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HW--76398

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IRRADIATION PROCESSING DEPARTMENT

MONTHLY REPORT

JANUARY, 1963

HANFORD ATOMIC PRODUCTS OPERATION

GENERAL ELECTRIC

RICHLAND, WASHINGTON

MASTER

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IRRADIATION PROCESSING DEPARTMENT
MONTHLY REPORT
JANUARY, 1963

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SUMMARY

MANUFACTURING

Reactor input production (MWD) was 0.2 per cent above forecast; 2.6 per cent below at the six old reactors and 3.2 per cent above at the K's.

Over-all time operated efficiency was 69.7 per cent (70.3 per cent forecast); 65.2 per cent at the six old reactors and 83.1 per cent at the K's.

There was no increase in reactor record power levels.

Nine ruptured fuel elements, seven I&E natural and two I&E enriched, were removed from the reactors. Three of the natural metal ruptures were at D, two at DR, one at F and one at H. The two enriched ruptures were at DR and H.

A total of 764 process tubes was replaced; 346 at H, 220 at B, 192 at F, five at D and one at DR.

Two tube and three Van Stone leaks were corrected. The tube leaks were at D and H, both caused by ruptures. This represents the best tube leak experience since February, 1959. Two of the Van Stone leaks were at F and one at D.

PRODUCTION FUELS

A total of 664 tons of natural and enriched fuel elements was produced for 101 per cent of the forecast.

Production through the AlSi Shop was at the rate of nine canning lines per day for the first three days of January and ten lines per day for the balance of the month. Canning line operating efficiency was 99.0 per cent for the month. This is a new record for 10-line operation.

Bare core inventory reflects a 1.5 months' supply and finished fuel a 2.1 months' supply. This does not include 288 tons of dingot material in the bare core inventory and 105 tons in the finished fuel inventory.

The AlSi slopover reject rate was lower during January. The reduced rate is attributed to the components used during the month.

During uranium quality control receiving evaluation, a core from 6-inch Lot CY-004, which is vacuum outgassed material, was found to have a 1.5-mil thick film of UO₂ on the I.D. Normal pickle operation would not entirely remove this film. Engineering is instituting further study of the vacuum outgassed material pending decision as to over-all quality. Feed site production of this material has been discontinued by Engineering.

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RESEARCH AND ENGINEERING

Reactor power levels were limited by tube outlet temperature limits at B, C, DR and F Reactors, by bulk outlet temperature limits at D and H Reactors, and by administrative limits at the KE and KW Reactors.

Production Test IP-549-A, "Half-Plant Low Alum Feed Water Treatment at F Reactor," was initiated on January 16. One half of the reactor will use water treated with the minimum alum feed required to provide reactor quality water, but not less than 6 ppm; the other half of the reactor will use water treated with alum feed at 18 ppm. Of the 20 fuel charges discharged and examined to establish pretest conditions, 26 per cent of the elements showed localized corrosion attack, nine per cent showed severe corrosion attack and one per cent showed penetration to the AlSi bond.

The special corrections imposed on DR Reactor total control procedures to compensate for the 1961 Ball 3X drop have been cancelled, as boron burnup has largely overcome the poisoning effect of residual balls in the stack.

Operation of the 1706-KER recirculation loop facility has continued in support of the NPR development program. An out-of-reactor recirculation test in KER-1 was completed; the objective was to evaluate problems of changing from lithium hydroxide to ammonium hydroxide for pH control. Testing of eight N-Reactor inner fuel elements with modified end closures continued in KER-2 Loop. KER-3 Loop remained shut down because of primary pump failures. The test of 14 N-Reactor fuel elements continued in KER-4 Loop.

FACILITIES ENGINEERING

Material and equipment are available for the first zirconium tube outage, currently scheduled to begin February 11 at the 105-KW Reactor. All material except the gas boots is on site; these latter are in transit. A total of 269 Zircaloy process tubes has been received, of which 198 have been accepted. One hundred twenty-five nozzles are on hand, and 150 are ready for shipment.

Laboratory testing of the production model boring bar contemplated for use at the 105-K Reactors for process tube channel renovation has been completed. Mockup tests at 108-D indicate a minimum of ten years of residence for zirconium tubes in all zones of the K Reactors based on current fuel concepts and exposure rates. Procurement of equipment for the first major outage is in progress and is currently scheduled for completion by early April.

The HAPO designed self-supported fuel element charging machine is being readied for use at the conclusion of the February Zircaloy tube replacement outage at the 105-KW Reactor. Additional delays in completion and testing of the two machines being procured from vendors in Seattle now indicate these

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will be delivered between the second and fourth weeks of February. At least three weeks of off-reactor testing will be required after they are received.

Testing and development of alternate VSR liner materials is continuing. Out-of-reactor drop testing of aluminum oxide, pyrolytic graphite, and polycrystalline graphite resulted in a life of approximately 1000, 170, and 100 rod drops, respectively. The 1000 drop life expectancy for the aluminum oxide liner material resulted from modifications of the VSR rod tip to reduce impact forces. Drop tests with the modified tip are planned for the graphite materials. The design of equipment to enlarge the VSR channels for sleeve installation is essentially complete. Fabrication and testing of components are well along with testing scheduled for early February.

The 100-K Area high pressure crosstie line was cleaned to reduce the friction loss to backup water flow provided under Project CGI-844 (100-K Area Coolant Backup). The capacity increase so achieved will be determined during the next outage by flow test.

RESPONSIBILITY

There were no significant changes in responsibilities.

FORCE SUMMARY

	<u>Exempt</u>	<u>Non-Exempt</u>	<u>Total</u>
General	7	1	8
Research and Engineering	84	55	139
Manufacturing	392	1403	1795
Production Fuels	108	425	533
Facilities Engineering	89	37	126
Financial	19	16	35
TOTAL	699	1937	2636

SAFETY, SECURITY AND RADIATION EXPERIENCE

There were 118 medical treatment injuries, one security violation, no disabling injuries or radiation exposures exceeding operational control.

INVENTIONS

All persons engaged in work that might reasonably be expected to result in inventions or discoveries advise that, to the best of their knowledge and belief, no inventions or discoveries were made in the course of their work during the period covered by this report except as listed below. Such persons further advise that, for the period therein covered by this report,

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notebook records, if any, kept in the course of their work have been examined for possible inventions or discoveries.

<u>Name</u>	<u>Date</u>	<u>Title</u>
R. J. Kasko	1-23-63	Reciprocating Pump Internal Packing Lubricator.
W. E. Cawley	1-30-63	Explosive Gas Forming of
C. E. Frantz		Van Stones on Process Tubes.
F. F. Vlacil		
E. R. Astley		
L. W. Lang	1-23-63	Synthetic Nuclear Fuels

General Manager

AB Greninger:DLD:ng

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MANUFACTURING

PRODUCTION OPERATION

There were no discharge goal exposure adjustments initiated during January; however, action was taken to increase unclassified plutonium production at B and F Reactors.

JANUARY DISCHARGE EXPOSURE (MWD/T)

<u>Regular Uranium</u>		<u>Enriched Uranium</u>	
<u>6 Old</u>	<u>2 K's</u>	<u>6 Old</u>	<u>2 K's</u>
616	736	847	905

Rail and truck shipments of Essential Materials were as follows:

Carload shipments for IPD	-	393
Carload shipments for other Depts.	-	179
Truck shipments for IPD	-	<u>150</u>
Total		722

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MONTH	REACTOR OPERATION STATISTICS - PROCESSING										TOTAL HOURS	% OF TIME
	B	C	D	DR	F	H	KE	KW	TOTAL			
JANUARY, 1963												
INPUT PU (% OF FORECAST)	92.7	100.2	95.6	121.5	81.5	87.9	103.2	103.2	100.2			
TIME OPER. EFF. (% OVERALL)	70.2	82.0	79.4	81.0	50.6	28.0	84.0	82.3	69.7			
NEW TUBES INSTALLED	220	0	5	1	192	346	0	0	764			
WATER LEAKS - TUBE	0	0	1	0	0	1	0	0	2			
- VAN STONE	0	0	1	0	2	0	0	0	3			
FUEL ELEMENT RUPTURE	0	0	3	3	1	2	0	0	9			
HELIUM CONSUMED - M CU. FT.	1012.9	934.6	798.5	361.6	545.5	214.2	322.6	241.9	4431.8			
UTILIZATION OF REACTOR OUTAGE HOURS												
CHARGE - DISCHARGE	20.6	22.9	40.6	43.0	28.2	117.0	28.9	75.2	376.4	6.3		
RUPTURE REMOVAL	0.5	5.8	13.1	25.0	14.4	17.6			76.4	1.3		
WATER LEAK		40.1	16.8		11.4	54.0						
TUBE REPLACEMENT	163.8	7.9	15.3	14.3	150.3	283.2		19.7	122.3	2.0		
MAINTENANCE	19.0	38.2	45.9	49.4	133.5	35.1	53.4	8.4	654.5	11.0		
STANDARD CHECK			0.9		1.5	13.4			382.9	6.4		
PRODUCTION TEST	4.7	18.2	19.8	5.4	13.5	7.7	14.9	2.8	15.8	0.3		
PROJECT WORK									87.0	1.5		
OTHER	13.1	0.6	0.5	4.3	15.0	7.7	22.2	11.2	11.2	0.2		
TOTAL	221.7	133.7	152.9	141.4	367.8	535.7	119.4	131.9	1804.5	30.3		
REACTOR OPERATION STATISTICS - POWER												
RIVER WATER PUMPED - M GALS.	8507.8		6947.4		2483.3	2095.5	8814.5	8739.7	37588.2			
WATER TREATED - M GALS.	3320.5	4617.5	3943.7	2856.4	2447.1	2017.7	8767.9	8739.7	36710.5			
WATER TO REACTOR - GPM NOR.	89200	97000	87200	86000	89200	83700	205900	205500	943700			
FINISHED WATER - ANALYTICAL DATA												
TURBIDITY PPM AVG.	.002	.004	.003	.003	.003	.003	.006	.005				
PH AVG.	6.90	6.90	7.00	7.00	6.95	6.92	6.60	6.60				
CL ₂ RESIDUAL PPM AVG.	.05	.004	.15	.15	.05	.11	.07	.05				
DICHROMATE PPM AVG.	.99	1.40	1.00	1.00	1.00	.97	1.77	1.77				
STEAM GENERATED M LBS.	71273		88962		46274	52664	28845	26969				
EVAPORATION RATE/LB. OF COAL	7.92		7.60		8.90	8.79	130.9	107.0				
EVAPORATION RATE/GAL. OF OIL							1501200	1573600				
TOTAL KW - HB GENERATED									3074800			

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B-C REACTOR OPERATION

An outage for metal processing and a tube replacement outage occurred at B Reactor. At month's end, 220 process tubes had been replaced.

Three significant outages occurred at C Reactor; two due to high water collection and one due to an unexplained Panellit gauge trip. Failure of obsolete thermocouple wells caused water to leak into the reactor through the rear face bellows on both water leak outages.

At B Reactor, all central zone tubes have been probologged since October. Those tubes showing evidence of accelerated corrosion were replaced and water mixers included in the recharged metal columns. At the conclusion of the tube outage now in progress, the enriched core will be retubed and recharged.

At the time of the unexplained Panellit scram at C Reactor, the upper limit switch on VSR No. 48 failed and was replaced during the shutdown.

The 10-inch Bell and Spigot cast iron line supplying fire and sanitary water to the 184-B Building broke adjacent to a shutoff valve and was repaired.

Pump No. 9-C was removed from Building 181-B for a Class "A" overhaul. After dismantling, inspection revealed that the throat on the discharge end of the first stage bowl was completely eroded away. Some of the 1 1/8-inch stud bolts used in the flanges of the first and second stage bowls were also eroded through 75 per cent of their diameters.

Decontamination of the rear face of B Reactor resulted in a 44 per cent dose rate reduction to an all time low of 24 mr/hr.

D-DR REACTOR OPERATION

D Reactor experienced four outages. Those which resulted in appreciable outage time were two outages due to failure of fuel elements and one outage due to a process tube water leak. DR Reactor experienced seven outages. Those contributing appreciable outage time were two outages due to failure of fuel elements, one outage due to an unexplained Panellit trip, and one outage to investigate a high tube outlet water temperature reading. Although unexplained, cause of the high reading is attributed to a fault in the instrumentation system. The four fuel element failure outages at D and DR Reactors are attributable to the unclassified plutonium program since the ruptured slugs were all in tubes which had exceeded the goal exposure for normal metal.

The feeding of sulfuric acid at Building 190-D to lower the pH of process water for one-half of D Reactor (PT IP-442-A) was started on January 30.

At DR Reactor, effort was continued to improve vertical rod drop times. In addition to broaching five rod channels, two B-D-F knuckle type rods, one B-D-F standard rigid type rod and one DR semi-flexible type rod were installed in place of the standard rods

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The number of idling process water backup turbine pumps was reduced from nine to seven at both Buildings 190-D and 190-DR. This change, made possible by a modification in Process Standards, is equivalent to steam savings valued at approximately \$500 per month.

F REACTOR OPERATION

Production during the month was limited by five reactor outages; a carry-over rupture outage from December, a scheduled tube replacement, a leaking spline cap, a vertical safety rod scram, and a ruptured fuel element.

During the tube replacement outage, 289 process tubes were probologged and, of these, 24 required replacement. Water collection from the reactor by the dryers in Building 115 remained between three and five gallons per day after equilibrium levels were reached.

As authorized by Production Test IP-549-A, "Half Plant Low Alum Feed Water Treatment, F Reactor," alum feed to the west half of the 183-F Building filter plant was reduced from 18 ppm to 6 ppm on January 16. Alum feed remains at 18 ppm for the east half of the plant in order to compare aluminum corrosion in the reactor under the two different feed rates.

