposure of 1060 MWD/T. The fuel elements operated at maximum specific powers of 130 kw/ft, which was 50 percent higher than the power obtained in adjacent normal fuel columns in the reactor and 35 percent higher than the specific power in the highest power normal fuel columns in the reactor. Selected fuel elements are now being destructively examined in the Battelle-Northwest Radiometallurgy Laboratory. Preliminary results of this examination confirm earlier observations made at the 105-C Fuel Examination Facility. These are:

- (1) The corrosion of the 8001-F cladding was uniform and occurred at predicted rates, or at the rate of about 7.5 mils of cladding lost per month of exposure at full power conditions. Only one of the five fuel columns in this test suffered pitting corrosion attack, and the pit depths were estimated to be less than 10 mils.
- (2) The uranium fuel cores remained dimensionally stable during the irradiation, and to date the destructive examinations have not revealed any adverse core behavior.
- (3) The AlSi-bonding layer between the aluminum cladding and the uranium core had been broken in both the internal and external surfaces of the higher powered fuel elements (centrally located in the process tube). There was no unusual corrosion attack of the cladding on these elements, so heat transfer evidently was unaffected by the broken bonds. It is not known when the bond

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cracking occurred. Because evidence of unusual heat transfer characteristics is not present, it is believed that the bond cracking did not exist for a significant period of time during irradiation, but that it took place either during the thermal cycle occasioned by the reactor shutdown at the end of the irradiation. It may have happened after discharge from the process tube, possibly during the initial stages of the destructive examination. Metallographic examination of the bond is scheduled and should reveal the answer.

In general, the results of this test are encouraging from the standpoint that they indicate the possibility that current standard fuel core and cladding alloys can be applied successfully at 150 percent power conditions.

An additional 20 columns of 125 Metal fuel were charged into the KW reactor on October 31, 1966, under the authorization of PTA-040 to evaluate the dimensional stability of five uranium core alloys and two heat treatments at 150 percent of base specific power. The results of this test will provide a basis for choosing a core alloy for testing at higher power densities. The iron, silicon, and aluminum compositions in the five uranium alloys being irradiated are given in the following table:

Alloying Constituents in Uranium Core Alloys-ppm
Production Test Authorization-040

<u>Alloy</u>	<u>Iron</u>	<u>Silicon</u>	<u>Aluminum</u>
1. Standard	150	75-140	35-50
2. Test	150	50-70	35-50
3. Test	350	40	800
4. Test	150	350	40
5. Test	350	350	800

The core alloys were heat treated in hollow form and quenched in oil. Half of the fuel cores of each alloy were fabricated from rods which were induction heated during the heat treatment step, and half were heat treated in a salt bath.

The alloys are being tested to evaluate their behavior in our reactor environment. Alloys to combat only the problem of cavitational swelling could be selected on the basis of experimental results from Oak Ridge and Savannah River. However, the current test is designed to determine that fuel elements made from these alloys will also be dimensionally stable throughout the irradiation and not only at the exposures and temperatures where cavitational swelling occurs.

10-B. Nuclear Safety and Criticality Control

Work on the technical basis for processing 210 Metal solutions in the fuel waste recovery facility continued. Actual density variations of the sodium diuranate filter press cake with water content were measured. Using these values, and detailed chemical analysis of the material,

criticality calculations were made. Such data on density variations will be obtained from all uranium-containing liquids entering the waste recovery process. Careful analysis of these actual density and atom concentrations over an extended period may permit reduction in present conservatisms. Extensive checks have been performed to verify the calculational methods employed. The agreement for various uranium compounds for enrichments at different H/U atom ratios has been very satisfactory. A report is being assembled displaying all the data, and discussing assumptions and methods of analysis.

The critical mass parameters for 210 Metal fuel were reevaluated using HAMMER computer code. The calculations verified previously calculated differences in parameters between clad and unclad elements. In previous predictions of critical parameters, the code overestimated the material buckling (B^2) values for the various H_2O/U volume ratios. Reducing the B^2 values by $5.0 \times 10^{-4} \text{ cm}^{-2}$ gave excellent agreement with measured values.

The B^2 values obtained from HAMMER were utilized in preparing the nuclear safety specifications for handling the 210 Metal fuel which will be tested in a K reactor. The original values recommended for 210 Metal safety analysis were based on extrapolation from I&E B^2 data at lower enrichments, and are in close agreement with the newer values as shown below:

Safe Parameters	$B^2(\text{HAMMER})$		$[B^2(\text{HAMMER}) - 5.0 \times 10^{-4} \text{ cm}^{-2}]$	
	Clad	Unclad	Clad	Unclad
Sphere				
Radius (in.)	6.4	5.9	6.6	6.1
Mass (lb.)	300	250	330	270
Hemisphere				
Radius (in.)	9.9	9.2	10.2	9.5
Mass (lb.)	560	460	610	510
Inf. Cylinder				
Diameter (in.)	8.9	8.1	9.2	8.4
Mass/Unit Length (lb./ft.)	210	180	220	200
Inf. Slab				
Thickness (in.)	4.6	4.1	4.8	4.3
Mass/Unit Area (lb./ft. ²)	120	110	123	117

Experimental measurements by Battelle-Northwest of the critical mass parameters of the 210 Metal K reactor fuel were started November 21. Completion and analysis of the C measurements may permit the less conservative nuclear safety specifications.

Mission 11 - Reactor Modernization

The objective of this program is to demonstrate the feasibility of operating a reactor with the process channels enlarged up to one inch, and to develop

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the physics, engineering, and fuel technology data needed to support modernized reactor operations. The program required includes a multi-tube test facility for fuel development, for testing channel enlargement equipment and new process tube assembly designs, and for defining lattice physics parameters for potential reactor loadings. In the laboratory, physics, heat transfer, and mechanical development experiments are required to obtain the data necessary to complete a safety analysis, to procure fuel design data, to establish reactor operating limits and loading patterns, to determine the channel conditioning requirements, and to evaluate charge-discharge behavior of the fuel.

The overbore test facility proposed for C reactor consists of a block of 44 tubes enlarged one inch over present size to approximately 2-3/4 inches diameter. Present plans are to install 34 smooth-bore Zircaloy process tubes and 10 ribbed aluminum process tubes.

11-A. Reactor Engineering

A comment issue of the test block design criteria, document DUN-1452, has been prepared.

Procurement activity has progressed to the point where orders have been placed for all engineered items of the process tube assembly except the Zircaloy process tubes and the inlet and outlet nozzles. Vendor contacts and liaison are proceeding vigorously on the latter items to develop interest and favorable bids. The longest delivery component to date is the Zircaloy-2 process tube order with promised delivery dates of about 40 weeks. Fuel component delivery may also be delayed beyond vendor promises, based on previous experience; consequently, fuel may not be available until the first or second quarter in FY-1967.

Development of most of the tools required for enlarging the reactor penetrations has reached the procurement stage. Final drawings of all major tooling required for the on-reactor installation are nearing completion with the exception of the welding equipment and the graphite cutting bar. No action can be anticipated on the latter for several months until final tube distortion data are available.

11-B. Fuel Development

One column of 24 XMIN rod-in-tube fuel elements fabricated by the hot-die-sizing process was charged into channel 3865-KE on October 20, 1966. This channel had a Zircaloy tube installed in 1961 to test N reactor fuel. The inside diameter of these tubes is about 1/4 inch larger than that proposed for the test block. Thus, the fuel being tested is slightly larger than that to be used in a modernized reactor.

Initial irradiation conditions are shown in the following table:

Average Power	2390 kw
Maximum Specific Power	155 kw/ft
Average Flow	117.4 gpm
Average T_o	92.4 C
Average ΔT	78.7 C
Average Annulus T_o	95.8 C
Average Hole T_o	91.5 C

The surrounding eight tubes of K5N fuel are operating at conditions which produce a maximum specific power of 80 kw/ft, 72 gpm per column, and an average coolant outlet temperature of 100 C.

No problems have been encountered with the irradiation to date. The operating conditions of the column indicate that the fuel design objectives have been achieved. The fuel will reach 500 MWD/T goal exposure in about 54 full-power days (late December).

About 100 rod-in-tube fuel elements are currently being fabricated for testing an end-spider concept for holding the rod in place, and for determining corrosion behavior. Presently, the inner rod is held in place by crimp-locking with supports similar to arch self-support projections. The end-spider concept is expected to prevent cladding damage that could be produced by chattering during irradiation, and permits end movement of the rod during irradiation as well as easier rod and tube assembly.

