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MONTHLY REPORT

NOVEMBER, 1967

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December 15, 1967

MONTHLY REPORT

NOVEMBER 1967

DOUGLAS UNITED NUCLEAR, INC.

Richland, Washington

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

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SUMMARY

REACTOR PLANT OPERATIONS

B, C & K Reactors

Reactor input production (Pu) was 326.2 KMWD, 243.6 at the two K reactors and 82.6 at B and C. U-233 input production was 3,676 equivalent MWD. Overall time operated efficiency was 81.4 percent, averaging 93.2 percent at KE and KW and 69.7 percent at B and C. At KW, uninterrupted operation at 4,400 MW (the administrative limit) resulted in a K reactor production record for a 30-day month.

Production of nondefense plutonium containing up to 12 percent Pu-240 is continuing satisfactorily at the B, C, and K reactors. The four fuel failures experienced in November were in columns being irradiated beyond the 12 percent Pu-240 exposure on a test basis.

N Reactor

This reactor remained down all month due to continuation of the labor strike which began September 1. Exempt and non-unit, nonexempt personnel continued to maintain the plant in a safe and orderly condition.

Primary coolant leak and equipment repair continued. The reactor fuel loading adjustment that had been interrupted when the strike occurred was completed. Storage of the acid used for reactor piping decontamination in October caused some corrosion in the waste storage tank, due probably to a breakdown of the inhibitor in the acid. Corrective steps have been taken.

J. A. Jones and Combustion Engineering personnel continued to use the neutral gate for area access. Construction activities progressed satisfactorily on Cell 3 restoration work, Cell 3 steam generator retubing, and installation of the backup boiler facility.

FUEL & TARGET FABRICATION

B, C & K Reactors

Production of primary fuels totaled 100.9 tons of natural uranium elements and 116.7 tons of 94 Metal elements. Fuel core inventory at month end was 1,093 tons, a 4.4 months' supply. Finished fuel inventory was 1,835 tons, a 5.1 months' supply.

Hot-die-size fabrication of the first two charges of one-inch overbore rod-in-tube fuel elements is being completed. The inner rod elements are finished except for the attachment of locking supports.

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N Reactor

The strike at N Reactor continued to affect N fuel production, and input was limited to six production extrusions. However, about 45 development extrusions were made before the press was taken out of service for planned modification. Final fuel assembly work and output production returned to near normal levels; the output of 40 tons represented 106 percent of forecast.

The 23 Chemical Workers temporarily laid off in October remained in lay-off status throughout November.

TECHNICAL ACTIVITIESB, C & K Reactors

The effect of graphite lubricant on the corrosion behavior of hot-die-sized fuel elements continues under investigation. Severe pitting of elements not de-lubed after cladding occurred in static water tests at room temperature. However, similar elements exposed in the KE flow loop to 130 C flowing water for 19 days showed essentially no corrosion.

Examination of the second group of no-mixer fuel columns irradiated in new tubes as part of the low-dichromate test at B Reactor revealed extensive corrosion, but no significant corrosion difference between the two dichromate levels. The test is continuing. Final results, upon which a dichromate concentration decision can be made, are expected in March, 1968.

Irradiation programs are progressing at B and C Reactors to determine fuel exposure limits and fuel parameters which are most limiting to high Pu-240 production as a result of extended exposure.

Twenty-six of the 50 neptunium-aluminum targets in the Pu-238 test irradiation at KE Reactor have been discharged and are awaiting transfer to the reprocessing cell. Two targets have been sampled and analyses are in progress.

Work was started to define conditions relating to fuel element chattering in a process tube, since diversification schemes in some cases involve the irradiation of relatively low density fuel elements. Initial results indicate that chattering with the self-support geometry may not be a significant problem, regardless of flow rate or fuel density.

N Reactor

Design criteria for Mark IV fuel have been reviewed and tentative dimensions have been verified by the FLEX code. Life testing of this fuel has been completed, and irradiation of 50 columns is scheduled to start after the first full operating period following the end of the strike.

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The Advanced Technology Case study has considered four levels of increased reactor power, and systems capabilities have been evaluated in relation to these levels.

Investigation of the three Mark II drivers which failed in August has continued. Failures 40 and 41 now appear to have been identical, with water entry identified on the inner cladding at the contact made by the solid support located between the locking clip and the stop. Failure 39 has yielded no new clues and is still judged to have had water entry in the same vicinity as failures 40 and 41.

The fission products released from one of the high-exposure N reactor fuel elements experimentally heated to failure last month have been analyzed. The krypton release was several times higher than the amount identified in previous tests, whereas the xenon release was anomalously low. Additional fuel elements have been discharged from the reactor to permit continuation of this failure study. Tests at higher temperatures in new Bldg. 324-D facilities are scheduled for late in February.

ADVANCED OPERATIONAL PLANNING

Eleven Case B cases were submitted to Oak Ridge for combinatorial processing.

An updated reactor overbore study was completed in which the prime purpose was to determine the date when the cumulative unit cost of plutonium following overboring returned to its value without overbore. For no funding available prior to FY-1971, the date ranged from early FY-1975 to late FY-1976 depending on the number of reactors overbored.

Agreement has been reached on costing of the AECOP Pu-238 studies. The change in the over-all production complex cost will be the determining criterion in comparing alternate site irradiation of Np-237.

IRRADIATION SERVICES

A neutron flux measuring capsule was charged into a front-to-rear test facility at KW Reactor. This capsule has a neutron cross-section designed to simulate that of a United Nuclear uranium carbide test capsule, and will provide data for the design of flux attenuating sheaths to be used in the UNC experiments.

FEATURE REPORT

This month the appended special report describes improvements made in nuclear safety instrumentation at the B, C and K reactors in recent years. The report relates these improvements to the total complement of reactor instrumentation, and summarizes current programs for the development of additional systems.

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GENERAL

Efforts to resolve the issue(s) involved in the work stoppage at N Reactor continued throughout November. Numerous meetings were held with the Hanford Atomic Metal Trades Council during the first half of the month, but these resulted in no progress. On November 17 the Federal Mediation and Conciliation Service intervened on their own motion. Joint meetings were then held under FMCS auspices. No progress was made, and on November 28 the FMCS recessed negotiations subject to call by the Mediator.

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

for Charles D. Harrington
President

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REACTOR PLANT OPERATIONS - B, C, AND Ks

PRODUCTION

General

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

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

Deactivation work at the D Plant reactor and power buildings is essentially complete except for project work. The shipment of irradiated uranium from the D Reactor storage basin will be completed in about four months.

OPERATING EXPERIENCE

Reactor Loadings

Front face maps showing the loadings of B, C, and the K reactors are reproduced on Figures B-1 through B-4, which follow page B-7. The tonnages listed are approximate; actual fuel totals are given on page B-2.

The special program for producing nondefense plutonium is continuing at the B, C, and K reactors. About 63 tons of uranium at B Reactor and 21 tons at C Reactor above 12 percent Pu-240 were discharged because of evidence of core swelling. (See Production Fuel Performance, Page B-6.)

PRODUCTION REACTOR STATISTICS — NOVEMBER, 1967

REACTOR	B	C	KE	KW	TOTAL
INPUT PRODUCTION — PU — KMWD	41.1	41.1	111.6	132.0	326.2
— U-233 — EQUIV. MWD	842	673	969	1192	3676
POWER LEVEL (MW)— MAXIMUM	2100	2225	4400	4400	13125
— AVERAGE	1855	2110	4311	4400	12676
TIME OPERATED EFFICIENCY — %	74.5	65.0	86.3	100.0	81.4
OUTAGE TIME ALLOCATION — %					
CHARGE — DISCHARGE	9.3	9.2	5.6	0	6.0
FAILED FUEL REMOVAL	1.0	16.6	0	0	4.4
WATER LEAKS	4.2	3.6	0	0	2.0
TUBE REPLACEMENT	6.9	0.3	0	0	1.8
OTHER MAINTENANCE	2.8	1.0	2.2	0	1.5
STANDARDS CHECK	0.4	1.4	0.6	0	0.6
PRODUCTION TESTS	0.5	1.1	3.2	0	1.2
PROJECT WORK	0	0	0	0	0
OTHER	0.4	1.8	2.1	0	1.1
TOTAL	25.5	35.0	13.7	0	18.6
NUMBER OF OUTAGES	2	2	2	0	6
NUMBER OF STARTUP INTERRUPTIONS	0	0	1	0	1
WATER LEAKS — TUBE	1	1	0	0	2
— VAN STONE	1	1	0	0	2
NEW TUBES INSTALLED	24	2	0	0	26
FUEL CHARGE—(TONS)—NATURAL URANIUM	176.8	151.5	281.1*	274.5*	883.9*
—ENRICHED URANIUM	40.1	60.1	161.8	166.9	428.9
FUEL ELEMENT FAILURES	2	2	0	0	4
HELIUM CONSUMED — M CU. FT.	196.1	197.4	228.0	126.8	748.3
WATER TO REACTOR					
NORMAL OPERATING FLOW — GPM	95 300	103 000	215 200	215 500	629 000
PH	6.58	6.41	6.51	6.51	--
DICHROMATE—PPM	0.75	0.50	0.90	0.90	--

*Includes the 46.2 tons (at KE) and 44.8 tons (at KW) of special depleted uranium in the E-D loadings (PITA-048).

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Power Levels

Power levels at the K reactors were restricted by the 4400 MW administrative limit. The C Reactor power level was restricted by the bulk outlet water temperature limit of 95 C. Early in the month B Reactor power level was restricted by reactivity control limitations associated with the nondefense plutonium program. During the remainder of the month, following the discharge of some of the high exposure uranium, the 95 C bulk outlet water temperature limit became limiting at B.

Fuel Element Failures

Four failed fuel elements were removed from the reactors, as noted in the outage summary below. The stuck natural uranium bumper element failure at C Reactor on October 27 required 112.5 hours this month (212.5 total) for discharge and removal of the process tube. This tube broke into several pieces during removal, and some graphite damage was experienced. The channel was left empty pending detailed examination at a later date.

Reactor Outages

Six reactor outages occurred in November. In addition, the C Reactor continued down on a fuel failure outage initiated in October.

<u>Date</u> <u>Down</u>	<u>Reactor</u>	<u>Outage Hours</u>	<u>Remarks</u>
Oct. 27	C	134.1	Reactor continued down for removal of a stuck natural uranium bumper element and process tube from channel 2562. Also completed some Production Test, leak test and charge-discharge work.
Nov. 6	B	67.8	Removal of a failed natural uranium watermix element from tube 3486. Also replaced seven process tubes.
8	C	70.1	Unexplained Panellit trip on over-bore tube 3166. Completed leak testing and replaced two process tubes.
17	B	116.1	Removal of a failed 94 Metal bumper element from tube 2589 and a failed natural uranium mixer element from tube 3262. Also replaced 17 process tubes and completed charge-discharge.

<u>Date</u> <u>Down</u>	<u>Reactor</u>	<u>Outage Hours</u>	<u>Remarks</u>
Nov. 22	KE	83.6	Reactor scrambled by High Speed Scanner because of unexplained fluctuation of electrical power to the scanner. Scheduled charge-discharge of fuel and maintenance work were completed.
28	C	48.1	Removal of a failed 94 Metal bumper element from tube 2890. Also completed the charge-discharge of fuel.
30	KE	14.3	Reactor scrambled by loss of the 230 kv power supply to the KE plant (see below). The outage continued through month end.

Zeta Potential Control - K Reactors

The zeta potential control of water coagulation was started at 183-KE and KW on November 8. This instrumentation affords a refined control of alum feed to the raw water. By month end the alum feed at both water treatment plants was reduced to 12 ppm from a normal feed of 15 ppm.

EQUIPMENT EXPERIENCE

Process Pumps and Motors

The rotor for the No. 1 3500 hp motor at 190-C was installed and the return to ten-pump operation was in progress at month end. The rotor for this motor had been off-plant for shaft straightening and chrome plating of bearing journals.

The KE Reactor resumed operation on November 25 with process water being supplied by only five pump sets at 190-KE. This made the No. 2 pump set available for a Class A inspection which was started November 27.

Electrical Power Failure - KE Reactor

The cause of the electrical power failure in KE plant on November 30 is being investigated by ITT/FSS. Prior to the failure, an extended critical power condition (Grade "W") was in effect at KE to permit ITT/FSS to complete modifications to 230 kv breaker No. 396. No. 2 bank was lost when 230 kv tie-breaker No. 394 opened. KE plant emergency equipment performed as designed.

Effluent System - K Reactors

The repair of 107-K effluent retention tank bottoms, outlet valves, and Dresser couplings is continuing. This work is approximately 25 percent complete at KE and 60 percent complete at KW.

Construction work has been resumed on the drainage system in 100-K Area to minimize the possibility of further displacement of the retention basin outfall lines during future high river level periods. The drain culvert has been installed in the ditch west of the two effluent lines, and gravel backfill is being placed. The major drainage ditch, which is intended to provide general drainage of the area below the 107-KE retention basins, is being excavated.

Immersion Type RTDs - KE Reactor

During October, 23 immersion RTDs failed on the KE Reactor. Six were considered to be open-circuit type failures. The other seventeen failed because of low resistance to ground. There is a correlation between hot and warm tube RTD assembly failures (18) and water leaks.