For corrosion control purposes, sodium silicate feed to the west high tank was started on January 14. The outlet piping of the tank was successfully descaled by mechanical means before the addition of the sodium silicate.

Instrument activities included the acceptance of the temperature monitor Analog-Digital Converter system and the installation of two additional rear face vertical thermocouple bundles; this program is now 54 per cent complete. In addition, three graphite stringers were replaced with Inconel sheathed thermocouples.

Electrical activities included the replacement of the clutches and associated bearings in four vertical safety rod clutch units and an operational check on two other vertical safety rods. In addition, the relays and motors on both 2.3 KV incoming lines to the 181 Building were calibrated, checks were completed on the No. 2 incoming 2.3 KV bus to the 182 Building, and the 1608 Building switchgear circuit breakers were overhauled.

Mechanical activities included replacing 192 process tubes, 191 during the tube outage and one additional tube due to a ruptured fuel element. Four crossheader to riser connections were repaired, new mattress plates were installed, three front face gunbarrels were replaced to provide for new graphite stringers, and the No. 18 vertical safety rod channel was broached for improvement of rod insertion time.

Efforts toward gas loss improvement included replacing the 115 Building No. 1 and No. 2 dryer drain lines, foaming the Omega seal between the 21 and 22 crossheaders and replacing eight gunbarrel boots.

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H REACTOR OPERATION

Reactor outages included a tube leak outage in progress at the start of the month, two outages caused by ruptured fuel elements and an extended outage for process tube replacement and metal processing. The process tube containing the rupture, 1975-H, was also found to be the source of a water leak.

Wall thickness of 754 tubes was measured by the probolog method during the extended tube replacement outage. The tubes for replacement were chosen on the basis of indicated wall thickness as extrapolated to the next scheduled tube replacement outage. In preparation for the extended outage, the reactor rear face hardware was decontaminated, thereby reducing the average exposure levels by 40 per cent to an average rate of 31 mr/hr.

A total of 346 process tubes was replaced.

During metal processing operations, ten per cent of the tubes processed required greater-than-normal pressures to discharge the metal columns. Seven of these tubes required replacement due to internal damage incurred from removing stuck fuel elements and dummies.

The following design changes were completed:

DC 613 - Installation of high level radiation alarm in the reactor discharge area.

DC 637 - Modification of the 105 console to relocate and accommodate additional instruments.

DC 601 - Modification of D platform electrical controls to permit the by-pass of all other control locations.

The impellers of the No. 7 pump (190 annex) were found to have cracks in the suction vanes and were replaced with a rebuilt assembly. The drive gear thrust bearings of pumps No. 5 and 7 were also replaced due to their cracked condition.

KE-KW REACTOR OPERATION

KE Reactor outages included four Panellit scrams and a manual shutdown to correct a leaking front face flexible connector. Causes of the Panellit scrams were a leaking Bourdon tube, a defective spline cap insert assembly, a defective spline, and a broken front face nozzle. The latter incident was formally investigated and reported in HW-76354, "Complete Failure of Front Face Nozzle 1162-KE".

KW Reactor outages were a scheduled shutdown for metal processing and KE-KW high pressure crosstie system cleaning work, and a Panellit trip.

The wall thickness of 309 process tubes at KW Reactor was measured by the probolog method. Results indicated that 12 of the tubes have a remaining wall

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thickness of 37 mils or less. Of the approximately 2,000 tubes which have been measured at KW Reactor to date, 13 tubes have indicated wall thicknesses between 33 and 35 mils. Results of 584 tubes measured at KE Reactor during the previous month indicated that 11 of the tubes have a remaining wall thickness of 37 mils.

Successful replacement of Horizontal Control Rod No. 19 at KW Reactor was achieved. Prior to removal, it was necessary to saw the rod into three sections. All three pieces of the rod, including the related step-plug, were removed and buried. Borescoping the channel revealed the presence of 3X system balls, rub marks and a broken tube bearing block. The rod channel was cleared, a new rod and step-plug were installed, and the rod was returned to normal service.

An Inconel sheathed geminol front-to-rear graphite thermocouple stringer was installed in process tube channel 2870 at KW Reactor. The installation is to support evaluation of the He-N₂ atmosphere and its effect on higher permissible reactor graphite operating temperature limits.

More than normal glycol heating system leakage problems were experienced. One particularly troublesome leak was located in the 150-KE heat exchanger. To correct the condition, the exchanger was isolated and temporary repairs were made. A total of 280 heat exchanger tubes has been plugged at 150-KE and similar repair has been made to 200 tubes at 150-KW. Plans were made to re-tube both exchangers under the Planned Maintenance procedure.

APPLIED REACTOR ENGINEERING

MAINTENANCE PRACTICES OPERATION

The inventory of B-D-F-type aluminum tubes is 3540. Other tubes on hand include 667 C-type aluminum and 319 K-type.

SPECIAL OUTAGE SERVICE OPERATION

During the splitting operation on the past 531 tubes, eight stuck splitters were experienced; seven at F Reactor and one at H Reactor. A total of 4.3 outage hours was required for removal.

A funnel, designed to prevent contamination spread in the rear face during tube pushing operation, was used successfully in F and H.

The personnel exposure per tube required for tube replacement remained at an average of 40 mr for craftsmen and 13 mr for supervisors.

MAINTENANCE ENGINEERING OPERATION

The new transistorized Analog to Digital Converter unit was placed in operation at 105-F on January 15, 1963.

Repairs were completed to the inlet chute in 107-KE C retention basin.

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During the January outage at 105-H, 340 nozzles were examined to determine the condition of the threads and fittings and six per cent of the nozzles showed varying degrees of thread damage.

OC Schroeder

Manager, Manufacturing

OC Schroeder:DLD:bm

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B-7

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PRODUCTION FUELS

MANUFACTURING

Current Month's Production

	NATURAL					ENRICHED					Total
	C	K	Reg.	Bumper	K5S	C	K	Reg.	Bumper	K5ES	
Acceptable Fuel Elements (Tons)	69	174.5	292.3	0	19.6	19.7	19.9	22.3	47.1	0	664.4
Per cent of Forecast	95	111	117	-	56	99	69	112	83	-	101

Manufacturing Yield

Current Month Per cent	90	87	88	-	71	91	90	91	81	-	-
Forecast Per cent	90	90	90	85	55	90	90	90	85	55	-
Previous Month Per cent	92	87	92	97	-	92	91	86	84	-	-

Bare Uranium Available for Processing (Tons) 902

Finished Products in Storage (Tons) 1180

Work in Process (Tons) 232

Special Products Finished (Pieces)
 4" Boll - 478
 6" Boll - 26

Uranium Utilization Per cent 98.1

Fuel Recovery

8" I&E (Pieces) 29,970
 6" Watermix (Pieces) 1,768

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MANUFACTURING

Production

A total of 664 tons of natural and enriched fuel elements was produced during January, 101 per cent of forecast.

Yields and efficiency for January were as follows:

	<u>January</u>	<u>Forecast</u>	<u>Variance</u>
Eight-inch I&E	89	90%	-1%
Six-inch I&E	91	90%	+1%
Operating Efficiency	99	96%	+3%

Projection elements produced during January totaled 47 tons of six-inch material and 20 tons of eight-inch fuel. The yield on the six-inch material was 81 per cent and 71 per cent on the eight-inch. The welding yield on this fuel was 92 and 91 per cent, respectively.

Production and reactor charging for January and to date are as follows:

	<u>January 1963</u>		<u>Nov. 1960 to Date</u>	
	<u>Natural</u>	<u>Enriched</u>	<u>Natural</u>	<u>Enriched</u>
Tons Produced	20	47	1851	972
Tons Charged	38	151	1738	863
Tons Finished Inventory	113	109	113	109

A total of 478 lithium-aluminum four-inch elements and 26 six-inch elements was placed in storage during the month.

Bare inventory is 902 tons, a 1.5 month's supply. This figure does not include 288 tons of dingot material.

Finished inventories total 1180 tons, a 2.1 month's supply. This amount excludes 105 tons of dingot material.

Scrap

One shipment of scrap was made during January, totaling 10 tons natural and 5 tons of enriched metal. One shipment of 420 empty boxes with lids was also made.

Canning Line Operation

Production through the AlSi Shop was at the rate of nine (9) canning lines per day for the first three (3) days of January and ten (10) lines per day for the balance of the month. Vacation relief people not currently needed are being utilized to run the tenth line.

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AlSi Slop Rejects

The AlSi Slop reject rate this month is lower than it has been in a year. The eight-inch rate is approximately .9% and the six-inch rate is .8% for both shifts. The rates are lower because of component scheduling.

Cap and Can Machine Problems

There have been several broken rails on the can machine this month. Although some of the broken rails were due to new operators who weren't completely familiar with the machine, the majority were due to bent can baskets and hangers that would not lock to the basket. A speed-up in basket repair and additional hanger alignment was instituted to alleviate the problem.

During times when the can machine is down for rail replacement, it is frequently necessary to raise the machine carriage and the solutions are then dried on the components. This results in stains on the can that often cannot be removed and it becomes necessary to scrap the cans. During the troubles encountered with the can machine this month, we have had to scrap several hundred cans due to these stains.

Change in Welder Turntable Speed

The turntable speed of the semi-automatic welders was reduced from 20 rpm to 12 rpm for the E-N load material. This was to improve the quality of the weld. Two of the welders have been changed back to 20 rpm in order to get a comparison of the two weld speeds. Samples from both speeds are being collected and the 20 rpm weld appears to be as good if not superior to the 12 rpm weld.

Pre-Set Tooling

Pre-set tooling has been put on lathe #3. The lathe operators change the rough face, finish face, and contour tool. The millwrights continue to change the ream and chamfer tools. Pre-set tooling will be put on the other lathes as the tool holders become available in production quantities.

Rail Welders

The rail welder operation has been unsatisfactory. It was very costly in both material (high reject rate) and downtime to get 2B and 2C set up on self-supports. 2B has not yet performed satisfactorily on self-supports. After welding most of the K5N's, it took three days to get 2B set up so we could weld bumpers. After a long shutdown on 2A, it was very difficult to get it operating properly. Under present conditions, the most profitable method of rail welding is to weld on one product for as long as possible. With O3E's, C5N's, K5N's and K5E's scheduled for railing in February, we can expect difficulties with re-etching, set-up, and storage problems.

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Ajax Furnace Failures

On January 20, 1963, Ajax Furnace Serial #1017 located in 7-C recovery position failed due to AlSi penetration through to the induction heating coil and core of the lower hearth. The furnace was removed from service for a complete upper and lower hearth refractory replacement. This is the first upper hearth replacement for this furnace since the initial start-up in October of 1954.

It is expected that the renovation will be completed within the next two weeks; after which time, the furnace will be fired, charged, and placed back in service.

On January 26, 1963, the fuses blew on Ajax furnace serial #1765, located in 4-D position. The electrician removed a large piece of AlSi slag wedged between the induction coil and the core. This apparently had caused a short circuit and the resultant fuse failure. The fuses were replaced and the furnace was restored to normal operation without visible damage to the core or coil.

Zig-Zag Conveyor Inspection

An engineering inspection of the canning and component Zig-Zag conveyor system was completed this month. The purpose of this inspection was to determine the degree of deterioration and wear of the over-all conveyor network and establish a listing of replacement parts necessary to renovate system and thus assure continuity of operation.

A cursory analysis of the inspection data has been completed and it is apparent that all 90° twist and adjacent 90° vertical curved sections in the component conveyor should be scheduled for replacement in the very near future. Other minor modifications and replacements are also being recommended to assure future trouble-free operation. The canning conveyor appears to be in exceptionally good condition and should require very little attention over and above the recommended productive maintenance.

Pressurized Quench Machines

On January 12, 1963, at the request of Quality Control Engineering and Operations, Cuno filters with a 5 micron filler were installed on the air lines to the pressure bells on canning lines 2 and 5. It was theorized that the welding pin hole rejects could be the result of air contaminants injected into the braze layer during the pressure quenching operation; however, information received from Quality Control indicates no measurable reduction in pin hole rejects in material canned on these lines.

UT-2 Testers

UT-2 bare core testers have operated with an availability of 96%. This represents an improvement of 6% over the December performance. There were no prolonged outages sufficient to cause any canning line down time.

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UT-4 Testers

UT-4 weld integrity testers have operated with an availability of 84%. This represents a loss of 4% over the month of December. There were several outages of long duration which made it necessary for the Finishing Operation to backlog material. However, there were no canning line outages due to tester malfunction. It has become obvious that the outages due to mechanical wear are more and more prevalent and it will soon be necessary to remove these tester tanks for major overhaul.

UE-1 Testers

The UE-1 bond and penetration testers have operated with an availability of 87%. This is essentially the same as the December performance. There were no outages of sufficient duration to result in canning line downtime.

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FUELS ENGINEERING

Spire Failures

Available data indicate that groove and/or ledge-type corrosion (both are extremely non-uniform types of attack) probably caused the majority of side other and hole failures incurred during the past six or seven months. Spire (hole) failures that are believed to be corrosion induced characteristically have incurred in high temperature tubes at higher exposures, say greater than 750 MWD/T. The spire (hole) failures thus far have been incurred exclusively in high temperature, high exposure bumper fuel charges containing water mixers. The fact that the spire failures have been confined to bumper fuel charges containing mixers can possibly be rationalized on the basis that the great majority of high outlet temperature tubes irradiated in the old reactors to exposures beyond 750 MWD/T, fall in this category.

Annular Failures

Annular (side other) failures that are believed to be corrosion induced begin appearing at somewhat lower exposures, say 550 MWD/T, but also in the higher temperature tubes. The annular type of failure has been incurred in both bumper and non-bumper fuel; to date, however, all failures of this type have been in charges irradiated in relatively new process tubes not containing water mixers.