11-C. - Physics

Calculations have been made using the MOFDA code to determine the enrichment necessary for the one-inch overbore test block. A uniform enrichment level of 0.8 percent U-235 will provide the same material buckling as natural uranium fuel in present size process tubes at hot operating conditions and average fuel exposure.

An experiment for the K lattice with the same process tube and fuel dimensions to be used in the C reactor test block is being designed for the PCTR. The primary functions of this test are to measure the reactivity of the lattice, wet and dry, and the reactivity gain on flooding. The tube blocks will be machined down on the sides and top to allow two 62.5-mil lucite pieces to be placed in the lattice to approximate partial flooding. Polyethylene will be used to surround the process tubes for the flooding measurements. Calculation with the MOFDA code has been made to determine the enrichment of the fuel to match the k_{∞} of the normal K5N fuel lattice ($k_{\infty} = 1.038$); the necessary enrichment level is 0.855 percent U-235. The results from this PCTR test can be used to make acceptably accurate extrapolation to the C reactor overbored lattice utilizing current calculational methods.

Mission 12 - Fundamental Studies

This Mission is inactive.

Mission 13 - Nuclear Safety Program

The objectives of this program are: (1) to develop improved information for defining the maximum credible accident for the reactors; (2) to develop analytical techniques to describe the sequence of events after an accident in evaluating the rate and amount of fission products released from the reactor fuel; and (3) to develop better methods for mitigating the consequences of an accident. Specific study areas to be addressed include: (1) fuel temperature transients under accident conditions; (2) fission product release from ruptured or molten fuel; and (3) seismic measurements.

Laboratory work on fission product release from irradiated fuel continued. No significant results are available to report.

Mission 14 - Columbia River Studies

The production reactors use treated Columbia River water as coolant on a once-through basis. Thus radionuclides, which are formed largely from the activation of river salts, are released to the river. It is the objective of this program to devise practicable methods of reducing the production and/or release of the radionuclides.

A column of fuel elements was discharged from D reactor at 60 percent of goal exposure, and selected elements and spacers were sent to Battelle-Northwest's Radiometallurgy Laboratory for defilming. Analysis of the film for radio and parent isotopes is in progress. Previously, fuel at goal exposure had been similarly treated. Obtaining and examining fuel at 20 to 30 percent of goal exposure will complete the effort to define film, radio, and parent isotope buildup as a function of time in the reactor.

Experiments continued in the water treatment pilot plant to define the efficiency of parent isotope removal as a function of pH and floc zeta potential when using aluminum sulfate as the flocculant.

The use of deionized water to cool two process tubes continued.

Mission 16 - Other Isotopes

The reactors are of large volume and possess the versatility to irradiate a large number of products. The excellent physics capabilities of these reactors make it quite simple to irradiate isotopes which have potential as energy sources. In addition, some reactors have capability for discharging isotopes without shutting the reactor down.

It is the objective of this program to study all possible radioisotopes, other than those which are included in specific Mission programs, to determine those for which a market interest can be gained. For likely candidates, the feasibility of producing them will be evaluated and demonstrated.

Four capsules of promethium produced by fission product recovery have been prepared for irradiation. The test irradiation will demonstrate the feasibility of burning out the promethium-146 which is the principal contributor to radiation dose. The result could be an energy source which requires less shielding for its applications. A capsule of natural neodymium will also be irradiated in this test. The purpose is to begin investigating the feasibility of producing promethium-146 by this method.

About 70 grams of high density thulium-169 is available for irradiation test purposes. It is planned to encapsulate this material and irradiate it to produce thulium-170. The effectiveness of a DUN-operated reactor for producing thulium-170 will be evaluated, and the quality of the product will be determined.

Mission 17 - Highly Enriched Fuel Program

This program has as its objective the investigation of the safety and economic feasibility of full and partial highly enriched fuel-target loadings. The availability of high speed computer codes, improved experimental methods and facilities, and proven calculational techniques will allow resolution of the questions on the safety of such loadings. The predominant characteristics of the highly enriched fuels which require safety analysis are the lack of U-238, which results in decreased negative Doppler coefficient, and the low melting temperatures of uranium-aluminum alloys.

There were no significant developments on this program during the month.

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ENGINEERING AND TECHNOLOGY--REACTORS

Graphite Studies

The fundamental objective of the graphite temperature limit is to prevent excessive oxidation of the graphite moderator blocks with carbon dioxide and water vapor in the reactor atmosphere. Laboratory tests have shown that 10 percent uniform oxidation of graphite results in a loss in strength of approximately 50 percent. This value of a strength loss of 50 percent has been established as the limit to graphite oxidation over the lifetime of the reactor stack. This limit has been further interpreted as an oxidation rate of not more than 2 percent per thousand days of reactor operation. At this rate limit, the reactor stack could be operated for 15 to 20 years before the graphite strength was reduced 50 percent.

As with any chemical reaction, the rate of reaction of graphite with oxygen, carbon dioxide, and water vapor increases with increasing temperature. While all three oxidants are present to some degree in the reactor gas system, the reaction between graphite and oxygen at the present temperatures is so rapid that all practical precautions must be taken to exclude this oxidant from the gas system. Reaction with water vapor is sufficiently low at present temperatures and concentrations that it does not significantly contribute to the overall stack oxidation.

There is an economic advantage to operating the reactors at higher graphite temperatures, because of the reduced need for helium in the reactor gas atmosphere. With a 50 C increase in graphite temperature, the helium loss is reduced by about 50 percent. The out-of-pocket savings are large enough that a testing program is being performed to evaluate the feasibility of relaxing present graphite temperature limits.

A test program which authorized an increase of 50 C (from 650 C to 700 C) in the graphite temperature limit at B, C, and D reactors has been in effect for about nine months. Preliminary results indicate that, from the standpoint of graphite oxidation with carbon dioxide, continued operation at a 700 C graphite temperature limit is acceptable at C reactor but questionable at B and D reactors because their lower purity graphite is more susceptible to oxidation by carbon dioxide than is the C reactor graphite. A temperature limit of 680 C is indicated by the test results as acceptable for the B and D reactors.

Zircaloy Process Tube Hydriding

The present downstream spacers in the Zircaloy tubes of the K reactors are being replaced with anodized aluminum spacers to prevent the further buildup of a zirconium hydride case layer on the surface of these tubes. The development program is currently directed toward determining whether the existing hydride case layer will have to be removed to prevent further buildup of hydrogen in the base metal, and to develop methods for removing the case layer from in-reactor tubes should this prove necessary. A complete review of the Zircaloy process tube hydriding problem to date is presented in the recently issued document DUN-1623.

Activities during November included the examination of tube 4372-KE which was removed from the reactor after operating eight months with stainless steel spacers. The downstream end of this tube had been sandblasted before it was installed. The object was to determine if stainless steel dummies would also protect tubes with sandblasted surfaces, which are believed to be very prone to hydriding. This is the kind of surface which the tubes will have if the existing hydride case layers are removed by a sandblasting technique which has been developed (as reported below). Hydrogen analyses and metallurgical examination indicated there was no hydrogen pickup in the tube, except for a few hydride needles right on the Van Stone flange. The aluminum Van Stone flange gasket was severely pitted, as has been found previously when stainless steel dummies were used.

In addition, one stainless steel and one 6063 aluminum coupon holder were removed from KE reactor for base metal hydride evaluation. The Zircaloy coupon in these holders were made from tubes with heavy case layers and significant base metal content, which were carefully measured before being installed in the holders and reirradiated. The coupons are presently being analyzed by Battelle-Northwest. These data will aid in the determination of whether base metal hydrogen buildup has continued even though the galvanic currents which resulted in the buildup of hydrogen initially are eliminated.

Although a decision regarding the need for removal of the existing hydride layer from the rear of the K Zircaloy tubes is not expected until April, 1967, development work has continued to define methods, procedures, and techniques for removal if required. Current effort is directed towards a second major on-reactor test of grit blasting machinery and techniques, with modifications as indicated from results obtained in the September trials.

Disposal of the highly contaminated grit wastes initially was done with dry filters only, but an alternate liquid system is being developed in the 108-D Laboratory. A specification and procurement request have been prepared to obtain an alternate vacuum producer, waste-discharge handling system. Fabrication of the third-generation lance drive has been completed and the unit is undergoing laboratory tests. The new drive unit represents the type which can be used in a full-reactor application.