After the high speed scanner scrambled KE Reactor because of an intermittent RTD on October 23, a review was made of intermittent RTD failures and it was decided to bypass the Technical High Limit feature of the scanner until a solution could be found to this RTD problem. This action was taken on November 14, with approval until May 1968.

REACTOR PERSONNEL TRAINING

Recertification examinations were given to the first of the four letter-shifts of Processing Supervisors and Specialists. The examination is based primarily on physical changes in the reactor plant and changes in operating procedures that have occurred since the initial certification program was completed three years ago.

PROCESS ASSISTANCE AND CONTROLProcess Physics

Reactor flux flattening efficiencies, although lower than for normal weapons-grade plutonium production due to the high Pu-240 plutonium production program, are gradually improving. B and C Reactors have discharged all fuel with exposures corresponding to greater than 12 percent Pu-240 plutonium. Exposures corresponding to approximately 14-15 percent Pu-240 plutonium were reached before irradiation had to be limited to the 12 percent Pu-240 mark because of fuel failures. However, both of these reactors are continuing to demonstrate successful 12 percent Pu-240 production with the present natural and enriched fuel models.

KE and KW Reactors terminated 30 and 35 days, respectively, of continuous operation this month. The E-D loads continue to behave as expected and are now extending minimum outage times to about 60 hours. Ten additional columns of depleted uranium (0.14 wt% U-235) have been charged into each K reactor. These elements are of hot-die-size design and are scheduled for irradiation to an exposure of 4000 MWD/T.

Operational physics data of interest are summarized below:

	Reactor			
	<u>B</u>	<u>C</u>	<u>KE</u>	<u>KW</u>
Effective Central Tubes (ECT)*	1432	1567	2167	2188
Flattening Efficiency** - November	0.76	0.83	0.71	0.71
- 12-Month	0.79	0.85	0.73	0.71
Average				
Equilibrium Scram Recovery Time - Minutes***	None	None	None	None

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

**Flattening Efficiency--ECT divided by the number of power generating tubes.

***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

Descriptive data for the four fuel failures which occurred during the period October 25 to November 25 are as follows:

<u>Fuel Type</u>	<u>Location</u>	<u>Failure Date</u>	<u>Column Exposure (MWD/T)</u>	<u>Failure Type</u>
C2N	2562-C	10/27/67	1975	Unclassified
O3W	3486-B	11/6/67	1479	Mechanical damage
O3E	2589-B	11/17/67	2137	Water entry
O3N	3262-B	11/17/67	1541	Groove corrosion

The failure in tube 2562-C was detected by a gradual increase in Panellit pressure of about 20 psi during October. The failed element, and two others from process tubes showing similar increases in Panellit pressure, are being examined in the Battelle-Northwest Radiometallurgy facility; preliminary examination indicates the probable cause of the failure to be swelling due to grain boundary tearing.

The three production fuel failures in B Reactor were being irradiated beyond the 12 percent Pu-240 goal (1400 MWD/T) on a test basis. Those in tubes 3486 and 3262 were water-mix elements in natural uranium fuel columns. Both resulted from corrosion penetration of the fuel cladding. There was evidence that one element had suffered mechanical damage during charging. The third failure occurred in a 94 Metal fuel column at 2137 MWD/T. The failure mechanism is believed to be water entry, although core swelling is being observed in high exposure fuel and may be related in this case. This is based on the fact that other fuel elements irradiated with the failure exhibited core swelling.

The following summary shows failure frequencies, as number/million elements discharged, for the 3-, 12-, and 24-month periods ending October 31:

	<u>3 Months</u>	<u>12 Months</u>	<u>24 Months</u>
Small Reactors - Natural U	50	28	19
- 94 Metal	0	11	13
K Reactors - Natural U	11	5	10
- 94 Metal	15	9	9
C Reactor - Nat. U (Overbore)	0	90	345

Reactor Effluent Activity Data

The following table shows the total kilocuries/month discharge for six significant radionuclides in the reactor effluent during October:

<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	1.4	0.13	0.07	0.010	4.1	1.2
C	2.5	0.23	0.26	0.063	9.3	2.1
KE	2.0	0.21	0.24	0.013	4.1	1.6
KW	<u>2.3</u>	<u>0.18</u>	<u>0.17</u>	<u>0.012</u>	<u>4.3</u>	<u>1.8</u>
Total	8.2	0.75	0.74	0.098	21.8	6.7

The progress of experimental work on effluent activity reduction is described in Section D of this report, under R&D Mission 10.

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Zone	Loading	
	Tons	Material
Central	103	Natural Uranium
Ring	5	94 Metal
	15	Natural Uranium
	9	94 Metal
Fringe	.59	Natural Uranium
Thoria Support	26	94 Metal
Thoria	5	Thoria

Figure B-1. Loading Pattern - B Reactor

Zone	Loading	
	Tons	Material
Central	113	Natural Uranium
	(Includes 6 tons in overbore block)	
	23	94 Metal
Ring	13	Natural Uranium
	19	94 Metal
Fringe	26	Natural Uranium
Thoria Support	18	94 Metal
Thoria	4	Thoria

Figure B-2. Loading Pattern - C Reactor

B-B

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Zone	Loading	
	Tons	Material
Central	46	Special Depleted Uranium (PITA-048)
	101	94 Metal (for depleted uranium support)
	6	94 Metal (for neptunium support)
	77	Natural Uranium
Ring	32	94 Metal
	26	Natural Uranium
Fringe	132	Natural Uranium
Thoria Support	23	94 Metal
Thoria	4	Thoria

Figure B-3. Loading Pattern - KE Reactor

B-C

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Zone	Loading	
	Tons	Material
Central	45	Special Depleted Uranium (PITA-048)
	112	94 Metal (for depleted uranium support)
	75	Natural Uranium
Ring	33	94 Metal
	25	Natural Uranium
Fringe	130	Natural Uranium
Thoria Support	22	94 Metal
Thoria	4	Thoria

Figure B-4. Loading Pattern - KW Reactor

B-D

REACTOR PLANT OPERATIONS - N

PRODUCTION

General

N Reactor was down throughout the month of November due to the labor strike which was initiated on September 1. The plant continues to be maintained in a safe and orderly shutdown condition by exempt and non-unit, non-exempt personnel.

By the end of the month, fifteen bargaining unit personnel had returned to work, five in maintenance and ten in operations.

Input production, time operated efficiency, and steam availability for the past six months are charted below:

Statistical Summary

Because of the work stoppage, N Reactor production and steam availability were zero in November. Fuel oil usage was 7,111 barrels.

The fuel charge at month's end was:

<u>Fuel</u>	<u>Tons</u>
94 Metal	4.1
125 Metal	2.3
210 Metal	215.8
Total	222.2

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These tonnages include some of the changes in in-reactor inventory made in late August but not fully accounted for in that month's report.

OPERATING EXPERIENCE

Charge-discharge work initiated at the beginning of the reactor outage (August 26) was resumed and completed. A total of 54 tubes were processed.

Following completion of the current loading adjustment, the reactor loading was as shown in the front face map which follows page BN-3.

EQUIPMENT EXPERIENCE

Plant Maintenance

Excellent gains continued to be made in the general area of plant maintenance by the strike coverage crews. As noted last month, these efforts have been devoted principally to primary coolant system leak and equipment repair.

Cell 3 Restoration

Restoration work is proceeding satisfactorily. Damaged sections of the water line and helium lines have been replaced. The specification for restoration of electrical equipment has been issued, and a cost estimate is being prepared by J. A. Jones. The minor amount of soot deposit in the pipe gallery has been removed.

Cell 3 Steam Generator Retubing

The program of retubing is proceeding satisfactorily in accordance with Combustion Engineering's schedule. All tube stubs are removed from unit 3A, and this work is under way in 3B. The removal of Incalloy-800 sleeves in 3B requires an additional step of drilling to permit stub removal.

PROCESS ASSISTANCE AND CONTROL

Plant Audits During Work Stoppage

Plant conditions during the third consecutive full-month outage, with emphasis on those relating to nuclear safety, were audited daily by technical personnel not assigned to the operating crews.

Tritium Concentration in the Primary Coolant

Tritium concentrations in the primary coolant have ranged from less than 10^{-5} to 10^{-4} $\mu\text{Ci/ml}$ during the outage, as compared to normal values (prior to the first target failure) of about 7×10^{-6} $\mu\text{Ci/ml}$ and to peak values of 5×10^{-2} $\mu\text{Ci/ml}$.

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Chemical Waste Storage Tank Corrosion

The inhibited 10 wt% H_3PO_4 solution used to decontaminate the N Reactor primary coolant piping on October 10, plus the system rinse water, totaled about 750,000 gallons. This dilute acid was added to the chemical waste storage tank which already contained about 100,000 gallons of residual solution from the Cell 3 decontamination on September 15. The pH of the combined solutions in the waste storage tank was about 1.5.

On October 24, the tank corrosion rate (as indicated by a corrosometer probe) started rising rapidly; i.e., from 1-2 to 3.5 mils per day on the 24th, 6 mils per day on the 25th, and 7.5 mils per day on the 26th. Between October 26 and October 31, 23,000 gallons of 50 percent caustic were added to the tank to raise the pH and reduce the corrosion rate. By October 31, the pH had been raised to 9.9 and the corrosion rate reduced to <0.1 mil per day. Between October 10 and October 31, the waste tank may have been pitted as deeply as 0.060 inch. Ultrasonic measurements of the tank wall showed a minimum uniform thickness of 0.490 inch in nominally 0.50-inch plate. The ultrasonic measuring device cannot detect pitting.

The recirculation pump piping in the 1310 Building showed considerably greater corrosion. Ultrasonic measurements showed wall thicknesses of 0.200 to 0.220 inch in nominally 0.365-inch wall, 10-inch diameter pipe. On October 25, a leak occurred in this pipe. The leak was patched, and no additional leaks have occurred.

The reason for the sudden acceleration of pitting corrosion rate on October 24 is not known, but it probably resulted from a breakdown of the inhibitor in the decontaminating acid.

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	41	42	43	44	45	46	47	48	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64	65	66	67	68	69	70	71	72	73	74	
34																																			34
33																																			33
32													76	76	76																			32	
31								B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B						31	
30								B	B	B		B		B	B	B		B		B		B		B	B	B	B	B						30	
29								B	B	B	B		C			C	C	C	C	C			C				B	B	B	B				29	
28								B	B	B						C	C										B	B	B	B				28	
27								B	4	B						C							C		C				B	4	B			27	
26								B	B	B	B										C	C			C	C	C		B	B	B	B		26	
25								B	B	B																	C			B	B	B	B		25
24								B	B	B				C		C					66							C		C	B	B	B	24	
23								B	B	B				C							66			66	66				C	C	B	B	B	23	
22								B	B	B				C							66	66						C			B	B	B	22	
21								B	B	B											66			66	66	C			C	C	B	B	B	21	
20								B	B	B	C											C						C	C		B	B	B	20	
19								B	B	B	94																C	C	C			B	B	B	19
18								B	B	B	C																				B	B	B	18	
17								B	B	B	C			D														C	C	B	B	B		17	
16								B	B	B	94			D							C		C		C	C	C			C	B	B	B	16	
15								B	B	B																					C	B	B	B	15
14								B	B	B																									14
13								B	B	B				C																	C	B	B	B	13
12								B	B	B																									12
11								B	B	B																									11
10								B	B	B																									10
09								B	B	B	B	C	C	C	C	C																		09	
08								B	B	B	B	C	C	C	C	C																		08	
07								B	B	B	B	C	C	C	C	C																		07	
06								B	4	B	B						C		C															06	
05								B	B	B	4	B	B			B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B			05	
04								B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B			04	
03																																			03
02																																			02
01																																			01
	41	42	43	44	45	46	47	48	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64	65	66	67	68	69	70	71	72	73	74	

Code	No. Tubes	Description	PT	No. Tubes	Description
	636	MK II Base 6% Program	04	5	Monitor Columns
B	212	MK II Spike 6% Program	66	10	Initial 2.1 Coproduct PT
C	123	MK II Base 12% Program	76	6	8" Meltdown Test
D	3	MK II Poison 6% Program	78	4	Increased Support Height
	974	Total	94	3	MK IV Fuel
	30	Total PT's		2	Blank Channel or Empty Tube
	1004	Grand Total			
				30	

Figure BN-1. Loading Pattern - N Reactor

BN-A

SECRET

FUEL AND TARGET FABRICATION - B, C AND K REACTORS

PRODUCTION

Acceptable Elements Produced

Type	Tons Input	Elements to Storage - Tons			Yields - Percent	
		Unrestricted Use	Upstream Use	Total	Current Month	FYTD
<u>AlSi-Bonded, Natural U</u>						
8" Regular	-	1.3	-	1.3	-	92.4
8" Bumper	-	88.2	-	88.2	-	95.2
8" Self-Support	55.9	4.1	7.9	12.0	87.7	90.9
<u>AlSi-Bonded, 94 Metal</u>						
6" Regular	-	-	-	-	-	100.0
6" Bumper	28.9	-	-	-	-	95.7
6" Self-Support	130.3	106.5	9.8	116.3	93.3	92.2
<u>AlSi-Bonded, Depleted U</u>						
8" Self-Support	-	0.8	-	0.8	-	-
<u>Hot-Die-Sized, Nat. U</u>	1.2	-0.2	-	-0.2	-	-
<u>Hot-Die-Sized, Depl. U</u>						
8" Self-Support	-	0.9	-	0.9	-	-
Thoria	-	-	-	-	-	-

Procurement and Inventories

Item	Tons Received	Tons Placed in Process	End-of-Month Stock	
			Tons	Mo's. Supply
Natural U Cores	80.5	55.9	767.0	4.9
94 Metal Cores	225.3	160.4	326.0	3.1
Thoria Powder	-	-	14.0	-

AlSi-Bonded Fuel

Production totaled 95 percent of the November forecast. The below-forecast production resulted mainly from the removal of all K5NS fuel from finished inventory for reinspection for can-base braze quality. The net effect of this reinspection has not yet been determined. For the fiscal year to date, production is 101 percent of forecast. Of the 219.3 tons produced in November,

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218 tons or 99.4 percent were fuel elements with bumper or self-support rails attached.