PT-423-Oil vs. Water Quenched Cores

The 14-tube designed portion of the oil vs. water quenched core test has been irradiated and measured, and a preliminary analysis of the results has been made. The test basically consisted of irradiating 14 tubes, each of which was alternately charged with water and oil quenched normal uranium elements. The cores for this test were fabricated at the National Lead Company of Ohio, and, in order to eliminate between ingot and within ingot effects, alternate blanks from each ingot rod were oil quenched following heat treatment. The other pieces were quenched in water as usual. Eleven (11) ingots were used in this test.

As originally designed, it was intended that the elements would be loaded in oil and water quenched in pairs from the same ingot, and that all ingots would be represented in each reactor tube. Unfortunately, this detail was overlooked during the test, and ingot to ingot differences were confounded.

The elements were irradiated to an average tube exposure of 800 MWD/T in F reactor.

Upon completion of irradiation, the elements were measured for OD, ID, and warp.

All indications are that the oil quenched material was definitely superior to the water quenched material. It had an average of 1.4 mils less warp and showed no increase in bumping (contrary to what might have been expected from the larger grain size).

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A serious weakness with the results to date is the lack of any information concerning the post-irradiation core dimensions (i.e., absolute dimensions -- relative dimensions on the two types of material are available). Representative samples are presently being selected for hot cell metallographic and density measurements. A final report will be issued upon completion of this work.

Bumper Fuel Elements

The irradiation status of bumper fuel elements in the old reactors is as follows:

	<u>Total Charged</u> (Tons)	<u>Total Discharged</u> (Tons)	<u>No. of</u> <u>Ruptures</u>
Natural Bumper	1738	1026	28
Enriched Bumper	863	606	34*

*Includes 15 ruptures in H reactor E-N load.

None of the ruptures sustained to date have been of the type bumper fuel was designed to prevent; an increased incidence of bumper fuel ruptures at high goal exposures during recent months has resulted in an assessed performance level that is not significantly better than non-bumper fuel. Most of the recent ruptures appear to be caused by either ledge or groove corrosion of the cladding. Failures attributed to this corrosion are largely dependent on fuel residence time and occur primarily in higher power factor tubes.

Self-Supported Fuel Elements

A summary of the irradiation status for overbore and regular self-supported fuel in the C reactor test facilities is as follows:

	<u>Total Pieces</u> <u>In Reactor</u>	<u>Total Pieces</u> <u>Discharged</u>	<u>Ruptures</u>	
			<u>Side*</u> <u>Other</u>	<u>Core</u> <u>Split</u>
Overbore (CVIN)	1386	4,509	2	5**
Regular (CIVN & CVN)	2208	14,488	1	0

*External cladding corrosion failures caused by broken or damaged rails of a former design.

**Includes three ingot core failures in a single column associated with a temperature cycling incident and two dingot core failures in separate columns.

As previously reported, the exposure of dingot overbore fuel was reduced to 75 per cent of goal (800 MWD/T) because dingot fuel has proven to be more rupture prone than ingot uranium. Use of dingot overbore cores will be discontinued as soon as about 10 tons remaining in process inventories are utilized.

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Alternate Aluminum Component Vendor Evaluation

About 147,000 non-bumper and 87,000 bumper OIIIN fuel elements have been clad in Harvey aluminum components, which were fabricated after major process and facility changes were made in this vendor's plant to upgrade product quality. The irradiation status of these fuel elements as of January 1, 1963, was approximately as follows:

<u>Fuel Type</u>	<u>No. Discharged at Goal</u>	<u>No. In-Reactor</u>	<u>No. to Be Charged</u>	<u>Ruptures</u>	
				<u>Test</u>	<u>Control</u>
Non-Bumper	85,000	43,000	19,000	7	3
Bumper	41,000	24,000	22,000	0	2

Based on ex-reactor tests and processing of the improved Harvey components, the metallurgical, chemical, and physical properties of this vendor's product are comparable to the control material. Irradiation data obtained thus far do not show either equivalence or a significant difference in the two vendors' products. Neither conclusion is likely to be reached through the completion of the present test. Considering this, the indicated low risk of accepting Harvey components, as well as the vendor's capability and interest shown toward providing a high quality product at competitive costs, Harvey has been qualified to bid competitively on all future aluminum component orders except the KVN and KVE component models. These models are sufficiently critical to delay the use of Harvey components until additional quantities of less critical models are irradiated.

K Reactor E-N Demonstration Loading

In order to provide the target ("Y") elements for a 51-tube KE reactor E-N demonstration loading by the requested April 1, 1963 delivery date, arrangements are being made to modify available aluminum components and use Li-Al alloy stock in inventory to assemble the target pieces. Forty-seven (47) columns will be of the KVE self-supported geometry and four (4) columns of the KIVE geometry. Four target elements will be charged in each column.

Cap End Braze Problem - Overbore Fuel

The production of the CVINS overbore fuel has been delayed due to difficulties with cap end braze integrity, as detected by the UT-4 tester. An investigation of the process has revealed two problem areas. The component AlSi reservoir design is inadequate to supply molten AlSi to compensate for shrinkage during quenching. The 30-degree tilt at the load station of the automatic quench machine allows a major portion of the AlSi to drain from the assembly. The second problem is serious non-wet cans and cap spires.

The development tests indicated the use of manual quench to lessen the AlSi reservoir problem. Production of the CVIN fuel with manual quench was not satisfactory due to the can and cap non-wet conditions. The components do not wet when they are cleaned either in the 306 Pilot Plant or the 313 Building facilities. Investigations are currently under way.

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E-N Program

During the fabrication of enriched fuel elements for the second H reactor E-N load, one canning line (line 2) suffered a high reject rate for internal fractured bonding. Ninety (90) per cent of all rejects of this category were from line 2. The fabrication of the second E-N load began during the month and the same canning lines were used -- lines 1 and 2. Again, line 2 produced the majority of the internal fractured bond rejects. Process Engineering initiated a test on line 2, designed to determine if the mode of AlSi freezeout on the cold core upon entry into the furnace effected the mechanism of the internal fractured bonds. Ten (10) lead plugged cores were cycled, each paired with one normal core through the canning process on line 2. Total internal bond count data was obtained from the elements after the normal finishing operations. The average internal total count for the test elements was 734, as compared to 3.1 for the normal control elements. These data prove conclusively that the mode of AlSi freezeout on the core greatly affects the bond produced. It has been determined from development tests that the intermetallic bond thickness will be 2.5 to 3.5 times greater than normal if the uranium is heated to bath temperature in the lead layer prior to AlSi contact. It has been suspected that the AlSi can freeze near the ends of the cores and "plug" the bore. This is governed by the agitation of the core upon entry through the AlSi layer of the duplex bath.

An investigation revealed that the duplex agitator for line 1 agitated the fuel elements as the basket passed through the AlSi layer; line 2 agitator did not. This explains the reason that line 1 did not produce internal fractured bonds. The agitation through the AlSi assures that the bore fills solid with AlSi and forms the normal thickness bond. The line 2 agitator was adjusted to be identical with line 1, and the internal bond rejects were reduced to below that of line 1.

New Model Development - KVE

KVE canning cycle optimization was completed and is presently being evaluated. Preliminary evaluation indicates that normal "F" conditions will be optimum. A 400-piece production run for reactor charging in February was completed with a satisfactory yield.

New KVE can-sleeve assemblies show an average temperature of 545°C at the end of the preheat cycle (47 seconds). Used KVE sleeves show an increase in temperature at this point of nearly 10°C (554.8°C). A thermocouple was held in place in the base well of the cans during temperature recordings.

Hot Press

Radiometallurgy examination of the fuel elements from the irradiation test IP-490-A-FP has been completed. This test was made to compare Sylcor hot press, HAPO hot press, and HAPO AlSi bonded fuel elements. Examination of the HAPO hot press rupture did not reveal the mechanism of failure. As reported previously, the fuel element had two separate ruptures beneath the self-support rails. Examination of sixteen support welds on other fuel elements from this irradiation test has not revealed any damage to the clad bond from the welding operation.

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The 1,000 CIVE enriched (.94%) hot press fuel elements that are being fabricated by Sylcor under contract DDR 149 were not received at HAPO in January as anticipated. Delivery is now expected in February.

Fluid Pressure Bonding

A contract (DDR 141) was made in July, 1962 with the Batelle Memorial Institute to pressure bond 128 fuel assemblies. The 128 fuel assemblies were prepared at HAPO and shipped to BMI in January. The preparation of these fuel assemblies required the assembly and sealing of the assembly by welding to protect the nickel plated surface of the uranium fuel core.

Hot Die Sizing

Evaluation of the test results completed to date has revealed two process deficiencies. A more sophisticated control of the inert gas shield used in assembly is required to prevent oxidation of the nickel plate in the closure zone. The oxidation of the nickel plate is resulting in a more uniform diffusion condition. Secondly, the containment die principle used in the first end bonding test resulted in cladding upset and lateral bond cracking in the nickel-aluminum diffusion structure.

The improvements in the inert gas purging of the fuel assembly have already been introduced to the process. The end bonding press has been converted to the differential expansion die concept. Initial tests using the differential expansion concept for end bonding have eliminated the metal upset problem. Sufficient dies are now being fabricated to conduct a variables test.

Evaluation of the first end bonding test is continuing. The following additional conclusions have been reached since the last report:

- 1) Lateral surface bond strength improved close to the ends, deteriorated at a point 1-1/2 inches from the ends. Each remained the same at the center for end bonded fuel elements. A summary of these results is contained in the following table.

Lateral Bond Strength*

<u>Location</u>	<u>3/4" From Cap</u>	<u>1-1/2" From Cap</u>	<u>Center</u>	<u>1-1/2" From Base</u>	<u>3/4" From Base</u>
Test 1* Preheated	740	585	931	570	934
Test 2 No Preheat	778	460	901	450	909
Controls No End Bonding	296	862	907	874	531

*Bond strength expressed in pounds force to pull a 0.190 inch diameter stud.

**Test 1 fuel assemblies were preheated in an auxiliary furnace for five minutes @ 680 C.

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- 2) Cracking within the nickel-aluminum diffusion layer was found to be the reason for the lateral bond deterioration near the containment dies. The cracking appeared to be more prevalent in pieces subjected to the highest temperatures. Statistical analysis of the first attempt to set up a grading system for the degree of lateral bond cracking verified the conclusion reached from optical examination.
 - a) An interaction between press time and temperature was apparent with combinations of high temperature - low time and low temperature - high time giving the least amount of cracking for preheated pieces.
 - b) For non-preheated pieces both time and time-pressure interaction were significant process variables affecting lateral bond cracking.

The above testing results have led to the conclusion that the lateral bond cracking problem, although manifesting itself as a time-temperature interaction, is in reality a stress condition caused by the degree of the interference of fit between the die and the fuel element during end bonding.

- 3) Statistical analysis of graded closure integrity after end bonding indicates that process conditions significantly influence the closure between the aluminum jacket and cap. The pressure and temperature of the end bonding press are the dominant variables with an interaction between pressure and temperature.

Analyses of the thermal heating data taken during sizing have proven very valuable. This information was taken at the interface of the nickel-plated core and aluminum jacket by the use of imbedded thermocouples. The significant temperatures from these data are those at the interface just after point passes through the die. At that time in the process the aluminum-nickel diffusion is initiated. The temperature measured varied from a minimum of 325°C to a maximum of 490°C. At various preheat conditions the temperature at the cap and base were the same while the center was slightly higher. This would indicate that the weak bonds on the cap end are not necessarily a function of temperature at present operating conditions, but more one of oxides and the absence of relative metal motion to break up these oxides.

Electroplating Nickel on Uranium

The major effort consisted of repair and replacement of equipment and installation of improved process monitoring facilities. The plating tank, which developed a serious leak during December, was replaced. The new tank is designed for improved solution maintenance, with both a bottom outlet and overflow weir discharging to the suction side of the filter pump. During plating, continuous filtration is employed; thus heavy particulate matter is continuously removed from the bottom of the tank, and light matter and organics from the surface of the solutions as it overflows the weir.

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Fabrication of the new prototype plating rack was completed, and testing initiated. At present eight cores have been plated and tested. The plate porosity has been confined to the base end contacts and these appear only intermittently. It is believed this problem can be corrected by minor modifications of the contact edges.

Installation of the demineralized water equipment has been completed. This central facility will provide demineralized water to the present plating line and any planned future installation.

KVN Self-Supports

During this period the KVN support has been redesigned increasing the as-fabricated height to a nominal 99 mils. Six hundred (600) KVN fuel elements are being held prior to rail welding until sufficient vendor manufactured rails arrive on the plant, and will be used to prove rail strength capabilities of the new design. A small quantity of the KVNS, Revision #3 rail was fabricated on site, and current testing indicates 95% confidence for sizing down.

A gauge has been designed to check rails quickly for minimum height on self-supported fuel elements. Fabrication of two gauges is scheduled for completion at the end of January. Electrical components may not be received by these dates. However, the gauges can be used with dial indicators.

A tool has been designed and fabricated to raise depressed rails to acceptable heights. Initial testing of the tool was favorable on KVN fuel elements. Loading capabilities of the raised rails is being determined at this time. This work was scheduled for completion on January 24, 1963. An adapter has been designed and fabricated to utilize a rail shear tester for sizing KVN self-supported rails after radiography.

KVE Supports

The KVE support rail die has been fabricated, and initial testing indicates satisfactory performance. Limited data indicates that the strength is generally equivalent to the three-dimple KVNS bridge support. Further testing on sizing dimensions is required. The H-3 engineering specification drawing of the KVE support has been prepared and approximately 4,000 KVE supports are on hand for further evaluation.

CV Support Problems

Difficulty of supports below minimum height has been experienced with CV and CVI supports. The engineering specification drawings of these supports have been revised to give increased support and dimple height. Orders have been placed on site for 50,000 CV supports and 16,000 CVI supports, incorporating these changes.

