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N - REACTOR DEPARTMENT

MONTHLY REPORT

JULY

1964

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HANFORD ATOMIC PRODUCTS OPERATION

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N-REACTOR DEPARTMENT
MONTHLY REPORT - JULY, 1964

By Authority of CG-PR-2

RM Item 8 27-92

By J. Tang 9-4-92

PM Eck 9-8-92

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General Manager

Work performed under Contract No. AT(45-1) - 1350 between
the Atomic Energy Commission and General Electric Company.

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TABLE OF CONTENTS

Operating summary	4
Summary	6
Reactor plant operation	7
Fuel Manufacture and Technology	21
Plant Modifications	33
Conversion	34
Results of Startup Testing	35
Technical Activities	41
Administrative Summary	47
Index	53

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HW-80559 7

OPERATING SUMMARY

Input production Pu (KMWD)	18.1
Input production T (KMWD)	0.
Maximum power level attained (MW)	1,600
Average power level while operating (MW)	1,140
Time operated efficiency (%)	56.9
Plant utilization (%)	62.6
Number of Shutdowns	
Scheduled	0
Unscheduled	11
Reactor Process Data	
Power at which computed (MW)	1,600
Primary coolant inlet temperature (°F)	350
Primary coolant outlet temperature (°F)	420
Maximum graphite temperature (°F)	650
Heat Loss	
Graphite cooling (MW)	26.8
Shield cooling (MW)	1.2
Rod cooling (MW)	1.0
Steam temperature (°F)	328
Steam pressure (PSIG)	86
Dump condenser flow (gpm)	164,000
Inlet temperature (°F)	64.5
Outlet temperature (°F)	139
Nuclear Steam Generating Rate	
Maximum (1000's lb/hr)	4,940
Average while operating at 1600 MW (1000's lb/hr)	4,895
Water Collected from Gas Atmosphere (gal/day)	Negligible
Demineralized Water Produced (gal/min.)	1,200
Helium Losses (1000's cu. ft.)	612
Fuel Oil Usage (1000's gal.)	1,321
Flattening Efficiency (%)	69

Fuel Conditions

Heat flux

Maximum (BTU/hr/ft. ²)	272,000
Average (BTU/hr/ft. ²)	185,000

Clad temperature

Maximum (°F)	465
Average (°F)	430

Uranium temperature

Maximum (°F)	610
Average (°F)	525

Fuel Failures

1 - Tube 3361

Fuel Balance	<u>.094</u>	<u>1.25</u>	<u>Co-Prod</u>	<u>Other</u>	<u>Total</u>
Fuel charged (Tons)	6.23	17.66	0	0	23.89
Fuel discharged (Tons)	23.82	0	0	0	23.82
Net Change (Tons)	-17.59	+17.66	0	0	0.07
Total in reactor at month end (Tons)	350.26	17.66	0	0	367.92

Fuel Element Output

0.947%	216
1.25% (driver)	324
1.25% (spike)	0
1.25%+0.947% (spike)	<u>1,008</u>
Total assemblies	1,548
Total tons	33.5
% of forecast	112

Uranium Utilization

70%

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HW-80559 7

SUMMARY

Power ascension and testing continued, with the power level being increased first from 800 to 1200 MW, and then to 1600 MW. Enriched fuel (1.25%) has been charged in 64 columns to provide the reactivity to operate at 2000 MW. The observed reactivity increase is $0.15 \pm 0.05\%$. The plant operated smoothly except for a number of shutdowns, several caused by electrical grounds in the control system for heat dissipation, and others by failures of valve actuators on dump condensers.

The first failure of a fuel element occurred when an end cap on an inner element ruptured. Both gross gamma and the gamma energy rupture detection systems responded as expected to the event, and the reactor was shut down. Examination of the fuel element disclosed that water gained entry through a pinhole in the end cap weld. No basic fault is indicated in fuel element design, materials or manufacturing technique, and repetitions should be readily preventable in the future.

A planned scram from 1200 MW demonstrated highly satisfactory performance of the "discontinuous actions" sequence involving steam generator bypass valves, dump condenser valves, boiler response to steam demand and the make up system for the surge tank.

A two group diffusion model used to calculate k_{∞} for the standard fuel from control rod data obtained at startup gave a value in good agreement with that obtained from the Physical Constants Testing Reactor.

Work has been in progress for several months on steam generator 4A, removing 100,000 ft. of tubing with intergranular corrosion, and installing new Inconel tubing. The retubing has been completed and the head is now being welded to the shell.

Nuclear safety specifications have been prepared for all aspects of handling fuel with up to 1.95 percent enrichment.

OPERATING PATTERN

Observations - Plant Behavior

Power ascension and N-3 testing continued during July with power level increasing from 800 to 1200 MW, and further to 1600 MW. Spike enrichment fuel was charged in a total of 64 enriched fuel columns to increase reactor reactivity for operation at higher power levels.

Total plant operation has been generally satisfactory although there have been some equipment failures and malfunctions. Several shut downs were occasioned by electrical grounds in the 125 volt DC electrical control system for the plant heat dissipation system. Several other shut down's were caused by heat dissipation system upsets caused by valve actuator failures on individual dump condensers. The causes for these malfunctions are understood and remedial actions to improve equipment component integrity are in progress.

Fuel Element Failure

A fuel element failure, the initial fuel failure for N Reactor, occurred on July 7 when the end cap on the inner element of a fuel element ruptured in tube 3361. The failure was detected by the rupture monitor system and reactor operation was suspended as soon as the presence of the failure was unmistakable. Both the gross gamma and the gamma energy rupture detection systems responded as expected. This event occasioned the first use of the process tube diversion system under actual operating conditions. This system also worked as expected. Following the reactor shut down, the rupture was discharged without difficulty. (Examination of the fuel is described on pp. 25-32.)

Although the fuel element failure resulted in some contamination of the primary loop, no chemical decontamination of the piping was necessary and the level of radiation is regarded as not posing any hazard to operating personnel near the loop.

While the reactor was shut down for removal of the failed fuel element, use of the outage was made for a planned charging of spike fuel.

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HW-80559 7

Planned Scram

A planned scram test from 1200 MW demonstrated the best operation of the automatic "discontinuous actions" sequence yet observed. Upon receipt of the scram signal, the steam generator by-pass valves opened in the four cells which were in use. As steam header pressure declined the dump condenser steam admission valves went closed, and the boiler, initially operating at 250,000 lb/hr increased its steaming rate over a three-to-four minute period by almost a factor of two. Closing of the dump condenser steam admission valves caused the dump condensers to fill with water from the surge tank; lowering of surge tank water level brought into play make-up water, thus maintaining surge tank level within appropriate limits. The systems for controlling the water level in the steam generator performed satisfactorily, although it may be desirable to effect some reduction in the lag time between the signal for level and the injection of feedwater.