K Reactor Downcomers

Downcomers are used in the reactor effluent piping systems to dissipate the kinetic energy of the effluent water. At the K reactors these are rectangular vessels. 48 x 78 inches in cross-section, which are subdivided into 30 rectangular flow channels.

Tests were performed at KE reactor to evaluate the effect of streamlining the entrances to these channels on the coolant flow characteristics in the downcomer. The results show a definite improvement in downcomer flow characteristics. There was a 20-25 percent increase in coolant flow rate in channels 3, 4, 7, and 8, a decrease in turbulence in the downcomer, and a decreased water level of about one foot in the approach box at current reactor coolant flow rates. These streamlining changes will also be made at KW reactor in view of this improvement in flow capability.

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RTD Response Computer Program

The program to check resistance temperature detector (RTD) response by analysis of data obtained from the high speed scanner is essentially completed. Input data from KW reactor has been obtained and is currently being processed and prepared for the initial test of the program; this is expected to take place in early December.

VSR Drop Times

An analysis of K reactor vertical safety rod (VSR) drop times has been prompted by the need to evaluate more closely the degree of conservatism afforded by present nuclear safety calculations in order to demonstrate the safety of the proposed E-D load. Measured drop times for the VSRs at KW reactor for the past two years were analyzed and the following conclusions drawn:

- (1) For all practical purposes, the minimum cold drop time and the maximum hot drop time are identical.
- (2) The new flexible VSRs have in every instance faster drop times than the older, rigid VSRs.
- (3) The geometric distribution of fast and slow VSRs is relatively uniform. In other words, there are no sections of the reactor in which the average VSR drop time is significantly different than in any other section of the reactor.

These conclusions indicate that it will be possible to meet the nuclear safety requirements for the E-D load, and that in many instances it will be possible to guarantee compliance with hot drop time requirements even though data are available only from cold measurements.

Safing Seal Demonstration

As requested by the Atomic Energy Commission, a demonstration of the Hanford-developed reactor safing seal was conducted at F reactor on November 16. It was attended by representatives of 43 of the United Nations General Assembly delegations, and by 18 representatives of various news media. AEC representatives included Commissioner S. M. Nabrit.

The demonstration consisted of: (1) a brief technical explanation of the safing seal concept and its adaptability to various types of nuclear reactors, (2) the actual installation of one safing seal assembly on the front face of the reactor, and (3) the inspection and subsequent removal of seals (previously installed on F reactor) to demonstrate methods of inspecting for attempted circumvention of the seal. Written material describing the seal concept was provided to the visitors in support of the oral presentations. Following the demonstration, tours were conducted through the reactor building. This was the first public opening of a DUN-operated production reactor to foreign nationals and the press.

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Ben Franklin Dam Study

Engineering efforts in support of the study to determine possible effects on Hanford facilities of the proposed Ben Franklin Dam have continued on both the 385-foot msl and the 400-foot msl pool elevation cases. Authorization was received from RL-AEC during the month to proceed with study of the 400-foot msl case.

Model verification tests in the R. L. Albrook Hydraulic Laboratory at Washington State University disclosed discrepancies between model and prototype data which necessitated the taking of additional river velocity measurements. Corrections have been made in the model to achieve the required similitude, and verification of the models is currently complete.

At the request of RL-AEC, a tour of the hydraulic model complex at WSU was arranged and conducted (on November 16) for interested personnel from several state and federal agencies. Included were the U. S. Army Corps of Engineers, the Washington State Department of Fisheries, the U. S. Fish and Wildlife Service, and the Federal Water Pollution Control Administration.

Soil stabilization studies are in progress by the firm of Shannon & Wilson, Inc., of Seattle, Washington, to determine the effects of changes in groundwater table elevations which would result from construction of the proposed dam.

On-Reactor Polymer Tests

A long-range program is under way to test the service capability of a limited number of specific polymer materials under the most damaging of on-reactor conditions, using a device which has been installed on the rear face of D reactor. Material selection is based on recommendations by vendors and users. The materials are fabricated as O-rings which are tested in the fixture under static and dynamic conditions while exposed to hot process water piped from a rear effluent line. This water is piped through a heater to increase its temperature approximately 15 C. Samples are installed as received from the vendors, and also with added radiation dosages of 5×10^6 and 5×10^7 R.

These tests will provide a direct comparison of materials against a silicone rubber control which has been used on these reactors successfully for a number of years. The program will eventually include other types of test equipment to evaluate other materials, such as sealants and coatings, for reactor rear face service.

Project Engineering--Reactor Facilities

Project Status Summary

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

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DAP-500, Increased Raw Water Pumping Capacity--181-B/C River Pumphouse

This project, managed by RL-AEC, covered the moving of two river pumps and their appurtenances from the deactivated 100-H Area to the 181-B River Pumphouse. Little design was required, and the plant drawings will be as-built to show the installation. The work was completed in approximately six months in spite of a one-month delay caused by the reactor work stoppage in July-August. The additional pumps increase the capacity of the 181-B pumphouse by 10,000 to 15,000 gallons per minute depending on the river elevation and the routing of the water.. The project was completed November 17 at a cost of \$56,000 compared to an authorized \$62,000.

DCE-505, Boiler Control Improvement--165-KE and KW

Directive No. EQT-058 was issued November 2 by RL-AEC authorizing \$10,000 for start of design. The work will consist of replacement and modifications to existing obsolete boiler control system components in the K plants at both 165-KE and 165-KW to provide: (1) Increased boiler safety, (2) increased reactor safety through improved reliability of boiler response to emergency steam demand, and (3) reduced maintenance.

The project schedule calls for detail design to start January 1, 1967 and be complete July 1, 1967. Physical performance in the field will be started about August 1, 1967 and the first area beneficial use should be attained by January 1, 1968; the second area beneficial use is expected by April 1, 1968. The directive completion date for the total project is May 15, 1968.

ENGINEERING AND TECHNOLOGY--FUELS AND TARGETS

Production Test Summary

The status of current production tests is summarized in Appendix B.

Electro-Reduced Uranium Ingots

The chemical composition specification for electro-reduced ingots cast at Mallinckrodt Chemical Works has been waived as they exceeded the specified limits for iron, nickel, chromium, and aluminum. These ingots have been received at National Lead of Ohio (NLO), and it was requested that they machine 0.50 in. from the OD to improve the surface quality. NLO has reported that the ingots were machined and all have good surface quality for rolling. Hot hardness tests are being performed on this alloy to see if any problems will arise during the rolling of this particular alloy to rod stock. Results of these tests are not yet available.

Induction Heat-Treating at NLO

The installation of induction heat-treating equipment and facilities at NLO will begin in February, 1967. As presently scheduled, the testing of this new heat treating equipment will be divided into three phases:

Phase A

This testing, which will be conducted at the equipment vendor's plant, will consist of a test on 1,000 cores of different fuel models to evaluate operation of the equipment before it is shipped to NLO.

Phase B

After the installation of the equipment at NLO, tests will be conducted on cores for a thorough metallurgical evaluation of the induction heat-treated product.

Phase C

This phase will be a ten-day throughput and capability test prior to the placing of the unit in production of I & E cores. Cores from this phase will be shipped to DUN for reactor charging.

Specification Revisions

DUN-1659, "Provisional Engineering Specifications, Anodized Aluminum Spacer Elements," was prepared and issued. These specifications define the now-known quality requirements for these treated spacer elements; however, use of the anodized spacers (dummies) is expected to provide better definition of reactor needs upon which revisions to the specifications will be based.

RL-REA-2177, "Engineering Specification, Thorium Oxide Target Elements," was revised to permit the use of a thorium wafer in these elements as a heat barrier to reduce weld rejects.

Anodizing Process

Process optimization tests for the aluminum dummy anodizing process showed that the best coatings were obtained with 70 minutes anodize time, 20 amp/ft² current density, 7 percent oxalic acid concentration, and 20 C temperature. The best cleaning process appeared to be a severe caustic etch followed by a nitric acid dip. It was predicted that a 20 percent increase in throughput capacity would be achieved, without jeopardizing the anodic film quality, by running at 16.5 amp/ft² current density, 60 minutes anodize time, 7 percent oxalic acid concentration, and 18 C to 25 C temperature. The predicted average film thickness and resistance values under these conditions were 0.0014 inch and 15,700 ohms, respectively, which are in good agreement with production data.