These changes were made due to low yield caused by supports being below the dimensions as specified in the process specifications. The changes will require 100% sizing of all self-support model fuels in order to get the circumscribed circle specification.

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Automatic Lathe Loader

The first production models of the automatic lathe loader and unloader stamper were installed and moved to #3 position in mid-January. The prototype loader in the #2 position is still in service pending completion of another stamper and scheduling the replacement of the loader.

Canning Furnace Pneumatic Vibrators

Canning line #4 canning furnace vibrators were equipped with an air supply lubricating system to maintain vibration frequency in accordance with QC standards. This system was developed in the Pilot Plant and is being proved on canning line #4. A 14-day trial run is under way, and based on results which at this point are very satisfactory, the remaining canning lines will be converted to this system. Work continues on an electrical vibrator to eliminate the possibility of atmosphere contamination.

QUALITY CONTROL

Yield Control

	<u>December Yield</u>	<u>January Yield</u>
8-inch I&E normal	90.0	88.7
6-inch I&E enriched	90.1	90.8
8-inch I&E bumpered	96.7	71.0
6-inch I&E bumpered	84.1	81.3

The monthly yield is down 1.3% this month on 8-inch material due to an increase in component preparation, pinhole, and braze layer contamination rejects. The 6-inch unbumpered yield is up due to the use of Diversey PX-1288 as an AlSi sloop preventative.

The 8-inch bumpered yield reflects the low yield of the KVN material through the rail welders. The 96.75 yield shown for December includes one lot of reprocess material.

The 6-inch bumper yield is low due to the more restrictive UE-1 and UT-4 tester settings that were employed during the production of E-N load material, to an increase in surface rejects at Final Inspection due to stains, and to an increase of approximately 7.9 per cent over the December figure for rail weld rejects.

Uranium Quality

During the routine uranium quality control evaluation of uranium cores in the Laboratory, a sample from a 6-inch core from Lot GY-004 was observed to have an unusual film on the I.D. which was not attacked by the concentrated hydrochloric acid used as an etchant. The GY lots are vacuum outgassed, followed by machining of the O.D. of the pieces. The I.D. is in the finished stage before vacuum outgassing.

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Examination of a cross section of this sample revealed a layer of film approximately 1.5 mils thick which was identified by x-ray diffraction as uranium dioxide. Another section of this core was processed through the normal pickle cycle to see if this treatment would remove the film. Examination of the pickled sample revealed that all of the UO_2 film had not been removed.

Other cores with this film on the I.D. have not been found. Bond count has indicated good internal bonding in the vacuum outgassed material.

Engineering has had production of this material discontinued at the feed site. Evaluation of ultimate quality will be implemented by an Engineering Production Test and test material to be evaluated on the basis of fuel quality characteristics and on irradiation performance.

OIVT Lithium-Aluminum Quality

Inspection of one shipment of lithium-aluminum OIVT rod stock has been completed according to PWR 120 (1962). The inspection showed that the internal diameter is too small and the outside diameter is too large. Visual inspection indicated that voids, inclusions, cracks, and pits will be the primary cause of reduced yields, as has been the experience in the past.

Test Reactor Operations

Routine fuel element production testing backlog at the Test Reactor continues at approximately seven days of reactor time. Presently, four hours per week are being devoted to spline testing which is also reducing the backlog of material.

Approximately 25 per cent of the reactor operating time was made available and utilized for special testing purposes. These were as follows:

- 1) SWR-727 - Measured dih of 212 spline samples for Facilities Engineering Operation, IPD.
- 2) SWR-728 - Measured dih of two bars of "F" size graphite for Tokai Electrode Manufacturing Co., Ltd. of Japan.
- 3) SWR-729 - Measured dih of 160 samples for Facilities Engineering Operation, IPD.

The completion of the fourth phase of the Hanford Test Reactor training program on Reactor Instrumentation has been delayed temporarily in order to eliminate the backlog of uranium and spline samples.

There was no reactor down time as a result of equipment failure or for maintenance.

Audit of Inspection Gauges

Concurrent with the gauge audit program, a study has been made of the Receiving Inspection tools. The purpose of this program is to update and improve currently used gauges, to provide new gauges needed, to institute a file of

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custom gauge drawings, and to check tolerances being used against drawing requirements. Talks with Receiving inspectors brought a number of deficiencies to light requiring redesign of several gauges and standards and correction of incorrect tolerances being used on a comparator template.

All currently known shortcomings have been either corrected or changes are being implemented (i.e., improved designs are in the fabrication stage) with the exception of an upset metal measuring device which has not yet been designed pending determination of the most appropriate approach.

Self-Support Fuel Elements

An evaluation has been made of the capability for producing self-support fuel elements within the specified dimensional tolerances. On the KVN model, the total variation in fuel O.D. and support height following the welding and sizing operations was found to be such that the capability limits of the process exceed the minimum support height and maximum circumscribed circle diameter specifications. The standard deviation in support heights due to inherent variation alone has been estimated as 0.000838 inches following the sizing operation at the weld station. Providing this can be maintained and with the support heights adjusted to provide 99 per cent confidence of 99 per cent conformance to the minimum height specification, the average circumscribed circle diameter would be about 1.710 inches; the maximum circumscribed circle diameter would be 1.716 inches (99 per cent conformance - 99 per cent probability). The capability limits can be brought within specification limits by reducing the support height tolerance on the finished fuel element to 0.0013 inches or less.

Deformation of the supports due to handling, and possibly autoclaving, has been the largest source of height variation. Changes in average heights of as much as 0.004 inches between welding and palletizing have occurred; this damage has been observed primarily on the cap end supports.

The present method of reducing support height variation is that of sizing. Initially, this was done after welding; however, since additional variation is introduced in handling and possibly autoclaving, this stage of the process was moved to Final Inspection. The sizing operation must be improved in order to provide the required uniformity in heights. The lack of uniformity is partially attributed to the elastic properties of the nylon sizing rollers and Fuels Engineering is in the process of replacing these with steel rollers.

Capability data indicate that an increase in a fabricated support height of 0.0025 inches or more is needed to provide adequate assurance of the supports on the finished fuel being above 0.092 inches. Preliminary test results indicate that increasing the support and dimple height to a nominal 0.10 inch and 0.093 inches, respectively, will provide adequate assurance of the supports on the finished fuel being above 0.092 inches; however, most of these will fail the maximum circumscribed circle diameter specification. Additional development work is needed in obtaining a method of sizing which will produce the tolerance needed on the finished support height.

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PLANT FACILITIES

Status of Active Construction Projects

CAF-954 - High Pressure Autoclave Development Facility

Issuance of the design criteria for Commission approval has been rescheduled for February due to delays in the review by NRD.

CAF-961 - Consolidated 303 Area Service Facility - Phase I

A directive has been issued authorizing interim funds of \$66,000 to the AEC for Title I and Title II services. A Work Authority authorizing \$10,000 to the General Electric Company was issued January 17, 1963, for the preparation of design criteria and related services. Work is under way in preparing the criteria.

CAF-973 - Steam Distribution Line Modifications - 300 Area

A Work Authority has been received for interim engineering work relative to design criteria preparation and other related General Electric engineering services. Preparation of the design criteria is under way and is scheduled for issuance to the AEC for their approval early in February.

CGF-979 - Pilot Scale Plating Equipment - 300 Area

Further project activity has been held in abeyance pending further AEC fund authorization. The design criteria prepared for this project received special commendation from the Commission.

CAF-980 - Boiler Replacement for Additional Steam Generating Capacity - 384 Building - 300 Area

Answers to the questions posed by the AEC in their official letter of transmittal returning the original project proposal, along with the revised project proposal, should be ready for transmittal by February 1, 1963. Funds will be available from the FY 1964 PAC Budget.

Drawings Produced

	<u>January</u>	<u>December</u>
New	62	44
Revisions	93	109
Small Charts	17	4
Large Charts	9	5
Miscellaneous	44	30

Duplicating Production

	<u>January</u>	<u>December</u>
914 Copier Masters Prepared	548	348
914 Copier Copies	16,285	14,040
Multilith Impressions	300,523	252,326
Ozalid Copies	9,613	6,439
Number of Orders Processed	1,297	1,120

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Utilities Statistics

	<u>January</u>	<u>December</u>
Average Steam Generated (M lbs/hr)	106	84
Maximum Steam Generated (M lbs/hr)	174	128
Total Steam Generated (M lbs)	78,496	60,323
Total Condensate Return (M lbs)	33,366	24,856
Coal Consumed (Tons)	3,730.7	2,902.3
Evaporation Rate	10.52	10.39
Efficiency - Actual	71.2%	69.9%
Efficiency - Optimum	74.8%	74.0%

No. of Boilers on: 3 4 3 2 3 4 5 6 4 3 4 3
26 27 30 8 9 10 11 13 16 17 22

Sanitary Water From 3000 Area (M Gals)	129.6	126.5
Peak Water Flow 3000 Area (gpm)	5,100	4,900
Average Water Flow per Month (gpm)	2,903	2,928
Peak Water Consumption for 24 Hours (M Gals)	5.0	4.5
Compressed Air Production (scfm)	53,549	45,339

Facilities and Equipment Experience

During the past period of high steam production it was noted that a serious piping restriction in the soft water piping to the deaerator limited pumping capacity to the extent that it is questionable whether or not adequate boiler feedwater would be available if the condensate pumps were not in service. This points out the fact that the operation of the boilers is relying more and more on the condensate return system.

Overhaul of the No. 2 air compressor has been delayed due to a shipping error by the manufacturer. The replacement piston and rings were not the size specified. It is anticipated that this unit will be back in service by the end of the month. Because of this delay, the completion schedule for installation of the new No. 6 compressor has been revised to February 28, 1963.

Installation of the new motor control center in the 384 Building is under way and is approximately 20 per cent complete. Completion is scheduled for March 30, 1963. This completion date is somewhat dependent upon the steam loads during the month of March.

Craft Training

A total of 208 man hours of training was given during the period December 20, 1962 through January 20, 1963. Two classes were conducted on the Flux Monitor manual for PRTR craftsmen and engineers. In response to a request for training needs of HL, NRD and IPD during CY 1963, all departments have indicated about the same level of service as was provided in 1962. A specific request to repeat a course on refrigeration conducted for the pipefitter craft has been received. It will be necessary to temporarily upgrade a craftsman to serve as instructor. This course is planned to start in February. It is also planned to have a representative of the Foxboro Company conduct classes on Foxboro instruments in service in the 300 Area.

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Cost Study

A study was conducted to develop a cost report for the Plant Facilities Operation that would more appropriately show expenditures as related to responsibilities. The portion of this operation cost to Production Fuels in terms of percentage, dollars and cents per fuel element was developed to relate performance to production and better understand the impact of charges as related to Production Fuels Cost.

Manager, Production Fuels

WN Mobley:PGR:mfj

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RESEARCH AND ENGINEERING

PROCESS AND REACTOR DEVELOPMENT

REACTOR FUELS

Projection Fuels

The two columns of KVNS elements charged in KW Reactor on 12-3-62 have accumulated an average tube exposure of 400 MWD/T. Goal exposure for these columns is 800 MWD/T.

REACTOR ENGINEERING

Zircaloy-2 Process Tubes

Approximately 270 tubes have been received; of these the first 100 tubes were inspected on site for dimensions, surface finish, and wall thickness. Of the remaining tubes one out of five were borescoped and vidigaged for wall thickness and all were checked for O.D. surfaces and dimensions. Of the total tubes inspected 18 have been rejected.

Channel 3586-D

Longitudinal cracks found at the rear end of the Zircaloy-2 ribbed tube in channel 3586-D were successfully welded on 1-11-63. The rear face hardware was modified such that the tube now has a welded collar in place of the former packing gland closure. This tube was installed on 7-7-55 and has performed satisfactorily except for rear face water leaks. Examinations of the tube indicate no deterioration in integrity.

KE Reactor Nozzle Failure

On 1-12-63 a tube inlet nozzle failure at the lug-ring snap-ring groove occurred and the reactor was automatically scrambled by the Panellit gage monitoring tube flow. Cooling water flow was from rear to front following the failure and was adequate to prevent fuel element burn-out. However, the reactor was at little more than half the operating power level when the incident occurred.

Corrosion

With the startup of F Reactor on 1-16-63, Production Test IP-549-A, "Half-Plant Low Alum Feed Water Treatment at F Reactor", (HW-75541) was initiated. The process tubes on the far side of the reactor will receive process water treated with sufficient alum to produce reactor quality water but the feed will not be lower than 6 ppm alum. The near side tubes will receive process water treated with alum at a feed rate of 18 ppm. The first group of 20 fuel charges, which will establish pretest conditions, have been discharged and examined. Visual examination of these elements provided the following results: (1) 26 per cent of the fuel elements exhibited localized corrosion

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STATUS REPORT OF PRODUCTION TESTS

Test No.	Type Metal	Tubes	Reactor	Goal Exposure	Current Exposure	Remarks
IP-216-A	Normal prod. natural & enriched fuel elements	65	All	Normal variable goal		Provides for monitoring the performance of a sample of all normal production material to assist in development of a Quality Index for production fuel. Test is continuous.
IP-272-A	I&E self-supported fuel	69	C	Variable +200		Provides for testing of CIVN geometry fuel in C Reactor ribless zirconium tubes.
IP-381-A & IP-431-A	I&E self-supported fuel overbore size	62	C	800 MWD/T		Provide preliminary qualitative data regarding irradiation behavior of larger fuel element designs.
IP-409-A	I&E self-supported fuel	2	KW	800 MWD/T	400 MWD/T	Provide qualitative data to confirm the fuel geometry design. Two columns discharged 12-3-63, short of attainment of goal exposure in test of new charging machine.
IP-423-A	Oil-quenched natural uranium I&E fuel elements	150	F	Variable		Provide a comparison of dimensional stability of standard water-quenched uranium fuel cores and oil-quenched cores

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as groove and ledge attack, and (2) about 9 per cent of the fuel pieces exhibited severe corrosion attack with 1 per cent of the total showing groove penetration to the AlSi bonding layer.