Spike Charges

The estimated excess reactivity held in the control rods at 1200 MW is 7 mk. The predicted excess reactivity at 1600 MW is 4 mk. Twenty-five spike channels have been charged into the reactor, making a total of 64, as previously mentioned, to provide the additional enrichment required to go to about 2000 MW.

Process Tube Monitoring

Mention was made in the June Report on page 8 of work being done in the Process Component Experimental Loop on the problem of chattering of dummies in process tubes. During the reactor outage of the week of July 13, process tubes in channels 2845 and 2946 were borescoped over the front and rear dummy sections to determine the effect of rattling of the dummy spacers. The front two dummies removed from these tubes showed considerable damage to the dummy supports. However, borescope observations revealed no evidence of damaged oxide film on the process tubes. Bright rings were observed at support locations and were interpreted as evidence of rotation of the dummies. Continuation of this rotation and the resultant burnishing

of the tube surface would probably lead to tube damage. Such damage is not occurring at a rapid rate, however, and corrective action is being taken. Production testing of the effect of removing the upstream spacers is now under way as the first step in the evaluation. (See page 18.)

Secondary System Oscillations

On July 12, while the reactor was operating, the oscillation of the condensate feedwater flow to the steam generator was essentially eliminated by manipulation of controller and time lag settings. As a result, steam flow from the generator smoothed out and overall system performance improved. Work is continuing to improve the level control system of the steam generator, to determine limitations in controller settings and to improve the response to small disturbances. Analog simulation studies will provide a basis for manipulating controller settings during reactor operation.

No apparent deficiencies in design have been uncovered, and only tuneup of the system appears necessary. (See also Production Test PT-NR-21 on page 18.)

Sensitivity of liquid levels in the surge tank and condensers has continued to be a problem, and careful operation, without sudden changes of important valve settings, is necessary if scrams due to low surge tank level are to be avoided. Corrective measures have been outlined and their installation is scheduled.

Boiler Operation

Status of the plant auxiliary boiler did not change significantly during the month. Operation at levels such that seven burners are on is still required, since the ability of the boiler to ramp up from idling steaming rates has not yet been demonstrated. However, the boiler was not responsible for any operating problems this month.

Confinement Fog Spray Control

In order to provide increased assurance of continuity of fission

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HW-80559 7

product rain-out by the fog spray and, at the same time, keep the Zone I driving force pressure at a minimum, it appears desirable to restrict fog spray shutoff to manual control. For this to be done safely, the vacuum venting capacity must equal or exceed the fog spray condensing capacity. At a negative pressure in Zone I of -0.5 psig, the calculated capacity of the Zone I vacuum relief valves is 177,000 cfm, and the estimated reverse flow capacity of the two banks of filters is 192,500 cfm. The limiting fog spray condensing capacity is 302,000 cfm. Thus, together, the vents and filters will prevent the possibility of damage to the filters by excess negative pressure caused by continuation of condensation by the fog spray.

Fog Spray Accumulator Relief Valve

The discharge from the fog spray accumulator relief valve is to be routed to a drain in the basement to prevent recurrence of severe flooding outside the building at times when the pumps are running with low demand on the system.

PROCESS PHYSICS

Control Evaluations

A two-group diffusion model has been employed in preliminary analyses of the control rod experiments conducted at startup. To-date this model has been used to calculate a $k_{\infty} = 1.065$ for the standard NPR fuel and a $k_{\infty} = 1.139$ for the spike fuel. The value for the standard fuel is in good agreement with that obtained in PCTR experiments (1.062 ± 0.004) and in fair agreement with the startup results of 1.055 ± 0.004 . This analysis is encouraging and justifies an investigation of the control rod experiments. This work is under way.

A 3-region cylindricized geometric model of the NPR lattice cell was employed in these calculations. The central region radius extends to the midpoint of the water annulus between the two-fuel annuli. The first annular region extends from that point to the outer surface of the process tube. The outer annulus extends to the outer boundary of the cylindricized

cell. Each region is homogenized for calculation of the respective, average, equilibrium neutron spectra. Group constants for each of the three regions are obtained by averaging microscopic cross sections over the respective spectra.

Graphite Temperature Evaluation

Preliminary calculations were made with the IBM N-Heat Program for low power level operation. The calculated temperatures were within 100 F of the measured temperatures. Newly calculated film coefficients and contact coefficients will be used to bring the calculated values into even closer agreement with the measured temperatures, and the modified program will be used to estimate graphite temperatures in advance of actual power level increases.

MECHANICAL EQUIPMENT - REPAIR AND MAINTENANCE

Approximately 1400 scheduled jobs and 500 unscheduled jobs were completed during the month. Equipment failures occurred in locations where trouble had not been previously experienced and repair or modification jobs of major scope required the assistance of the construction contractor, J. A. Jones.

Repair of Steam Generator 4A

Repair of steam generator 4A continued, with installation of the last of new Inconel tubing to replace the corroded stainless steel tubing. The 7-1/2 ton vessel head was lifted into position on July 24, and welding of the head to the shell is now in progress.

At a meeting of representatives from the vendor, the construction contractor, the Atomic Energy Commission and General Electric Co., presentations were given by two corrosion consultants regarding test results and analyses of probable causes of intergranular corrosion in all of the steam generator units and stress corrosion cracking in unit 3B. These technical presentations were given to provide information pertinent to negotiations regarding repair costs and future warranties.

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HW-89559 7

Rupture Monitor System

The right side rupture monitor pressure-differential control valve began leaking through its body casting. Investigation revealed erosion of the body metal, immediately downstream of the valve seat, to have occurred to the point of complete wall penetration. Inspection of the left side valve revealed similar erosion attack. Both valves were weld-repaired and reversed in the line when reinstalled. Smaller valves with stellited throats were ordered as replacements.

Rupture monitor sample line transport time data were reviewed to determine what transport times fall outside of the 60 (± 6) second limit. Four sample lines in the left turret room and two sample lines in the right turret room had transport times in excess of 70 seconds. Drain headers for each line found outside the limits will be relocated to re-establish the correct transport time.

An increased reading on the left side diversion header was detected by the gross gamma monitor. Investigation revealed that gross gamma readings on the right and left side diversion headers are related to reactor heat shifts. If reactor heat shifts from the right to the left side, the gross gamma reading for the left side diversion header increases.

Rupture Monitor Toggle Valve Assembly

The rupture monitor toggle valves have been leaking and a suitable method of repairing them is being investigated. A device for removing the valve stem liner has been developed and may prove a satisfactory solution to the problem.

Horizontal Rod Suction Lines

An equipment modification procedure is being prepared to reroute suction lines of the rod drive low-volume pumps in the outer rod rooms to prevent air locks from occurring in the lines. The air locks cause the pumps to lose prime and to knock severely, resulting in unnecessary wear on moving parts.

Fog Spray Control Valve Replacement

The ten 105-N fog spray control valves were replaced with similar valves of higher pressure rating in order to prevent seat blowout problems that had been previously experienced. A test line will be installed to provide means for routine testing of a control valve and of fog spray nozzles. This latter modification will require an extended outage.

