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DECEMBER 10, 1964

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N - REACTOR DEPARTMENT

HW--80559-11

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MONTHLY REPORT

NOVEMBER 1964

HANFORD ATOMIC PRODUCTS OPERATION

RICHLAND, WASHINGTON

GENERAL ELECTRIC

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OPERATING SUMMARY

Input production Pu (KMWD)	55.71
Input production T (KMWD)	0.
Maximum power level attained (MW)	3,600
Average power level while operating (MW)	2,932
Time operated efficiency (%)	63
Plant utilization (%)	65.3
Number of Shutdowns	7
Scheduled	1
Unscheduled	6
Reactor Process Data	
Power at which computed (MW)	3,600
Primary coolant inlet temperature (°F)	350
Primary coolant outlet temperature (°F)	481
Maximum graphite temperature (°F)	1,190
Heat Loss	
Graphite cooling (MW)	43
Shield cooling (MW)	1.5
Rod cooling (MW)	1.4
Steam temperature (°F)	313
Steam pressure (PSIG)	81
Dump condenser flow (gpm)	300,000
Inlet temperature (°F)	58
Outlet temperature (°F)	134

Nuclear Steam Generating Rate

Maximum (1000's lb/hr)	12,100
Average while operating (1000's lb/hr)	11,700
Water Collected from Gas Atmosphere (gal/day)	Negligible
Demineralized Water Produced (gal/hr)	63,532
Helium Losses (1000's cu. ft.)	341
Fuel Oil Usage (bbls.)	32,882
Flattening Efficiency (%)	83

Fuel Conditions

Heat flux

Maximum (BTU/hr/ft. ²)	510,000
Average (BTU/hr/ft. ²)	425,000

Clad temperature

Maximum (°F)	550
Average (°F)	515

Uranium temperature

Maximum (°F)	805
Average (°F)	735

Fuel Failures

0

Fuel Balance	<u>0.95</u>	<u>1.25</u>	<u>Coproduct</u>	<u>Other</u>	<u>Total</u>
Fuel charged (Tons)	0.37	0	0	0	0.37
Fuel discharged (Tons)	0.37	0	0	0	0.37
Net change (Tons)	0	0	0	0	0
Total in reactor at month end (Tons)	327.75	39.74	0	0	367.49

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Fuel Element Out (Assemblies)

0.95%	972
1.25% (driver)	0
1.25% + 0.95% (spike)	48
Normal uranium	540
Total assemblies	1,560
Total tons	29.3
% of forecast	83.7
Uranium Utilization (%)	73.0

SUMMARY

The reactor operated at 2800 megawatts for a week, after which the power level was raised in two steps to 3600 megawatts. Authorization was received from the Atomic Energy Commission to proceed to full power of 4000 megawatts. Performance has been such that a careful assessment shows that the plant has the capacity to go safely somewhat beyond full power without exceeding any existing process standards. The preliminary Hazards Summary report on conversion to dual purpose operation is nearly ready for distribution. p 8

Operation of the boiler was interrupted by rupture of small steam lines from vibration caused by a broken valve stem. One of the steam generators shows increasing evidence of leakage, and is being watched closely. The repairs to the dump condensers have solved the immediate problem of vibration and leakage. p 9

The reactivity in the control rods is 7 mk compared to a predicted 5.5 mk. Coolant flows through the east and west halves are unbalanced by five percent, about double the predicted difference; but flow through the left side is still higher than expected. p 10

Extrusion of 1.95 percent enriched driver fuel elements has been started. It has been shown that fuel elements with as much as 50 percent out of standard variation in clad thickness could be charged in fringe positions without resulting in excessive swelling. Beryllium zircaloy alloy wire has been produced by swaging. p 21

Meltdown experiments on Li-Al target with Zircaloy show that a reaction takes place that retains the lithium in such a way that a damaged target may still be an adequate poison for reactivity control. The first lithium-aluminum alloy for the coproduct program has been extruded and target assemblies have been processed. p 26

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REACTOR PLANT OPERATION

Plant performance was generally excellent. The reactor operated at or near its maximum authorized power of 3600 megawatts for 18 days. On November 8 and 15 the successive increases up to this power level were attained without difficulty.

On November 17 a manually initiated scram from this power level was performed as part of the Power Ascension Program. All systems responded satisfactorily to the transient, with the exception of two of the drive turbines that did not reduce to low speed, and three control rods that were slow. The turbines were tripped off manually to pony motor drive, and the three rods were backed up by automatic drops of six rows of safety balls.

The primary system repressurization after the scram was limited to 165 psi above the initial level of 1400 psi, the two standby burners on the boiler lit off automatically, and the secondary system response was extremely good.

No problems have appeared that would limit continuation and completion of the Power Ascension Program.

Dump condenser performance has been excellent since modifications were made to the tubing.

Reliable results are being received from most of the automatic pH and conductivity instrumentation. The hardness analyzer has provided valuable information in monitoring for dump condenser leaks. The chloride analyzers will require filters to remove iron oxide particles unless a filter soon to be installed in the demineralized water supply is sufficiently effective.

In addition to the planned scram, operation was interrupted by a number of incidents: a valve stem failure in the 2A steam generator block valve, a false fuel element rupture monitor indication, a fuel oil leak, a flow monitor component failure, loss of vacuum to two drive turbines, and a valve stem failure on one of the two boiler steam pressure reducing valves.

The valve stem failure was similar to that in another valve last July. Both were of 17-4 PH heat treated steel. The mechanism of failure is believed to be stress corrosion cracking.

The false rupture indication was traced to a crud burst in the monitor. Backflushing cleared it.