Hot-Die-Sized Fuel

The plating station input totaled 826 cores, 17 of which were for development purposes.

OPERATING EXPERIENCE

AlSi-Bonded Fuel

Time operated experience was 98.3 percent. Of the downtime, 0.7 percent was charged to operations and 1.0 percent to equipment. Canning line operations totaled 60 line-shifts, at the rate of three line-shifts per day.

The total manufacturing yield was 90.5 percent. Although 0.5 percent above forecast, the yield was lower than in recent months, due principally to the K5NS reinspection noted above. Of the total rejects in each category, the percent reclaimed or released for upstream use was: marred surface - 62 percent, closure weld - 82 percent, rail weld - 100 percent, bond - 94 percent, and AlSi slopover - 100 percent.

Uranium utilization, exclusive of special development but including defective material, was 99.3 percent.

Hot-Die-Sized Fuel

Fabrication of the outer tubular elements for the rod-in-tube overbore fuel has been completed. Autoclaving, final inspection and the measurements required before assembly with the inner rods is under way. Overall yield of the CMLE outer tubes was 84.2 percent. Also completed is the rebonding of the inner rod ends to increase the bond strength.

Spacers for Overbore Use

Anodizing and the attachment of self-supports to the aluminum spacers for use with overbore fuel is in progress. An ultrasonic welding process for attaching the spacers was developed. It was found that the 6061-T6 aluminum alloy spacers had to be annealed prior to welding to produce good ultrasonic support welds.

FUEL AND TARGET FABRICATION - N REACTOR

PRODUCTION

General

The strike at N Reactor continued to curtail input production, which was limited to the extrusions required for the outer preshape billet development. No extrusions had been forecast for November.

The 23 Chemical Workers temporarily laid off as a result of the strike were in lay-off status the entire month. The majority of the Chemical Workers who were retained for production were assigned to finishing and assembly operations to reduce in-process inventories, and permitted output production to return to normal levels.

Statistical Summary

Input (Billets Extruded)

Mark I, Outer, 9 $\frac{1}{4}$ Metal	6 (1.2 tons)

Output (Finished Assemblies)

Mark I, Outers, 125 Metal	1271
Inners, 9 $\frac{1}{4}$ Metal	
Mark IV, Outers & Inners, 125 Metal	210
Mark IV, Outers & Inners, 9 $\frac{1}{4}$ Metal	<u>241</u>
Total Assemblies	1722
Tons - Total	40.3
- % of Forecast	106.1
Uranium Utilization - %	77.7

OPERATING EXPERIENCE

Mark IV Outer Supports

Outer support fabrication problems for the Mark IV assembly continued during November. A major problem is the occurrence of longitudinal cracks on the weld projections. Working closely with the vendor, new fabrication techniques are being developed to eliminate this problem.

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Manpower Requirements - Final Assembly

A study of final assembly manpower requirements and improved material handling methods has been completed. As a result of this study, a roller device has been added to the top of the fuel carts which permits inspection, sizing of supports, locking and shoeing without removing the fuels from the carts.

Extrusion

Only six production extrusions and forty-four development extrusions were made during the month. The latter are described on page CN-3. Following these operations, the press was taken out of service for continued work on the press improvement program (see below).

EQUIPMENT EXPERIENCE

Extrusion Press

Most of the approved standards for piping, tubing, and high pressure hose materials are completed. Major conversion work on the hydraulic system will start on December 4 and continue through December 15. The remaining work can be accomplished in 2- or 3-day periods to accommodate production and development extrusion schedules.

A lock-out device to prevent accidental closing of the die locks during stem-to-container measurement checks is scheduled for installation during the above outage. At the same time, all pressure gauges located on the pump deck will be relocated to the outer south wall of the pump deck enclosure. All gauges have been calibrated, equipped with snubbers and readied for installation.

A piping completion report will be issued at the time the press modification is completed.

PROCESS ASSISTANCE AND CONTROL

Mark IV End Closures

A weld penetration audit revealed that the inner clad of some Mark IV fuels, thin clad case, were excessively thin in the area at the base of the end cap. Subsequent destructive examination of first-run fuels revealed that the clad in the braze zone will be below the minimum specification limit of 0.009 inch approximately 75 percent of the time.

An examination of two recut fuels indicates that some rework clads may be only 0.0025 inch thick in this area. This study will continue and appropriate action taken to reject or waiver the fuels involved.

Control Charts

A study has been completed proposing the use of control charts at pre-mill inspection for die wear, heat treat for inner fuel warp, locking clip weld for clip position, and outer fuel support weld for as-welded support height. Control of the process at all of these stages has presented problems and control charts would result in faster detection of problems. Examples of each control chart have been prepared based on existing data, and trial control limits have been calculated.

Gauge Audit

An audit of production dimensional gauging has been initiated. -This will be the first complete such audit that has been made in several years and will take at least two months to finish. To date, all length gauges have been completely checked and standardized as to types of contacts, dial indicators, and height of bearing blocks to measuring contacts. The audit of the warp gauges has been completed.

Quality Audit Procedures

Fourteen new audit procedures have been written, and all existing audit procedures revised. Conformance to each standard will be included in all routine audit reports.

DEVELOPMENT EXTRUSIONS

More than forty special engineering development extrusions were made during the month, as follows:

- Twelve outer and ten inner billets were coextruded to provide flow loop test fuels and material to establish brazing and welding parameters for the modified Mark IV program.
- Two Zircaloy extrusions for the Mark IV program were made as part of a program to develop the process and technique for making Zircaloy components on site. Both extrusions were successful and work on this program will be continued.
- Nine extrusions of aluminum rods and tubes were made for use in further studies of aluminum for target cladding.
- One extrusion to produce a ribbed Zircaloy process tube was made in the study of dimensional control in ribbed tubes. Preliminary examination of a 12-foot length indicates that an extruded ribbed tube of this size is feasible. Some deformation of one rib was noticeable, but ribs appear straight and no twist is visually evident. Accurate dimensional measurements of the tube and ribs are in progress.

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- Eight extrusions were made for development programs relating to B-C-K fuels. Four of these were coextrusions to produce I & E fuels, and four were Zircaloy tubes for the hot-die-sizing program.

TECHNICAL ACTIVITIES - B, C & K REACTORSRESEARCH AND DEVELOPMENT

The progress on research and development work conducted by Douglas United Nuclear is reported by Mission number and title, here for the B, C, and K reactors, and for N Reactor beginning on page DN-1. Unless otherwise noted, the Missions are as defined in the July 1967 issue of this report. No coverage is accorded the two Hanford R&D Missions (6 and 9) in which DUN does not participate.

Mission 1 - Basic Production1-A. Fuel DevelopmentNon-Uniform Cladding Corrosion - Hot-Die-Size Fuel

The effect of graphite lubricant on the corrosion behavior of HDS fuels was investigated by charging into the KE Reactor flow loop both cleaned fuel elements and as-sized fuel elements from which the graphite lubricant had not been removed. Tests in static water at room temperature had shown severe pitting of fuel elements not de-lubed; however, in the flow loop, the fuel elements whose surfaces had graphite lubricant showed no (or extremely minor) evidence of corrosion after 19 days in 130 C water flowing at 33 gpm.

Prior flow tests of nickel plated fuel elements from which the graphite lubricant had not been removed resulted in severe pitting of the fuel cladding. To duplicate these results, additional nickel plated fuels were tested, including test elements whose nickel plating had been defected. Both nickel plated fuel types were severely corroded; however, the defects had no discernable effect on the corrosion. Unplated control fuel elements showed little or no evidence of corrosion attack.

To evaluate the effect of lubricant from the die sizing operation entrapped in the surface of hot-die-sized cladding on non-uniform corrosion, production fuel elements were charged into KW Reactor on October 27 (under PTA-088). The test consists of 15 columns of natural uranium fuel and includes standard AlSi canned K5N fuel, standard hot-die-size K5AN fuel, and K5AN fuels which were sized with five mils of excess cladding which was then removed by chemical etching.

K Reactor Fuel Support Corrosion

A device is being designed that should provide quantitative data on fuel support corrosion by slicing the arch section off of the supports, and then determining the weight loss on the arch section. The major problem to overcome is the close indexing required to remove only the proper portion of each support.

PTA-057 - Low Dichromate Test - B Reactor

Examination of the second group of no-mixer fuel columns irradiated in new process tubes has been completed. The irradiation of these eight columns had been terminated by a corrosion failure at an average exposure of 1250 MWD/T. More than 75 percent of the elements in the downstream 16 positions exhibited localized corrosion, and 50 percent of these elements showed penetration to the AlSi braze layer. Although the corrosion was extensive, there were no significant differences in localized corrosion frequency and severity or in uniform corrosion penetration between the two dichromate levels. The test is continuing with 20 mixer columns in medium-age tubes scheduled to 1400 MWD/T. Final results of this test program should be available by March 1968; these should enable a decision concerning dichromate concentration to be made.

1-B. High Pu-240 Program

Irradiation programs are being performed at B and C Reactors to establish fuel exposure limits, and to determine which fuel parameters are most limiting to high Pu-240 production as a result of extended exposure of production fuel elements.

The program was started at B Reactor on September 18, with 514 tubes of natural uranium fuel and 97 tubes of 94 Metal fuel. The test, originally written for a goal exposure of 2000 MWD/T, was revised in October to allow three ruptures in natural fuel and one in enriched fuel. The change was made to assure adequate fuel failure statistics, to establish fuel performance limits, and to provide a firm upper limit for the test operation. Currently, the natural fuel has achieved an average exposure of 1450 MWD/T and the enriched fuel 2050 MWD/T. Four natural uranium fuel failures have occurred since the test began at B Reactor, as shown below:

Date	Tube	Exposure (MWD/T)	Failure Type	Comments
Sept. 30	2682	1237	SG	Dichromate test monitor column
Oct. 18	1066	1370	EM	HDS Fuel (PTA-011)
Oct. 23	2265	1395	CP	Core split
Nov.-6	3486	1479	ED	Mixer mechanical damage

Of these, only the core split failure can be considered as related to fuel quality. Thus, the authorization in effect calls for up to two more natural fuel failures and one enriched fuel failure. The test will be terminated regardless of fuel failure experience whenever enough information is at hand to establish a real fuel performance limit. Such a limit may be indicated by recent C Reactor experience which is summarized below.

The program was initiated at C Reactor on October 1, with 424 tubes of natural uranium fuel and 100 tubes of 94 Metal fuel. One failure has occurred at C Reactor since the start of the test. This failure was in tube 2562 on October 27; the natural fuel column was at 1957 MWD/T. The failure was characterized primarily by a 10-15 psi Panellit pressure increase, with little gamma activity being experienced. The ruptured element was so badly

stuck and mangled by the discharge process that its failure mechanism could not be established. Twelve other tubes of high exposure (about 1900 MWD/T) fuel which had shown similar Panellit pressure increases were discharged for visual examination.

Four of the C Reactor columns (including the rupture column) have been examined to date. Approximately eight elements with >20 mils swelling were found. Two of these, when sectioned in Radiometallurgy, showed the problem was one of core swelling or growth as opposed to water entry and uranium corrosion. All indications point toward a core exposure limitation with the present core model, at least with the current batch of C fuel. All except one of the above 13 columns (including the rupture column) were from cores canned February 21-27, and eight of the 13 were from the same lot; this suggests that a poor batch of cores may have caused the swelling problem.

On November 8, C Reactor was scrambled by a spurious Panellit trip on a 0.5-inch overbore tube. At the time, there were about 180 natural fuel columns with exposures in the 1800 to 2000 MWD/T range, and about 50 of these had shown Panellit pressure increases of 5-15 psi during the previous 30-50 days. These fuel columns were discharged and the high exposure test of natural fuel at C was discontinued for the near term. However, about 50 natural fuel columns were left in; these will reach 2000 MWD/T in December, and will provide further verification of the fuel swelling/growth behavior. Also, data will be obtained at B Reactor and with enriched fuel at C (current average exposure of 1600 MWD/T) to provide additional information as to whether the swelling problem is general with present fuels at high exposure.

The C Reactor test also included an evaluation of the benefits to a high exposure program of 6.2 pH (PTA-084, Sup. A). The incentives for continuing this phase of the program are under study, with the conclusions depending significantly on the results of irradiations currently in progress.