Valve Actuators

A series of problems developed in the control and hydraulic system for actuators of the 16" valves which control steam flow to the 16 dump condensers from the 46" steam header on the roof of 109 Building. These problems cause malfunction of the valves and require removing the dump condensers from operation. A corrective program was outlined and action started to upgrade these systems for reliable operation. Installation of the auxiliary steam line to the actuators is about 80 percent complete.

Primary Pump Shaft Seal Confinement

Work was initiated to provide a felt wiper-type seal on No. 4 primary pump shaft in place of the existing confinement water seal. The same type of seal will be installed on the other units during the next outage if tests on No. 4 prove satisfactory. The new seal eliminates the breakdown bushing and also alleviates shaft alignment problems.

CRWS Discharge Valves

Production testing of these valves in the river pump house circulating raw water system has continued at specified intervals throughout the month with no hydraulic system malfunctions to date. In one instance, a valve refused to open. A portion of the neoprene seat had dislodged and caused the butterfly to wedge. The total elapsed time between actuations of a valve has been gradually increased, with 17 days being the longest interval to date. The interval between actuations will be increased until the valves exhibit the "sticking" previously experienced. (See also Production Test, PT-NR-20, on page 17.)

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HW-80559

Rupture Monitor Differential Pressure Control

The rupture monitor system pressure differential control system must be improved to provide greater stability and accuracy. Recent leakage experience with the differential pressure valves points up problems with the size of the valves as well as with their control. Smaller sized valves were ordered for trial application.

Secondary System Valves

A program was initiated to improve the vibration and contaminated hydraulic oil problems associated with the control system components of the high pressure reducing valves. Procedures and corrective instructions have been issued to cover hydraulic oil cleanup, potentiometer mount, hydraulic pressure switches, and relocation of the pressure gages. Twenty-micron filter assemblies and cartridges have been ordered for use in system cleanup and to replace the existing cartridges following cleanup.

Cell 3 Valve

The stem in the 24" block valve in the cell 3 discharge line from No. 3 primary pump broke inside the valve housing above the disc. Isolation of a section of primary system and disassembly of this valve was required. The loose seat was removed. The loop integrity was restored so that No. 3 cell could be operated without an outlet block valve, since delivery time for the replacement stem is four to six weeks.

Header-to-Header Components

Inspection of the inlet and outlet connector arrangements revealed leakage from the stem packing of the inlet valve for channel 1971. Diversion leg Grayloc's were found leaking at the diversion valve locations for channels 1772, 1170, 1265, 3264, and 1161. Repair of these leaks, using the header arm freeze technique, has been scheduled for the August outage.

Fuel Handling

Work on underwater fuel handling equipment, and on activities involved

with metal handling as a result of enrichment charging, brought to light a significant weakness in the electrical system of the segregation area crane. Causes of malfunction have been worked on continuously and the problem has been partially solved. Side links of the underwater fuel handling conveyor chain are being replaced with new less brittle material.

Helium Loss

The loss of helium gas from the unit has been of concern and a survey indicated leakage in the vicinity of the rear face. A program to tighten the bolts on the rear nozzle gas seals of all process channels was completed.

Confinement Solenoid Valves

Intermittent sticking of the three-way solenoid valve in the confinement valve hydraulic system has been experienced and efforts are being made to correct this problem.

Water Leak

During water-to-water operation of the secondary system a leak developed under the basement floor of the 109 Building, beneath the turbine bay area adjacent to cell 2. The leak was in a 12" line used in the water-to-water mode of operation. The construction contractor is excavating, and making repairs.

Vibration Snubbers

Approximately 30 per cent of the vibration snubbers fastened to the 28" steam discharge lines (roof of 109 Building) from the steam generators in the 109 cells have failed from movement of the 28" pipe lines. This problem is being studied and repairs to the snubbers are underway.

Instrument Air

The instrument air system in the reactor building failed because of inadequate lubrication of four-way plug valves on the Electrodryer. Recommendations have been made to improve the reliability of the system.

Sump Pumps

Difficulties with sump pump operation in the steam generator cells have caused the cell sumps to fill to a deep enough level for the water to boil ~~and~~ contact with some primary loop piping. This condition has

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HW-80559 71

in turn resulted in the generation of large quantities of water vapor within the cells and the pipe gallery causing higher than normal cell humidities and many of the electrical ground faults that have been experienced. Modifications to the pump shaft bearings, which appear to be the main weakness in pumps, are in progress.

Water Quality

Work continues to place the remainder of the water quality sampling and analysis systems into operation, to correct initial operating difficulties and to establish more reliable and consistent sample analysis results.

ELECTRICAL EQUIPMENT AND INSTRUMENTATION - REPAIR AND MAINTENANCE

125 V DC Ground Detector

Detection of grounds in the 125 V DC instrument power supply system has proven important; and an improved detector has been developed and is being prepared for installation. The principal superiority of the new detector is that it is able to detect a grounded condition without providing a relatively low impedance path to ground through itself.

AK-2 Breakers

The undervoltage solenoid for tripping the AK-2 breaker can be improved. The certification testing of modified breaker operation has been completed on the two spare breakers. Each breaker was tripped 300 times using the undervoltage trip mechanism and 300 times using the shunt trip mechanism, without a failure to trip. The trip times are averaging approximately 40 milliseconds for one breaker and 60 milliseconds for the other, which indicates acceptable response.

Afterheat Removal Pump Controls

The starting circuit on after heat removal pump No. 1 was changed by placing a contact in series with the automatic start feature so that the pump can be restarted after a manual trip of the breaker when the auto-start is still being demanded. It had previously been found that one sequence of operations could result in failure of this pump to start upon demand.

Safety Circuit Improvement

Two modifications improving the reliability of the safety circuit were completed. The circuit starting the timers which initiate a ball 3X trip on reactor supercritical with rods in, was modified to start the timers on a rod scram signal rather than a rod-in signal. The rod cooling water low flow trip circuit was modified to eliminate breaking of the safety circuit by relay contact bounce.

Flux Monitor Chambers

Some erratic operation of individual chambers in the nuclear high level flux monitor system was experienced during the month. Two of the twenty-four chambers have failed, one in the reactor and one in the galvanometer summing circuit, two others still in service are unstable. The failed chambers will be replaced as soon as spares are available; and correction of system difficulties continues as time permits during reactor outages.

SPECIAL TESTS PERFORMED

Plant testing continued with the performance of the following:

PT-NR-2 - Routine Graphite Sample Irradiations

A full complement of samples was charged into a process channel to monitor oxidation by and reaction with O_2 , CO_2 and other gases of the reactor atmosphere. The test is part of an overall problem of prescribing operating conditions for the reactor graphite.