Unusual Incidents

During emergency maintenance on the boiler forced-draft-fan pressure switches, a short blew a fuse in the fan-damper control circuit and resulted in the dampers closing on both forced-draft-fans. The auxiliary burner fan remained in service. The resulting loss of combustion air to the boiler dropped off four of the six main burners in service. Before manual fan damper control could be restored and a purge initiated of the unburned gases in the boiler to permit reignition of the main burners, (a period of 32 minutes) the boiler pressure dropped from 475 psi to 340 psi and the steam flow dropped from 180,000 lb/hr to 30,000 lb/hr. There was no loss in frequency on the B bus turbine-generator since reactor-generated steam continued to supply both it and also the drive turbines in the 109 Building. The reactor was not shut down.

During operation at maximum-authorized power, the motor starter on the hydraulic pump for No. 13 dump condenser steam pressure control valve burned out, blowing the breaker in the motor control center. Since this center supplies the power to hydraulic motors for the steam pressure control valves for all four dump condensers on the No. 4 steam quadrant, the loss of hydraulic pressure caused all four steam pressure control valves to begin closing. Reactor power was immediately reduced to a level which could be supported by the twelve remaining dump condensers. The motor control center was restored to service in a few minutes and all of the valves except one were returned to service. The transient which resulted from this accidental upset was readily controlled by the combination of automatic and manual actions.

During an attempted startup, an unstable oscillation occurred in the drive turbine header pressure control valves at the boiler, and caused rupture of three instrument lines sensing steam pressure. As a consequence steam escaped into the boiler house and the boiler shut down, causing loss of B bus and scrambling the reactor. Failure of a valve stem caused the incident.

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PROCESS PHYSICS

Equilibrium Conditions

November 23, 1964

Reactor Power Level	3600 MW
Reactivity in Rods	7 mk (predicted = 5.5 mk)*
Flattening Efficiency	83 percent
Axial Power Distribution	Peak-to-average ratio is equal to or lower than a cosine distribution. This ratio was lowered from a ratio about 20% higher by changing the pattern of the rods.
Horizontal Rod System Worth	67 $\pm$ 4 mk
Spike Column Loading	144 columns (no change)
Reactivity contribution of spike enrichment	6 mk

The power ascension sequence is being controlled to provide the required minimum shutdown margins of 10 mk or greater, computed on the standard assumption of a cold-water injection accident at hot turnaround, as directed by the Atomic Energy Commission.

The startup of November 21 was from an almost xenon free condition. Turnaround occurred at a power level of 33 percent with a shutdown margin of about 14 mk which is considerably in excess of the 10 mk required.

*The reactor is being operated with the inlet temperature approximately 15 F lower than the design value. This results in approximately 1 mk more reactivity than expected.

Axial Flux Distribution

The rod pattern being employed since the startup of November 21 has been changed from that previously employed in such a way as to produce much greater axial flattening of the power. The axial flux distribution peaking has thus been made equal to or less than that obtained from a cosine distribution. The reactor is not cycling but the new power distribution, while favorable from the point of view of increasing the total power level to which the reactor may be operated without exceeding power density limits, has narrowed the margin from cycling.

Primary Coolant Flow Distribution

On the basis of process tube flow monitor data taken during five-loop operation of the reactor at the latest power level plateau, the coolant flows through the east and west halves of the reactor were found to be unbalanced by about 5 percent. This is about twice the amount predicted during design. However, the average rate of flow through the reduced flow side is still higher than expected, because the total rate of flow through the reactor is higher than originally calculated. This bulk flow unbalance does not appear to represent a limit to full power operation. Individual tube low flows may be found to be limiting, but they can be individually corrected.

MECHANICAL AND THERMAL EQUIPMENT PERFORMANCE

Dump Condensers Repair

It was reported in October that the dump condensers were being fitted with devices for holding down the tubes to prevent destructive oscillation from the steam flow.

This repair program appears to have been successful in bringing the units back into sound operation. They have been operating well since the repair job was completed. After approximately two weeks of operation, condenser No. 1 was opened for inspection, as planned. The external appearance of the tube bundle and bands was excellent, with no evidence of fretting or chafing.

Horizontal Rod Seals

Some of the horizontal control rod drive piston seals have been leaking, and deforming by extrusion after many cycles of use. Various types and combinations of rod seals have been tested in an attempt to find a durable non-leaking seal. The most successful seal configuration consists of a

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bushing in the gland bottom ahead of the packing. This combination apparently compensates for the natural sag of the rod itself, giving a better sealing action and also minimizing wear and seal extrusion.

Another successful seal tested this month employed 3 rings of black teflon and one ring of 90 durometer Buna N, alternating with brass spacers. After 120 cycles the leakage was still well within the acceptable limits and there was no visible sign of extrusion. After an additional 100 cycles the leakage was still within acceptable limits but there was quite extensive extrusion.

Grayloc Repair

Grayloc connectors are used to connect throttling butterfly valves into the inlet connector piping and to connect diversion valves and diversion piping into the outlet connector system. These couplings permit removal of valves for maintenance. Stresses have caused some of the couplings to leak, and improved 'O' rings have been sought for sealing the flange faces of the couplings.

Hollow metallic 'O' rings were tested for over 1000 thermal cycles, after which the four test assemblies were disassembled for inspection. Although there had been no evidence of leakage during the thermal cycle tests, there were signs of very slight leakage between the 'O' ring and the 'O' ring groove seat. Four new test assemblies with modified 'O' ring grooves are currently being thermal cycled between 395 F and 555 F, with a combined bending moment of 11,000 inch-pounds being applied on the cooling cycle, to simulate cooling stresses on the coupling.