Ledge and Groove Corrosion

Although out-of-reactor simulated ledge and groove corrosion could be accomplished in the past, duplication of results had been unsuccessful. Recently an accelerated corrosion test, utilizing a caustic solution, produced corrosion patterns strongly resembling the ledge and groove corrosion observed on irradiated fuel elements. Additional testing continues to show

duplication of corrosion patterns and indicates validity of the test. Results to date are summarized as follows:

- (1) A normal autoclave film adds a measure of corrosion protection to the fuel surface.
- (2) The autoclave film is delicate and its protective properties are easily destroyed with normal handling.
- (3) The presence of oil on surfaces prior to autoclaving, produces a less corrosion resistant film. Corrosion patterns produced resemble those observed on irradiated fuel. (Fuel projections are coated with oil before ultrasonic welding.)
- (4) Working or stressing the fuel element after autoclaving destroys the film in localized areas and reduces the quality of the autoclave film over the entire piece. Fuel elements subjected to working or stressing before autoclaving resisted accelerated corrosion attack very well.
- (5) Weekend autoclave cycle films (approximately 70 hours) show more protection than weekday films (20 hours), but only when they were not damaged. Increasing the cycle to 150 hours, by using three 50-hour cycles, did not add to corrosion resistance of the film.
- (6) Application of black marking ink (Flow Master) after autoclaving enhances the corrosion resistance of an autoclave film. The marking ink appears to act as a sealant to the porous autoclave film.

Overbore Fuel Process

Process parameters have been developed and verified for cladding C6ANS (C reactor overbore) fuel with tubing components. No major problems were encountered using the standard hot-die-size processing conditions. The test results showed that increasing standard end-bond pressure by 12-1/2 percent improved both top and base end cap-to-clad bond strength. Use of the slightly increased end-bond pressure has been initiated in the current C6ANS production run. Preliminary tests show that the base closure bond using the increased pressure is equivalent to the top closure bond. No unusual dimensional changes have been observed on the finished fuel as a result of the slightly increased end-bond pressure.

ADVANCED CONCEPTS AND PLANNING

MFCs AND OTHER PROPOSALS

MFC-10 - Centralized Reactor Operating Personnel Training Program

A preliminary report of the MFC-10 study was issued. It is currently being reviewed by interested DUN and AEC personnel.

MFC-12 - New Capabilities for Production Reactors

There were no significant developments on this MFC.

Study of Application for Hanford-Produced Co-60

The latest revision of this MFC proposal was issued as DUN-849 REV1. The estimated costs are \$20,000 for FY-1967. The proposal is for a study to evaluate present and potential applications of Co-60 as produced from cobalt-coated process tubes.

Space-Dependent Transient Tests in a Deactivated Production Reactor

This proposal was discussed with the staff personnel of the AEC's Division of Reactor Development & Technology early in the month. Subsequently, representatives of the SPERT program at Idaho Falls visited Richland to discuss the proposal. DUN has been encouraged by both parties to send the proposal to the Director of DRD&T, and it is being rewritten to eliminate the MFC format.

OPERATIONS RESEARCH

CAGE Upgrading

The CAGE-Technical program upgrading and debugging is complete. The new version permits flexible shutdown-startup reactor operations. Shutdowns of one or two consecutive quarters for the purpose of reactor modification are identified and coded as to quarter and year for use by CAGE-Financial.

Provisions now exist to enable the use of variable assay cascade enrichment (for blend) for use in conjunction with discounted cascade cost parameters. The input data editing has been expanded and input feedback systems have been added to screen out obviously erroneous data and warn of incomplete or missing input parameters.

A subprogram has been added to allow computation of oralloy values based on its use as a blend stock and the value of Pu-238.

Storage requirements have been reduced and computation time per case has been reduced by about seven seconds by realigning the core storage and subprogram interlinkage. The logic of the decision-making apparatus used to determine the best feed stock makeup and disposition of unused fuel materials has been improved.

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Upgrading and debugging of CAGE-Financial is now 90 percent complete. Remaining effort is centered around several minor adjustments and final debug. The debugging process will extend though actually making some decremental analyses before any case cards will be shipped to Oak Ridge. The new program includes a much improved intermediate printout.

Oralloy Value

Study continued on the value of returned oralloy if used as a sweetener to re-enrich Hanford reactor tails for recycle. Streams down to 0.947% U-235 assay have now been tested, assuming unlimited availability of oralloy starting in 1968. The preliminary result of these calculations is that the incremental neptunium thus produced would cost about \$80/gram. The cost of the oralloy used each quarter of the campaign was based on the value to the AEC if the oralloy was used to sweeten natural uranium stock to 2.6% U-235.

The test was repeated for a higher oralloy cost based on its value if blended with 2.0% U-235 cascade output to produce 2.6% U-235 which is also under consideration in the Pu/Oy study. In this case, the incremental cost of the neptunium rose to \$280 per gram, which means that oralloy is too expensive to enhance Pu-238 production. If Pu-238 had a higher value (say \$500/gram), then the oralloy (even under these circumstances) would be of value to the production reactors.

Cascade-Reactor Data Linkage

To provide the desired linkage with the cascade computer model, additional CAGE-Technical output is required. DUN is proceeding to provide this feature.

Executive Model

DUN work on the executive model has been negligible during the month. The current main effort is concentrated on developing a feedsite model which will adequately treat uranium recycle. DUN's input to this segment has been completed earlier. A test problem will be formulated at a Richland meeting in December.

Cm-244 Model

The Richland portion of the curium-244 production model has been completed. The model calculates production and incremental costs for production campaigns utilizing both Hanford and Savannah River facilities. The production and cost mathematical relationships for Savannah River are still being developed by duPont personnel. Processing schedules are being developed for analysis with the model which will evaluate the effects of various operating modes and plutonium supply assumptions on both production quantities and costs of Cm-244.

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ADVANCED PLANNING

Long-Range Plan

Input has been prepared and presented to assist in the evolution of a long-range plan for the AEC production programs falling under the scope of Douglas United Nuclear. The following planning elements have been completed to date: environment (including competition, market appraisal, and special factors), strengths and weaknesses of the DUN-operated facilities and DUN manpower, and capital requirements to permit meeting all foreseeable future opportunities. The next steps are to consider and select opportunities, evaluate alternatives, and select the plan. The plan will cover ten years.

Participation in AECOP

A Senior Engineer from the Company's Advanced Concepts and Planning Section will be Hanford's first representative to the newly-formed AEC Combined Operational Planning Group (AECOP). This new organization will be headquartered at Oak Ridge.

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IRRADIATION SERVICESIRRADIATION ACTIVITIES SUMMARY

The following irradiation services were performed for other contractors and agencies:

Battelle-Northwest

<u>Reactor</u>	<u>Test Hole</u>	<u>Facility</u>	<u>TISR</u>	<u>Number of Irradiations</u>	<u>Material - Purpose</u>
KE	2D	Quickie	172	46	Activation analysis of various materials to determine submicrogram quantities of isotopes (irradiation times: minutes to days).
KW	2A	Quickie	172	12	
KW	2A	Quickie	184	27	Activation analysis in support of AEC-designated program (irradiation times: minutes to days).
KE	2D	Quickie	184	35	
KW	3D	General Purpose	236	2	One metallic creep rate capsule was charged in support of a creep rate study program (irradiation times: months to years).
KW	4C	Snout	243	2	Two cooled tensile specimen capsules were charged and discharged in support of fast neutron damage to metals program (irradiation times: days to years).
KW	2A	Quickie	254	8	Activation analysis of river water samples in support of a program to reduce water activity (irradiation times: hours to days).
KE	2D	Quickie	254	6	
KW	2A	Quickie	271	1	Depleted uranium samples were irradiated to provide radiation counting source standards (irradiation times: days to months).
D	-	PCFF	271	1	

<u>Reactor</u>	<u>Test Hole</u>	<u>Facility</u>	<u>TISR</u>	<u>Number of Irradiations</u>	<u>Material - Purpose</u>
KW	2A	Quickie	346	2	Various materials were irradiated primarily for activation analysis (irradiation times: hours to days).
KW	2D	Quickie	346		
KW	2B	Snout	351	3	Controlled temperature graphite capsules were irradiated in support of studies of fast neutron damage to graphite at various temperatures (irradiation times: hours to days).
<u>Oak Ridge National Laboratory</u>					
KW	2B	Snout	601	1	One capsule containing 126 cobalt disks was charged for production of cobalt-60 (irradiation time: 2 months).
<u>NASA-Lewis Research Center</u>					
KE	2B	Snout	701	2	After operating at design conditions for two weeks a UO ₂ -Mo capsule failed and was discharged and returned to the customer. A second capsule was charged and is operating at design conditions. This program is in support of development of space propulsion reactor fuel (irradiation goal: 1000 hours).