1-C. Zircaloy Hydriding

Analyses of samples from the 0.5-inch overbore Zircaloy process tube removed from C Reactor in September are still not complete. Base metal hydrogen content from the rear flange to 180 inches upstream has been determined, however; the maximum was 139 ppm at the 20-inch location. It was interesting to note that there was the usual hydriding throughout the spacer pattern; however, in this case, the spacer pattern extends to about 175 inches due to the short fuel charges used in these tubes. This is a further indication that the reactor flux of itself does not prevent hydriding in the active part of the tube.

Tube 3671-KW, which was removed after the failure of a 210 Metal fuel element, is still being analyzed. A large number of samples were taken from the area occupied by the rupture, but no sign of hydrogen was found. The hydride pattern in the downstream end of the tube was normal in shape but unusually high in concentration. Maximum total found was 3684 ppm at six inches upstream from the rear Van Stone flange; the maximum base content was 170 ppm. The tube had been operated a total of 47 months, the last eight months with anodized spacers. A base metal content of about 100 ppm would have been

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expected. The analyses are being repeated to make sure there has been no error. The significance of these data is not known, and will not be put into perspective until tubes in the next group are analyzed (in January or February).

1-D. Computational Techniques

CLUMSY

Difficulties with safety circuit time constants were encountered with the 1108 version of CLUMSY during the 2.1 E-N fuel core loading safety analysis. The problems resulted in returning to the use of the 7090 version of the code, which requires approximately five times the running time and involves considerably greater input.

HAMMER

The main effort with the HAMMER code has been to condense all the options into one working program with expanded libraries. This is a problem of overlaying certain sections of the program when they are not in use without destroying the input for the next section of the program. Considerable progress has been made, and an early completion date is envisioned. The expansion of the libraries which was made last month was found to be incorrect. The problem was traced to the program ETOM which processes the ENDF/B Library Tape.

HAMMER-EXTERMINATOR-2 - Comparison of PCTR Tests

The specifications and results of six PCTR tests with mixed lattices of fuel and targets were obtained from Battelle-Northwest. The test dimensions and specifications were inputted into HAMMER and the resulting smeared cross sections used as input for EXTERMINATOR. The different cases being evaluated are:

- o 125 Metal fuel - thoria target
- o PuAl fuel - thoria target
- o Undersized 125 Metal fuel - thoria target
- o 94 Metal fuel - bismuth target
- o 210 Metal fuel - lithium target

The preliminary computer results agree very closely with the experimental results. Results of k_{∞} measured and calculated for the 125 Metal fuel - thoria target and the 210 Metal fuel - lithium target mixed lattices are shown below:

<u>Lattice</u>	<u>Lattice Status</u>	<u>EXTERMINATOR Geometry</u>	<u>Calculated k^∞</u>	<u>Measured k^∞</u>	<u>Δk^∞</u>
1.25 E thoria	Wet	X-Y	1.0751	1.0626 + .0092	.0125
"	Dry	X-Y	1.0879	1.0702 + .0094	.0177
"	Wet	R-Z	1.0750	1.0626 + .0092	.0124
"	Dry	R-Z	1.0876	1.0702 + .0094	.0174
"	Wet	R-O	1.0751	1.0626 + .0092	.0125
"	Dry	R-O	1.0878	1.0702 + .0094	.0176
2.1 E lithium	Wet	R-Z	.9811	1.0008	.0188
"	Dry	R-Z	.9715	.9969	.0254

The R-Z geometry cases require the minimum running time and provide fully consistent results as indicated by other geometry cases.

Mission 2 - Coproduct

2-A. 2.1 E-N Test Irradiation

Plans are proceeding for the isotopic analysis of the fuel and lithium targets from the 2.1 E-N test demonstration, discharged from KW Reactor on September 15.

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2-B. Full Core 2.1 E-N Demonstration Load

The final draft of the technical evaluation document has been completed. The first draft of the safety analysis review is expected to be issued early in December.

2-C. Lithium Spline Development

The vendor has completed fabrication of the 20 required splines. Delivery is expected in December.

Mission 3 - Transplutonium Technology

Correlation of the CUPID code with the measured isotopic yields from the plutonium irradiation are nearing completion. This analysis will provide the basis for predicting the isotopic buildup in the planned irradiation of 10 kg of plutonium.

Preliminary arrangements are under way to initiate fabrication of test quantities of Pu-Al target elements. Studies are also under way to establish criteria for this fabrication work.

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Mission 4 - Pu-238

4-A. Test Irradiation

Twenty-six of the 50 neptunium-aluminum targets have been discharged from KE Reactor and are now awaiting sampling, mandrel removal, and shipment to the reprocessing cell. Processing of these targets should start in mid-December.

Two targets have been sampled, and analyses are in progress. Reduction of the initial analytical data has been completed; results indicate the isotopic content is well within prediction uncertainty for one element. Results from the other element are not consistent.

The following table describes the scheduling and production of each of the target columns irradiated:

Tube No.	Planned No. of Samples to be Taken	Np-237 Burned (Gm)	Pu-238 Produced (Gm)	Percent Pu-238	Discharge Date	Target Data Available	Dissolver Sample Available
2857	3	54.45	44.30	87.8	9-27-67	11-13-67	12-1-67
3157	2	70.50	55.20	86.3	10-10-67	11-17-67	12-10-67
2955	2	76.91	58.75	84.6	10-13-67	11-17-67	12-15-67
2759	3	121.54	87.83	81.4	10 more FPD*	?	?
2960	3 or 4	131.99	91.16	79.0	25 more FPD*	?	?
		455.39	337.24				

* Full power days

The compacted graphite targets will be discharged during the next KE Reactor outage. The impregnated graphite neptunium targets will remain in the test block until the Production Test is completed (sometime in mid-December). It is planned that four elements and the dissolver solution from each of the columns will be analyzed particularly for Pu-236 content.

4-B. Np-237 and Pu-238 Cross Sections

The effective neutron cross sections of Np-237 and Pu-238 are known to be dependent on the neutron energy spectrum. The cross section of Np-237 increases both with higher epithermal-to-thermal neutron flux ratios, and with higher thermal neutron temperature. On the other hand, the Pu-238 cross section decreases with the same changes in neutron spectrum. A lower ratio of Pu-238 cross section to Np-237 cross section results in decreasing the burnout of Pu-238 relative to production and, hence, increasing the Pu-238 yield as a function of Np-237 burnout.

A study has been initiated utilizing the HAMMER code to calculate these cross sections as they are influenced by various reactor conditions. Initial results indicate the calculated effective cross sections in a K Reactor agree quite well with those predicted by the Westcott formulation which assumes a uniform slowing-down density and Maxwellian thermal neutron energy distribution unperturbed by neutron absorptions in the lattice. Expanding the work

to a wider variety of reactor lattices has been hampered by the delay in placing neutron cross sections and moderator scattering kernels for a variety of temperatures on the HAMMER cross section library.

4-C. Irradiation of Uranium Containing a High U-236 Concentration

Four tubes from PTA-069, which authorized the irradiation of 72 Metal containing 400 ppm U-236, have been discharged from B Reactor: two at approximately 300 MWD/T, and two at 500-600 MWD/T. Eight samples are being prepared for analysis. Preliminary isotopic data for assessing the U-236 neutron cross section should be available by mid-December.

94 Metal containing 1000 ppm U-236 has been received, and a Production Test for utilizing this enriched uranium in a K Reactor is being written.

4-D. Alternate Target Element Matrices

Tests have been initiated to evaluate the use of a magnesium oxide matrix for neptunium irradiations as an alternate to a Np-Al alloy target core. The study thus far has involved evaluation of annular NpO₂ powder blend. Pellet strength, density, reactivity, and dissolution characteristics are included in the properties being evaluated.

Mission 5 - Other Isotopes

5-A. U-233

One column of test thoria target elements has been fabricated and awaits charging into C Reactor under PTA-076. These elements were fabricated from recycled thoria, the processing of which was performed on site. The test column contains elements with compacted light thoria powder having a nominal density of 5.7 gm/cc as well as elements containing sintered thoria pellets, the nominal density of which is 8.1 gm/cc.

5-B. Co-60

The vendor has completed fabrication of the Haynes 25 alloy process tubes for irradiation to determine their feasibility for use in the cobalt-60 program. Further development work remains to be done to perfect welding and flanging operations. It is expected that a tube will be ready for reactor installation by the end of the year.

Mission 7 - Target Space Enhancement

7-A. High Power Density Fuel

Hydraulic Behavior of Low Density Fuel

Work has been initiated on a program to define the conditions which result in fuel element chattering in a process tube, since diversification schemes involve the irradiation of relatively low density fuel elements in some cases.

Initial efforts have concentrated on evaluating eddy-current based instrumentation for identifying chattering in laboratory investigations. It appears that such instrumentation will be quite useful. It was established that such equipment would respond to fuel element movement in the process tube. However, coolant flow rates up to 90 gpm failed to induce chatter of solid aluminum (SA) spacers. Also, bored SAs, SAs without supports, and I&E elements were tried; none yielded significant motion. An element weighted to the density of water was allowed to float in the tube, and this motion was detectable both by ear and with the instrument. These results indicate that chattering in the self-support geometry may not be a significant problem regardless of flow rate or fuel density.

Zircaloy Clad Fuels

Delivery of the billets for fabricating Zircaloy clad fuel elements for K Reactor tests will be delayed until December. This pushes the completion date for these fuel elements into February. The preliminary work required to prepare for the program will be conducted by using natural uranium billets which are available.

Preliminary tests were initiated in an effort to form end closures on Zircaloy clad fuels by the HDS end bonding process. Two brazing materials were evaluated, tin and a nominal 10 w/o silicon-aluminum alloy, as intermediate metallic bond layers between the uranium core and the Zircaloy end caps. Several temperatures were employed, with the highest causing both braze materials to bond the die cup to the workpiece (at braze line interface only). Over the range of parameters tested, none produced a bond between either braze material and the Zircaloy. Some evidence of bonding to the uranium was present only with the tin. Samples of fuel containing each of the brazing materials were welded satisfactorily. Evaluation of this concept is continuing.

Alternate Aluminum Cladding

In studies directed toward development of alternate aluminum cladding alloys, two billets of X8001 aluminum alloy were cast to prove the operation of the direct chill casting machine. This machine is one in which molten aluminum is poured into a short chill mold from which the aluminum is withdrawn into a water spray where the billet solidifies. The result is a continuously cast aluminum ingot with a very fine dispersion of the second phase material.

Previously, considerable difficulties were encountered in operation of the casting machine. After modifications to the mold and hydraulic drive mechanism, satisfactory operation was obtained. Metallographic examination of the cast ingots revealed many oxide inclusions. A temporary flux was made of equal parts by weight of NaCl and KCl pending receipt of a commercial aluminum cleaning flux. Fluxing the molten aluminum yielded clean castings, but some heats showed a visible evolution of hydrogen gas. A system for degassing with argon has been installed at the melt furnace and is ready for use on the alloys to be cast.

One of the X8001 cast billets was extruded into tubing. The product appears to have a satisfactory microstructure with no trace of the as-cast eutectic network. Casting of the alloy billets for processing into tubing and sheet bar is being planned.

Uranium Core Alloy Evaluation

Production Test PTA-098 is planned for charging into C Reactor during December or January to evaluate the effects of uranium composition and heat treatment on core stability at high exposures and normal tube powers. The test will include the following compositions and heat treatments:

<u>Composition (ppm)</u>			<u>Heat Treatments</u>		
<u>Fe</u>	<u>Si</u>	<u>Al</u>	<u>Core Blank Form</u>	<u>Heat Method</u>	<u>Quench Medium</u>
150	120	--	Solid Solid	Salt Induction	Oil "
350	--	800	Solid Hollow	Induction "	Oil "
350	350	800	Solid Hollow	Induction "	Oil "
--	350	--	Solid Hollow	Induction "	Oil "

A total of 160 columns, 20 of each treatment, will be irradiated to varying exposures ranging from 1000 to over 3000 MWD/T.

7-B. Reactor Modernization

All of the 22-foot sections (half-length) of welded smooth-bore Zircaloy tubes have been received from the fabricator for the 14-tube Reactor Modernization Test Facility at C Reactor. The sections are currently being joined to produce full-length process tubes by butt welding (using DUN personnel and equipment). Weld procedures and equipment specially prepared for fusion welding the sections together have been very successful as determined by inspection, including 100 percent radiography, as required by the tube specification. Van Stone flanging of the upstream ends has been accomplished readily. The Zircaloy tube fabricator is now forming spline tube sections from blanks with guide rail strips already welded in place.

Nine of the best aluminum tubes in the first lot to be formed have been received. Six of the tubes with rib twist up to 14 degrees are being reserved for test facility installation, and the three other tubes will be used for necessary tests and development efforts. A new lot of aluminum tubes have been formed with initial inspections disclosing at least 10 tubes with acceptable rib twist; however, the tensile strength of the aluminum for this lot appears to be lower, approximately 14-15,000 psi vs. about 18,000 psi for the first lot and 19,000 psi for the original specification. A 10-foot section of tube from the second lot has been received for evaluation of the

effects of reduced strength on Van Stone load limits and fuel sliding resistance in comparison with results of the same tests on a tube section from the earlier lot.

The extruded aluminum nozzles for the test facility are presently being machined. All of the flanged end surface has been machined including the gasket seat. Inspection of the parts at this stage has disclosed some minor problems with eccentricity of the ribs along the barrel in relation to the axis of the barrel as established by the machined base. It is planned to use a special plug gage to check the tapered bore at the cap end of the nozzle. The critical port counter bore and threaded area will be checked by the use of the molding material and a comparator. The special profile counter boring tool for the port will also be checked on a comparator to assure the tool is properly ground prior to production.