Butterfly Valve Operator

A low flow condition in a process tube was caused by the inlet butterfly valve being in the 50 percent lift position. All attempts to move the vane to the full-open position failed, and it was necessary to replace the actuator to obtain normal operation. Malfunction of the operator was caused by galling of the piston rod.

Graphite Cooling System Surge Tank

Equipment has been modified to reduce the possibility of an explosive gas mixture accumulating in the graphite cooling system surge tank. A portion of the clean makeup water is now routed directly into the top of the surge tank, so that the water in the bottom of the tank is free of gas at all times.

Steam Generator Leak

Steam generator 3A continues to show an increased leakage rate based on our isotope carryover measuring technique. The most recent data show a leak rate of about 550 gallons per hour. A development program is underway concentrating on tube welding techniques in anticipation of the possibility of resleeving this generator during the March outage should this be necessary. Meanwhile leakage data are being taken twice weekly during operation. Similar measurements of blowdown water from the other generators show that none of them have any appreciable leakage.

Incoloy-800 for Steam Generators

The sleeve material used recently for repair of unit 3B was Incoloy-800. A review made of recent Incoloy-800 technology and of test data on prototype repair joints indicates a significant superiority of Incoloy-800 over 304 stainless steel for sleeve-type repair of steam generators. No instances were found of Incoloy-800 failure by stress corrosion cracking under service conditions that can be expected for the steam generators.

Valve Stem Failures

Three valve stems have failed in the past few weeks in different types of valves. Two of the failed stems were made of 17-4 PH stainless steel. Preliminary analysis of these two breaks shows a type of cracking usually associated with stress corrosion. Metallurgical tests of the shafts indicate that they were heat treated at 900 F instead of the specified 1025 F. This improper heat treatment would leave 17-4 PH stainless steel sensitive to stress corrosion. An unfailed 17-4 valve stem was removed for inspection and found to have incipient cracks at the same location.

Sump Pumps

Each of the five steam generator cells has a sump that can be filled with nine feet of water to separate it from the pipe gallery when maintenance is required in the cell. The cells also serve to collect certain drainage.

The sump pumps continue to be a problem. Seizing of the pump shaft bearings and rubbing of the impellers have been experienced, along with blowing of electrical fuses because of motor overloads. The cooling water to the bearings, originally obtained from the pump discharge, is now obtained from a filtered water line. This change has benefited the bearings, but appears to be creating a differential thermal expansion between shaft and housing, which may be causing the impellers to wear. The original impeller clearances have been reset.

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Control on Discharged Coolant Routing Valve

Occasional drifting of a swing disc valve and partial unsealing of the disc from the low-pressure discharge port has been observed. When this occurred recently, it permitted undesirable backflow into the quench basin from the spill cooler effluent line which connects with the low-pressure flush line about 22 feet downstream of the valve and normally discharges to crib or river. This inadvertent backflow into the quench basin is undesirable because the capacity of the basin is only adequate to dissipate the energy from an emergency dump of the primary loop coolant, and its effectiveness is reduced when backflow is permitted. Suitable changes in a solenoid control circuit should correct the condition.

Filter Tests

The stack effluent filters for trapping radioactive particles have been provided with a cooling system to prevent overheating from the particles.

High-temperature tests were performed on the particulate filters that had previously been used in the filter cooling cycle tests. One of the filters was baked for a total of 32 hours at 300 F in an air-circulating oven. The wood frame and the adhesive darkened slightly and the gasket hardened, but the filter media were unaffected and the filter effectiveness was undiminished. The test results indicate that temperature control lags in the filter cooling system will not jeopardize the integrity of the filters.

Other Mechanical Maintenance

Other maintenance included:

- (1) Cleaning of servo motor pilot valves to provide improved control of three of the primary drive turbines,
- (2) Centrifuging lube oil of one primary drive turbine to remove water (source unknown), and
- (3) Modification of the No. 1 safety circuit to initiate a rod scram upon loss of any one circulating raw water river pump, instead of any two pumps. The revised circuit permits an increase in maximum allowable dump condenser raw water outlet temperatures.

Dump Condenser Instrumentation

In the inspection of the No. 1 dump condenser two weeks after it had been repaired, it was discovered that the four distance transducers installed to measure amplitude of vibration of the tubes had been completely destroyed. This is believed to have resulted from softening of the epoxy potting compound. The transducers are being replaced by high temperature ceramic units.

Additional instrumentation has been installed, including two tri-axial accelerometers, nine level probes and ten thermocouples.

Data collected from the tri-axial accelerometer mounted on one of the more loosely supported cooling tubes on the periphery indicate that the tube motion in the vertical-east-west plane may exceed one-inch. The tri-axial accelerometer mounted on the main support strap indicated very little strap motion except for a few surges occurring several seconds after the scram.

The level probes were installed at intervals of six inches to provide a check on the differential pressure-level indication system.

The thermocouples will measure the inlet and outlet temperatures of the cooling water and also the temperature of the steam chest.

Safety System

The electrical circuitry was installed to permit introduction into the No. 1 safety circuit of a rod scram trip from primary system low head-pressure. This circuit will serve to limit the minimum primary system pressure during equilibrium reactor operation and will permit an increase in maximum allowable process tube outlet temperatures, since without this circuit process tube outlet temperatures have to be held low enough to prevent in-reactor boiling even though primary system pressure might fall to a relatively low level.