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<u>Reactor</u>	<u>Test Hole</u>	<u>Facility</u>	<u>TISR</u>	<u>Number of Irradiations</u>	<u>Material - Purpose</u>
KW	2D	General Purpose	701	1	The UO ₂ -Mo capsules failed after 1150 hours and 1100 hours respectively, at design conditions, and were pulled out of the flux region of the reactor into the biological shield (irradiation time: run to failure).
KE	2A	General Purpose	701	1	

Reed College

KE	2D	Quickie	801	2	A set of samples containing blood serum was irradiated in support of a cancer research program (irradiation time: three days).
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NASA-Ames Research Center

KW	2A	Quickie	802	1	775 milligrams of recovered strontium nitrate were received from Battelle-Northwest and charged. 1200 milligrams of material were discharged, reencapsulated by Battelle-Northwest, and shipped to NASA at Cape Kennedy. These irradiations are for the purpose of providing NASA with Sr-85 radiation sources for the biosatellite program.
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ADMINISTRATION-- GENERALBENEFIT PLANS

The Company has been advised that the Pension Plan, Salaried Savings Plan, and the Wage Savings Plan have been approved by the Internal Revenue Service.

AEC APPROVAL LETTERS

At month-end the following approvals had not been received from RL-AEC:

<u>Approval Number</u>		<u>Date of Transmittal to RL-AEC</u>
AP-39	Pension Plan portion of letter entitled, "Pension Plan, Salaried Savings Plan and Wage Savings Plan"	January 12, 1966
ATD-89	Trailer Movement Expenses for Newly Hired Exempt Employees - Appendix "B" Modification No. MO-10	October 25, 1966
ATD-90	DUN Participation in AEC Combined Operations Planning Committee	November 23, 1966
ATD-94	Reclassification of 105-C Loop	November 9, 1966
ATD-97	Approval Request for General Ledger Manual Revision - Account 02CR-400202, 02AR-400200, 02BR-400201	November 18, 1966
ATD-98	FY-1967 Budget for Attendance in Meetings of Professional and Trade Societies	November 23, 1966
ATD-99	FY-1967 Budget for Attendance at Off-Site Courses and Seminars	November 23, 1966

SAFETY PROGRAM

In April, 1966, the Company inaugurated a special safety program under the slogan "Let's Make a Million", with an initial goal of 1,000,000 man-hours without a disabling injury. That goal was achieved on May 8, at which time the program was continued with the slogan "Let's Make It Two". The 2,000,000 mark was reached on September 2, and the promotional phrase was changed to "We're Shooting for Three". On November 30 the man-hours total exceeded 2,800,000, and hopefully the 3 million mark will be reached in mid-December. Appropriate recognition of this achievement is planned.

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APPENDIX APROJECT STATUS SUMMARY--REACTOR FACILITIES

<u>Title</u>	<u>Authorized</u>	<u>Percent Complete</u>		<u>Status</u>
		<u>Design</u>	<u>Construction</u>	
CAI-105, Modifications for Use of Bauxite-KE and KW Water Plants	\$395, 000	100%	95%	ATPs discontinued. Revised acid and product line installations needed and being installed. Dry unloading system vendor on-site to study equipment problems.
CGI-125, Revision 1 Replacement and Improvement of Fixed Gamma Radiation Monitoring Instrumentation-Existing Hanford Production Reactor Areas	\$385, 000	100%	90%	Partial beneficial use was obtained in 105-C, 105-KE and 105-KW. ATPs are being conducted on the 105-KE and 105-KW systems.
CAI-174, High Speed Scanning System for Temperature Monitoring-KE Reactor	\$861, 000	100%	98%	Beneficial use has been achieved. The system is being operated on test prior to start of the six-month reliability ATP and safety circuit tie-in.
CAI-175, Grounding of the 2400-Volt Systems-100-B and D Areas	\$150, 000	100%	78%	Construction at 181-B ("B" side), 182-B, 183-B, 190-B, 182-D, 183-D, and 190-D has been completed.
AEC-191, Fence Relocation-100-B and D Areas	\$300, 000	100%	85%	Patrol will move into new patrol buildings at B and D Areas in the next two weeks.
DAP-500, Increased Raw Water Pumping Capacity-181-B/C River Pumphouse	\$62, 000	100%	100%	Complete 11-17-66.

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<u>Title</u>	<u>Authorized</u>	<u>Percent Complete</u>		<u>Status</u>
		<u>Design</u>	<u>Construction</u>	
DCP-503, "C" and "D" Work Platform Safety Improvements-Five Reactors	\$465, 000	100%	0%	Work to start in December. Material and equipment pro- curement is under way.
DCE-505, Boiler Control Improvements-165-KE and KW	\$240, 000	0%	0%	A total of \$9, 000 was released on 11-21-66 to the on-site Architect- Engineering for start of detail design.
DAP-507, Work Area Fog Spray Systems- 105-B, C, D, KE, and KW	\$130, 000	75%	0%	Design is in progress.
CGI-967, High Speed Scanning System for Temperature Monitors- KW Reactor	\$1, 100, 000	100%	99%	Electrical tie-in to the No. 1 Safety Circuit was completed 11-26-66.

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APPENDIX B

STATUS OF PRODUCTION TESTS (as of 11/20/66)

Test No.	Type Element	Tubes	Reactor	Goal Exposure	Current Exposure	Remarks
PITA-32 Supp. C	Natural U and 94 Metal HDS fuel for ribbed tubes	27(N) 36(E)	B	Normal	Variable	*Irradiation continued with- out incident. Fuel charged in intermediate power tubes outside enrichment ring.
	Natural U and 94 Metal HDS fuel for K Zirc tubes	1(N) 30(N)	KW KE	Normal	Variable	*Same as above. 26 natural U columns discharged from KE during October.
PITA-32 Supp. B	Natural U HDS fuel for C over- bore facility	60	C	Overbore Variable	Variable	Irradiation continued with- out incident.
IP-766 Supp. A	Pu-Al alloy fuel	1	B	3750 MWD/T*	3030 MWD/T	Test to obtain data on build- up of higher Pu isotopes. Irradiation continued with- out incident.
IP-805	NpO ₂ elements	1	KW	180 oper- ating days	91 days	Test to obtain data on Pu- 238 production. Irradia- tion continued without incident.
PTA-011	HDS and AlSi- bonded natural uranium	10(2)***	B	Normal	0 MWD/T	Program for routine moni- toring of HDS and AlSi- bonded fuel performance. First cycle discharged from B reactor on 11/4. Second cycles were discharged from KE (10/21) and KW (10/31) at sub-goal exposure due to groove corrosion problems with HDS fuel. Third cycle at KE is all AlSi- bonded fuel; no PTA-011 fuel is now in KW.
		10(1)	D	Normal	480 MWD/T	
		10(3)	KE	Normal	120 MWD/T	
		10(2)	KW	Normal	460 MWD/T	

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STATUS OF PRODUCTION TESTS (contd)

Test No.	Type Element	Tubes	Reactor	Goal Exposure	Current Exposure	Remarks
PTA-039	AlSi-bonded natural uranium	20	KW	600 MWD/T	550 MWD/T	18 columns discharged 10/31/66, currently awaiting post-irradiation measurement in C Basin.
PTA-021	HDS rod-in-tube natural uranium	1	KE	500 MWD/T	250 MWD/T	One column of a possible total of five charged in test channel 3865-KE on 10/20/66. The test column consists of 24 8.870-inch long-rod-in-tube fuel elements of the XMIN design. A discharge date in late December is anticipated.
PTA-023	HDS natural uranium	20	B	800 MWD/T	250 MWD/T	Test to evaluate dimensional performance of solid oil quenched cores as compared to hollow, oil-quenched cores. Two columns to be removed at 400 MWD/T for early evaluation. Test charged on 9/23/66. Entry in last month's report inadvertently omitted.
PTA-040	AlSi-bonded 125 Metal	20	KW	1200 MWD/T	370 MWD/T	Test to evaluate dimensional performance of five alloys at approximately 150 percent of normal uranium specific power. Charged 10/31/66.