A Production Test is now circulating for approvals to authorize fueling and operation of the C Reactor overbore block.

7-C. Highly Enriched Fuel

Criticality Studies

Correlation of experimental and calculational results for the technical bases for handling highly enriched fuel is proceeding, but was temporarily delayed due to problems encountered in running the EXTERMINATOR-2 computer code. However, preliminary correlation of experimental and calculational results indicate that the HAMMER-EXTERMINATOR system will provide a workable base for the highly enriched fuel in-reactor safety studies.

Using Highly Enriched Fuel to Supplement Slightly Enriched Uranium

Production has been calculated using MOFDA for various ratios of 94 Metal and highly enriched fuel with LiAl targets in a K Reactor. Calculations for specific cases are shown in the following table. The highly enriched elements contain 35 grams/ft of 93 percent U-235 uranium. The produced plutonium contains six percent Pu-240.

<u>Fuel Columns</u>		<u>Production (Kg/yr)</u>	
<u>Fraction</u>	<u>Fraction</u>	<u>Plutonium</u>	<u>Tritium</u>
<u>Highly Enriched Fuel</u>	<u>94 Metal</u>		
0.0	1.0	1035	3.13
0.1	0.9	930	4.14
0.2	0.8	812	5.25
0.25	0.75	750.	5.86

Work is continuing on the safety analysis of this fuel system. The study is being extended to evaluate highly enriched fuel supplementing 72 Metal.

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Mission 8 - Nuclear Safety

8-A. Heat Decay and Radiation Dose Rates from Freshly Discharged Fuel

Calculations indicate that fog spray application within 10 seconds following discharge is necessary to prevent fuel-clad melting of a central high power element should it fail to enter the discharge basin. If there was no application of the fog spray system, fuel-clad melting would be expected to occur in about three and a half minutes following discharge of the fuel. If fog spray application is delayed for one minute, fuel-clad melting would occur between four and five minutes after fuel discharge.

A second series of fog spray cooling experiments in the Battelle-Northwest heat transfer laboratory indicates that the cooling curves are altered significantly when a corroded surface representative of reactor fuel elements is used. The heat transfer rates at low surface temperatures are reduced, but these increase when the fuel temperatures exceed about 300 C. These tests are continuing but indicate that the calculations discussed above are conservative.

8-B. Riser Failure Analyses

Work is continuing on programming an iterative solution to the riser failure analyses. The basic nodal equations for the reactor charge have been prepared, programmed, and initial cases are being run. Two general cases are being considered: (1) the loss of the A and B risers with a continued high pressure pumping flow, and (2) the loss of the A and B risers with the simultaneous loss of BPA electrical power. The first case provides more flow, but the possibility of additional piping failures due to very severe pump cavitation is in question. The second case provides a smaller flow, but the initial severe pump cavitation diminishes and should disappear after 30 seconds.

8-C. Charcoal Filter Tests

At the confinement filter test facility in Bldg. 189-D, a new current type charcoal filter was tested for its efficiency to adsorb elemental stable iodine. This was done by generating stable elemental iodine vapor and injecting it into the air stream well upstream of the filter, and then sampling the air stream immediately upstream and downstream of the filter and passing the sampled air through small charcoal-loaded cartridges. The cartridges were then placed in a reactor test facility for activation by neutrons, removed and analyzed for radioiodine content.

The filter tested by this means showed 94 percent efficiency for retention of iodine. The charcoal filter canister was disassembled for inspection. It was found that the charcoal in it had settled, leaving voids in the charcoal media. Additional charcoal was placed in the canister to fill the voids. The subsequent efficiency test was inconclusive because of lack of care in performing the test. The test will be repeated using another new charcoal filter.

Mission 10 - Columbia River Studies

Detailed observations of the 100-K Area water plant operations were made. Emphasis was placed on determining the causes of the variation in performance between filters with respect to head loss and effluent turbidity. In both the KE and KW treatment plants, information was gathered on variations in operation of all unit processes, i.e., chemical addition to filtration. All phases of the backwashing cycle were timed and bed expansion measured for each filter. Attempts to obtain core samples from the filters during normal operation were not successful. Alternate methods will be tried.

A test to continue the evaluation of ground disposal of reactor effluent was initiated on October 30 in B Area. By month end the disposal trench, which is 500 feet long, 40 feet wide at the bottom, and 200 feet wide at the top, was receiving 49,000 gpm of B Reactor effluent. Test wells surrounding this facility and the springs along the river bank are monitored regularly.

ENGINEERING AND TECHNOLOGY - REACTORSFuel Support Hydraulic Characteristic Tests

Hydraulic tests were performed on 51-element K5E fuel columns fitted with 50-mil arch supports and with the CMLE support. Preliminary analysis of the data indicates that the 50-mil arch supports have relatively little effect on fuel column hydraulics, only increasing the fuel column pressure drop from 280 to 290 psig at operating conditions. The CMLE support, which is 40 mils thick but wider than the 50-mil supports, has a greater effect; it increases the column pressure drop to 335 psi at operating conditions.

Shipping Container Evaluation

An evaluation of shipping containers assigned to DUN, which are used for offsite shipments of radioactive and fissile material, shows that the containers do not presently meet AEC Manual Chapter 0529 criteria.

An investigation of means for modifying these containers to meet the above criteria has led to the design of a plywood buffer container to go around the present casks (HAPO-13 and -14 and the Reed College casks). This buffer should allow the existing containers to withstand the 30-foot fall and fire test outlined in the criteria, without loss of shielding in the container itself. The buffer is basically a layer of Douglas Fir plywood six inches thick over all surfaces of an existing container. Two such buffers are being built for test purposes at a cost of about \$600 each. This contrasts with an estimated cost of about \$9,000 for an all-metal buffer container designed to perform the same function.

Ball 3X Modifications - C & K Reactors

The Ball 3X bellows serve as a flexible passage for the 3X balls from the biological shield to the thermal shield. The thermal shield is supported by the reactor graphite, and continued graphite contraction from irradiation effects is causing the thermal shield to sag. The sag causes the bellows to stretch

and will eventually threaten the structural integrity of that portion of the ball channel.

The solution successfully demonstrated on October 13 for the K Reactors is to sever the bellows, to preclude their supporting the thermal shield, and then to wrap a steel shroud around the severed bellows to preserve the constrained passageway for the balls. (Reference: May 1967 issue of this report)

A similar solution has been proposed for the C Reactor Ball 3X bellows, and design of equipment is nearing completion.

Universal Flexible VSR

Failure of two universal flexible VSRs at KE Reactor led to a reevaluation of the VSR design. Analytical work has confirmed that the present design should be satisfactory, and was not involved in the failures. Adjustment and testing of the VSRs at KE and KW has shown marked improvement of rod deceleration characteristics.

A test program to verify the stress levels determined analytically is in progress at Bldg. 195-D.

Columbia River Hydrology

Work is under way to complete the analysis of the CY-1967 Columbia River cooling program, with a view toward early publication of the report.

Personnel from the R. L. Albrook Hydraulic Laboratory, WSU, were here on November 9, in connection with the river modeling program being conducted at WSU under Contract DDR-112. A boat excursion on the river was taken for close examination of river flow conditions, particularly in the Coyote Rapids area. In addition, field survey data were gathered to determine the precise location of the river head loss which occurs between Coyote Rapids and the 181-KW Building. The information obtained will be utilized in the river model facility at WSU to establish a single model control point for use during various test programs.

Immersion Type RTDs - KE Reactor

Examination of 34 RTD assemblies removed from the KE Reactor showed the following:

Nine RTDs were from hot effluent water locations; three of these showed strong evidence of having failed due to water traveling down the cable conductors; four had failed due to water traveling down the conductors or by water migrating through the rubber into the pin cavity; and two had failed due to water entering along the aluminum reinforcing structure and into the pin cavity. Most of these units showed large quantities of clean water, which implied that the water which caused failure had entered only a short time before. Twenty of the 34 RTDs failed due to low resistance to ground and were from cold tubes. The remaining five RTDs were classed as open circuit type failures.

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On-reactor failure information was not obtained on a group of seven RTDs removed from the reactor. These seven contained three RTDs thought to be open circuit, but bench tests in the lab did not verify this supposition. Sectioning of two which registered the highest resistance to ground showed one of these units to have no weld between one pin and the connected braid. All other RTDs showed good welds. A number showed some corrosion due to water in the pin-braid junction.

Two RTDs removed because of intermittent-type circuit failures were sectioned. During sectioning, the electrical circuit in the assembly was monitored continuously for resistance changes. No open circuits were found. Examination by three independent observers showed all welds to be satisfactory. The Monel bodies of the RTDs were sectioned to determine whether the resistance element could be contributing to the open circuit; these elements were found to be in excellent condition. The rubber-to-rubber bands and the rubber-to-metal bands in both RTD assemblies also were excellent.

It has been established that under certain conditions, water can migrate through the silicone rubber and condense on the inside of an integral connection. A search has been initiated for coating material and water repellents which could act as barriers to this water migration. Of the materials tested to date, only ethylene-propylene has showed significantly low moisture migration to be of interest. Other materials for testing are on order. Work is also under way to locate a consultant on moisture barrier coatings and the RTD failure problem.

It is possible that a deflector applied to the integral connection of an RTD assembly would be a means of alleviating failures on hot tubes in water leak locations. To evaluate this approach, four hundred metal deflectors have been fabricated and are ready for installation on RTD assemblies at KE Reactor.

Project Engineering - Reactor Facilities

Project Status Summary

The status of approved construction projects relating to B, C, and K reactor facilities is summarized in Appendix A.

ENGINEERING AND TECHNOLOGY - FUELS AND TARGETS

Component Cleaning

As an interim step, while redesigned cans are being developed and procured to replace the present stock, an improved component cleaning process has been in effect for the past two months for the K reactor fuel models. Early testing showed manual cleaning to be superior to the automatic cleaning machine as measured by the ultrasonic base end closure tester, and large scale testing is underway to determine the reason for this difference.

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Manual Cleaning Plus Automatic Machine

Eleven baskets totaling 1300 cans have been processed through the manual line, followed by automatic machine cleaning. Comparison with cans cleaned only manually shows the following:

Fuel Model	No. of Days	Braze Integrity Test Reject Rates - %	
		Manual and Machine Cleaned	Manual Cleaned Only
K5E	4	1.1	2.1
K5N	8	0.7	1.0

As shown above, the combination of manual and automatic cleaning appears to yield best results.

Without Deoxidizing Step in Manual Cleaning

In a second test, the same procedure was followed as above but without the deoxidizing step in the manual cleaning phase of the combined manual and machine procedure. In four days of processing, the cans which had not received the deoxidizing step in the manual cleaning line had 1.5 percent rejects; this compared to 0.7 percent for those receiving all process steps in the group given only manual cleaning.

The third test, which is not yet completed, consists of degreasing the cans on the manual cleaning line only with the rest of the cleaning process steps occurring in the automatic machine.

Hot-Die-Size Process Variables Test

Material for testing hot-die-size process variables under PTA-060 was charged into KE Reactor late in the month. This test, consisting of 27 fuel columns, is designed to determine the effects of clad thickness, reduced end bonding time with increased pressure, aluminum cladding annealing, and their interactions upon ledge and groove corrosion. Three of the test columns are scheduled for discharge at about one-half goal exposure for preliminary evaluation.

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

Design criteria for the Mark IV fuel, which have been reported previously, are reviewed as follows:

- Reduce failure potential by minimizing surface heat fluxes and peak uranium core temperatures.
- Reduce tube power restrictions under the non-boiling criterion by minimizing imbalance of subchannel enthalpy rise.

Tentative dimensions for base case and thin-clad fuels conforming to the above criteria have been verified by FLEX, a fuel element design and analysis code. Final verification must await the results of measurements to be made on test columns during reactor operation and on test columns in the flow loop. Fabrication of test columns of Mark IV fuel having the modified dimensions should be complete by mid-December.

The fabrication of 50 columns of Mark IV fuel of the initial design is nearing completion. Irradiation of this fuel is scheduled to start after the first full reactor operating period following the end of the strike.

Life testing of Mark IV fuel in the flow loop has been terminated after approximately 2,400 hours duration at a flow rate of 340 gpm. Results indicate that a wider restraining stop for the inner fuel cylinder is required to ensure axial positioning of the inner and outer fuel tubes.

1-B. Advanced Technology Case

The Advanced Technology Engineering Feasibility Study is being updated for incorporation of comments and probably will be issued in December. This Study considers four levels of increased reactor power in the anticipated power ascension program. Systems capabilities have been evaluated in relation to these levels.

Mission 2 - Coproduct2-A. Mark II Driver Failures

Investigation of the three Mark II drivers which failed in August was continued. The findings show a strong similarity among these failures and augment previously reported similarities of their in-reactor operation and post-discharge appearance.

Two of the failures, No. 40 and 41, now appear identical in every respect. Both have points of water entry on the inner cladding at the contact made by the solid support that was fixed between the locking clip and the stop. Failure 39 has yielded no new clues, and the point of water entry is still judged to be in the vicinity of the stop. Findings and conclusions about each of the three failures are as follows:

Failure 39

Careful examination has removed three possible causes from consideration:

- End associated; i.e., closure porosity.
- Fretting corrosion by wires or chips.
- Fretting corrosion by the W-spring or solid supports.