Rupture Monitor System

Indication of a gross failure of the GM tubes on the gross gamma system was detected during a pre-startup check. It was found that the anode voltage was too low for the GM tubes to start. When the voltage was adjusted, the system was returned to service.

On November 24, a false fuel element rupture signal, which resulted in a scram, proved to come from crud deposition in the monitor. Spurious signals of this kind will be reduced or eliminated by raising the signal channel window of the gamma energy monitor to make it less sensitive to crud indications and still retain sensitivity to identify fission products.

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Temperature Monitor System

The 6.5V DC power supply for the temperature monitor and data logger system resistance temperature detector bridge circuits overheated and began to drift. The spare unit was installed and showed the same symptoms. A new power supply with better characteristics was installed in a better ventilated location and is holding the load with no measurable drift.

SPECIAL TESTS PERFORMED

Status of Production Tests:

PT No.	Doc. No.	Title	Status			
			Approved	In Progress	Completed	Final Report
2	80369	Routine Graphite Sample Irradiations	X	X		
3	81478	Routine Monitoring of Graphite Oxidation in N-Reactor	X	X		
6	81339	The Use of Isotope Producing Rods in N-Reactor	X	X		
7	81029	NPR Shield Evaluation; Shield Plug Measurement During Start-up Test N-2				X
8	81327	Coproducer Demonstration Test (1.25%)	X			
9	81609	Low Goal Irradiation of Fuel Elements with Varying Amounts of White Oxide on Surface	X	X		
10	82136	N-Reactor Corrosion Monitoring	X			
11	82074	Fog Spray Control Valves				X
12	82241	Copper Base Alloy Corrosion Program	X	X		
14	82385	Irradiation of Thorium-Uranium Crud Monitor Elements in N-Reactor	X	X		

PT No.	Doc. No.	Title	Status			
			Approved	In Progress	Com- pleted	Final Report
15	82659	Primary Loop Pressure Relief Valves - Relief and Reseating Pressure Test	X	X		HW-83175 (Interim Report)
20	82933	Hydraulic Actuators for Valves			X	
21	83200	Optimization of Secondary Cool- ant Supply Operational Modes	X	X		
21 SUP	83216	Supplemental Testing Procedure, Surge Tank Pressure Control Test			X	
21 SUP	83472	Supplemental Testing Procedure, Turbine Generator Tests			X	
21 SUP	83857	Supplemental Testing Procedure, Pressurizer Level Control Test			X	
23	83389	Horizontal Rod Scram No. 2, Power Ascension Program, N-3			X	HW-84066
29	84239	Full Load Dump Condenser Characteristics				(Test cancelled by RLOO-AEC)
30	84232	Evaluation of Ammonia for Con- trolling Oxygen in the Graphite Cooling System	X	X		
32	84367	Exposure of Corrosion Coupons in N-Reactor Process Tubes	X			
33	84401	In-Reactor Enthalpy Imbalance Measurements	X			
34	RL-NRD-26	Horizontal Rod Scram No. 5, Power Ascension Program, N-3			X	

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OUTAGES

<u>Date</u>	<u>Type</u>	<u>Duration (hours)</u>	<u>Cause</u>
11/5/64	Manual Shutdown	54.6	Failure of PCRV-203-3 valve stem.
11/17/64	Manual Scram Test	81.3	Perform scram test and planned maintenance.
11/24/64	Manual Scram	101.0	Spurious rupture indication on Process Tube 1459.
11/28/64	Manual Scram	3.7	Repair fuel oil leak which occurred during plant start-up attempts.
11/28/64	Manual Scram	3.3	Loss of vacuum to two drive turbines during startup attempts.
11/29/64	Automatic Scram	2.4	Process tube low flow trip from faulty flow monitor component.
11/30/64	Manual Scram	19.9 as of 11/30/64	Failure of valve stem on HPV-107.

FUEL MANUFACTURE AND TECHNOLOGY

PRODUCTION STATISTICS

	<u>Current Month</u>	<u>CYTD</u>
1. <u>Input (Extrusions)</u>		
Outer Billets-0.95%	0	879
0.95% Single	0	8
1.25% Driver	0	85
1.25% Spike	0	378
Inner Billets-0.95%	161	1,086
1.95% Driver	<u>43</u>	<u>43</u>
Total Extrusions	204	2,479
Total Tons	32.8	454.2
% of Forecast	99.3	92.7
2. <u>Output (Finished Production)</u>		
Assemblies-0.95%	972	11,600
1.25% Driver	0	655
1.25% Spike	48	3,125
Normal uranium	<u>540</u>	<u>540</u>
Total Assemblies	1,560	15,920
Total Tons	29.3	311.0
% of Forecast	83.7	88.9
3. <u>Uranium Utilization</u>	73%	73%
4. <u>Shop Yield</u>		
Outers	75.4%	80.7%
Inners	38.4%	78.4%
Total	76.0%	80.0%

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PRODUCTION SUMMARY

Input production for November was 100 percent of forecast. On November 19, the scheduled extrusion of 1.95 percent enriched driver tubes was initiated. These fuel elements are to be used in further development of the coproducer fuel program. No unusual extrusion problems were encountered and the extrusion rate was increased by approximately one third. As a result of this, it is now possible that the 15 extra uranium billets received for this program can be processed within the original extrusion schedule.

Finished fuel assemblies in inventory equal 235 tons or 65 percent of a reactor load.

PROCESS DEVELOPMENT

Re-evaluation of Reject Fuel Elements

Fuel elements rejected during fabrication because of large variation in clad thickness are being re-evaluated. The variations occur mainly as localized clad thickening. The areas with more than the allowable 12 mils variation are no greater than 2 square inches and the variation is gradual. No step transitions in thickness have been observed in the fuel elements being considered for reactor charging.