PT-NR-17 - Horizontal Rod Scram No. 1, Power Ascension Program

This test was successfully completed. System response was essentially as predicted by previous studies. The boiler plant service and process systems ramped and responded as designed, such that the post-scram conditions obtained were as expected. The summary report is nearly completed.

PT-NR-20 - Hydraulic Actuators for RWSV 805 Valves

Function testing of the 66" discharge valves on all four large

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HW-80559 7

river pumps was conducted. During a test of the No. 1 pump valve, a reactor scram occurred after the valve was closed, through actuation of the low pressure monitoring circuit on the No. 3 pump. This circuit monitors all operating pumps. Further testing of these valves during reactor operation has been temporarily suspended until the circuit trip settings can be analyzed and reset to values which will allow testing during operation.

PT-NR-21 - Optimization of Secondary Coolant System Operation Modes

This test authorizes the connection of auxiliary recorders to control circuits to facilitate adjustment. Under the provisions of this test adjustments were made to achieve better control system response for steam generator liquid level control and improved surge tank liquid level stability. Signals were recorded from the level controllers and new, simulator-generated settings were applied to them. The system response was noted and the settings were optimized for overall system control. Initial investigation of the control response in the surge tank level instrumentation was begun.

PT-NR-22 - Modified Process Tube Spacer Patterns

Front dummy spacer patterns were altered in six fuel columns for an operating period to see if this affected the chattering or rotation of the dummies noted early in the month. After the completion of this operating period, these dummies were removed and inspected. Evidence of rotational motion was still present. Evaluation of this problem is being expedited by a series of out-of-reactor test loop experiments.

189-D Loop

Operational responsibility for the N-Reactor Process Component Evaluation Loop in 189-D Building was transferred to N-Reactor Department on July 1. Since then the loop has operated 553 hours, mostly on spacer chattering tests and environmental tests.

OUTAGES

Date	Time	Duration Hours	Type	Explanation
*6/30	0912	2.4	Scram	HPV-201-6 valve closed after hydraulic pump had failed and pressure in hydraulic system had bled down.
*6/30	1357	2.5	Scram	Dump condenser No. 2 gave indication of having broken tube.
*6/30	1929	3.6	Scram	Reactor started down to take No. 1 primary pump turbine off the line because of overheated thrust bearing and drive shaft vibration.
*6/30	2341		Shutdown	To allow putting No. 1 turbine and pump back into reactor loop.
7/5	2210	13.9	Unplanned Outage	Loss of 105 instrument air supply which caused graphite coolant system to dump and go on once-through raw water cooling.
7/6	1208	1.9	Scram	Low surge tank level.
7/7	0745	117.5	Shutdown	Ruptured fuel element in tube 3361.
7/13	1059	2.5	Scram	Low surge tank level due to opening and closing of dump condenser steam admittance valve.
7/15	0927	1.9	Scram	Process tube flow monitor trip.

* Not included in statistics in Operating Summary

OUTAGES - continued

Date	Time	Duration Hours	Type	Explanation
7/17	1030	21.7	Shutdown	Ground on a bus of 125 V DC system in 109 Building. At the same time a planned scram from 1200 MW was carried out to test the performance of the "discontinuous actions" sequence.
7/18	1207	16.8	Shutdown	Steam leakage in 109, and 109 electrical grounds.
7/21	1011	3.4	Scram	Pressure trip on No. 3 river pump while testing No. 1 river pump discharge valve.
7/22	0537	5.6	Scram	Low surge tank level when HPR-201-13 (secondary system modulating steam dissipation control valve) opened wide causing secondary system pressure to cycle.
7/22	2028	9.6	Shutdown	Flow discrepancy in Cell 3 traced to broken cell isolation valve stem.
7/26	2257	36.9	Shutdown	Faulty temperature detector on process tube charged during outage.

FUEL MANUFACTURE AND TECHNOLOGY

PRODUCTION STATISTICS

	<u>Current Month</u>	<u>CYTD</u>
1. <u>Input (extrusions)</u>		
Outer Billets - 0.95%	30	495
1.25% Driver	0	85
1.25% Spike	156	355
Inner Billets - 0.95%	<u>81</u>	<u>596</u>
Total Extrusions	267	1511
Total Tons	51.5	280.9
% of Forecast	99	89
2. <u>Output (Finished Production)</u>		
Assemblies - 0.95%	216	8252
1.25% Driver	324	324
1.25% Spike	0	0
1.25% + 0.947% Spike	<u>1008</u>	<u>1355</u>
Total Assemblies	1548	9931
Total Tons	33.5	191.3
% of Forecast	112	91
3. <u>Uranium Receipts</u>	<u>Current Month</u>	
Total Billets	<u>473</u>	
Total Tons	<u>83.9</u>	
4. <u>Uranium Utilization</u>	70%	71%
5. <u>Shop Yield</u>		
Outers	<u>82.3%</u>	<u>83.0%</u>
Inners	<u>76.3%</u>	<u>83.2%</u>
Total	<u>79.7%</u>	<u>83.0%</u>

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HW-80550-7

FUEL MANUFACTURE AND TECHNOLOGY

PRODUCTION SUMMARY

Input production for July was almost exclusively devoted to extrusions required for the spike enrichment program, namely, 1.25 per cent enriched outer billets (now 95% completed) and normal .95 per cent enriched inner billets. An additional 30 outer billets of .95 enrichment were extruded under an engineering program of evaluating the use of tipped mandrels in primary extrusions. It is expected that this type of extrusion will provide more uniform material structure near the inner surface of the extruded piece and will further improve clad thickness variability. Input tonnage for the month was 98 per cent of forecast.

Finished production consisted primarily of spike assemblies and co-producer test elements. The target elements for assembly with the co-producer driver tube were received from Hanford Laboratories, and assembly of these elements is approximately 75 per cent complete. Finished production was 100 percent of forecast.

Finished fuels in storage amounted to approximately one-third of a reactor load. Deliveries to the reactor site consisted of 1600 1.25 per cent enriched spike fuel elements and 400 co-producer fuels. The remainder of both the spike assemblies and the co-producer fuels will be shipped to the reactor site in early August.

The feedsite completed the shipment of 1.25 per cent enriched billets by mid-month. Current inventory of onsite billets has increased to approximately 3-1/2 months supply. Although this inventory level is somewhat above normal, it will enable the feedsite to devote the major part of their billet production effort during the next two months to producing 50 billets of 1.95 per cent enrichment, which have been requested for delivery during September-October.

This material will be extruded for use in a small demonstration loading of co-producer fuel.

PROCESS DEVELOPMENT

Unbonded End Closure

Work has continued on the development of an unbonded tungsten-inert-gas braze end closure. The current problem area is that of obtaining proper centering between the V on the bottom of the chevron cap and the V-shaped recess in the uranium without imposing excessively restrictive machining tolerances. This centering can probably be obtained by providing a radius in the root of the notch in the uranium.