REACTOR PHYSICS

E-N Conversion Ratio

Plutonium yield measurements from 18 batches which are representative of the E-N load were correlated with previous data and adjusted to expected values in a reactor operated routinely with this type of loading. The data agree with calculated predictions that the plutonium formation is a function of loading radius and is lower in the fringe. Under the actual exposure and loading conditions a net gain of 13.5 per cent in conversion efficiency is indicated relative to a natural uranium load with 650 MWD/T goal exposure.

Uranium Recycle

The per cent of recycle fuel in the enriched fuel present in the reactors at the end of December and discharged during December are given below:

	Per Cent of Recycle Fuel in Enriched Fuel							
	<u>B</u>	<u>C</u>	<u>D</u>	<u>DR</u>	<u>F</u>	<u>H</u>	<u>KW</u>	<u>KE</u>
December 31 Status	100	100	100	99	100	99	100	100
December Discharge	100	100	100	--	100	89	100	100

Spline Testing and Zircaloy-2 Tube, K Reactor Physics Measurements

Preparation was started for reactivity measurement of splines which have been irradiated to various exposures. Specifications were prepared of the needs of zirconium tubes to be used in the K Reactor lattice mockup in the Physical Constants Test Reactor. The PCTR measurements are designed to determine the wet and dry k_{00} for the zirconium-tubed K Reactor with KV geometry fuel and also complete the spline burnout measurements.

Reactor Kinetics

The debugging of the IBM program CLUMSY is continuing and so far the level trip and the power rate trip which are based on outlet temperature have been checked out.

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RADIOLOGICAL ENGINEERINGRadiation Control Experience

The following table summarizes the 1962 radiation exposure experience for critical IPD classifications:

<u>Classification</u>	<u>Total Dose</u>	<u>No. of Employees</u>	<u>Average Dose/Employee</u>	<u>Maximum Dose</u>
Radiation Monitors	222352 mr	81	2745 mr	3056 mr
Processing Operators	624987	270	2315	3058
Pipefitters	295711	122	2424	3011
Millwrights	225298	103	2187	2997

Lapse of Radiation Control

Lapse of Radiation Control
Distribution by Reactor and Component

	<u>B</u>	<u>C</u>	<u>D</u>	<u>DR</u>	<u>F</u>	<u>H</u>	<u>KE</u>	<u>KW</u>	<u>IPD Totals</u>
Processing	0	1*	0	0	1	2*	0	0	4
Maintenance	0	0	0	0	1	1	0	0	2
Suppl. Crews	0	0	0	0	0	0	0	0	0
Applied Reactor Engr.	0	0	0	0	2	1	0	0	3
Research and Engineering	0	0	0	0	0	1	0	0	1
Facilities Engineering	0	0	0	0	0	0	0	0	0
Central Maintenance	0	0	0	0	0	1	0	0	1
Reactor Areas	0	1	0	0	0	0	0	0	1
Assigned Totals	0	1	0	0	3	4	0	0	8
IPD General		1							1

* One lapse involved Central Maintenance, Applied Reactor Engineering - SOS, H Processing, and C Processing. Because the problem originated at 105-H, it is reported in the column of H statistics.

Vertical columns do not add up to the indicated totals because in some cases a Lapse of Control may be chargeable to more than one component.

Effluent Activity Data

The table below shows average reported concentrations of selected radionuclides from reactor effluent samples taken during December, 1962. All units are 10^{-12} curies/ml.

<u>Reactor</u>	<u>P32</u>	<u>As76</u>	<u>Zn65</u>	<u>Np239</u>	<u>Cr51</u>
B	4.4	51	22	80	430
C	4.6	23	11	70	210
D	6.1	61	32	130	510
DR	4.2	80	22	110	460
F	7.6	45	34	185	790
H	6.7	50	9	90	340
KE	10.9	40	30	80	420
KW	6.8	32	11	40	320

All reactor water plants were at a nominal feed rate of 18 ppm.

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PROCESS TECHNOLOGY OPERATION

REACTOR POWER LEVEL LIMITATIONS

During this report period, the primary limits to reactor power levels for B, C, DR, and F Reactors has been tube outlet temperature limits to control tube corrosion. D and H Reactors' power levels have been restricted by bulk outlet temperature limits. Power levels at the K Reactors are currently restricted by the AEC administrative limits.

PROCESS STANDARDS

HW-46000 B, Process Standards - Reactor

Three revised standards were issued during the month. These were:

Process Standard A-030 - "Pressure-Flow During Reactor Shutdown"

Changes made in this revision to the standard are as follows:

1. Minimum flow requirements to the reactor were lowered.
2. Initial flow reductions to crossheaders must now be made according to ΔP gauges in conjunction with the ΔP tables.
3. Crossheader throttling valves are now required to be opened upon failure of ΔP gauge during throttling operation.
4. A setting was specified for the ΔP gauge low flow alarm.
5. The ΔP table was expanded to include higher tube powers.
6. Anticipated pressure drop values were given for each process tube cap removed.
7. A rear riser water level was specified for use when on the crossunder line.
8. ΔP gauge settings for the bottom three crossheaders were specified for use while on crossunder line.
9. References to the Process Guides were deleted. Basic considerations attending a ruptured tube were included in this revision.

Process Standard C-110 - "Ball 3X System"

Specifications on the frequency and method for functionally testing components of the ball 3X system were revised.

Process Standard F-010 - "Process Material"

The standard contains specifications on fuel elements, dummies, poison

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materials, and chemicals, and authorizes use of certain of these materials in reactor. The standard was completely rewritten. A new chemical decontaminating agent was added, and a previously used agent was deleted. Print numbers on other materials were updated.

HW-46000 F and HW-46000 H, Process Standards - Reactor

One revised standard was issued for each of these manuals during the month. This was:

Process Standard F-010 - "Process Material"

Revisions to these standards are identical to those for HW-46000 B, above.

HW-41000, Process Equipment Standards - All Reactors

Four Process Equipment Standards were issued during the month. These were:

Equipment Standard A-010 - "Process Water System"

High tank flushing requirements were updated and a high tank screen cleaning schedule was specified.

Equipment Standard D-010 - "Properties of Horizontal Rods"

Serviceability requirements for the horizontal rods and a minimum number of horizontal rods required to be serviceable prior to reactor startup were specified in this revision.

Equipment Standard D-030 - "Process Tube Replacement and Stuck Charge Removal"

A new section limiting the diameter of process tube channel broaches and cutting tools was added to the standard.

Equipment Standard D-060 - "Reactor Confinement"

The standard specification requiring emergency fans to be operated for a four-hour period when being tested was revised to reduce the test period to one hour.

PROCESS CHANGE AUTHORIZATIONS

The number of process change authorizations issued this report period were: 17 to permit temporary deviation from Process Standards - Reactor, HW-46000; three to permit temporary deviation from Process Standards - Water Plant, HW-27155 Rev1; and one to permit temporary deviation from Process Equipment Standards, HW-41000. These were:

PCA #2-152 - "Continuous Cooling of Fuel Charges, C Reactor"

The process change authorized cooling of tube charges by back flow from the rear headers to facilitate replacement of operational charge-discharge

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facility hardware with standard C-Reactor front-face hardware.

The reactor had been shut down five days. Cooling by reverse flow was limited to one tube at a time. Visual flow monitoring and capability to restore front-to-rear flow was required in the process change. The process change expired on December 6, 1962.

PCA #2-153 - "Removal of High Sensitivity Seismoscope, B Reactor"

To provide a seismoscope starter for use in developing revised seismoscope circuits, the process change authorized removal of the high sensitivity seismic starter from B Reactor. The high sensitivity starter is not a safety circuit instrument, but is used to provide a low level seismic activity alarm. The process change specified that C-Reactor personnel notify B-Reactor personnel of any low level seismic activity detected on the C Reactor high sensitivity seismic starter. The PCA will expire on March 29, 1963, or on return of the high sensitivity starter to B, whichever is sooner.

PCA #2-154 - "Bottom Thermal Shield Backup, KW Reactor"

The process change authorized use of service water coolant to the bottom thermal shield in place of process effluent coolant with service water backup, provided the low pressure crosstie between the two K Reactors' service water systems is maintained in service. The PCA expires on February 6, 1963.

PCA #2-155 - "Top Thermal Shield Temperatures, K Reactors"

The process change authorized an increase of 50 C in the high temperature limit for the top thermal shield as read by thermocouples TTS-8 and TTS-9. The change was contingent upon maintaining top thermal shield temperatures as read by TTS-6, TTS-7, and TTS-13, 40 C below the normal limit specified in Section 70.01-2.a of Standard A-070. The specification was designed to maintain thermal shield temperatures below the temperatures which would cause damage to the thermal shield. An immediate reactor shutdown was required on exceeding the temperatures specified in the PCA.

PCA #2-156 - "Last-Ditch System Backup, K Reactors"

The process change authorized removal of the #2 diesel pump from service for repair for approximately five hours without operating the remaining two diesels as required by Process Standards A-080 and 190-B-090. The process change specified that personnel were to be stationed near the other two diesels for the purpose of starting them manually if they failed to start automatically on demand.

PCA #2-157 - "VSR Withdrawal, C Reactor"

Design Change #614 was approved to provide an interlock which prevents HCR removal until VSR's are withdrawn. The design change was recently completed at C Reactor.

It was economically attractive to test the revised system prior to startup

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and prior to raising water pressure. The process change provided instructions for safe removal of VSR's during a reactor outage. The PCA expired on December 20, 1962.

PCA #2-158 - "Last-Ditch Backup, K Reactors"

The process change authorized removal of the K Reactors' last-ditch diesel pumping system from service for a 96-hour period with one K Reactor in shut-down status. Specific alternate backup systems were delineated in the process change. Although PCA #2-158 was issued, all pertinent portions of PCA #2-158 were subsequently authorized in PT-IP-559-I, HW-76064, "Authorization for Operation of a K Reactor Without Last-Ditch Backup," issued January 2, 1963.

PCA #2-159 - "Water Leak, B Reactor"

B Reactor was authorized to continue operation with a water collection rate in excess of the rate permitted by standards. Specifications designed to prevent moderator damage were stated in the PCA which expired on December 23, 1962.

PCA #2-160 - "Flux Monitor Range Switch Settings, B Reactor"

Following a water leak, the amplifiers on two flux monitors read at equilibrium below the value specified in Process Standard C-030. The process change provided instructions and alternate specifications for operation of the flux-monitor system during the period required for the amplifier reading to increase to the value specified in standards. The process change expired on December 26, 1962.

PCA #-161 - "Storage of Enriched Slugs, H Reactor"

The process change authorized a storage arrangement for enriched metal charges not presently covered by standards. The purpose of the change was to provide for more efficient use of storage space. Specifications necessary to preclude accidental storage of enriched uranium which could become critical on water flooding were included in the process change.

PCA #2-162 - "V-73 Valve, K Reactors"

The process change authorized deletion from Standard A-022, Section 22.01, of all references to the V-73 valve. Following completion of Project CG-844, the specifications referring to the V-73 valve were no longer applicable.

PCA #2-163 - "Water Shutoff Times, B, D, F, and K Reactors"

The purpose of this process change was to establish that water shutoff times for cored or overbored process tube channels or zirconium process tubes specified in Standard A-030 are not applicable and to authorize the calculation of water shutoff times for such channels by Process Technology Operation personnel when required.

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PCA #2-164 - "C Work Platform Control Circuit, B Reactor"

The process change authorized use of the front work platform at B Reactor during reactor operation with one of the two photocell installations out of service. The photocells are designed to prevent or stop movement of the work platform if obstructions are present between the platform and front-face components. Specifications and procedural controls were listed in the PCA designed to provide acceptable alternate protection against component damage. The process change expired on December 28, 1962.

PCA #2-165 - "Last-Ditch System Backup, K Reactors"

On December 29, 1962, it was found necessary to remove the #3 diesel pumping unit from service to adjust the governor control. The standards require all three diesels to be on automatic start status or alternately one may be out of service if the remaining two are started and operated. It has been found that operation of two diesels in a non-pumping status results in heating of the last-ditch coolant supply. It has been deemed more acceptable, on a short-term basis, to provide personnel to stand by and start the two remaining diesels manually in case of need and in case the automatic start feature fails. The process change authorized removal of #3 diesel from service for an eight and one-half hour period for adjustment of the governor control without operation of the remaining two diesels on the bases discussed above.

PCA #3-1 - "Slow VSR's, DR Reactor"

On December 29, 1962, at DR Reactor the drop speed of several VSR's were slower than specified in process standards. The process change authorized reactor startup following renovation of the VSR channels. Power level vs. graphite temperature curves and power level vs. metal exposure curves, both based on a 2.6 second VSR drop time, were supplied and specified for use during the startup and operating period. The PCA expiration date was specified as the next minimum outage.

PCA #3-2 - "Thermal Shield Coolant Temperature Monitoring, DR Reactor"

The process change authorized temporary alternate means for monitoring the temperature of the thermal shield outlet coolant following failure of one of the two temperature sensors. The alternate means were based on specifying minimum thermal shield coolant flow and fringe zone tube powers with some dependence placed on the biological shield temperature monitoring sensors.

PCA #3-3 - "Water Leak, C Reactor"

Following reactor startup on January 4, 1963, and during the next seven days of reactor operation, the water collection rate at C did not drop to within the standards limit of 24 gallons per day. From the data, it was deemed uncertain that a new water leak existed. Authorization was given to operate four additional days provided the water collection rate did not exceed 65 gallons per day.