Acceptance of outer fuel tubes with inner clad thickness variations in the 12 to 21 mil range would increase the first and second load fabrication yield by two to three percent, and would save about \$110,000 in FY-1965.

Studies indicate that these fuel elements could be safely irradiated in the two upstream and two downstream positions of each fuel column in the reactor. Thermal gradients in fuels in these positions are less than one-half the maximum, and the uranium temperatures and exposures would be sufficiently low to avoid excessive swelling.

Fuel Element Contamination

It has been established that most, if not all, of the rejection of fuel elements for braze or weld contamination is the result of a daughter product contamination in the weld rather than uranium metal contamination. This conclusion was reached after retesting of reject fuels at least 60 days after brazing. A new procedure will permit retesting of rejects after uranium daughter product equilibrium has been reached, and will result in rejecting only those fuels that are true weld contamination rejects.

Unbonded End Closure Development

The process for tungsten inert gas (TIG) braze end closure has now been developed for the simulated driver tube geometry as well as the regular N outer tube production model. Ten full length simulated driver tubes have been prepared with TIG braze end closures. No large differences exist between the two geometries except that it is necessary to make a preheat weld pass on the top of the massive driver cap to assure adequate preheating.

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Beryllium-zircaloy alloy wire for TIG braze closure has been successfully produced by the swaging process. Quantity fabrication by this process on a semi-manufacturing basis should result in a cost of about \$1 per foot. The only other potentially practical process is fabrication by multiple coextrusions of bundles of rods. This may produce wire at 75¢ per foot.

1.95% Enriched Uranium Driver Tubes

The coextrusion campaign of the 1.95 enriched uranium billets consisted of the following:

1. Seventeen billets of British alloy (350 ppm Fe, 800 ppm Al) with a .300 inch wall thickness highly worked uranium sleeve inserted inside a standard outer tube billet and the uranium billet assembly welded into a zircaloy subassembly.
2. Sixteen billets of the standard uranium alloy (150 ppm Fe, 100 ppm Si) with .300 inch wall thickness inner uranium sleeves but without zircaloy subassemblies.
3. Thirty-two billets of the standard alloy without inner uranium sleeves or zircaloy subassemblies.

All billets were fabricated from uranium ingots that were either single or triple beta heat-treated before primary extrusion.

Warp

Tests and production data relating to inner fuel element warp have been reviewed. A review of data on warp of inner fuel elements results in the following conclusions:

1. An increase in wall thickness variation (within specification limits) does not increase warp.
2. There is no correlation between position at heat treatment and warp.
3. There is no strong correlation between warp before and after heat treatment.

CONVERSION

Secondary System Surge Tank Simulation

The first runs from the recently completed simulation program point to acceptable stability of the 3-vessel surge tank system. Under conditions of low operating pressures in the secondary system, the surge tank system responded satisfactorily with moderate overshoot and with reasonable time of return to normal setpoints for cases of extreme out-surge and moderate insurge.

OXYGEN CONTROL IN THE GRAPHITE COOLING SYSTEM

Introduction

The graphite cooling system is a low pressure, low temperature recirculating system in which demineralized water with the pH adjusted to 10.0 - 10.2 passes through the active zone of the reactor to cool the graphite, through heat exchangers where the heat removed from the graphite stack is given up to river water, through the graphite cooling system pumps, and back to the reactor. The coolant temperature ranges from 100 to 200 F, and the system pressure is normally about 150 psi. Purity is maintained by feed and bleed with a normal make-up of 40-50 gal/minute in an 8,300 gallon system. The principal materials in the system are 304 stainless steel in the heat exchanger tubes, Zircoloy-2 in the graphite cooling tubes, carbon steel in the piping, and copper or copper alloys in valves, pressure sensing lines, and fittings.

The demineralized water is deoxygenated and has a base added to give a pH of 10.0 to 10.2. The low impurity, low oxygen, and high pH conditions were specified to minimize both corrosion of carbon steel piping and formation of activated crud. The original intent was to operate the system with an oxygen level of 7 ppb, adding hydrogen as necessary to suppress the formation of radiolytic oxygen in the neutron flux of the reactor. Lithium hydroxide was specified as the base to maintain the high pH.

Radiolysis of Coolant Water

As expected, nuclear operation caused radiolytic decomposition of the coolant water with consequent oxygen formation. It was not expected, however, that addition of hydrogen to the system would prove ineffective in

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reducing the oxygen concentration. As power levels increased, oxygen in the graphite cooling system coolant increased to equilibrium concentrations of 50,000 ppb, as compared to the 7 ppb originally intended. Hydrogen added at the specified concentration of 2,200 ppb during low power operation had no effect on oxygen formation. Addition of hydrogen was stopped as the reactor reached higher powers because the hydrogen formed by radiolytic decomposition of the water produced hydrogen concentrations of up to 50,000 ppb in the coolant - twenty times the concentration originally planned - coexisting with the 50,000 ppb oxygen. Hydrogen peroxide, another product of the radiolytic decomposition of water, reached concentrations of up to 100,000 ppb. The addition of hydrazine was ineffective in scavenging the oxygen in the coolant because the hydrazine was radiolytically decomposed to ammonia as soon as it was added. The limit of 500 ppb ammonia in the system imposed to protect the copper and copper alloys precluded the addition of sufficient hydrazine to remove appreciable oxygen.