*PITA-32 Remarks are as of 10/31/66.

**Accumulated exposure to adjacent columns.

***Number in parenthesis refers to cycle currently in reactor.

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DUPLICATE

APPENDIX CSIGNIFICANT REPORTS ISSUED

<u>Number</u>	<u>Classification</u>	<u>Author(s)</u>	<u>Date</u>	<u>Title</u>
DUN-849 REV1	Secret	LE Kusler	11/23/66	Study of Application for Hanford-Preduced Co-60
DUN-925	Secret	JP Schmidt and CG Organ	11/9/66	PTA-021, Preliminary Irradiation of Rod-In-Tube Fuel
DUN-1282	Secret	SS Jones	11/7/66	Review of the Results of PT-IP-813, Modified Export System and D Reactor Tripout Test
DUN-1300	Secret	RE Baars	11/18/66	PTA-040, High Power Density Core Alloy Evaluation
DUN-1587	Secret	KL Hladek	11/16/66	Heat Transfer Calculations for PTA-021
DUN-1623	Confidential	WK Alexander	11/7/66	Review of the K Area Zircaloy Process Tube Hydriding Problem
DUN-1653	Secret	R&E Staff	11/15/66	Technical Evaluation of E-D Loading, K Reactors
DUN-1659	Confidential	HF Tew	10/1/66	Provisional Engineering Specifications--Anodized Aluminum Spacer Elements
DUN-1686	Confidential	GF Jacky and HF Tew	11/2/66	Exploratory Studies on End Bonding During Sizing, Hot-Die-Sizing Fuels Fabrication Process
DUN-1701	Secret	WN Adachi	11/10/66	Completion Report, Production Test IP-761, Installation of Flat Horizontal Control Rod-KW Reactor

<u>Number</u>	<u>Classification</u>	<u>Author(s)</u>	<u>Date</u>	<u>Title</u>
DUN-1712	Secret	CR Zook	11/14/66	Ball Safety System Entry Modifications Program
DUN-1714	Unclassified	GL Erickson	11/15/66	Transport Characteristics of the 107-C, D, and KW Basins
DUN-1723	Secret	R&E Staff	11/18/66	Comparison of Production Reactors to Nuclear Power Plant Design Criteria
DUN-1725	Secret	HG Spencer	11/17/66	Trip Reports
DUN-AOP-11	Secret	RD Jensen	11/3/66	Decremental Studies Using Savannah River V/R Cases

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APPENDIX D
EMPLOYMENT SUMMARY
(As of 11/30/66)

	<u>Exempt</u>	<u>Non- Exempt</u>	<u>Total</u>	<u>Change from 10/31/66</u>
<u>Operations</u>				
General	1	1	2	0
Manufacturing	243	808	1 051	-4
Production Fuels	77	290	367	+1
Research and Engineering	66	40	106	-2
Facilities Engineering	<u>55</u>	<u>18</u>	<u>73</u>	<u>-1</u>
Sub-Total	442	1 157	1 599	-6
<u>Finance and Administration</u>				
General	1	1	2	0
Finance and Administration	64	63	127	0
Employee and Community Relations	17	9	26	0
"Technical Graduates"(1)	<u>23</u>	<u> </u>	<u>23</u>	<u>0</u>
Sub-Total	105	73	178	0
<u>Advanced Concepts and Planning</u>	8	1	9	0
<u>Company General</u>	<u>6</u>	<u>2</u>	<u>8</u>	<u>0</u>
Total	561	1 233	1 794	-6

(1) Employees in this interim classification are on the E&CR Section roll during their rotational training.

SAFETY STATISTICS

Disabling Injuries: November	0
CY to Date	1
Days Since Last Disabling Injury	310
Man-hours Worked Since Last Disabling Injury	2, 872, 862

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SAFETY AND CONTROL ROD SYSTEMS
OF
THE HANFORD PRODUCTION REACTORS

The nuclear control system for the five production reactors operated by Douglas United Nuclear for the AEC differ in detail, but all follow the same general concept. There are four separate control systems at each reactor; three are routinely active, and one provides backup if the other three fail to perform satisfactorily. The active systems consist of a horizontal rod system to provide level, or operating, control; a vertical safety rod system to provide shut down and hold down control; and a supplementary multipurpose manual system to help compensate for xenon burnout, to provide small reactivity adjustments, and to provide bulk poison for total control. The backup system, if called upon, fills the vertical safety rod channels with small balls of poison to ensure the reactor will become and remain subcritical.

This report will describe both the horizontal control and the vertical safety rod systems. These systems have much in common: Both require passing long, slender elements through channels in the reactor core; both have required accommodation to similar problems from graphite moderator movement; both now reflect similar solutions to these moderator movement problems. In addition, the horizontal control rods today reflect improvements that increase their ability to withstand prolonged in-reactor exposure, a capability not demanded of vertical safety rods. This report will also describe these improvements.

REACTOR MODERATOR CONDITION

Before describing the two rod systems and their evolution, a brief consideration of reactor moderator condition should be helpful. All five reactors are similar, differing in size rather than in concept. With the exception of moderator temperature, describing any one essentially describes them all.

The reactor moderator consists of what once was a neat cubical stack of graphite blocks, criss-crossed, keyed, and otherwise geometrically interlaced to hold itself together. Two or three thousand process tube channels pierce this stack from front to rear, forty or fifty open channels run from top to bottom for vertical safety rods, and ten to twenty open horizontal channels run from one side to the other for horizontal control rods.

When the reactor was new, all of these channels were straight and true and lined up exactly with their matching holes in the reactor shields. But irradiation causes graphite to both grow and shrink, depending upon temperature. A Hanford moderator does both: It grows in the outer, cooler regions, and it shrinks in the central, hotter zone. This action has caused the graphite stack to change its shape and distort the once true channels.

From the side, the horizontal control rods in the lower regions see relatively little distortion because the dimensional changes, which accumulate from the reactor base, do not add up to anything significant. Near the top, however, these changes have accumulated to several inches--as much as six inches for the two larger K reactors--and are manifest primarily as a central, hotter zone subsidence; growth at the cooler sides has not been serious. These upper rod channels are fairly flat across the central zone and slope upward for about seven feet at either side. Rods entering these channels must negotiate these inflections and remain free enough to move easily on demand.

From the top, vertical safety rods see relatively little change in the central, hotter zone because there the stack has simply subsided. But at the front and sides, cool zone growth surrounding hot zone shrinkage has caused the channel tops to move toward the center, creating rather serious channel curvature as well as misalignment with mating holes through the top shield. Toward the rear, stack growth has not been as noticeable because the stack operates somewhat warmer there than at the front where cold cooling water enters.

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Moderator movement has caused difficulties other than channel distortion. Graphite is not very ductile, and when stresses from this movement get high enough, the blocks break and create additional problems. For example, pieces either fall or project into rod channels, fault lines develop across significant portions of the stack and produce steps in the rod channels, and gaps open all across the stack that in themselves cause little harm but do collect balls from the backup poison system whenever they drop. These balls inevitably get into lower horizontal rod channels and interfere with rod operation.

Finally, this movement is continuous; moderator distortion will increase with time. Fortunately, the rate at which these changes proceed is quite predictable, and no surprises should arise. Because of this predictability, it has been possible to develop adequate means to accommodate this moderator distortion both now and in the foreseeable future.

THE VERTICAL SAFETY ROD SYSTEM

The vertical safety rod system consists of several dozen uncooled poison elements about forty feet long that can fall under their own weight into channels provided for them in the graphite moderator. All are held raised by power latches that release and permit the rods to fall should the power be interrupted.

The vertical safety rod system serves two general purposes, poisoning to shut the reactor down and poisoning to hold it down. Shutdown criteria allow about two seconds after the safety circuit trips for the rods to insert 90 percent of their length. To satisfy this requirement, the rods must fall freely when unlatched, and they require open and unbound channels. Holddown criteria merely require sufficient poison to be in the core to keep the reactor subcritical. As long as enough rods will go into

their channels, the holddown criteria can be achieved. As the moderator becomes more and more distorted, these criteria become more and more difficult to meet. For example, some of these channels had become so curved, crooked, and obstructed by broken blocks that occasionally the rods would only partially insert and then doggedly resist withdrawal.