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PCA #3-4 - "Last-Ditch Backup, K Reactors"

The process change authorized having one diesel pumping unit out of service without operation of the two remaining diesel units as required by standards. The basis and justification for this process change is essentially identical to that reported in PCA #2-156 and 2-165.

PCA #3-5 - "Graphite Temperature Stringer Requirements, H Reactor"

Process Standard C-040 was revised to require seven graphite thermocouple stringers instead of five. The process change authorized a delay in meeting the more restrictive requirements of Process Standard C-040 to permit time for procurement, fabrication, and installation of the additional stringers.

PCA #3-6 - "Slow VSR's, DR Reactor"

The process change authorized operation at DR Reactor with VSR insertion speeds slower than specified in process standards. The process change included authorization to install two flexible BDF-size VSR's, a semi-flexible DR-size VSR, and a rigid BDF-size rod. Based on the VSR drop time following renovation work, power level limit curves as a function of graphite temperature and metal exposure were supplied and authorized for use.

REACTOR AUDITS AND INSPECTION

One engineer audited conformance to process standards on all Processing Operations' shifts by making 15 inspections at each reactor during the report period.

RUPTURE EXPERIENCE

<u>Failure Date</u>	<u>Tube Number</u>	<u>Lot Number</u>	<u>Type of Material</u>	<u>Tube Power at Failure (kw)</u>	<u>Exposure MWD/T</u>	<u>Type of Failure</u>
1/2/63	3786-D	KZ-473-D	I&E NB	1177	672	Side-Other
1/6/63	1180-H	KY-246-D	I&E EB (0.94%)	1138	837	Side-Other
1/9/63	1975-H	KZ-308-D	I&E NB	1101	605	Side-Other
1/14/63	1987-DR	KY-239-D	I&E EB (0.94%)	1058	826	Hole
1/17/63	1167-D	KZ-450-D	I&E NB	1329	840	Hole
1/17/63	4075-D	KZ-450-D	I&E NB	932	794	Hole
1/28/63	1468-DR	KZ-443-D	I&E NB	1102	724	Hole
1/29/63	1362-F	KZ-092-A	I&E N	1263	569	Side
1/30/62	2473-DR	Unknown ⁽¹⁾	I&E NB	1013	743	Hot-Spot Split-Long.

(1) Six-inch internally and externally cooled natural uranium fuel element with attached water-mixer spool. Unable to obtain lot number of mixer pieces.

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Legend.

I&E NB - This is the symbol for internally and externally cooled production reactor fuel elements of natural uranium which have projections (bumpers) welded to the fuel element jacket.

I&E EB - This is the symbol for internally and externally cooled production reactor fuel elements which have projections (bumpers) welded to the fuel element jackets. The uranium cores are enriched in U-235 and the weight per cent U-235 in the core material is stated.

I&E N - This is the symbol for internally and externally cooled production reactor fuel elements of natural uranium.

Side-Other - Failure probably caused by corrosion or water penetration of the external can wall or other unknown mechanism.

Hole - Failure on the internal surface of an I&E piece probably caused by water penetration through a weld or other unknown mechanism.

Split-Long. - Failure probably caused by uranium cleavage along the axis of the core.

Side Hot-Spot - Failure probably caused by accelerated corrosion of the external can wall in a localized region of high temperature.

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OPERATIONAL PHYSICS OPERATION

FILE PHYSICS ASSISTANCE

As measured by over-all flattening efficiency and nonequilibrium losses, operating efficiency during this report period was near average. Administrative limits based on the average of the ten hottest tubes provided continuing flattening incentive at five of the reactors.

The shearing-off of a nozzle by elevator operation at the KE Reactor resulted in a close examination of risks associated with the procedural aspects of supplementary control use, and the imposition of more restrictive procedures.

The special corrections imposed on DR Reactor total control procedures to compensate for the 1961 Ball 3X drop have been rescinded, as boron burnup has largely overcome the poisoning effect of the residual balls in the stack.

SUMMARY OF OPERATING DATA OF PHYSICS INTEREST
FOR THE MONTH OF JANUARY, 1963

Reactor	B	C	D	DR	F	H	KE	KW
ECT In January (1)	1505	1640	1495	1490	1485	1485	2420	2510
12-Month Average ECT	1495	1665	1485	1490	1480	1520	2510	2540
Equil. Scram Time (2)	23-27	18-22	18-22	18-22	15-18	14-16	20-25*	25-30*
Recording Period -								
From:	12-18	12-23	12-18	12-17	12-18	12-17	12-17	12-19
To:	1-23	1-23	1-22	1-21	1-22	1-23	1-21	1-21

*Equilibrium scram recoveries are not attempted at the K Reactors.

- (1) Effective Central Tubes: This value is defined as pile power level divided by the average power of the ten most productive tubes in the reactor.
- (2) This is defined as the maximum time which may elapse between scram and first indication while still permitting a successful scram recovery.

B Reactor - J. R. Fredsall

Operating efficiency has been exceptionally high at B Reactor during this period. The long runs, with accompanying high average exposure, have led to better than average ECT. Power level is restricted by an administrative limit on the ten hottest tubes.

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C Reactor - D. W. Constable

Due to a 28-day outage (which terminated 12-28-62) plus subsequent water-leak problems, operating efficiency was low during the report period. During the extended outage, 300 process tubes were replaced; however, an HCR change indicated in last month's report to be planned during the outage (number five half-rod to be replaced with a full rod) was postponed. A computer code for routine startup transient calculation has been completed and tested; results look promising. Power level is limited by an administrative limit on the ten hottest tubes.

D Reactor - D. G. Albertson

During the report period, maximum level was at first restricted by administrative tube limits, but later by a bulk outlet temperature limit of 93.5°. Flattening efficiency was good, although continuity was marred by several rupture-induced shutdowns. One of these had particular physics import, in that apparently a tube leak resulted from a severely cocked rupture, releasing an estimated 300 c-mk (reactivity effect) of water into the stack; operation was unstable during the drying-out process. Despite the high preponderance of nonequilibrium operation, efficient equilibrium operation has been attained, attributed at least in part to a more optimum E-ring arrangement recently completed.

DR Reactor - S. M. Skidmore

A 30-day operating run (interrupted by one hot startup), terminating on 12-31-62, highlighted DR operation during the first part of the report period; the scheduled shutdown was due to BPA peak load requirements. During the outage, several VSR channels were broached (for speed of control reasons), and PT IP-535-C for measuring relative strengths, normal DR VSR versus B-size VSR, was run. There are presently four B-size VSR's in the system. Residual ball burnout has progressed to a point where special total control considerations are deemed unnecessary. Maximum power level is restricted by an administrative limit on the average of the ten hottest tubes.

F Reactor - A. W. Medcalf

The reactor operated at equilibrium during the first half of the report period, until a rupture on 12-30-62 interrupted a 19-day run. Subsequent operation has been plagued by a spline seal failure, a single VSR drop, and finally by inability to discharge a full column of temporary poison in the reactor at equilibrium. The latter resulted from the failure of a special test nozzle for ensuring piece-by-piece PCCF discharge to function as intended.

H Reactor - E. L. Conner

Several hot startups, a significant water leak, and sharply restricted spline utilization (following the KE nozzle break) contributed to the rather low operating efficiency over the report period. The reactor will be started up

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with its third E-N loading late in January; new HCR calibrations, based on data from previous runs, will be used. No changes have been found necessary in the temperature coefficients as used for the previous load. Adoption of the new HCR calibrations coupled with an ~300 c-mk predicted reactivity loss in loading changes has led to a planned extra-conservative approach to critical.

KE Reactor - L. L. Grumme

This report period, characterized by sporadic operation and resultant relatively low flattening efficiency, was also marked by an operating mishap of high potential hazard. During routine splining operations, in which the spline inserter was being utilized, the elevator was moved with the inserter inadvertently still attached to process tube 1162. The movement resulted in the shearing-off of front face nozzle number 1162, and an automatic scram from 2400 MW.

This item had far-reaching physics implications in that special restrictions were temporarily imposed upon all splining operations by the Processing operations, the most important one being restriction of the use of the spline inserter in several areas. Splining continued on a manual basis, with minor production loss (in the form of nonequilibrium startup losses). By report time, however, the temporary restriction on inserter usage had been removed with the qualification that extra procedural control be exercised to prevent untimely elevator movement.

KW Reactor - R. A. Chitwood

With the exception of a four-day scheduled outage (1-9-63 - 1-13-63) operation at KW was continuous. Flattening efficiency was good, and power level was restricted by the AEC administrative bulk power limit. A slight cycling tendency toward the end of a 29-day operating run was effectively controlled by close attention and timely efforts by the operating specialists with assistance from the reactor physicist.

PROCESS PHYSICS STUDIES

Reactor Safety Studies

Interim calculational measures used to credit residual Ball 3X effects in total control administration at the DR Reactor have been rescinded by the publication of HW-75407, Final Report, PT IP-447-C, Rod Drop Transient Comparison - D and DR Reactors.

PT IP-535-C, Test of Smaller VSR's in DR Reactor, was carried out on January 1, 1963, as a check on the control capacity of B-type VSR's relative to the larger DR rods initially installed. The primary concern in the use of smaller VSR's are the effects of response times and control capacity on speed-of-control; during normal operation when the Ball 3X hoppers are free to trip, the use of smaller VSR's would have no effect on available total control capacity.

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Total control requirements for the full-scale H and small-scale K E-N loadings have been reviewed. The intended production test for the K reactors has been modified to take account of the slightly larger water loss effect in the zirconium lattice. Supplementary control needs at the H Reactor for meeting total control requirements will be more than taken care of by the operating requirements for supplementary poison in the wet reactor during an extended outage.

Reactivity Studies

The accountability tables for production yield and quality predictions are currently being revised to include recent changes in fuel types, latest theoretical parameters, and possibly slight revisions in normalization factors. The effect of these changes is not expected to cause an appreciable difference in net predicted yields. A discussion of CPD measurements methods was presented to IPD Research and Engineering and to Nuclear Materials personnel by the CPD chemist working in this field.

Control Efficiency Studies

Full E-N loading of the H Reactor is again being resumed; the recently authorized third core loading was being charged at the end of the report period. This loading is being carried out under a Process Improvement Transition Authorization rather than a production test as formerly. Approximately 100 splines will be needed during the initial startup; there will be 14 fewer PCCF columns in use during this loading than during the previous E-N loadings.

The K reactor E-N block test, calling for the irradiation of approximately 50 columns in the zirconium-tubed lattice of one of those reactors, is currently circulating for signature.

A production test is currently being circulated for signature to provide some preliminary information on reactor response for small uncompensated rod movements. This production test is in support of a second phase calling for installation and trial of a prototype system for automatic distribution control in the D Reactor in the spring of 1963.

Reactor Fundamentals Training

The Reactor Specialist and Pile Operator training classes on Reactor Fundamentals, provided during the past several years by Operational Physics, were concluded at the end of December. In their place class sessions were begun on Reactor Physics under the Operator Certification Program sponsored by Manufacturing Section. In addition to the preparation of lecture and text materials, 16 class sessions were given during the month of January with an average attendance of 16 at each two-hour class. The first two lectures of the four-lecture series were covered during the month in classes for Supervisors and Reactor Specialists.

Of 77 operating personnel who had taken the extensive Refresher Quiz on reactor fundamentals in the previous program, 18 had eventually completed that test under audited conditions with a grade of 95 or better.

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DECLASSIFIEDTESTING OPERATIONIRRADIATION TESTING OPERATION

Irradiations - Sample and experiment irradiations were handled as follows:

Reactor	Test Hole	Facility	Request No. of Number	of Samples	Material-Purpose
KE	2D	Quickie	HAPO-184	24	Washington Designated Program
KW	2A	Quickie			
KE	2D	Quickie	HAPO-172	8	Effluent water (study to reduce radioisotope level in river)
KW	2A	Quickie			
KE	2D	Quickie	HAPO-254	1	Effluent water salts (effluent water radioisotope reduction study)
KE	2D	Quickie	HAPO-252	6	Arsenic (tracer isotope production)
KW	2A	Quickie			
C		Process Channel	HAPO-098	50	Graphite (burnout rate study)
D		Process Channel	HAPO-098	20	Graphite (burnout rate study)
KE		Process Channel	HAPO-098	20	Graphite (burnout rate study)
KE	2D	Quickie	HAPO-272	8	GRAPHIKOTE (radioactive buildup of process tube lubricant)
KW	2C	Hot Graphite	HAPO-177	4	Graphite (radiation damage study)
KW	4C	Snout	HAPO-177	1	Boron-graphite (heat generation rate and radiation damage study)
KW	3674	General Purpose	HAPO-222	2	Uranium (swelling test capsule)
KW	3C	General Purpose	HAPO-236	1	Zirconium (creep rate study)
KW	4C	Snout	HAPO-236	1	Heater lead material (radiation damage study)
KW	4B	Snout	HAPO-243	9	Iron (fast fission damage study)
KW	2A	Quickie	HAPO-249	1	Flux monitor foils (fast and thermal neutron flux study)
F		Process tubes	HAPO-279	4	Uranium (cladding studies capsule)
		*Gamma	HAPO-171	2	Plastics and rubber (radiation damage study)

*Gamma irradiation facility is located in 105-KE Building storage basin

H-1 Loop

Corrosion test H-1-2 was removed from the H-1 Loop on January 15 after 836 hours of operation at reference conditions. The loop was transferred to process water cooling on December 22 because of a serious leak in a valve seal. Since the introduction of process water would reduce the value of the corrosion and crud data gained from the test, it was decided to discharge the test short of the goal exposure of 2160 hours.

Vertical Bowing

Tube No. 4676-F

Up .09" since 8-10-62

Down .04" at 20' since 8-10-62

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Process Channels 2355, 2356, and 1585 in H Reactor were examined prior to retubing to determine their condition. Channel 2355 was clear and in good condition. Channel 2356 was blocked with broken graphite at 20 feet in from the inlet tube sleeve. There were several pieces of aluminum and a splitter blade in the channel. Channel 1585 had a stuck thermocouple stringer which had some balled up wire 13 feet in from the outlet tube sleeve.