The basic reactions involved in the radiolytic decomposition of water are:

Recombination Reactions

The recombination reactions, which convert hydrogen and hydrogen peroxide to water by a chain reaction between the H and OH radicals and the molecular H_2 and H_2O_2 , are temperature dependent and are accelerated above about 150°C.

The combination of low coolant temperature, which inhibits the recombination reactions, and the absence of ammonia, which breaks down slightly to hydrazine in a radiation field, probably accounts for the excessively high concentrations of hydrogen peroxide and oxygen observed in the graphite cooling system coolant and not observed in the higher temperature, ammonia conditioned primary system coolant.

Corrosion Effects of Oxygen

Excessively high concentrations of oxygen in the coolant lead to accelerated localized corrosion (crevice or pitting) of the carbon steel piping or fittings. Pits up to 10 to 15 mils in diameter and 5 mils deep have been observed in the 40 mil tubing from one graphite cooling pigtail.

Ammonia for Oxygen Control

At present the only readily available technique that shows promise for controlling oxygen in the graphite cooling system is the addition of ammonia in concentrations of 10 to 20 ppm for pH control. Experiments at low pressure and low temperature have shown that operation at pH 10.0 - 10.2 with ammonia maintained oxygen levels low (80 ppb) as compared to similar operation with lithium hydroxide for pH control where oxygen levels were greater than 15,000 ppb. This has been the only directly applicable experimental data known to us to indicate ammonia's effectiveness under these conditions. At the higher primary system temperatures, it has been demonstrated that oxygen control was possible with either pH control material.

Ammonia was not originally specified for pH control in the graphite cooling system, as it was in the primary system, because of the fear that it would corrode the copper and copper alloys in the system excessively. Since that time, corrosion tests have shown that corrosion of the copper alloy components is no greater in contact with ammoniated pH 10 water than the corrosion in contact with pH 10 lithiated water. The data, although not extensive, indicate that there is little risk of increased damage from uniform corrosion of copper alloys in contact with the liquid coolant as a result of changing from lithium hydroxide to ammonium hydroxide for pH control of the graphite cooling system. This situation is not true for those copper alloys exposed to air into which the ammoniated coolant may leak. Ammonia vapors in air can cause stress cracking of stressed copper alloys. The components most susceptible to this type of damage appear to be the brass fittings that hold the copper pressure sensing lines to the individual graphite cooling tube flow monitor venturis located in the inner rod rooms. Leakage of ammoniated water from one fitting could lead to cracking of other fittings.

Preliminary results from about 20 hours of operation with ammonia (derived from the breakdown of hydrazine) in the coolant showed oxygen and hydrogen peroxide concentrations below their detection limits, 5 ppb and 25 ppb, respectively. The hydrogen concentration in the coolant was high, 30 cc/l, as was expected. If continued operation with ammonia confirms the initial results, and the results of corrosion tests are favorable, ammonia may be adopted permanently for pH control in the graphite coolant.

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TECHNICAL ACTIVITIES

NEW PRODUCTS - COPRODUCT

Physics Design - 1.95 percent Coproduct Test

Several alternate physics designs are being studied for this test loading. The presence of an unshielded 21 tube coproduct block in the reactor would cause a flux depression in the base load. At the same time the outer edges of the block would leave an excessively large power-density. One means of balancing the load would involve the introduction of a buffer layer of 24 natural uranium channels around the coproduct block. Another means involves the addition of a second ring of natural uranium to completely suppress power peaking. Other expedients, such as loading of lithium in the coproduct block, are also being studied.

Coproduct Fuel Meltdown Experiments

Heating tests have been conducted on Al-Li alloys to determine behavior of a target element during a loss of primary coolant - whether the Zircaloy cladding would sustain expansion stresses of the melting core, and the rate at which the Zircaloy would be consumed by the molten lithium and aluminum. Targets were heated by induction coils for about 30 minutes and then cooled. In two of the tests a strong exothermic reaction took place during cooling and a complex reaction product formed. The reaction started at 870 C and the temperature increased to 970 C. Comparison with other tests shows that the reaction is promoted by slow cooling and by the presence of air-oxidized surfaces, rather than vapor blasted surfaces, on the aluminum cans.

Coproduct Hazards Studies

Melting time calculations made earlier on the 1.25 percent test fuel element and on the 1.95 percent twenty pound per foot element indicated that lighter weight fuels would melt in shorter times on total loss of cooling. Subsequent computer runs made to define lower weight limits based on melting time criteria for use in parametric studies contradict the earlier calculations.

The principal factors contributing to the apparent anomaly are the effect of heat capacity of the target and the geometry of the assembly. For a given pressure drop in the active zone, varying the weight of the driver requires changing the size of the target to maintain approximately the same flow cross section in the inner annulus. Thus the lighter weight driver designs involve larger targets having greater heat capacities which increase nearly in direct proportion to the increase in volume. Also, the increase in target diameter results in a decrease in annular gap (thus a decrease in conductive resistance) as well as an increase in radiative heat transfer area.

Coproduct Alloy

The lithium-aluminum alloy for the coproduct demonstration target rods has been primary extruded and coextruded onsite. One hundred forty target assemblies from 20 coextrusions have been processed through assembly and zircaloy welding and are ready for final testing of the zircaloy end closure. Yields are about 65 percent.

NEW PRODUCTS - POLONIUM-210

Potential Methods for Po-210 Production

Four possible techniques have been identified for production of Po-210 in the reactor. These are (1) use of peripheral channels to irradiate metallic bismuth, (2) irradiation of Bi_2O_3 in a single tube, (3) substitution of bismuth for uranium in some of the normal process channels, and (4) irradiation of bismuth in the cross coolant channels. Comparisons of the four cases have not been completed.