Uranium Oxide Burner

The uranium oxide burner operated on a two-shift basis and the backlog of lathe turnings has been reduced from about eighty barrels to a more normal level of ten barrels. Replacement of the circulating water pump in the air exhaust system and mechanical overhaul of the burning drum have eliminated downtime from faulty equipment. It has been demonstrated that the oxide burner facility is capable of handling the increased load imposed by the change from chemical milling to machining of the end recess. Large quantities of 1.25 enriched lathe turnings that have been accumulated by Hanford Laboratories will be burned to oxide in early August.

Hangers for Heat Treatment

Evaluation of the new hangers used for inner fuels in the heat treating operation indicates no significant difference in warp reject rate between the test and the control materials.

Braze Rings

An order of 20,000 sets (outer and inner rings) of beryllium-zircaloy braze rings has been placed. The bid price of each set was 14 per cent below the previous order price and 33 per cent below the purchase price of one year ago. This reduction represents a savings of \$17,000 on the current order, or about \$41,000 annually.

Fuel Supports

Routine deliveries of 50 mil thick outer supports of satisfactory

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HW-80559 7

quality are now being received. It appears that the vendor can supply zircaloy strip with sufficient ductility to permit forming without cracking.

A sample of the inner support strip obtained from the vendor indicated that it was equal in quality to the material rolled onsite. If a large scale fabrication test yields material of acceptable quality, production quantities of inner support strip will be sought from the vendor.

Uranium Billets for Co-Product Demonstration

Detailed specifications have been developed and an order placed for 12-1/2 tons of 1.95 per cent enriched uranium billets for fabrication of the co-product driver fuel elements. Recent irradiation test results indicate that fuel elements made from uranium containing 800 ppm of aluminum and 300 ppm of iron exhibit less swelling than fuels made from the standard alloy (nominally 150 ppm iron, and 100 ppm silicon). Sufficient uranium to fabricate one-half of the fuel elements needed as part of the reactor charge for co-product fuel will be made from this new iron-aluminum alloy. The remainder of the fuels for this reactor charge and all of the fuel for the exponential pile testing will be fabricated from the standard alloy.

Uranium Specifications

The specifications for uranium billets have been revised to incorporate a shorter length Class III billet. Up to 5 per cent of each geometry of the billets shipped during any one month may be two inches shorter than the nominal length of Class I billets. These short billets occasionally occur at the feedsite because of mechanical defects in the primary extrusion surface, or because additional cropping of an ingot is required. This change will result in a reduction in cost to the feedsite. The feedsite cost of scrapping a short billet is \$75 for the N-Reactor inner element and \$110 for the N-Reactor outer element billet. The prorated cost of co-extruding the shorter billets is \$58 and \$91 for the N-Reactor inner element and N-Reactor outer element billets, respectively.

New Process Sequence

Development work indicates that it may be desirable to heat treat the extruded fuels immediately after cut-off, ~~in place~~ with the copper cladding still on the fuel. This change in process would eliminate the abrasive cleaning operation, and on the basis of theoretical analysis and minor work with the NOE fuel element, would introduce no metallurgical problems any more difficult or complex than those in the current process.

1.95 Percent Driver Tube Status

Designs for 7-1/2 and 8-inch container liners for components for all billet sizes, and for end cap and braze rings have been completed. Standards for clad thickness have been written. Extrusion press tooling has been designed and ordered for coextrusion and extrusion of inner zircaloy components. Fabrication of forged zircaloy-2 supports has been started by Hanford Laboratories. Design modification to the 333 Building for the 1.95 percent production run is 10 per cent complete.

FUEL FAILURE INVESTIGATION

A fuel element failure occurred in N-Reactor on July 7. The rupture was on the upstream end of the inner element of the most downstream fuel assembly in tube 3361. The exposure was only 33 MWD/T, less than 2.5 percent of target. The uranium temperature was 410°F. Elements tested in the KER loops have successfully surpassed 3100 MWD/T, temperatures above 800°F, and power levels averaging more than full design. Subsequent examination of the failed fuel element made it apparent that the failure is not due to any inherent, generally present, defect in the fuel.

The entire tube 3361 contained fuel elements with contaminated brazes on one or both ends of either inner or outer elements. The braze on one end of the ruptured element contained uranium contamination in the range of 5,000-10,000 ppm; however, it was the uncontaminated end which suffered the rupture.

Visual examination of the rupture showed a circumferential crack

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HW-80559 7

FACING FIGURE 1

Views of the inner element as it was separated from the outer element.

The cap is almost completely detached by a crack at its base and the cracks formed by the blisters.

Figure 1

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HW-80559 7

FACING FIGURE 2

Top side of the cap.

The path of water was through the tiny blow hole at 5 o'clock on the inner weld, very near the transverse surface of the cap.

The two cracks in the outer weld are thought to have been caused by bending forces exerted by oxide formation under the cap.

Other apparent cracks are mostly surface discolorations.

HW-80559 7

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Figure 2

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[REDACTED]

HW-80559 7

FACING FIGURE 3

The blowhole and three microcracks.

The edge of the cap may be seen at upper right.

[REDACTED]

HW-80559 7

X 150

Figure 3

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HW-80559 7

in the outer cladding at the base of the cap; the crack extended over 300° of the circumference. At one end, the crack changed direction and ran longitudinally down the outer cladding for 1 1/2 inch, and then changed direction and ran circumferentially for 30°. (See Fig. 1.) It left the cap in contact with the outer clad along two 1/2 inch lengths - one circumferential and one longitudinal. Examination of the bore of the inner tube showed it to be full of Zircaloy cladding debris, and direct sight through the bore was blocked.

Examination of the cross-section below the damaged area showed nothing unusual in the cladding, the core, or the uranium-to-clad bond.

Two small spots of uranium metal were found adhering to the bottom side of the cap and they were opposite each other. These pieces discount the possible mechanism that the failure was caused by cap-to-core shear of an unbonded end cap.

The defects were followed by successive light grinding, polishing, etching and inspection to determine the depth of penetration. What appeared to be a tiny blowhole (See Fig. 2) on the inner weld adjacent to a stained area on the cap surface was seen to become gross gas void in the braze area and the void was connected to a flaw in the cap (See Fig. 3) by a series of microcracks.

As a result of these discoveries the failure has been attributed to water entry through a pin hole in the weld and braze metal, and continuation along a flaw on the inner surface of the cap, with final entry to the uranium through the braze metal under the cap. A void in the end cap probably originated as a forging burst in the original billet, this defect then being carried through the extrusion process and resulting in the defective end cap.

Work is under way to evaluate the possibility that other fuel elements may be similarly affected at some future time. Special testing programs are being initiated to more precisely identify any such defective fuel elements in order to insure their rejection.

PLANT MODIFICATION

METAL EXAMINATION FACILITY - PHASE I

The directive has been received on the Metal Examination Facility authorizing both design and project management by the General Electric Co.