HCR Channel 19 at KW Reactor was examined to determine why the rod had stuck. There were a number of balls in the channel and the usual amount of shifted graphite. There was no apparent obstruction in the channel which would cause the rod to stick.

COOLANT TESTING OPERATIONKER Loop Operation

KER-1 - Out-of-reactor recirculation testing to evaluate problems of switching from lithium hydroxide to ammonium hydroxide for pH control was completed. The out-of-reactor portion of KER-1 was chemically cleaned January 11 and was left open to the atmosphere from January 14 to January 28. The out-of-reactor portion of the loop was chemically cleaned a second time to remove the oxidation products. The objective of this test, as authorized by PT IP-551-A, is to obtain information to be used in establishing the required care and cleaning of N Reactor primary loop piping prior to startup.

KER-2 - Testing of eight 17-inch NIEL fuel elements with modified end closures as authorized by PT IP-536-A, Supplement A continued. The objective of the test is to evaluate the irradiation behavior of N Reactor inner fuel tubes at conditions equivalent to those expected at N Reactor.

KER-3 - Remained shut down because of primary pump failures.

KER-4 - Testing of fourteen 23-inch NAEL fuel elements as authorized by PT IP-477-A continued. The objective of the test is to evaluate the irradiation behavior of N Reactor fuel elements at conditions equivalent to or more than those expected at N Reactor.

1706-KE Single-Pass Tube Operation

Tube(KE Reactor)	Production Test	Water Supplied	Corrosion Inhibitor	pH
SP-1 (2952)	IP-476-AL	Deionized		
SP-2 (3050)	IP-524-AL	Normal process	1.8 ppm dichromate	6.6
SP-3 (4355)	IP-520-A	Pilot plant	1.8 ppm dichromate	6.6
SP-4 (4456)	IP-520-A	Pilot plant	1.8 ppm dichromate	6.6
SP-5 (4457)	IP-524-AL	Service	0.2 ppm quachrom glucosate	6.6
SP-6 (4863)	IP-524-AL	Service	0.2 ppm quachrom glucosate	6.6
SP-7 (4963)	IP-449-A	Deionized		
SP-8 (5063)	IP-449-A	Deionized		

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KE Reactor Outage Time Requirements

No unscheduled outages were charged to production testing. A total of 11.8 outage hours was charged to production tests as listed below:

<u>Production Test</u>	<u>Description</u>	<u>Hours</u>
IP-477-A, IP-536-A	Installed KER tray discharge equipment	11.8

Out-of-Reactor Facility Operation

TF-1 - Cyclic decontamination test No. 8 was started on January 22. The objective of this test is to determine the effect of extended decontamination solution contact times and extended periods of high-temperature operation on various decontamination processes. The prefilming and normal operating conditions are 270 C, 1700 psi and pH 10.0.

TF-2 - Testing to investigate the use of spring-loading to reduce the impact corrosion of Zircaloy-2 was continued. Operating conditions are 300 C, 1800 psi, and pH 10.0.

TF-3 - Testing to evaluate the effect of nucleate boiling in a crevice obtained by welding an NPR support on the surface of a heater element continued. Operating conditions are 596 F, 1985 psi and pH 10.0.

TF-4 - The cyclic decontamination testing using the alkaline permanganate-dibasic ammonium citrate procedure was completed. A new series of cyclic decontamination tests was started to evaluate a single-step carbon steel decontamination procedure using ammonium citrate as the decontaminant. Normal operating conditions are 300 C, 1650 psi and pH 10.0.

TF-5 - Testing to compare the corrosion inhibition of ammonium hydroxide versus lithium hydroxide in a carbon steel system was continued. Operating conditions are 290 C, 1400 psi and pH 10.0.

TF-7 - The test, specifically designed to evaluate fretting corrosion of PRTR process tubes at PRTR operating conditions was completed. The section of PRTR process tube was removed for examination and replaced with a new section. Fretting corrosion studies will continue at other than PRTR operating conditions. A new test which is primarily designed to evaluate the integrity of mechanical seals on special Zircaloy-2 capsules was initiated. The operating conditions are 200 F to 500 F (cyclic, 1125 psi, and pH 10.0. The test sections contain the following materials:

1. Five Zircaloy-2 clad capsules for mechanical seal testing.
2. A PRTR fuel element for fretting corrosion studies.
3. Two NIN fuel elements and two dummy fuel elements for corrosion testing.
4. A KSE3 fuel element for corrosion testing.
5. Six Zircaloy-2 clad fuel elements for braze closure testing.

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TF-8 - Uniform corrosion testing of A-212 carbon steel, 304 stainless steel and Zircaloy-2 was continued. Operating conditions are 700 F, 3300 psi and pH 10.0.

TF-13 - Heat transfer studies continued at 700 F and 125 psi.

TF-15 - Corrosion testing at 900 F, 4800 psi and pH 9.0 was completed. Corrosion testing was started at operating conditions of 1100 F, 3300 psi and pH 9.0 with ammonium hydroxide and hydrazine added. Coupons of 304 and 316 stainless steel, welded 316 stainless steel, Zircaloy-2, zirconium alloys, Hastelloy-X, Stellite and D-979 alloy will be changed after a conditioning run is completed.

TF-17 - Instrumentation revisions are in progress.

Design Test 1134 - Hydrostatic Cycling of NPR Pipe Sections - Testing of the eighteenth specimen is in progress. Testing of one specimen was completed during the report period.

COMPONENT TESTING OPERATION

Irradiated IPD Fuel Element Examination - 105-C Facility

Examinations were completed on fuel elements from 57 tubes requiring 212.1 fuel examination units. Total number of individual examinations completed was 6535.

Number of Tubes

Production Test

1	IP-12-A, Evaluation of Aluminum Components From Alternate Vendors
43	IP-216-A, Quality Certification Program
2	IP-468-A, Evaluation of Alpha Extruded Fuel Cores
1	IP-475-A, Corrosion Testing on 1706-KE Single-Pass Tubes with Low pH and Reduced Dichromate
10	IP-549-A, Half-Plant Low Alum Feed Water Treatment at F Reactor

Process Tube Corrosion Monitoring Program

In-reactor measurement of 2271 tubes was completed. The number of tubes measured for each reactor and document numbers of the reports issued are as follows:

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<u>Reactor</u>	<u>Number of Tubes Probologged</u>	<u>Report No.</u>	<u>HW-Number</u>
KE	584	86	76071
D	43	87	76097
DR	99	88	76111
B	188	1	76149
F	289	2	76150
KW	309	3	76275
C	32	4	76276
H	468	5	76277
DR	145	6	76278
D	114	7	76331

Visual examination was completed on ten process tubes.

<u>Reactor</u>	<u>Number of Tubes</u>	<u>Reason Removed</u>	<u>HW-Number</u>
B	4	Corrosion studies	76334
C	2	Leaker, Corrosion studies	76335
D	4	1-Rupture 1-Leaker 2-Tube outlet flange failure	76336

Critical Reactor Component Examination

All of the Truarc snap rings on the front face of KW Reactor were examined. None were broken.

Panellit System Programs

Gauges repaired, calibrated by Maintenance	174	
Gauges meeting calibration criteria	170	(98%)
-previous month		(78%)
In-board gauge reliability examination	870	
Gauges meeting examination criteria	852	(98%)
-previous month		(88%)
-average for past year		(94%)
Response time determination	870	
Gauges meeting response criteria	870	(100%)
-previous month		(100%)
-average for past year		(99.7%)
In-board Bourdon coil examination	12,708	
Non-leaking coils	12,673	(99.7%)
-average rate for past two years		(99.8%)
Gauges receiving failure analysis	114	

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2B-X1 switch examination	600
-acceptance rate	(83%)
In-Bourdon tube examination	90
Non-leaking tubes	81
Exchange gauge program (B, D, DR, F Reactors)	
Gauges received and examined	23
Accepted	13

O. H. Greager

Manager, Research and Engineering

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FACILITIES ENGINEERING

DESIGN ENGINEERING

Graphite Distortion Problems VSR Design

Fabrication of a vertical feed gear box has been completed and final mockup testing of the vertibore machines for VSR channel enlargement is scheduled in the near future. The design of the rear drive chain cutter reamer has been completed and fabrication of the equipment has been initiated. The new cast iron reamer is designed to operate the cutter at an anticipated cutting speed of one-inch per minute.

Out-of-reactor drop testing of aluminum oxide VSR liner sections indicates the sleeve sections will withstand approximately 1000 rod drops without appreciable damage. Similar tests of pyrolytic graphite sleeves resulted in a life of approximately 170 rod drops before failure as compared to a life of 100 drops for polycrystalline graphite liners. Modifications are being made to the VSR tip to reduce the impact on the channel liner in a distorted channel, in order to extend the life of the graphite liners. A modified tip of this type was utilized in the aluminum oxide sleeve tests. Experimental work in evaluating the thermal cycling characteristics of the currently available pyrolytic sleeves and the evaluation of irradiation damage to aluminum oxide samples is continuing.

Rear Face Crossheader Examination

The final report of the destructive and non-destructive examination of the rear face crossheader which was removed from 105-DR Reactor has been completed and is being readied for issuance. In support of possible future maintenance replacement of selected rear face crossheaders, design has been completed on a universal replacement rear face crossheader. The collection of reactor flow, pressure and vibration data on the replacement crossheader installed at 105-DR Reactor is continuing.

Ball Recovery System

The preliminary engineering study document was completed and issued (HW-75494). It provides concepts and justification for improving the Ball 3X vacuum recovery systems at 105-B, D, F, DR, and H Reactors. The proposal includes an improved vacuum system, VSR tip handling facilities, and conveyor-hopper system for ball handling and storage.

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EQUIPMENT DEVELOPMENT

105-K Reactor Zirconium Tube Program - Development

Laboratory testing of the production model boring bar contemplated for use at the 105-K Reactors for process tube channel renovation has been completed. 108-D mockup tests indicate a minimum of 10 years of residence for zirconium tubes in all zones of the K Reactors based on current fuel concepts and exposure rates. Procurement of equipment for a major outage is in progress. Alternate methods of channel reconditioning under investigation include abrasive cutting, flame cutting, plasma arc cutting, and electrolytic machining.

Self-supported Fuel Element (SSFE) Charging

The HAPO-designed SSFE charging machine is being readied for use at the conclusion of the first zirconium tube replacement outage at the 105-KW Reactor. Final testing of the two vendor-designed prototype charging machines is in progress in Seattle.

Prototype High Speed Scanner

The high speed scanning system prototype has been run continuously for 17,331 hours without incident.

Design Test Service

The N-Reactor program utilized 703 hours of engineering effort for design test service.

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PROJECT ENGINEERING

CGI-839, Modification of Fuel Element Test Facilities, 1706-KER

Parallel testing of two modified 275 HP recirculating pumps is scheduled by the vendor for the very near future.

A review of design progress scheduled for the safety circuit system portion of this project has been transmitted to the RLOO-AEC for approval. This schedule revision extends the design completion date from February 1, 1963, to April 1, 1964. The extended scheduled completion is contingent upon the placement of the purchase order for the loop safety circuit equipment by April, 1963. The preparation of design has been suspended since May 1, 1962, pending placement of the purchase order for this equipment. The Commission has contacted three vendors and requested proposals for purposes of negotiating a contract for procurement.

CGI-844, 100-K Area Coolant Backup

The 100-K high pressure cross-tie line was cleaned in January. The National Water Main Cleaning Company was the contractor and 100-K Maintenance forces assisted with the work.

CGI-883, Increased Process Water Flow, 100-K Area

The second 1500 HP low lift pump motor to be reconnected was installed at 190-KW in position No. 5. The vibration level is higher than expected but operable. The need for additional modification to the 190-K low lift pump motors is being evaluated.

CGI-939, Analog to Digital Converter System for Temperature Monitoring,
B, C, D, DR, F, and H Reactors

The first area installation of an Analog-to-Digital Converter was made at 105-F during the early part of January and readied for use January 15. Field preparation is being initiated at 105-H and has been completed at 105-D and DR.

CGI-960, C and D Work Platform Safety Improvements, All Reactors

The design criteria have been approved by the Design Council and forwarded to RLOO-AEC for approval. The certified design progress schedule indicating the start of detail design on December 26, 1962, and completion by December 31, 1963 (three and one-half months later than the proposal) has also been submitted to the RLOO-AEC for approval.

The platform synchronization system described in the design criteria is based upon the use of a rotary transducer-wheatstone bridge system which will be used to detect angular misplacement between the platform hoisting

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mechanism shafts. Protection against failure of the platforms will be provided by installing new drive brakes on the hoisting shafts or by new guide rail brakes installed on the platforms, depending on the particular location.

CGI-966, Safety Circuit Trip Identification Systems, 105-B, C, D, DR, F and H Areas

The resubmitted bid package for the engineered equipment was returned by RLOO-AEC with the statement that the justification was not sufficient for the restrictive specification.

CGI-967, High-Speed Scanning System for Temperature Monitors, KW Reactor

Revision No. 2 to the purchase specification for the high-speed scanner are in RLOO-AEC Purchasing for procurement action.

R-20520, K Retubing

Two hundred sixty-nine zirconium tubes have been received, of which 198 have been accepted in all respects except that chemical properties have not been certified on 138 tubes.

One hundred twenty-five nozzles have been received from the vendor and 150 additional nozzles are ready for shipment.

The only item of material remaining to be delivered for the first installation outage is the gas boots, which have been shipped by the vendor.

CPFF Construction Service Contractor-Liaison

Issued six new work orders and supplemented 11 old jobs for a total of \$92,825 to J. A. Jones Construction Company.