To remedy this condition, two complementary and rather obvious paths were chosen: First, enlarge, straighten and clear the channels; and second, provide poison elements flexible enough to negotiate the most crooked channel. Early diagnoses showed that even after a channel was enlarged and cleared it would still require sleeves to hold broken and loose graphite blocks out of the channel and to keep poison balls out of gaps in the moderator. Also, a practical limit to ultimate straightness was recognized; a completely straightened channel would cut through too many process tube channels. Consequently, flexible rods would also be needed, even in some straightened channels.

Each vertical rod channel mates with holes through the reactor's two shield layers. The first layer, the thermal shield, consists of about one foot of cast iron resting on top of the graphite moderator. The second layer, the biological shield, is self-supporting and (at C and the K's) consist mainly of about five feet of heavy aggregate concrete loaded with steel punchings. Between these two shield layers is a gap, or plenum, of nominally 7 inches at C reactor to 16 inches at the K reactors; as the moderator subsides, this gap increases. The holes through the two shields require enlarging to pass both channel renovation tools and new graphite sleeves.

Vertical Channel Renovation

Although the channel renovation procedures at the C and K reactors are similar, the actual operations at the K reactors are more difficult, primarily because enlarging C reactor's biological shield holes does not require cutting any concrete. Because they are the more demanding, operations at the K reactors will be described.

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CONCRETE BIT

EXPANDABLE PULL CUTTER

BIOLOGICAL SHIELD BORING MACHINE

GRAPHITE REAMER

THERMAL SHIELD
ENLARGEMENT MILL

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The sketch shown at the right will help visualize the operations as they are described. The lower step diameter in the upper biological shield is about 4-3/4 inches; the next step measures about 7 inches diameter. The first task removes this lower step. To do so, two of the four tools illustrated on the preceding page were developed; the two together are called the "Biological Shield Boring Machine". The first operation employs the "Expandable Pull Cutter; which light skin-cuts the lower step ID with the cutter shown. Once through the shield, the tool is held centered in the step by the small carbide wear pads visible along the tool's side. Then, two fly cutters extend from the tool's side (they are retracted in the illustration) to cut away the steel plates which lie beneath the concrete as the tool rotates and is pulled back up. Cutting this plate away allows concrete debris to escape during the final operation.

This final operation employs the "Concrete Bit" to cut everything, steel sleeve and concrete, from 4-3/4 to 7 inches diameter in only one pass. This "bit" appears deceptively simple in the illustration; however, developing a combination of carbide cutters and tool steel to make this cut was quite a task.

Once through the biological shield, the passage through the cast iron thermal shield also requires enlarging. This hole tapers from a round, near-6 inch diameter entrance at its top to a 4-inch square

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exit at its bottom which lies over the rod channel. The first operation passes a pilot drill slightly over 4 inches diameter through the hole, cutting small round segments from the square hole sides. Next, the hole is cut to an oblong shape in a series of cuts with the "Thermal Shield Enlargement Mill", shown just entering a cast iron block in the illustration. The final passage is 4-1/4 by 5-7/8 inches.

When these operations are completed, the channel down through the graphite is enlarged. To do this, a special "Graphite Reamer" was developed to both enlarge and straighten the channel in one pass. This tool is also pictured. The two mill-type cutters at either side of the reamer cut the short sides of the rectangular hole while a series of bar cutters attached to a roller chain cut away the long sides. Final channel dimensions are 4-1/8 by 5-13/16 inches. In the last channel renovation operation, a series of graphite sleeves are installed in the enlarged channels. A typical sleeve is shown below.

To illustrate more graphically how successful this renovation effort has been, two photographs, one taken before and one after, appear below. The very evident steps, gaps, and protruding blocks are quite typical. Also

BEFORE REAMING

AFTER SLEEVING

TYPICAL SLEEVE

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typical is the "Holland Tunnel" after sleeving. These pictures are of actual vertical rod channels in a K reactor.

Since channel renovation commenced in 1963, the operations have become relatively routine. All of the techniques and equipment necessary to do this work are now developed and have proven quite satisfactory. So far, 27 channels, 18 in the K reactors and 9 in C reactor, have been cleared, enlarged, and sleeved. Each year, these numbers will increase. The reactors are now in a good enough condition to permit diverting some of this effort to correcting horizontal control rod channel deficiencies which will be described shortly.

Vertical Safety Rods

Vertical safety rods for the first three Hanford reactors originally consisted of long, hollow, cadmium-plated steel tubes just under 2-1/4 inches diameter and about 40 feet long. These rods were entirely satisfactory as far as nuclear control was concerned, but they rusted very badly. Early replacements were necessary, and these were made with a mechanically similar rod made of stainless steel alloyed with about three percent boron. Subsequently, these stainless steel alloy rods came into use at all reactors. Two reactors, DR and C, used larger rods of 3 inches instead of 2-1/4 inches diameter.

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About five years ago, when channel distortion began to have an appreciable effect on vertical safety rod operation, a few rods were fitted with joints to see if a little more flexibility would help. It did. Ultimately, the rod design was changed to a series of two-foot long sections connected together by a loose knuckle, giving about five degrees of angular freedom at each joint. The sketch on the preceding page shows one of these joints. Inside the tubes, between the joints, lie slugs of a boron carbide mixture that add to the rod's poison inventory. These rods have performed very well, and today they are standard replacement stock for all five of the production reactors. Only the number of sections change to accommodate different reactor core heights.

THE HORIZONTAL CONTROL ROD SYSTEM

The horizontal control rod system consists of nine to twenty long, slender, cooled poison elements that slide in horizontal channels provided for them in the reactor core. All these elements enter from one side of the reactor and are driven, electrically or hydraulically, both in and out. This system provides the major flux shaping and power level control required in operating the reactors.

Quite a variety of these rods are employed today; the five reactors, though quite similar in many other respects, do have differences which prevent a common rod design from serving all. For most of the rods in use today, there are newer, better designed rods awaiting to be installed when the old rods become too troublesome.

Graphite distortion problems affect this system in several ways. Upper channels suffer from pure distortion: They slant downward from the near side shield ports, dropping about six inches in seven feet in the worst case (the K reactors), then become level across the central core zone, then slant upward for about seven feet at the far side, mirroring the channel entrance condition. Lower channels suffer indirectly from distortion: Open gaps between graphite blocks permit poison balls to filter into the channel and cause the rod to bind.

Both K reactors and C reactor suffer from distortion-induced rod binding much more severely than the older B and D reactors for two reasons: First, B and D have significantly less distortion; and second, B and D already have flat rods of considerably greater flexibility than the round rods of C and K. Actually, distortion problems at B and D reactors to date have not been serious enough to require any remedial efforts. Such has not been the case at the C and K reactors.

The standard C and K reactor horizontal control rods are similar in most respects, differing primarily in diameter. The rods ride in "U" shaped graphite track blocks that line each rod channel. These blocks hold the rod rather high in the channel and leave only 1/4 to 3/8 inch vertical clearance between rod and channel top. Because of this small clearance and the relatively stiff round rods, these rods begin to bind significantly as graphite distortion advances--the upper channels are first and worst.

To cope with these problems, two paths were chosen. In the first, both more flexible control rods and more channel clearance were provided for the upper, crooked channels; in the other, troublesome lower channels were lined to prevent ball infiltration.

Coping with Distortion

To eliminate rod binding problems, two approaches were adopted: First, replace the "U" shaped track blocks with relatively thin flat plates to provide more vertical channel clearance; second, design a new control rod much thinner in its vertical dimension to give both greater vertical clearance and a more flexible rod.

To open the channels at either K or C reactors, the old "U" shaped track blocks were replaced with particularly hard, flat, graphite plates keyed into the channel side walls and resting on the channel bottom. These plates are necessary not only to provide a hard, smooth surface for the control rod to ride on, but also to keep the channel sides from shifting into and blocking the channel. Cutting this keyway required developing a special tool that would cut slots at both channel sides and also follow channel curvature. The picture below shows this tool, on its side, during a mockup test. The cutter moves to one side during its trip into the channel and moves to the other for its trip out. A vacuuming attachment sucks out cutting debris to prevent the tool from jamming in the channel.