Function of the facility will be to provide information on the in-reactor behavior of the current fuel element and of developmental fuel elements. This information will be used to guide R & D work to improve fuel designs and materials behavior.

Design has been started and principal effort is being directed toward the special handling equipment that will be required in the rear face area of the reactor.

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HW-80559 7

CONVERSION

CONSTRUCTION

Construction continues on form work for the pipe gallery extension, on the water-to-water system piping in the sixth cell, on the 12-inch piping assemblies for the condensate return line tie-in's, and on conduit and cable in the reactor instrument and electrical room.

ADDITIONAL SURGE CAPACITY

An addendum to the Phase II Conversion Design Criteria was issued for comment to define the functional requirements of a three-vessel surge tank system necessary to obtain stable response and control during all modes of Phase II operation. Design work has started on the modifications to the existing surge tank for agreement with the requirements of the converted system.

CONDENSATE QUALITY MONITORING INSTRUMENTATION

Bids were received by the AEC on the equipment covered by Specification HWS-6890, "Automatic Condensate Quality Monitoring Equipment, 182-N and 109-N," from two firms, Montek Division of Memcor, and Beckman Instruments, Inc. The bids were forwarded to General Electric for review. Both quotations met the requirements of the bid request. Bid award has not yet been announced.

INSTRUMENTATION

Thirty-one revised and two new electrical drawings were issued for comment to show part of the General Electric design covered by the Conversion Design Criteria for primary and secondary system instrumentation. Specification HWS-6648 was revised to reflect the latest information on the control systems, control boards, and instrumentation for the heat dissipation system in order to provide Burns and Roe with an updated reference basis for conversion design. Drawings and specifications prepared by Burns and Roe for the conversion project are continuing to be reviewed for compatibility with reactor plant requirements.

RESULTS OF STARTUP TESTING

It is desirable to present a summary of the results of the N-1, N-2 and N-3 testing programs without waiting until the experiments can be fully reported upon. Accordingly some selected data, calculations and conclusions are given for physics, operation and engineering performance of the N-Reactor at Zero power, low power and under power ascension.

PHYSICS PARAMETERS

Basic Reactor Parameters - Cold

The basic reactor physics parameters for the N-Reactor lattice were measured during the zero-power physics tests. A uniform temperature of 70°F was arbitrarily chosen as the basis condition. The base lattice parameters are:

material buckling:	$101.6 \pm .4 \times 10^{-6} \text{ Cm}^{-2}$
migration area:	$541 \pm 8 \text{ Cm}^2$
k_{∞}	$1.055 \pm .001$

Measurement in the fully loaded reactor further provided the following reactor parameters:

augmentation distance:	
side to side	80.6 Cm
top to bottom	85.3 Cm
front to rear	44.3 Cm
total geometrical buckling:	$33.8 \times 10^{-6} \text{ Cm}^{-2}$
$k_{\text{effective}}$	$1.0359 \pm .0009$
$\Delta k/k$	3.47% k

Reactivity Changes with Temperature

The change in reactivity with reactor temperature was experimentally determined. Since the temperatures of the primary loop and the graphite moderator are separable and will be different during different aspects of reactor operation, the reactivity effects of temperature change in

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HW-80559

each were separately determined. The reactivity effect of increasing the graphite temperature to the operating point at full design power level (1000°F) was found to be $+2.0 \pm .17\%$, and the reactivity effect of increasing the primary loop temperature to the normal reactor inlet temperature (370°F) is $-2.1 \pm .1\%$. The reactivity effect of the additional coolant and fuel temperature increases within the reactor associated with power level increases is found to be $-.83\%$. These values are based in part on theoretical extrapolations from test conditions, which were limited to a maximum temperature of 475°F and performed under zero-power conditions.

Reactor Control and Safety Systems

Tests were performed both in the critical-sized loading and in the fully-loaded reactor to measure the control worths of the horizontal rod system and of the ball safety system. The values obtained are:

horizontal rods (86 rods) $6.7 \pm .4\%$

ball system (108 channels) $6.5 \pm .5\%$

In addition to these measurements, one-half of the horizontal rods were calibrated in a sequential manner by introducing reactivity increases by uniformly withdrawing other poison material (boron-carbide splines) from the fully-loaded reactor. This test provided a calibrated rod sequence for future measurements of excess reactivity. By summing the incremental changes in reactivity introduced, the excess reactivity of the reactor was found to be 3.47% , which confirms the value obtained from the basic reactor parameters.

OPERATIONAL REACTOR DATA

Enrichment Requirements and Flattening Efficiency

Integrating the results of the zero-power physics tests with calculated values for parameters which could not be measured under those conditions, (poison effects of fission products and the reactivity effects of buildup and burnout of fissionable isotopes), the expected reactivity behavior under operating conditions has been calculated:

cold, green reactivity 3.47%k
graphite temperature effect +2.00%k
inlet coolant temperature effect -2.10%k
power coefficient to design level -.83%k
xenon poison effect -3.0%k
net effect of fission product *0%k
buildup, plutonium buildup, and
U²³⁵ burnout.
operating excess reactivity -.46%k

This reactivity balance indicates a need for additional enrichment in order to achieve full design power level.

It is estimated that an excess reactivity of about 0.5%k should be invested in control rods at equilibrium power in order to provide sufficient operating flexibility, thus, enrichment sufficient to provide about +0.9 to 1.0% will be required.

The results of the zero-power measurement indicated that the reactor would operate with a flattening efficiency (ratio of average to peak tube power) of about 65 percent. This is a significantly lower value than expected (80-90%), the difference being attributable to the lower reflector efficiency found to exist. A flattening efficiency of 65 percent would result in maximum tube powers of 6150 kw at full design power level. Since all design studies and hazards analyses anticipated a maximum tube power of 5000 kw, further flattening will be required in order to meet the conditions of those studies. The requirement for addition of higher enrichment fuel provides an opportunity to selectively locate such fuel so as to enhance the flattening efficiency, as well as provide the required reactivity boost. It is calculated that charging fuel columns with the outer tube of the tube-in-tube assembly enriched to 1.25% U²³⁵ in the third and fourth tube rows in from the edge of the reactor will improve the flattening efficiency to 80-90 percent while providing the required increase in reactivity.

*from theoretical calculation using FLEX III code.

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HW-80559 7

On July 26, the reactor was charged with 64 columns of "spike" (1.25% enriched) fuel in the third lattice unit. The observed reactivity increase from these charges is approximately $.15 \pm .05\%$.

Operational Reactivity vs. Power Level

The observed excess reactivity invested in control rods is shown in the following table as a function of equilibrium power level; also shown is the approximate effect of the spike fuel additions:

<u>Power, MW</u>	<u>Kex, Including Spike</u>	<u>Kex, Without Spike</u>
400	1.4%	1.4%
800	.85	.85%
1200	.65	.57
1600	.50	.35

Extrapolation to full design power level without additional enrichment indicates a reactivity of approximately $-.2\%$, indicating an additional "reactivity boost" requirement of $.7\%$. Since approximately $.15\%$ of reactivity boost has already been added, a total reactivity boost of $.85\%$ is indicated by extrapolation of observed operating data. This is not inconsistent with the value of $.9 - 1.0\%$ previously indicated.