NEW TECHNOLOGIC DEVELOPMENTS

Fuel Melting Times

The fuel melting times for a loss of coolant from a break in an inlet header have been recalculated, using more recent values of power decay and graphite temperatures. At a given specific power, the time to melt is about 60 seconds longer - or, for a given melting time, the specific power can be about 25 kw/ft higher than was originally considered possible.

Gamma Analysis of Enrichment Level

It has been demonstrated that a gamma activity analyzer can differentiate between enrichments of unirradiated N type fuel elements. The method uses the 0.184 Mev gamma activity characteristic of U^{235} decay. Design of an instrument for use while loading monotubes is starting.

Graphite Program

Tests were conducted to determine the manner in which the keys on the tube blocks in the reactor might fail as a result of longitudinal contraction of the graphite bars. The test setup simulated in-reactor conditions using

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full-size tube and filler blocks. In the four tests performed, the keys failed in shear at a 45 degree angle about as predicted. In one test, the filler block failed first and the corresponding tube block key failed by wedging rather than shear. In each test considerable separation of the tube block was observed. These test results tend to confirm the theory that the reactor stack would rise rather than fall as a result of contraction of individual bars.

Control Rod Enhancement

Increases in the reactivity worth of the present control rods through the use of B^{10} in place of the boron in B_4C may be impractical because of both cost and shortage of the isotope. The price is \$3 a gram, and the available supply is probably limited to 100 pounds. Replacement would require the equivalent of 2400 pounds of natural boron, or about 3.3 million dollars worth.

ADMINISTRATIVE SUMMARY

NOVEMBER, 1964

EMPLOYMENT

Section	Permanent		Temporary		Total		Change	
	E	NE	E	NE	E	NE	E	NE
General	1	1	-0-	-0-	1	1	--	--
Finance	14	13	-0-	-0-	14	13	-2	--
Fuels	43	98	5	-0-	48	98	-1	--
Plant	64	213	3	1*	67	214	--	-4
Project	69	21	-0-	-0-	69	21	+1	--
Research & Engineering	47	8	-0-	-0-	47	8	+1	--
Employee Relations	5	2	2	-0-	14	2	+2	--
Total employees	243	356	17	1	260	357	+8	-4
Net change						617		+4
*1 Training								

SAFETY

Days without a disabling injury	731
Hours worked without a disabling injury	2,366,193

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VISITORS

<u>Name</u>	<u>Company</u>	<u>Contact(s)</u>	<u>Date(s)</u>	<u>Purpose</u>
T.S. Jerstad	Anchor Equip.Co.	R.C. Walker J.P. Sluka	11/5/64	300 Area Meeting on 17-4 PH Valve Stem Failures
E.E. Denhard	ARMCO Corp.	R.C. Walker J.P. Sluka	11/5/64	" " "
E.R. Hall	ESCO Corp.	R.C. Walker J.P. Sluka	11/5/64	" " "
H.S. Carmichael T.J. Graf	Reactive Metals Ashtubula, Ohio	R.H. Scanlon J.W. Nickolaus T.D. Naylor	11/11/64 11/12/64	Discuss primary extrusion development
E.F. Storer	American Brake Shoe Company	R.D. Crosier	11/11/64	Consult on HPV-201 valves
M. Harwell A. McLeon	Gray Tool Co. Houston, Texas	J.H. Fastabend	11/12/64	Discuss Grayloc leakage
M. Harwell A. McLeod	Gray Tool Co. Houston, Texas		11/12/64	Worked on Grayloc repair
H.S. Isbin (Consultant)	Univ. of Minn. Minneapolis, Minn.		11/12/64	Discuss thermal hydraulics and two-phase flow
H.S. Isbin	Univ. of Minn. Minneapolis, Minn.	R.H. Shoemaker	11/12/64 11/13/64	Discuss heat transfer studies
V.E. Miller	Power & Controls, Inc.	G.R. Klingler	11/13/64	Consult on rupture monitor control valves

VISITORS

<u>Name</u>	<u>Company</u>	<u>Contact(s)</u>	<u>Date(s)</u>	<u>Purpose</u>
D.R. Resner	GE Installation and Service Engineering Portland, Oregon	G.K. Dyekman	11/17/64	Discuss turbines
W.G. Wehmeyer H. Hovila	Engineered Indust. Systems Seattle, Wash.	A.A. Maupin	11/18/64	Discuss bridge cranes under C & D Elevators
H.C. Neilson	Boeing Co. Seattle, Wash.	M.M. Hendrickson	11/23/64	Discuss radiological consequences of possible release of tritium
Mr. Riegel	Eanco Company Chicago, Illinois		11/24/64 11/25/64	Working on the Travelling Wire Flux Monitor

VISITS

<u>Name</u>	<u>Company</u>	<u>Contact(s)</u>	<u>Date(s)</u>	<u>Purpose</u>
B.S. Kosut	ASTM Officers Mtg. Vancouver, B.C.		10/27/64	In conjunction with the Canadian Inst. of Mining & Metallurgy and Pacific Northwest Metals & Minerals Conference
J.R. Bolliger R.E. Trumble	Bonneville Power Auth. Portland, Oregon		10/29/64	Discuss Phase II Hazards Review.
M.C. Leverett R.E. Trumble	ACRS Washington, D.C.		11/5/64 11/12/64	Presentations on N-Reactor production test criteria and coproduct production test
R.C. Hoffman	GE-IPS Schenectady, N.Y.		11/9/64	Attend engineering conferences covering relaying, grounding & instrument transformers
R.C. Hoffman	Nife, Inc. Long Island, N.Y.		11/10/64	Discuss battery chargers, power supply systems and battery supports
R.C. Hoffman	Burns and Roe Hempstead, N.Y.		11/11/64	Discuss conversion data
C.A. Munro	GSA Seattle, Wash.	R.M. Ferbrache	11/10/64 11/11/64	Examine surplus electronic equipment from Bomark Missile Program closeout
R.E. Hall	Philadelphia Elec. Co. Philadelphia, Pa.		11/11/64	Discuss water treatment and storage problems