Plant Forces Work Review

The Labor Standards Board approved four jobs for assignment to plant forces, estimated to cost \$275,365.

Manager, Facilities Engineering

RT Jessen:dgm

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FINANCIAL

ACCOUNTING

The Department's research and development FY-1963 midyear budget review was revised to provide for the division of responsibilities between the former Fuels Preparation Department and the N Reactor Section and to isolate the Production Fuels Section's requirements.

The Department's FY-1965 research and development budget preparation has been initiated.

Department overtime, during six months of FY-1963, was 41 677 hours compared to 91 706 hours in the same period of FY-1962. This reduction represents about a \$400 000 cost improvement in salary and continuity of service costs.

Historical information on water flow and chemical consumption rates was supplied to N Reactor personnel to assist them in estimating N Reactor requirements. In addition, general information was provided on the existing cost accumulation system and its possible application at N Reactor.

BUSINESS AND INFORMATION SYSTEMS

During the month, a master file of all IPD exempt people has been established and the procedures developed to prepare the IPD organizational directory. This directory will be issued monthly on a very limited basis, and a semi-annual HAPO distribution will be maintained.

At the request of the Accountant-Production Fuels, the services of the Maintenance Cost Specialist were made available to review the clerical procedures in practice in the Production Fuels Section for the purpose of evaluation and recommendations for simplified methods. As this work can only be performed on a part-time basis, no recommendations have been forthcoming as yet.

AUDITING

Revisions of seven of the Department's Organization and Policy Guides were issued during the month. One guide was cancelled and replaced by a HAPO-wide guide on the same subject, Illegal Copying of Documents. This replacement was in response to our request and conforms with our practice of encouraging HAPO guides for those policies and procedures which may be uniform throughout HAPO. The OPG index was updated as of the end of the year.

Comments relative to a revision to HAPO Policy Guide 8.3, Property Management of Special Equipment, providing for the stocking in Stores of stoves, refrigerators and food freezers as replacement units, were submitted.

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Two revisions to AEC Manual Chapters were received and reviewed for effect on IPD operations.

Tests were made of production data submitted by the reactors during November 1962. No significant discrepancies were evident.

Continuing attention was given to area assigned and area pool vehicles parked in Richland without overnight parking tickets. Originally, information on pool vehicles was obtained from Transportation. Violations have been materially reduced.

SH Small

Manager, Finance

SH Small:slb

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SIGNIFICANT REPORTS ISSUED

<u>Number</u>	<u>Class'n.</u>	<u>Author</u>	<u>Date</u>	<u>Title</u>
HW-75494	Secret	JE Boyd RH Somerville	11/13/62	Preliminary Engineering Report Improved Vacuum Ball Recovery System, 105-B,D,DR,F,H Reactors.
HW-75890	Unclass.	DF Watson	12/13/62	Provision for Sodium Silicate Inhibitor Facilities - 100-K Coolant Backup System.
HW-75941	Unclass.	JE Boyd	12/19/62	Study Report on Reactor Process Work Platforms - Safety and Synchronization of Drive Equip- ment - All Reactors.
HW-76154	Secret	VG Blanchette	1/3/63	IPD Mid-Year Electric Power Forecast FY-1963 - FY-1967.
HW-76190	Unclass.	KD Coughren	1/10/63	Design Test 1154 - Zirconium Tube Mockup Testing of Radiation Distorted Fuel Elements, K Reactors.
HW-76254	Unclass.	MH Schack DH Curtiss	1/16/63	Production Test IP-565-AM - In- stallation of Zirconium Process Tubes 105-KW Reactor.
HW-76023	Unclass.	DF Watson	1/18/63	Project Representatives Meeting No. 3, Project CGI-905, Interim Modifications for Improved Coolant Backup, 100-B, C, D, DR, F and H Reactors.
HW-74705	Unclass.	JP Cooke RR Henderson	12/18/62	An Analysis of Hanford Reactor Atmosphere Systems.
HW-75973	Secret	JP Cooke	12/20/62	DT-IP-558-T Gas Sampling Facility Correlated with Front to Rear Thermocouple Stringer - KW Reactor.
HW-74443 Sup. 6	Unclass.	DF Arnold	12/31/62	Installation Steps for CGI-939 Operation Instructions
HW-76031	Unclass.	RR Cone	12/31/62	Summary Report - Linear Power Rate of Rise Instrumentation.
HW-75957	Unclass.	BW Herrman	1/2/63	Testing of Production Model Barton Metrisite Differential Pressure Transducer for NRD Primary Coolant Monitor.

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<u>Number</u>	<u>Class'n.</u>	<u>Author</u>	<u>Date</u>	<u>Title</u>
HW-75751	Unclass.	FE Haun, Jr.	12/19/62	Pressure Switch Testing Ball 3X System Application - Final Report.
HW-76004	Unclass.	MP Black	12/28/62	Design Test 1151 - Evaluation of New B Reactor Size Flexible VSR Joint Design - Final Report.
HW-76016	Unclass.	M Pociluyko	12/28/62	Flow Test Report on Eight NPR Dummies of Most Recent Design (SK-1-27701).
HW-74338	Unclass.	KG Hawkins	1/10/63	Design Test 1135 - K Self-Support Outlet Nozzle Strength Test Report Supplement.
HW-75805	Unclass.	HF Jensen	12/11/62	DT-IP-553-E Chemical Descaling of Intertie Piping in 105-KE.
HW-75729	Unclass.	KG Hawkins	1/10/63	Design Test 1145 - K Outlet Nozzle Strength Test - Final Report.
HW-72260 Sup. 1	Unclass.	FE Haun, Jr.	1/11/63	Flow and Energy Loss Characteristics of 3/4 Inch Aluminum Spiral Rear Connectors Test Report Supplement.
HW-76270	Unclass.	WEB Lander	1/18/63	Design Test 1148 - Pressure Reducing Valves for NPR Water Sampling - Test Report.
HW-76318	Conf.	WEB Lander	1/23/63	Life at the 25th, 40th, and 50th Tube Rows in the K Reactor.
HW-76186	Unclass.	RL Woodward	1/23/63	Design Test 1141 - K Nozzle Cap Gasket Test Report.
HW-76187	Unclass.	RL Woodward	1/23/63	Design Test 1142 - K Nozzle Insert Flow Erosion Test Report.
--	Unclass.	JM Goff, Jr.	1/14/63	B-D-F Van Stone Flange Pull Test Report.
--	Unclass.	RK Smith	1/21/63	Vacuum System for Tube Replacement.
--	Unclass.	RL Woodward	12/27/62	Cast Splitter Fatigue Test.

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<u>Number</u>	<u>Class'n.</u>	<u>Author</u>	<u>Date</u>	<u>Title</u>
--	Unclass.	KW Keefer	12/28/62	Zirconium Tube Push Mandrel Test Report.
HW-76073	Unclass.	FJ Mollerus	12/29/62	230-KV Automatic Reclosing.
HW-75407	Secret	AD Vaughn	10/10/62	Rod Drop Transient Comparison - D & DR Reactors.
HW-76130	Conf.	GB Hansen	1/7/63	Uranium Quality Control Analytical Results.
HW-76241	Conf.	GB Hansen	1/15/63	Uranium Quality Control Analytical Results.
HW-76306	Conf.	WG Jolley	1/21/63	Report of Uranium Accountability in Manufacturing Operation Processes for the Quarter Ending 12-31-62.
--	Unclass.	RH Scanlon	1/15/63	Billet Evacuation Test.
HW-76233	Secret	WN Mobley	1/22/63	Production Forecast.
HW-76370	Secret	HF Tew	1/28/63	Production Assumptions and Schedules.
HW-76362	Secret	WN Mobley	1/25/63	Production, Receipts and Inventories.
HW-76132	Secret	JT Stringer JE Minor	1/7/63	Report to the Working Committee From the General Electric Company, HAPO.
HW-76354	Secret	RS Bell	1/25/63	Complete Failure of Front Face Nozzle 1162-KE.
HW-74785	Unclass.	RD Duncan	9/4/62	Scope and Justification Report, Increased Capacity for the Reactor Atmosphere Dryer System, Buildings 115-B, D, F, and H.
HW-75674	Conf.	JR Young	12/20/62	Quarterly Report, The Reactor Process Tube Problem, December, 1962.

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TRIPS

<u>Name</u>	<u>Firm & Location</u>	<u>Date</u>	<u>Purpose</u>
G. F. Jacky	Sylvania Corning Nuclear Corp. Hicksville, New York	1/7/63	Observe performance of nickel plating equipment.
	Meaker Corporation Nutley, New Jersey	1/8-9/63	Same
	Hanson-Van Winkle- Munning Company Matawan, New Jersey	1/8/63	Same
	General Electric High Voltage Switchgear Philadelphia, Pa.	1/9/63	Same
	General Electric Communications Prod. Lynchburg, Virginia	1/10/63	Same
	Udylite Corporation Detroit Michigan	1/11/63	Same
	Crown Rheostat and Supply Company Elk Grove Village, Ill.	1/14/63	Same
J. T. Stringer	Savannah River Plant Aiken, South Carolina	1/21-25/63	Attend FEDC Working Committee Meeting
E. Hilgeman	Plant Engineering and Maintenance Show Chicago, Illinois	1/26-31/63	Latest maintenance techniques and use of work standards.
L. L. Samford	Alcoa Edgewater, New Jersey	1/7-8/63	Discuss quality and delivery of aluminum caps and cans.
	J. B. M. Tool & Die Bryon, Ohio	1/9/63	Discuss delivery of NPR dies.
	Cliff Mfg. Wickliffe, Ohio	1/10/63	Discuss quality and delivery of aluminum caps and cans.
	I & W Mfg. Wickliffe, Ohio	1/11/63	Discuss delivery of water mix spools.
	Brush Beryllium Cleveland, Ohio	1/11/63	Discuss quality and delivery of NPR beryllium rings.
	Harvey Aluminum Torrance, California	1/14/63	Discuss quality and delivery of aluminum caps and cans.

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<u>Name</u>	<u>Firm & Location</u>	<u>Date</u>	<u>Purpose</u>
P. B. McCarthy	Union Machine Works Pacific Steel Products Seattle, Washington	12/22/63 1/2-3/63 1/17-18/63 1/24-25/63	To discuss self-supported fuel element charging machines being fabricated by these vendors.
J. A. Dear	Union Machine Works Pacific Steel Products Seattle, Washington	1/2-3/63	Same as above.
H. F. Jensen	Dow Industries Piqua, Ohio Turco Products, Inc. Los Angeles, Calif. Consolidated America San Francisco, Calif.	1/2-5/63 1/16-18/63 (Twice to each)	To discuss cleaning of NPR piping.
F.A.R. Stainken	Western College Placement Association San Francisco, Calif.	1/16-18/63	Participant in Thirteenth Annual Conference.
F.A.R. Stainken	Hi-Y Club Yakima, Washington	1/30/63	Present talk under sponsorship of Hanford Speaker's Bureau.
S. S. Jones	Baldwin-Lima-Hamilton Pelton Division San Francisco, Calif.	1/20-21/63	Discuss export system. (CA-352)
W. E. Vetter	San Francisco, Calif. Automatic Switch Company San Francisco, Calif.	1/20-27/63	Attend National Symposium on Quality Control. Consultations on Panellit gauge mercury switches.
L. L. Samford	Western Filter Gardena, California	1/15/63	Discuss quality and delivery of water mix spools.
	R & R Tool Company Hollywood, California	1/16/63	Discuss quality and delivery of aluminum supports.
	Bridgeport Brass Riverside, California	1/17/63	Discuss quality and delivery of aluminum caps and cans.
M. P. Johnson	Baldwin-Lima-Hamilton Corporation San Francisco, Calif.	1/21/63	Discuss consultant contract and the proposed export system testing program.
R. S. Bell	University of Idaho Moscow, Idaho	1/10-11/63	Recruit technical personnel.
J. C. McKay	Chicago, Illinois	1/28-30/63	Attend National Plant Engineer- ing and Maintenance Conference.

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<u>Name</u>	<u>Firm & Location</u>	<u>Date</u>	<u>Purpose</u>
C.F. Quackenbush	New York City, New York	1/17-18/63	Attend committee meeting on nuclear piping.
A. McDonald	Westinghouse Electric Co. San Jose, California	1/18/63	Consultation on K Area high lift pump motors.
R. H. Wheaton	Oregon State University Corvallis, Oregon	1/17-19/63	Discuss Industrial Engineering concepts with members of the Mechanical and Industrial Engineering Department.

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VISITORS

<u>Name</u>	<u>Firm & Location</u>	<u>Date</u>	<u>Purpose</u>
D. G. White	Haughton Elevator Co.	1/21-22/63	Consultation on the proposed safety and synchronization system for reactor platform drive equipment in accordance with terms of Consultant Agreement CA-293.
W. F. Rehmke	Union Machine Works Seattle, Washington	1/8/63	To discuss self-supported fuel element charging machines.
C. Tonjas	Cryenco, Inc. Denver, Colorado	1/10/63	Alternate helium source for Hanford.
J. R. Eliason	General Electric Co. San Francisco, California	1/16-18/63	Preliminary discussions on forthcoming audit of IPD electrical facilities.
C. L. Liakos	Foxboro Company Seattle, Washington	1/10/63	Discuss instrumentation.
M. Thompson	Bridgeport Brass Bridgeport, Connecticut	1/24/63	Discuss quality and delivery of aluminum caps and cans.
W. R. Thomas	Bailey Motor Company Seattle, Washington	1/11/63	Field service at the 384 Building.
L. Bowman R. Meyers	Betz Laboratories Olympia, Washington	1/11/63	Consulting on nickel plating equipment.
J. A. Fellows	Mallinckrodt Chemical Works	1/16-17/63	Discuss uranium technology.
J. W. Ryerson	Meaker Company Nutley, New Jersey	1/11/63	Consultation on nickel plating equipment.

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