ENGINEERING PERFORMANCE

SYSTEMS OPERATION

The test programs have demonstrated that the integrated reactor plant performs satisfactorily up to 1200 MW, (30% of design power), both in stable operation and in scram transients. The plant has performed satisfactorily at 1600 MW, but has not been scrambled from this level.

No fundamental system inadequacies have been found, however, a number of design and construction features have required, or may require, modification. The more significant of these are highlighted in the following discussions of performance of critical subsystems.

CRITICAL SUBSYSTEM PERFORMANCE

Primary Loop System

The pressure drop across the primary loop, at a given flow, is about

65% of that predicted by design. This provides flexibility for orificing process tubes to improve coolant utilization should this become desirable. The reduced pressure drop currently prohibits operation with fewer than three heat-exchanger cells (since this would put drive pumps out of their normal range of operation), until such time as additional resistance may be provided in the primary loop.

A greater than expected variation in flow between individual process tubes has required that process tube flow monitor trips be individually set.

Secondary Loop System

The additional surge capacity provided by "stand pipes" in the condensate header proved to be inadequate. They have been valved off to improve system stability. The existing surge tank responds very rapidly to changes in steam header pressure or dump condenser capacity. This rapid response was predicted by analog studies. It is not anticipated that secondary loop in stability on response characteristics will prevent achieving design power levels, but the subject is under continuing study. Selected dump condensers have been operated to full design capacity. Selected heat exchangers have been operated to about 50% of capacity.

Considerable "tuning" of the secondary loop control system adjustments has been required. This was expected.

Graphite Cooling System

Normal techniques (hydrogen addition) for scavenging oxygen have not been effective in controlling the water quality in the graphite cooling system. This problem is under study.

Plant Service Boiler

Difficulties with the automatic burner light-off system combined with the immediate need for steam after a scram (with fuel at low irradiation exposure) has resulted in boiler operation at about 50% of design capacity with 7 of the 8 main burners lit. Characteristically the steam demand has risen quickly to about 75% of design capacity after a scram. In no instance has the boiler failed to provide the steam required after a scram.

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HW-80559 7

Nuclear Control and Safety Circuit Systems

During each of the scrams of the Engineering Test Program (N-2 and N-3), the safety circuits and the control rod system functioned as designed. On two occasions a single ball hopper (of the 107 in the ball safety system) failed to release balls on receipt of a trip signal. Both hoppers have operated satisfactorily several times since being repaired and restored to service.

Emergency Raw Water System

Each of the five ERWS diesel driven pumps has provided backup-coolant after each scram of the N-2, N-3 series.

On two occasions when the graphite cooling system was inadvertently "dumped", two diesel driven pumps (used also for the "fog spray" supply) provided adequate coolant backup.

TECHNICAL ACTIVITIES

NUCLEAR SAFETY

Nuclear safety specifications have been prepared for the pilot plant, the oxide burner and for transportation of fuels to the 100 area. These specifications include all enrichments up to 1.95 per cent. A nuclear safety review of the N Fuels Fabrication Plant for the processing of uranium enrichments up to 1.95 was made in collaboration with the critical mass physics representative.

NEW EQUIPMENT

A facility for performing development work was set up in the 105-N Building construction airlock which has now been designated as the 105-NA Building. A production type N Control Rod Mock-up and a Diversion Valve Gallery Mock-up are now in operation. An N-Reactor Graphite Distortion Mock-up is 90 percent complete.

Equipment Development Program

A program of equipment development has been laid out and will be executed and added to as funds permit. Work is proceeding on the following items.

Dynamic Gas Seal

The present process tube assembly dynamic gas seal (See Helium Loss, page 15) must be evaluated under existing ambient conditions and a back-up seal concept developed or the seal redesigned. Design of a test station was initiated which will permit testing of various seal packing materials and arrangements in an environmental chamber. Testing of the mica impregnated asbestos packing rings continued in the Process Component Experimental Loop.

Horizontal Rod Seals

Relocation of the rod drive test assembly from 189-D to 100-N has been completed and a test procedure has been issued to test a seal assembly which was installed in the return end of the cylinder while the drive was at 189-D. The seal is identical to the original seal furnished with the drive except that the first ring is made of leather impregnated with

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HW-8056 7

plastic. It is not planned to give the seal a life test now, but only to verify that the leather has better resistance to extrusion than the original teflon material.

Half Rods

In order to obtain better flattening of the reactor, half rods are being designed. These will contain poison material in only the forward half of the control rod. When this half rod is inserted to act as an absorber in the central zone there will be no absorption in the fringe regions. The absorbing half of the rod will be filled with target materials that produce useful isotopes as well as acting as neutron absorbers. The local control strength will approximate that of the present boron carbide rod, and the materials will be formed so that they will be both internally and externally cooled. The existing rod tip outer shells will be used, but the materials will have to be removed with a spline instead of by flushing.

Thermocouple Trains

Instrumentation design has been started for measurement and recording of approximately 96 temperatures in special thermocouple trains to be inserted in selected process tubes downstream of co-product and tube-in-tube fuel elements. It is proposed to use fine wire stainless steel clad thermocouples. The significant instrumentation design problems are related to bringing the thermocouple leads out of the process tube and through a suitable thermocouple disconnect for use inside the thermal barrier.

Graphite Monitoring Facilities

The facilities will be used to monitor graphite oxidation from an in-leakage and to measure irradiation distortion on small samples and full-size bars of graphite.

Fabrication of hole modification parts and sample handling tools for the three graphite monitoring test holes has been completed. Installation is complete for the test holes in the process tube channel and ball channel.

For the rod channel, removal of the thimble and gunbarrel and installation of test hole parts will be accomplished during the scheduled shutdown in August.

All tool work has been completed except for testing of the retrieving tool for the rod channel sample. Results from tests conducted using the ball channel sample handling tool to remove machined graphite samples from or insert them into a simulated ball channel were satisfactory.

Special Reactor Tools

Proper special equipment continues to be necessary for maintenance of N-Reactor. A second generation freeze-can for water flow stoppage from the branch arm of a vertical header during inlet or diversion valve replacement or Grayloc repair has been completed. The prototype cutter for irradiated tubing is approximately 85 per cent fabricated. Special castings, now being machined, are the only items remaining to be completed prior to assembly of the prototype pusher-puller machine for use in process tube insertion and stuck-slug removal.

Filter Cooling System

All punch list items for the 117-N filter cooling system were completed and the work order for installation of this system has been closed out.