Difficulty in identifying the point of water entry is due primarily to the extensive physical damage and distortion incurred by the cladding when the element failed.

Failure 40

Examination has shown that the previously reported inclusion is a plug of dense crystalline ZrO_2 . The reason why the plug could not be seen on the water surface of the cladding seems to be that a thin covering cap of metallic zircaloy exists over the oxide plug, and the water entrance point was around the perimeter of the 0.025-inch diameter plug.

The mechanism by which so selective a corrosion reaction occurred is not obvious. It appears that the covering cap of Zircaloy was protected whereas the perimeter of the cap was attacked. At the present time, the possibility that the Zircaloy cladding was selectively contaminated cannot be ruled out. However, the possibility also exists that a reactor-initiated failure occurred by reason that (1) the in-reactor positions in the columns (position 3 or 4) are common to all three failures, (2) the geometry of the locking clip and stop may result in inadequate cooling of the contact point of the solid support and the inner cladding, and (3) there are regions where there is an excessively thick layer of black ZrO_2 on the cladding.

Failure 41

Examination of the cracked end cap was halted when it was observed that the crack surfaces were free of ZrO_2 . It was apparent that the crack was a result of the failure, and not a prior condition that had caused the failure.

With the inner cladding separated from the uranium core metal by selective corrosion of their interface, it was relatively easy to slip the inner cladding out of a 6-inch long section cut from the end of the

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element. Attention was promptly directed to the region of the locking clip and stop, and an identically appearing inclusion was found on the uranium-contacting surface. Again, there was no visible defect on the coolant surface. The cladding was sectioned to remove the area containing the inclusion and to orient it precisely with the contact surface of the solid support.

The cladding has been examined on a metallograph and the coolant contact surface shows a crazing of the black ZrO_2 layer. The crazing shows itself as patches of black bounded by white lines. There is an overall imposed smear of red when viewed with polarized illumination. The area of cladding that was contacted by the other solid support shows a similarly crazed surface, but the overall deterioration is not of comparable severity.

Mission 3 - Transplutonium Technology

There were no significant developments in this program.

Mission 4 - Pu-238

There were no significant developments in this program.

Mission 5 - Other Isotopes

There were no significant developments in this program.

Mission 7 - Target Space Enhancement

There were no significant developments in this program.

Mission 8 - Nuclear Safety

8-A. Behavior of Overheated Fuel

Last month's report described the behavior of two high-exposure N Reactor fuel elements heated to their failure temperatures. The fission products released from one fuel piece that was heated in a steam atmosphere to a maximum temperature of 1,994 F, and then reduced to about 1,832 F, have been measured and analyzed. During the test, the cladding ruptured near the center of the fuel at 1,860 F and 50 to 60 percent of the uranium was forced from the cladding. Oxidation of the hot uranium foam created localized temperatures above the uranium melting point. The fission products released from this fuel element were as follows:

<u>Element</u>	<u>Percent</u>
^{85}Kr	39
^{131}I	<1.8
^{137}Cs	0.7
$^{140}\text{Ba-La}$	0.4
$^{95}\text{Zr-Nb}$	0.02

The krypton released was several times higher than the amount measured in previous tests. This most probably reflects the fact that there was a greater degree of uranium foaming and extrusion during this test than previously. The xenon released appears anomalously low, so the measurements and calculations are being rechecked. The long delay period and short test precluded the release of detectable quantities of iodine; however, the results do not disagree with indications from previous tests that the release of iodine from failed fuel is low.

Additional fuel elements have been discharged from the reactor to permit resumption of the failure testing if arrangements can be made to transport the fuel to the Bldg. 292-T facility during the strike.

8-B. Status of 324-D Fission Product Release Laboratory

Some specific construction jobs fell slightly behind schedule, but overtime will be used by J. A. Jones to meet the December 12 construction completion goal. The manipulators are to be received January 19 and the motor-generator set will arrive about mid-February. Cell shakedown will be done during January and February, with hot tests still expected to start late in February.

ENGINEERING AND TECHNOLOGY - N REACTOR

Process Tube Monitoring

Six process tubes were borescoped after the reactor had been decontaminated during the current strike outage. All tubes displayed their normal black oxide coating. It was surprising to observe, however, that a substantial area in each tube was still covered with brown crud deposits. Each of the tubes also had some downstream spacer fretting marks.

Effluent Control Program

A meeting was held with Professor Earl R. Parker at the University of California, Berkeley, on November 14, to discuss brittle fracture failure and the provisions in the piping specification to prevent its occurrence. The new

emergency raw water supply line was discussed, and methods for evaluating the existing emergency raw water supply header at the reactor were also considered. The general conclusion expressed by Professor Parker was that the required safeguards are being taken to prevent brittle fracture failure.

Project Engineering - Reactor Facilities

Backup Boiler Facility

The first of the two package boiler units is now on site. Some damage occurred in shipment and approximately 20 to 30 tubes will require repair or replacement. All tie-in work to the existing boiler systems has been completed.

Project Status Summary

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

ENGINEERING AND TECHNOLOGY - FUELS AND TARGETS

Mark IV Fuel Fabrication

The Mark IV (modified) fuels require a completely different end cap and braze ring; however, it is believed that the welding parameters used for the standard Mark IV fuels will be completely acceptable. The end caps being used have been designed with a lip on the top edge. This additional metal will permit intimate contact between the cladding and the cap after crimping, and should eliminate nonflow defects.

One phase of the Mark IV program is the utilization of a large Mark I billet inventory. Because the Mark IV billet has a much smaller inner diameter than the Mark I, a uranium sleeve is required. This sleeve has been successfully extruded from a Mark I billet and later successfully coextruded as a fuel piece.

ADVANCED CONCEPTS AND PLANNINGMFCs AND OTHER PROPOSALSMFC-12 - New Capabilities for Production Reactors

Nothing to report this month.

MFC-13 - Study of Application for Hanford-Produced Cobalt-60

The final report on this MFC, which reviews potential applications for cobalt-60, is now in preparation.

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OPERATIONS RESEARCHCAGE Applications

The eleven Case B cases were submitted to Oak Ridge for combinatorial processing. These cases are being used to test the shutdown-startup capability of the CAGE program.

Overbore Study

An updated overbore study of the B, C, and K reactors was completed. The objectives were to determine (1) the time when the cumulative unit cost of plutonium after overbore will equal the corresponding unit cost without overbore, and (2) the date when all information would be needed for budget decision, assuming funding would not be available prior to fiscal 1971.

The major conclusions of the study were as follows:

- Considering all constraints and lead times, overboring of the reactors could not begin earlier than the third quarter of FY 1972.
- Information for budget decision would be required by mid-FY 1969.
- Reasonable assurance exists that the technology and equipment for overbore will be fully developed and demonstrated by mid-FY 1969, which means that the overbore schedule could be advanced at least one year.
- Final plutonium unit costs obtained in the study are about \$4 a gram lower than unit costs without overbore on a total Hanford basis.

- The time when the cumulative unit cost of plutonium for the overbore cases equals the pre-overbore unit cost ranges from early FY 1975 to late FY 1976, depending on the number of reactors overbored and the sequence.

The study also reminded that solely overboring these reactors is not considered as being the ultimate in means of reducing unit cost. It would, however, equip them to take advantage of other major cost improvement steps which are now under study.

Advanced CAGE Programming

Mark IIa

Changes dealing with fuel cycle times, fuel materials evaluations, and nondefense plutonium tracking were made a permanent part of the program. Simplification of fuel preparation calculations was added whereby no preference is given to the B, C, or K reactors insofar as recycle U-236 content is concerned. Several changes have been made to the Mark IIa financial program. The new N Fuels variable cost equations were implemented in the program and they appear to be programmed properly. The current technology equation for research and development has been deleted from the program. In future CAGE cases, the same equation will be used for both current and advanced technology.

Mod-2

Programming for the technical portion of Mod-2 is nearly complete. The remaining work involves inventory calculations, and it is the intent to proceed with it. Mod-2 descriptor cards are now being generated routinely. A print of these card images is received with each case presently run.

MINIMODEL

Costing equations for the AECOP MINIMODEL have been submitted to Oak Ridge. Production data for seven reactor modes have been sent to date. Ultimately data will be submitted on 26 modes covering plutonium production from 6 to 30 percent Pu-240 and tritium production at enrichments up to 2.1 percent. The production data will be expanded later as better definition is obtained on future enrichment levels.

N Reactor Advanced Technology Evaluation (AOP Study)

The data-gathering phase of this study is proceeding at a satisfactory pace. CAGE cases are being prepared for primary options. Capital and R&D funding schedules also are being prepared.

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IRRADIATION SERVICESFUEL TECHNOLOGY

A neutron flux measuring capsule was charged into a bottom front-to-rear test facility at KW Reactor. This capsule, whose neutron cross-section is designed to simulate that of a United Nuclear uranium carbide test capsule, will provide data for the design of flux attenuating sheaths to be used in the UNC experiments.

An uranium oxide capsule was discharged from a B Reactor fringe process tube after an irradiation of four years. The objective of this test, conducted by Battelle-Northwest, is to determine the mechanism and extent of irradiation damage to sintered UO₂ exposed at low temperatures for extended periods of time.

ISOTOPE PRODUCTION

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SPECIAL TESTING

Calorimetry measurements of in-reactor heat generation rates for selected non-fissionable materials have been completed in a KE Reactor facility.

The materials measured were chosen either because of a specific customer irradiation need, or to provide data supporting the method for calculating heat generation values. The values obtained compare favorably with those predicted. The measurements were made using a unique calorimeter design. Both the calorimeter development and the measurement results will be reported in detail.

ROUTINE IRRADIATIONS

The following routine irradiations were performed:

- One hundred fifteen activation analysis samples were irradiated in the KE and KW Reactor Quickie facilities for Battelle-Northwest.
- One wave guide neutron detector assembly and one diamond X-ray cell capsule were irradiated in the KE Reactor Snout facility.
- Three cooled tensile specimen capsules were irradiated in the KW Reactor Snout facility.

Two packs of cigarettes were irradiated to produce radiobromine tracer in the cigarettes. These cigarettes will be used by Battelle-Northwest in studies of smoke residue retention in the lungs.

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ADMINISTRATION - GENERAL

UNION RELATIONS

The work stoppage which started on September 1 at the N Reactor continued throughout November. Numerous meetings were held with the Hanford Atomic Metal Trades Council during the first half of the month, but these resulted in no progress.

On November 17 the Federal Mediation and Conciliation Service intervened on their own motion. Joint meetings were then held under FMCS auspices. No progress was made, and on November 28 the FMCS recessed negotiations subject to call by the Mediator.

The "Red Circle" rate arbitration case was heard on November 29. A decision is not expected for 30.- 60 days.

MONTHLY PENSION ROLL

The first monthly pension to be paid by DUN was established with Bankers Trust Company. This required implementation of the Bankers Trust procedures for reporting the addition to the pension roll.

COMPUTING AND DATA PROCESSING COSTS

Average monthly computing and data processing charges from Computer Sciences Corporation, for the four months ended October 31, were 30 percent higher than the average monthly charges for FY 1967.

It was expected that with the use of the larger computer configurations, the historical pattern of reduced unit costs would follow. However, unit costs increased rather than decreased; this, together with reassignment of programmers prior to completing systems programming for DUN, contributed to increased charges. The increase also reflects work on the AEC MIS system.

HANFORD INFORMATION SYSTEMS (HIS)

A report "Scope of DUN Subject Areas" was issued to the HIS Task Force members and DUN personnel early in November. The multicontractor HIS Task Force is working on a number of specific assignments related to the overall conceptual design of the system. These include: (1) the development of a clearing house system concept for process transfers between contractors, (2) the classification of individual contractor interests as to common, related and unique interests, and (3) a plan for undertaking the conceptual design. All three areas of work are to be completed early in December and presented to the multicontractor Steering Committee for approval prior to start of the overall conceptual design.

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AEC MANAGEMENT INFORMATION SYSTEM (MIS)

The MIS System is now operating on a regular monthly basis with updated tapes sent to Washington each month. This has made possible the streamlining of routines.

COST REDUCTION

The Company has received recognition for its cost reduction activities. The September issue of "Cost Reduction Abstracts," published by the AEC, includes four cost-saving actions initiated by DUN.

A review of DUN records in the Integrated Cost System has been made in an effort to decrease the overall computer running time. Input changes affecting 54 liquidation records should give a measurable time saving.

INVENTORIES

An audit of Special Reactor Materials and Other Special Materials showed excellent control being maintained over inventories; all items were readily accounted for. The only variance between book and physical inventory was a small amount of platinum loss attributable to normal process deterioration.

An audit of B-C Processing Essential Materials showed that adequate control is being maintained over the inventory despite a lack of centralized storage space at both reactors. The problem will be alleviated at C Reactor early in 1968 when additional space becomes available.

A report detailing findings of the 1967 physical inventory of Special Process Spares and Spare Parts was issued to appropriate DUN and AEC-RL personnel. Actual variations were minimal; revisions of the sampling technique were recommended.

Adjustments to the inventory accounts for the variations were made in November.