VISITS

Name	Company	Contact(s)	Date(s)	Purpose
R.E. Hall D.W. Leiby M.C. Leverett R.E. Trumble	Burns and Roe Hempstead, N.Y.	Ivan Gabel	11/11/64	Discussions on N-Reactor conversion
R.E. Hall D.W. Leiby	Bailey Meter Co. Cleveland, Ohio	D. Wurster	11/12/64	Discuss N-Reactor conversion
G.T. Haugland	Combustion Engrg. Chattanooga, Tenn.		11/12/64	Inspect sixth cell steam generators
P.W. Hood	Western Pneumatic Tube Kirkland, Wash.	Lou Calzi	11/16/64	Raw material evaluation
J.L. Benson M.C. Leverett J.W. Riches	ANS-AIF Joint Meeting San Francisco, California		11/30/64 through 12/2/64	Meeting

INVENTIONS AND DISCOVERIES

HW-IR-1316, Inert Gas Welding Torch with No Seals on Water and Gas, by I. V. Nelson

SIGNIFICANT REPORTS

Report Number	Title	Author	Date
HWS-8324	Uncl Specifications for Lithium-Sulphate Ultrasonic Transducers	R.W. Gilmore	November, 1964
HW-80719 9	Secret Fuel Element Production and Inventory Report	H.P. Kraemer	November 6, 1964
HW-83828	Conf N-Reactor Summary Scram Reports	E.E. Leitz	September 30, 1964
HW-83892	Uncl Copper-Base Alloys in the N-Reactor Graphite Cooling System	W.E. Gurwell B.S. Kosut	November 30, 1964
HW-84209	Uncl Upgrading of the Circulating Raw Water System to Increase Operational Reliability	L.G. Henke	November 3, 1964
RL-NRD-13	Uncl The Effect of Goal Exposure on Phase III Electrical Power Generation	R.K. Robinson	November 18, 1964
RL-NRD-32	Uncl Preliminary Results and Recommendations on Copper-Base Alloys in the N-Reactor Primary System	B.S. Kosut	October 28, 1964
RL-NRD-35	Uncl Monthly Activities Report	C.A. Munro	October 29, 1964
RL-NRD-51	Uncl Specifications for Uranium Metal Billets for N-Reactor Fuel Elements	R.H. Scanlon	November 16, 1964
RL-NRD-52	Conf Chemical Specifications for Uranium Metal Billets for N-Reactor Fuel Elements	R.H. Scanlon	November 16, 1964
RL-NRD-56	Secret Proposal for Single Tube Fuel Loadings	L.M. Loeb	November 6, 1964

SIGNIFICANT REPORTS (Continued)

Report Number	Title	Author(s)	Date
RL-NRD-57	Uncl N-Reactor Immersion and Clamp-On RTD's	J.H. Fastabend	November 23, 1964
RL-NRD-59	Conf Predicted N-Reactor Plant Heat Balances	V.J. DeJong A.J. Ebens B.E. Smith	November 6, 1964
RL-NRD-60	Secret Reactor Design Analysis Monthly Record Report	H.R. Kosmata	November 9, 1964
RL-NRD-61	Secret Summary Report: N-Fuel Failure #1	L.M. Loeb	November 13, 1964
RL-NRD-62	Uncl 17-4 PH Valve Stem Failures in N-Reactor Primary Coolant System	R.C. Walker J.P. Sluka	November 21, 1964
RL-NRD-64	Conf N-Reactor Primary Loop Blowdown from Main Header Ruptures - Phase II	D.D. Stepnewski	November 12, 1964
RL-NRD-68	Uncl Explosion & Fire Hazards in 100-N Area Compressed Air Systems	G.K. Dyekman	November 16, 1964
RL-NRD-69	Secret Shop Yield	B.R. Black	November 14, 1964
RL-NRD-72	Uncl Modification of the 105-N V-23 Valves Control System to Nullify an Observed Operational Deficiency	L.G. Henke	November 17, 1964
RL-NRD-77	Uncl 109-N Dump Condenser Number 1 - Interim Inspection Report	J.W. Helton	November 23, 1964
RL-NRD-78	Conf Plans for Improvement of the Fully Brazed End Closure	W.M. Scott	November 19, 1964
RL-NRD-83	Uncl Confinement System Stack Monitoring Modification Requirements	J.W. Vanderbeek	November 23, 1964
RL-NRD-86 RD-2	Uncl 100-N Conversion - Construction-Plant Forces Work Definition - Electrical and Instrument Systems	J.D. Fogelquist	November 12, 1964

SUGGESTION PLAN PARTICIPATION

	<u>NOVEMBER</u>	<u>CY 1964 TO DATE</u>
Number of Eligible Employees	357	363
Number of Suggestions Received	58	557
Number of Suggestions Acted Upon	31	501
Number of Suggestions Adopted	7	188
Net Annual Savings	\$ 326	\$134,546
Amount of Awards	\$ 105	\$ 7,760
Percent of Awards to Savings	32.2	5.8
Average Amount of Awards	\$15.00	\$ 42.29

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