One other special channel cutting tool was required: This tool cuts away one channel side near the entrance to permit flopping the flat plates from their corner-to-corner insertion position to the channel bottom before they are pushed down the channel in their newly cut keyways.

It is not possible to straighten horizontal rod channels because of the process tube channels immediately above and below. Straightening these rod channels would have cut into too many of the process tube channels and

HORIZONTAL ROD CHANNEL
KEYWAY CUTTER

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removed them permanently from service. Widening the rod channels would serve no purpose as there is no front-to-rear curvature through the central zone where these channels pass through the moderator.

To provide more flexible control rods, one essentially basic philosophy was adopted: Thin the rods in their vertical dimension. But here the design effort for new C reactor rods departed from that for the K reactors. Because the K reactors operate at higher moderator temperatures than C reactor, and have the potential of going still higher, it was decided to abandon the conventional rod structural material (aluminum) in favor of a higher temperature alloy for any new generation rod designs. This requirement did not apply to new C reactor rods, and in the interest of standardization, new flat rods for C were designed to be of aluminum and compatible with those for B and D reactors. As will be noted later, B and D reactors also needed a new rod design for a reason not related to distortion, and one new rod was designed that satisfied the needs of all three reactors. The evolution of this new "universal" BCD rod is described in a later section.

For the K reactors, a new rod only 1-1/8 inches thick and 3-1/4 inches wide with Inconel 600 structural parts was designed. This rod provided not only considerably more vertical flexibility, but also considerably more channel clearance than the older 2-5/8 inch diameter round rod. The Inconel 600 structure gave an element that will not melt if it somehow sticks in the reactor and loses coolant, even at temperatures of over 2000 F. This rod also is described more fully in a later section.

Before these flat rods of either type can be installed, their channels must be equipped with flat plates. The "U" shaped track blocks will only pass a round rod; the flat plates will pass either round or flat rods.

These steps, opening up the rod channels and providing thinner, more flexible control rods, will restore the most distorted upper horizontal rod

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channels to a comfortably operable condition which should last for at least ten years. So far, relatively few channels have been modified because the renovation techniques and new control rods have just recently passed their design, development, and on-reactor tests. As the efforts devoted toward the vertical rod system ease, as they will in a few months, they will be turned toward horizontal channel work in earnest.

Coping with Balls

The other horizontal rod channel operation, that of sealing channels against balls, sleeves the channels with hollow graphite blocks very similar to those inserted into vertical rod channels. The holes in these sleeve blocks are round, however, instead of oblong, because no useful enlargement of the square channels is possible. To install these sleeves, the "U" shaped track blocks are removed identically as they would be for installing flat plates. Next the channel is surveyed and any obstructions are either pushed back into place or removed. The sleeves then are simply pushed into place to provide a continuous, gap-free channel. As with installing flat plates, these operations are not routine yet, but as more and more effort is diverted to the horizontal rod system, they will become so soon.

Horizontal Control Rods

The story behind the current production reactor control rods is a long one indeed; however, for this report, only the more recent developments will be described.

B and D reactors use identical horizontal control rods of an oblong cross section 3-5/8 inches wide, 1-3/8 inches thick, and about 35 feet long. They

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A Horizontal Control Rod Tip. Crushing Poison Rings Have Forced the Rod Housing to Bulge.

consist basically of three cooling channels surrounded by rings of a boron carbide aluminum mixture and enclosed in an aluminum housing (see sketch). The center tube carries water down the rod; the outer tubes carry the return flow. Two problems arose with these rods: First, the cooling channels to cool it effectively; second, the boron carbide aluminum poison rings swelled. These two effects combined to cause the rod to blister at the ends of the poison column inside the rod. As the column of poison rings tried to lengthen, it was restrained by fittings at the column ends attached to the cooling tubes. The rings then crushed and forced the already overheated housing to bulge. (as pictured above). This swelling did not become appreciable until something greater than four years irradiation had elapsed, then some rods would

B and D Rod

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begin to bind in their seals and generally operate very badly. Exactly how long a rod would operate properly depended upon how the rod was used. Those rods that extend only a small distance into the reactor during equilibrium operation would last indefinitely. But those rods that normally run deeply inserted would need replacing after about five years. As for temperatures, the housings exceeded 750 F over most blisters, a temperature far in excess of any reasonable expectation for any grade of aluminum.

The round rods for C and K reactors consist of long multiport aluminum extrusions that carry their poison as a pack of particulate boron carbide inside mid-wall channels. Water enters through a coaxial central tube and returns between this tube and the extrusion ID. C reactor rods are very similar to those for K except that they are 3 inches instead of 2-5/8 inches diameter and have four poison channels instead of two. Because cooling water is in direct contact with a continuous path to the housing outer surface, overheating has not been a problem for these rods. Also, the boron carbide pack has not shown nearly the tendency to swell as have poison rings in the BD design.

K Rod

In designing the new aluminum rod for B and D reactors to eliminate the swelling and overheating problems, the lessons learned from C and K rods were put to good use. The new rod employs boron carbide particles contained between two of its three aluminum extrusions; cooling water passes down through the center of the rod and returns just beneath the outer housing. In this way, all the aluminum structural parts are directly cooled. A rod of this design has been in service for more than

New B, C, and D Rod

one year and has lived up to expectations without exception. As long as cooling water never fails, this rod should last until the poison burns out, a period of something over ten years.

As mentioned earlier, C reactor required a more flexible rod for its more distorted upper channels. C reactor's operating conditions more closely approach those of B and D reactors than those of the K reactors, consequently going to the expense of equipping it with the new K rod design appeared unwarranted. What C reactor needed was simply a flat rod that would fit its channels. The BD rod, with a two-foot extension to its connection end, would do very nicely; this accommodation was provided. Thus, a new rod design now suffices to satisfy the needs of B, D, and C reactors.

Because of K reactors' higher graphite temperatures and the potential to go still higher, retaining aluminum as the rod structural material for any new designs appeared unwise. Consequently, Inconel 600, an iron, nickel, chromium alloy, was chosen for the cooling tubes and housing. The poison elements still are boron carbide aluminum compacts, but of a much heavier section than the old BD rod rings to restrain swelling. Even if they do swell, the Inconel housing's higher temperature tolerance will minimize the effect.

This new K rod design has survived a wide variety of performance tests, from negotiating repeatedly a mockup channel with distortion set for the worst condition expected ten years hence, to a year-and-a-half performance test in one K reactor's most distorted channel. The design appears good for at least ten years.

New K Rod

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Another program is now under way to develop the next generation K reactor control rod. All the present rod designs require cooling; providing this cooling accounts for most of these rods' design complexity. From an operating standpoint, rod cooling systems are difficult to service and have demanded more than their share of maintenance. The new generation design will not need cooling. This rod simply consists of 40-inch sections of dysprosium oxide impregnated nickel filler, clad with Inconel 600 and connected together by hinges.

Uncooled K Rod

Besides not requiring cooling, this new design also employs a new, more efficient poison. Dysprosium not only has a higher thermal neutron cross section than boron, it also can absorb more neutrons per atom. For a given amount of poison, dysprosium will burn out about one half as fast as boron. These rods will be smaller, simpler, and much less trouble to use than existing rods.

SUMMARY

Both vertical safety and horizontal control rod systems have required channel renovation to resolve problems of curvature, blockage, and poison ball infiltration resulting from moderator distortion. The techniques of renovation differed for each system, but the end results were the same: Channels were cleared and fitted with new internal components, and new rods were developed with sufficient flexibility to cope with present and future, unremovable channel curvature.

For those lower horizontal control rod channels plagued with ball infiltration, a channel clearing and sleeving technique was developed to prevent balls from getting into the rod channels and jamming the control rods.

The horizontal control rods of B and D reactors suffered from blistering and overheating, both of which were corrected in a new design that uses a poison material much less prone to swelling and a new cooling philosophy that insures direct cooling of all the rods' aluminum structural parts. This new BD rod was also designed to fit C reactor, thereby satisfying that reactor's need for a flat, more flexible rod. This one new rod replaces two older designs.

The K reactors required not only a flat, flexible rod to cope with their curved upper horizontal rod channels, but also a rod with higher temperature tolerance to accommodate their higher moderator temperatures. An Inconel 600 clad rod now serves these needs.

In summary, all of the known problems that have arisen over the past twelve to twenty-three years, problems with rods as well as rod channels, have either been corrected or corrective measures have been developed for them which are waiting to be executed as the need arises.

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