FUNDING OF RETIREMENT LIFE INSURANCE PREMIUMS

In a letter dated October 23, AEC-RL disapproved ATD-117, "Funding of Retirement Life Insurance Premiums." The letter gave no reason for this disapproval, but stated that alternate proposals would be considered by the AEC. As this same problem exists with other contractors, it was recommended to AEC-RL in a letter dated November 1 that they develop a proposal which would contractually provide for (1) an acceptable method of prefunding, or (2) payment of the actual cost as incurred, with costs allocated to work performed during the periods of service which qualify employees for the benefits.

APPROVAL LETTERS

At the close of the reporting period, the following approval requests had not been acted upon by AEC-RL:

<u>ATD Number</u>	<u>Subject</u>	<u>Date of Transmittal to AEC-RL</u>
AP-39	Pension Plan portion of letter entitled "Pension Plan, Salaried Savings Plan and Wage Savings Plan"	January 12, 1966
ATD-90 Add. #1	DUN Participation in AEC Combined Operations Planning Committee	August 23, 1967
ATD-98 Add. #1	FY 1968 Budget for Attendance at Meetings of Professional and Trade Societies	September 1, 1967
ATD-99 Add. #1	Off-Site Courses and Seminars Program	July 17, 1967
ATD-127	N Reactor Safety	July 14, 1967
ATD-132	Production Computer - Data Transmission	September 26, 1967
ATD-133	Off-Site Transfer of Douglas United Nuclear Employees (MO-15)	October 30, 1967
ATD-141	Douglas United Nuclear, Inc. Annual Report	November 7, 1967

EMPLOYMENT SUMMARY

DUN employment as of November 30 is summarized in Appendix C.

SAFETY

No disabling injuries or property damage accidents occurred during the period. No radiation exposures exceeded operational control limits.

Month end safety statistics were:

Disabling Injuries:	November	0
	CY to date	2
Days since last disabling injury		157
Man-hours since last disabling injury		1,500,000

APPENDIX A

PROJECT STATUS SUMMARY - REACTOR FACILITIES

Number & Title	Authorized	Percent Complete		Status
		Design	Construction	
<u>B, C & K Reactors</u>				
DCE-505, Boiler Control Improvements - 165-KE and KW	\$240,000	70	0	Factory testing of the control system power supplies has been scheduled for the week of November 27. Drawing prepared for installation of instrument air filters.
DAP-507, Work Area Fog Spray Systems - 105-B, C, KE and KW	\$149,000	100	99	Acceptance test procedure work is all that remains. This is complete at one reactor; other three scheduled on next reactor outages.
DAP-510, Discharge Chute Clearing Equipment - K Reactors	\$190,000	87	0	Detail design in progress. Electrical and piping drawings complete. Mechanical drawings being reviewed.
DAE-512, Replacement of Turbine with Diesel Drive - 181-B Pump	\$87,000	61	NS	Design held up pending decision to reuse existing reducer gear. Reuse will require rebidding on diesel engines. Design has slowed awaiting selection of a vendor and vendor design information.
DAP-513, Deactivation of Hanford Production Reactor (formerly DAP-509)	\$50,000	100	2	Design complete but construction (except for token start) is awaiting frozen funds. AEC-RL is obtaining estimates from J. A. Jones.

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PROJECT STATUS SUMMARY - REACTOR FACILITIES (contd)

Number & Title	Authorized	Percent Complete		Status
		Design	Construction	
<u>N Reactor</u>				
GAP-401, Upgrading Fire Protection - 100-N	\$150,000	100	10	Physical performance by Fire Detection Services started November 17. Overall construction progress approximately 19% behind schedule.
GCP-404, Fuel Spacer Disposal System and Refuse Cask - 100-N	\$77,000	100	56	Pouring of lead shielding by National Lead has been delayed but is now scheduled for November 30.
GCE-405, N Reactor Temperature Monitoring System Improvements	\$189,000	83	57	Bids for new RTD cable were received November 27 and are being evaluated.
GCP-406, Safety Platforms and Accesses	\$90,000	95	35	Work on 109-N cell platforms and 105-N pipe space platforms continuing ahead of schedule.
GCE-407, Fast Indexing of Rupture Monitoring System	\$50,000	100	45	Installation of fast-drive mechanisms is in progress in the left-side turret room. Overall construction is about 14% ahead of schedule.
GCE-408, W, C, D Elevator Safety	\$90,000	40	0	Vitro/HES detail design proceeding, but 50% behind schedule. Overall design completion still expected by January 1, as scheduled.
GCP-409, Fuel Handling Improvements	\$310,000	45	0	Detail design is slightly behind schedule. Fabrication work by J. A. Jones held up due to current freeze on construction funds.

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PROJECT STATUS SUMMARY - REACTOR FACILITIES (contd)

<u>Number & Title</u>	<u>Authorized</u>	<u>Percent Complete</u>		<u>Status</u>
		<u>Design</u>	<u>Construction</u>	
<u>N Reactor</u>				
GAP-410, Decontamination Waste Loading Facilities	\$65,500	43	0	Detail design work by Vitro/HES proceeding on schedule. Authorized funding increased by \$500.
GCP-411, Effluent Control Program - 100-N	\$1,670,000	40	0	Detail design is proceeding on schedule.
DCP-514, Air Conditioning, Water Quality Lab., 109-N	--	--	--	Project Proposal reviewed by AEC-RL on November 9. Release of directive deferred pending termination of temporary freeze on construction funds.

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APPENDIX B

SIGNIFICANT REPORTS ISSUED

<u>Number</u>	<u>Classification</u>	<u>Author</u>	<u>Date</u>	<u>Title</u>
DUN-2904	Secret	JP Schmidt	8/14/67	PTA-060 - Hot Die Sizing Process Variables Test.
DUN-2911	Unclassified	KJ Matty	8/7/67	PUPA - A Code for the Calculation of Pu-238 Production
DUN-2953	Confidential	HF Jensen CH Gydesen RL Woodward	11/6/67	Operation Guide for K Reactor Zircaloy 2 Process Tube Renovation
DUN-3039	Secret	JP Schmidt	9/1/67	PTA-088 - Hot-Die-Sizing Die Lubricant Removal Test
DUN-3112	Secret	WJ Dowis DL Renberger RD Shimer	11/3/67	N Reactor Power Level Capability
DUN-3145	Secret	GF Bailey	9/22/67	PTA-080 - Physics Tests at D Reactor Deactivation Interim Report - Experimental Data
DUN-AOP-65	Secret	RW Bown TS Dodge	11/17/67	Production Data and Budget-Type Costs for Planning Estimate Case PEA-3C.
DUN-AOP-69	Secret	WJ Dowis	11/20/67	Analysis of Savannah River Matrix Given in DUN-AOP-32, June 1, 1967
DUN-AOP-70	Secret	Edited by WA Blanton	11/21/67	Reactor Overbore Study
RL-GEN-1067 SUP24	Secret	RD Shimer	11/2/67	PT-NR-75 - Fuel Exposure Probe

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APPENDIX CEMPLOYMENT SUMMARY
(as of 11/30/67)

	<u>Exempt</u>	<u>Non- Exempt</u>	<u>Total</u>	<u>Change from 10/31/67</u>
<u>Operations Division</u>				
General	1	1	2	0
Manufacturing	212	705	917	-3
N Reactor	68	201	269	-5
Production Fuels	78	249	327	-2
N Fuels	37	71	108	+1
	396	1,227	1,623	-9
<u>Technical Division</u>				
General	1	1	2	0
Research & Engineering	79	42	121	+4
N Research & Engineering	45	19	64	+2
Advanced Concepts & Planning	10	2	12	0
Facilities Engineering	55	18	73	+1
N Project Engineering	55	18	73	-1
	245	100	345	+6
<u>Finance & Administration Division</u>				
General	1	1	2	0
Finance & Administration	72	76	148	+1
Employee & Community Relations	19	11	30	0
"Technical Graduates"(1)	24	.0	24	-2
	116	88	204	-1
<u>Company General</u>				
Management	5	1	6	-3
Total	762	1,416	2,178	-7

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

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FEATURE REPORT

NUCLEAR SAFETY INSTRUMENTATION - B, C & K REACTORS

INTRODUCTION

The purpose of this summary report is to describe improvements made in nuclear safety instrumentation at the B, C and K reactors in recent years, to relate those improvements to the overall complement of nuclear instrumentation at these reactors, and to show how these instrument systems' parameters and characteristics have been chosen to fulfill nuclear safety criteria. Also summarized are current programs for the further improvement of reactor instrumentation.

The speed-of-control criterion, which these systems and associated controls must satisfy specifically, means in effect that reactivity introduction rates must be controlled such that no situation of excessive specific power can develop relative to the available heat removal capacity. In other words, the relationships between reactivity ramp, specific heat generation rate, and coolant integrity are the bases for determining the necessary characteristics of reactor safety instrumentation systems, including their types, sensitivities, redundancies, interdependence, and dependability for assuring response when required. The many other reactor-associated instrument systems utilized in monitoring such process parameters as gas and water quality and graphite and shield temperatures, which have a potential bearing on speed-of-control but are less directly involved in accident prevention, are not covered in this brief report.

BASIC CONSIDERATIONS

As recently as the early 1960s, few instrument systems significantly more sophisticated than those included in the original World War II designs had been added to these Hanford production reactors. Use of the flow monitor systems had been upgraded to the extent of replacing the procedure of manual trip, subsequent to alarm, with automatic trip upon threat to individual tube flow. Power level safety protection, from low level to high, was provided by the flux monitor system; this system had automatic trips, but was procedurally dependent upon the setting of ranges and trip levels during operation, and was limited in the geometrical coverage afforded.

Technological advances in other areas have facilitated the solution of the problem of defining and automatically protecting against excessive specific power generation rates. High speed digital computers were becoming available for the solution of large-scale scientific problems, permitting analyses of complex and interacting reactor kinetics parameters with controls activated by instrument systems for nuclear safety purposes. As a result, it was possible to design new instrument systems, which, together with special interlocks and limits placed upon the existing systems, could provide a complete package to ensure the required degree of reactor protection. The

second area of technological advance which permitted the substantial upgrading of reactor safety instrumentation was the development of reliable solid state devices, and of more accurate, precise, and specialized analog instruments. The instrumentation at each of the B, C, and K reactors has subsequently been upgraded to meet the performance requirements derived in the above studies.

The primary systems in the current instrumentation package may be broken down into two major categories: (1) those which protect against loss (or potential loss) of coolant, and (2) those which protect against excessive (or potentially excessive) heat generation rates. Items in the first category are: the seismoscope and all-tube pressure monitor systems, power failure relays (to detect loss of coolant pump power), low pressure-very low pressure detectors (to detect too rapid a pressure decay), and an extremely-low-pressure detector (to detect sudden and essentially complete flow loss). Functions included in the second category are: startup flux monitoring, control of power ascension rates, and localized high-level trips. These functions are covered by the flux monitor system, the zone temperature monitor (ZTM) system, and the linear rate-of-rise system.

At the two K reactors, the functions of the ZTM and linear rate-of-rise systems are covered by the high speed scanning (HSS) systems recently installed there. Since a description of this sophisticated new instrumentation was featured in the January, 1967, Douglas United Nuclear Monthly Report (DUN-2011), these systems are not discussed herein except to note when they provide protection at the K reactors comparable to that afforded by the integrated composite of other instrument systems described for the B and C Reactors.

Figure 1, appended, shows by analytically derived results how the interaction of the various systems provides overlapped protection such that a safety circuit trip is assured prior to the occurrence of boiling for any postulated reactivity ramp. The three numbered curves on the graph apply as follows:

Curve 1 applies to the ZTM system (or the HSS system zone trip at the K reactors), where the region of protection is the area to the left of the curve.

Curve 2 applies to the linear rate-of-rise system (or the HSS system rate-of-rise trip at the K reactors), where the region of protection is the envelope below and bounded by the curve.

Curve 3, applicable to the flux monitor system, is really a band - the relationship between level and allowable reactivity ramp being a function of reactor flux distribution. The area of protection provided by the flux monitor is the area to the left of and below the applicable level-ramp relationship.

Further protection is assured by augmenting the automatic instrumented trips with interlocks which (1) prevent withdrawal of more than two horizontal control rods at a time, (2) automatically unbypass the linear rate-of-rise system at power levels above two megawatts, (3) prevent withdrawal of

horizontal control rods until all vertical safety rods have been fully withdrawn, and (4) prevent withdrawal of more than two vertical safety rods at a time at B and C Reactors. If two B or C Reactor VSRs are withdrawn simultaneously, they cannot be adjacent to each other or in the same longitudinal row. (At the K reactors the equivalent of no more than two VSRs may be withdrawn in comparable time because of the system pneumatic limitations).

As a result of installation of the ZRM system, the linear rate-of-rise system, and various interlocks, the reliance upon operational procedure for assuring automatic reactor protection from fast periods, fast rates-of-rise, and localized excessive levels has been essentially eliminated.

Figure 2, appended, indicates the effective power level ranges of operation for the various primary instrument systems. Each of these systems is described briefly in the following sections.

COOLANT LOSS PROTECTION SYSTEMS

Continuity of coolant flow is vital to nuclear safety, and immediate shutdown in the event of threatened coolant loss is considered essential for minimizing meltdown and radiation release effects in the event of coolant loss. This protection is afforded primarily by two instrument systems, as described below.

Seismoscope System

The seismoscope system has the function of detecting potentially damaging earthquakes (of Modified Mercalli intensity scale MM4 and above) and of initiating a reactor scram by tripping the No. 1 safety circuit.

The system consists of three separate pendulum starter switches (located adjacent to the reactor), each with an associated solid state relay unit which operates into a two-out-of-three coincidence or progressive trip logic to actuate the safety circuit. Following a trip, the seismoscope relay remains de-energized until manually reset. Each seismoscope is equipped with a separate annunciator in the reactor control room. The power supply on each of the three units is backed-up by a battery supply which would last a minimum of four hours after complete loss of normal power.

A further refinement, at the K reactors, is a No. 3 safety circuit (Ball 3X) trip in the event that at least two flux monitors are not at low-trip points within five seconds after a seismoscope-initiated scram. Loss of power to the flux monitors during this five-second period also would result in a No. 3 safety circuit trip.

The seismoscopes at all four reactors include recent modifications to improve reliability to trip-on-demand and to reduce shutdowns due to false trips. Improved reliability to trip on demand was achieved by installing solid-state relay circuitry with a backup battery power supply. Spurious

trips were eliminated by changing the trip matrix from two-out-of-two to two-out-of-three coincident or progressive (if not reset) trips.

Pressure Monitor System

This primary instrumentation is commonly referred to as the "Panellit System." Although the pressure monitor gauge on the inlet to each process tube, with its high and low trip capability, is intended as a flow protection device operating over the heat producing range of the reactor, the gauge also provides high power level backup protection.

The trip-after-instability (TAI) philosophy of limit application is administered by setting high and low pressure trip limits on each of the individual process tubes to cause a reactor scram prior to the occurrence of "burnout" conditions which might lead to fuel melting. Although this protection is intended primarily to guard against flow reductions, it is also effective in scrambling the reactor in event of tube powers sufficiently excessive to initiate boiling.

The circuitry is arranged such that a coincidence of both the row and column trips of any one gauge actuates the No. 1 safety circuit. A trip of any pressure monitor gauge is annunciated in the control room and the gauge responsible for the scram is identified by a trip identification system. Axial flux distribution currently has only a second order effect on fuel column burnout limits with the degree of pressurization available in these production reactors.

The pressure monitor system may be bypassed during outages and at low power levels (≤ 2 MW) of reactor operation. However, the system is interlocked with the flux monitor system through reliable solid state logic that the pressure monitor system is automatically unbypassed for reactor operation above 2 MW.

EXCURSION PROTECTION SYSTEMS

The nuclear instrumentation range requirement is from normal background or shutdown levels to approximately ten decades above normal background. Because a single instrumentation system is not presently available which would adequately cover both rate and level requirements over this large range, several different systems are employed to monitor reactor operation at the various power levels (see appended Figure 2). To provide adequate sensitivity at high levels, rate-of-change and high-level instrument systems with thermal sensors are used which overlap nuclear instrumentation protection in the upper ranges (see appended Figures 1 and 2).

Low-Level Neutron Monitor System

The low-level neutron monitor is used from shutdown or background levels (subcritical) to well up into the intermediate power range, approximately nine decades. During reactor outages for refueling, the low-level neutron monitor is utilized continuously to monitor for significant loading errors.

This system has two separate and identical channels, each of which consists of: a movable fission chamber (mounted on a screw-driven mechanism for chamber positioning over a total travel in reflector and shield of 60 inches), an amplifier, a five-decade logarithmic count rate meter, and period and level indicators and recorders. The system provides visual and audible alarms if an excessively short rising period occurs.

The primary use of this system, which has been available for procedural use during the past 10 years, is to determine when the reactor achieves criticality, and to indicate the magnitude of the rising period during low power level operation.

Intermediate and High-Level Systems

The instrumentation discussed under this heading provides protection in the intermediate and high power level ranges (see appended Figure 2). The first three such systems cause automatic safety circuit actuation; in addition, several others are used to inform the operator concerning power level, rate-of-rise, and flux distribution.

Neutron Flux Monitor System

The neutron flux monitor system, commonly referred to as the "Beckmans," is composed of four redundant, uncompensated ion chambers, picoammeters, recorders, and controllers for the setting of trip points. High trip-point setting is procedurally controlled, at full-scale or lower ranges and at a fixed percentage above the steady-state instrument reading on the operating range. To assure automatic rate protection in low and intermediate level ranges, the low trip-point is mechanically limited to a setting no less than two percent of full scale.

The four ion chambers are located in instrument passages at riser locations beneath the reactor and are positioned for equilibrium power level calibration. The effective range of operation is the top six decades of the ten-decade range of neutron flux. Both high and low level trips are connected into the safety circuit through a matrix logic circuit such that any two flux monitors in the tripped position, high and/or low, will trip the No. 1 safety circuit.

Each flux monitor has a manual bypass. Interlocks are provided so that the bypassing of more than one monitor will trip the No. 1 safety circuit. In addition, interlocks are provided which cause automatic unbypassing of the pressure monitor system and the linear rate-of-rise system whenever two or more flux monitors are ranged to a power level of two megawatts or greater.

Linear Rate-of-Rise System

At the B and C Reactors, the linear rate-of-rise system provides reactor trip protection against excess power rates in the intermediate and equilibrium power ranges (see appended Figures 2 and 3). The HSS systems serve this function at the K reactors.

The linear rate-of-rise system consists of three separate channels, each of which monitors the strap-on resistance temperature detectors (RTDs) located on the outlet connectors of ten process tubes in geometrically representative zones of the reactor. An average rate of tube outlet temperature rise (which is directly proportional to the average rate of power rise) is calculated by each channel. The system will cause a No. 1 safety circuit trip whenever coincident trips occur on any two of the three channels. The linear rate-of-rise system is automatically unbypassed by the flux monitor system at reactor power levels above two megawatts.

The linear rate-of-rise system is used to monitor startup power rate-of-rise after the subcritical monitors have been discontinued in the low megawatt range. The power rate-of-rise is displayed at the reactor control console.

Zone Temperature Monitor System

The Zone Temperature Monitor (ZTM) system provides a trip of the No. 1 safety circuit at the B and C Reactors in the event that excessive process tube outlet temperatures occur in any reactor zone. At the K reactors, this function is provided by the HSS systems.

The ZTM system consists of 91 channels on a three by six process channel grid, with one point every six columns and every three rows within the central zone of the reactor. Each channel includes a strap-on RTD, a magnetic amplifier, and a controller with a fixed high trip set point. Tripping of any point on a row will trip the entire row, and tripping of any two adjacent rows will trip the No. 1 safety circuit. Separate power supplies are utilized for alternate rows. A signal is annunciated at the reactor control console when any temperature point approaches trip.

A lighted visual display panel permits the operator to select any temperature he desires for monitoring both the tube outlet temperatures and the distribution of the hottest tubes. This capability is especially useful during startup.

Octant and Galvanometer Systems

Systems in the original design, for providing operator control information, include an octant flux monitor system at the C and K reactors (non-gamma-compensated neutron sensitive ion chambers located in each of the eight corners of the reactor) to provide trend information, and a galvanometer system at the B, C, and K reactors which affords an indication which is proportional to reactor power. The galvanometer is driven by the octant system at the C and K reactors, and by a single ion chamber in a central side test hole at B Reactor.

SAFETY CIRCUITS

The instrument systems described above are connected to one or more of the three automatic safety circuits, as listed in Table 1 and shown schematically in Figure 3 (appended). Each of the safety circuits has a supplementary manual trip and each is designed for fail-safe operation. They are constructed such that a specific protective instrument system will de-energize a pair of relays which in turn will actuate control and safety devices (vertical safety rods, horizontal control rods, and/or the Ball 3X system) under emergency conditions to scram the reactor. Ground detectors are provided on each of the safety circuits.

The primary protective safety circuit, appropriately designated No. 1, is counted upon for fulfilling fast shutdown (speed-of-control) requirements. It is actuated by the instrument systems which monitor coolant flow and temperature, power rate-of-rise, and flux level, and it also provides protection against electrical power loss and severe seismic activity. Shutdown is accomplished by the safety circuit release of the vertical safety rods (VSRs) which drop into moderator channels by gravity.

The No. 2 safety circuit is the control circuit for the horizontal control rod (HCR) emergency drives. It is normally tripped as a secondary action when the VSRs leave the withdrawn position (following a No. 1 safety circuit trip). It can also be used separately when it is not necessary or desirable to have the VSRs enter the reactor.

The No. 3 safety circuit controls the third safety system, commonly known as the "Ball 3X". A trip of this safety circuit causes hoppers of 3/8- and 7/16-inch steel and boron-steel balls to be released into the VSR channels. This automatically provides additional poison and ensures, through the fluid-like action of the balls, the insertion of poison in the event of a damaging earthquake which might result in rapid or complete loss of coolant, and/or deactivation of the VSRs and/or HCRs. The Ball 3X system also acts as a manual backup system should the VSRs fail to enter the reactor following a No. 1 safety circuit trip.

CURRENT NUCLEAR INSTRUMENTATION PROGRAMS

Nuclear instrumentation at the B, C and K reactors has been upgraded during the past three years to meet the systematic performance requirements determined from earlier analytical studies. As a result, instrument development effort currently is directed primarily toward the improvement of process flexibility and efficiency. Improvements in these areas will, however, enhance reactor safety through better knowledge of the nuclear process, and through more exacting operational monitoring and response. Two such development programs are summarized below.

In-Core Flux Monitors

The B, C, and K reactors are not now equipped with in-core nuclear instrumentation for dynamically monitoring central zone flux distribution.

Table 1. Safety Circuit Instrumentation

<u>Trip Signal</u>	<u>Trip Requirement (No./total)</u>	<u>Reactor</u>	<u>Protection Against</u>	<u>Safety Circuit Actuated*</u>
Pressure Monitor	1/2004 1/3220	B, C Ks	Flow Reduction	No. 1
Flux Monitor	2/4 2/4 2/4	B, C, Ks B, C, Ks Ks	High Flux Not Properly Ranged Flux Decay Too Slow**	No. 1 No. 1 No. 3
Seismoscope	2/3	B, C, Ks	Seismic Activity	No. 1
Zone Temp. Monitor	2/13 rows (adj. rows, 1/7 in a row)	B, C	Localized Excursion	No. 1
Rate-of-Rise	2/3	B, C	Excess Rate	No. 1
High Speed Scanner	6/28 15/3220 1/3220	Ks Ks Ks	Excess Rate-of-Rise High Outlet Temperature Very High Outlet Temperature	No. 1 No. 1 No. 1
Power Failure	1/8 1/10 1/12***	B C Ks	Loss of Pump or Pump Power	No. 1
Low Pressure - Very Low Pressure	2/2	B, C, Ks	Too Rapid Pressure Decay	No. 3
Extremely Low Pres- sure	2/2	B, C, Ks	Flow Loss	No. 3

* The No. 2 safety circuit is actuated whenever the No. 1 system trips.

** Seismoscope intertie (see text).

*** Each K has 6 low-lift and 6 high-lift pumps.

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Such an in-core system would provide the axial flux distribution data required for accurate calculation of local specific powers and local burnout safety factors.

Development work has been in progress in this area for several years, and the testing of prototype systems using both gamma sensitive and thermal neutron sensitive in-core flux monitors has already been conducted.

A development program is currently under way to evaluate rhodium self-generating thermal neutron sensitive detectors. Potential advantages of this latter type of detector are low cost, lack of polarizing voltage, and small physical size; possible disadvantages, relative to the previously tested chambers, are longer response time and lower signal strength.

The current testing of both the rhodium self-generating detectors and a traveling fission chamber calibration system is targeted to permit selection of a system for installation as a full-scale demonstration in one reactor by the first quarter of 1970.

On-Line Computers

Recent studies of on-line digital process computer application to the B, C, and K reactors indicate a highly promising potential for the use of such devices as aids to reactor operation. Applications intended for these reactors fall into five general categories: (1) data logging and report generation; (2) calculational aids to reactor operation; (3) sensor and instrument monitoring and checking; (4) outage communications and pre-startup checks; and (5) calculation and notification to the operator of appropriate control rod movements. An ultimate possibility in this area could be direct digital control of the entire nuclear process. Acquisition of the first system is targeted for late in FY-1969, with beneficial use before the end of FY-1970.

Safety benefits from this computer application would include a significantly better knowledge of the process, continuous calculation and monitoring of operating limits, extrapolation of operating trends to predict potential "problems", and automatic display of reactor startup status and conditions following each outage.

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Trip Functions: (as marked)

1. High tube outlet temperature (High Speed Scanner or Zone Temperature Monitor)
2. Rate of change of tube outlet temperature (HSS or Linear Rate-of-Rise)
3. High flux level (Neutron Flux Monitors)

* Typical limiting transient (continuing ramp at indicated level)

Figure 1. Safety Trip Limit Curves

Subcritical Region	Criti- cality Region	Intermediate Range	High-Level	← Equilibrium Power Level

Figure 2. Effective Ranges of Reactor Instrumentation

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Number 1 Safety Circuit (Vertical Safety Rod Drop and HCR Insertion)

* Dual at Ks only

Number 3 Safety Circuit (Ball 3X Activation)

Number 2 Safety Circuit (Horizontal Control Rod Insertion)

Figure 3. Reactor Safety Circuits - Schematic
(at all reactors unless otherwise noted)