Pressure Reducing Valves

The existing steam pressure reducing and control valves between the auxiliary boiler and the secondary loop have had a mediocre performance record since installation. Under certain conditions, the valves become unstable and oscillate violently until manually closed. A specification for replacement valves requires that the new valves be designed to prevent vertical instability and jumping of the valve plug, and that an in-line silencer be used to dissipate the energy of the expanding steam and reduce the noise. The piping modifications required to install these new valves feature a minimum number of changes to the existing layout.

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HW-80359 7

Steam Header Pressure Control

An equipment modification procedure is being issued for replacement of the boiler steam header pressure transmitter with an instrument having a pressure range closer to the operating range. The purpose is to improve the control of the steam header pressure and make the operation more satisfactory.

Boiler Purge Control

An equipment modification procedure has been issued for installation of necessary equipment to permit manual initiation of a boiler air purge from the control room after a flame-out.

Dry-Film Lubricants

Suitable dry-film lubricants must be established for use at the high temperatures of the N-Reactor environment. The present dry lubricant on the inlet butterfly valves and on the diversion valves of the process tubes is regarded as temporary, and is showing signs of failure.

CO-PRODUCT DEMONSTRATION PROGRAM

The fabrication of the 1.25% enriched co-product fuel elements for the production test (PT-8) is essentially complete and the fuel will be delivered to the N-Reactor for charging during the outage planned for August 6-20. The Hazards Analysis (HW-82162 REV) for this test has been completed and forwarded to the AEC.

Target material for the 1.95% enriched co-product test has been specified and the procurement order written. Nine tons will consist of the uranium-silicon alloy and 3-1/2 tons will consist of the uranium-aluminum-iron ("British") alloy.

Rewelding Program

A process has been developed for rewelding the co-product driver tubes with an angled electrode, and without machining the weld areas prior to

rewelding. Of 195 rewelded ends, 85 per cent were recovered. It appears that this recovery rate can be boosted to 92 per cent. The information obtained from this rewelding program is also being implemented to determine the feasibility of rewelding both outer-enriched and inner-enriched fuel elements and thus eliminating one process step in the recovery of weld rejects.

PROCESS PHYSICS

Heat Generation in the Loss-of-Water Accident

Studies are under way to determine the heat generation in the lattice of the 1.25% co-producer in the event of a loss-of-water accident. The results will be used to determine the time to melt down for the co-producer element. An important factor is the amount by which the reactor goes sub-critical, since this determines the fall-off of fission heating. Earlier calculations took no credit for loss of reactivity due to water expulsion; this factor is being included in the present calculations.

Analysis for Loss of Coolant

Fuel element temperature transients were calculated, assuming a total loss of the primary and emergency cooling systems, but with continued operation of the moderator cooling system. The purpose of the study was to determine the maximum tube power which would permit such a coolant loss without danger of fuel element melting. The power decay curve used in the study neglected the loss of reactivity from the loss of water in the core. The results of the study have been documented. Further work is continuing, including consideration of less severe decay curves.

NEW TECHNOLOGIC DEVELOPMENTS

Thermal Hydraulics Study

A report (HW-80692) has been issued by Hanford Laboratories covering about a year of thermal hydraulic experimentation on the N-Reactor process tubes and fuel. The results show that the thermal hydraulic design of the reactor is even more conservative than originally thought. The margins of flow rate, tube powers and total system pressure which exist between normal operating conditions and the onset of boiling instability are

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HW-80559 7

somewhat larger than previously believed. This fact tends to relieve somewhat the problem of setting safe but operable trip limits in the above parameters.

Nuclear Power for Desalination

A continuous effort has been maintained over the past year in studies on the adaptation of graphite-moderated, pressure-tube reactors to desalination service. Potential improvements have been realized in comparison with the modified N-Reactor concept submitted for the Inter-Agency Task Group study. Studies of the adaptation of the Hanford Graphite Superheat Reactor (HGSR) to desalination service have shown encouraging results in terms of cost of energy delivered to the desalination plant. The results of these design and evaluation studies have been documented in a report, HW-83081.

Physics Evaluations by Consultants

Doctors T. M. Snyder of Vallecitos and P. F. Gast of Argonne, spent several days reviewing the physics of N-Reactor. No serious problems were identified.

ADMINISTRATIVE SUMMARY

JULY, 1964

EMPLOYMENT Section	Permanent		Temporary		Total		Change	
	E	NE	E	NE	E	NE	E	NE
General	1	1			1	1	-0-	-0-
Finance	14	8			14	8	-1	+1
N-Reactor Fuels	46	98	5		51	98	+2	-1
N-Reactor Plant	65	224	3	1	68	225	+5	+8
N-Reactor Project	69	22			69	22	-0-	-1
Research & Engineering	46	10	1		47	10	-3	-0-
Employee Relations	4	1			4	1	+2	+1
	245	364	9	1	254	365	+5	+8
Total employees								
Net change						619		+13

PERSONNEL CHANGES

Dr. P. F. Nichols, Manager of Reactor Physics, left for employment elsewhere.

SAFETY AND SECURITY

Days without a disabling injury 609

Hours worked without a disabling injury 1,978,752

VISITORS

Name	Company	Contact (s)	Date (s)	Purpose
R.A. Graham	Wah Chang Albany, Oregon	H.P. Kraemer L.H. Steves R.R. Studer	7/2/64	Discuss Zr -2 strip quality problems.
T.M. Snyder	G.E. Co. Vallecitos Atomic Power Laboratory San Jose, California	M.C. Leverett & Staff	7/6-9/64	Consulted with M.C. Leverett and Staff on physics problems
R.P. Bryden N. Durham	Consolidated Electro- dynamics Corp. Pasadena, California	I.V. Nelson	7/7/64	Service leak detector.
C.E. Bussert P.N. McCreery C. Ryan	National Lead Cincinnati, Ohio	R.H. Scanlon J.W. Nickolaus	7/7/64	Discuss uranium specification proposal.
P.F. Gast	Argonne National Laboratories Chicago, Illinois	M.C. Leverett & Staff	7/9-10/64	Physics consultations.
W.D. Kelly	Sciaky Bros. Los Angeles, California	T.B. Correy	7/21-22/64	Service welding equipment.
M.C. Leverett	Technological Hazards Council, Philadelphia, Pennsylvania		6/16-17/64	Attended Technological Hazards Council Meeting.
D.R. Stenquist	US-AEC Washington, D. C.	W. Devine L. Murphy	6/30-7/2/64	Discuss application of LP-90 Computer Program to Commission Production Studies.

VISITS

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JUL 15 1964

VISITS (continued)

Name	Company	Contact(s)	Date(s)	Purpose
Frances McBarron	NSA-International Convention Washington, D.C.		7/7-12/64	Attended National Secretaries Association (International) Convention.
L.H. Steves R.R. Studer	Renton Coil Spring Renton, Washington	B. R. Pepka	7/14/64	Order for forged supports.
L.H. Steves R.R. Studer	Rem, Inc. Albany, Oregon	C. A. Brown	7/15/64	Turk's head and coil rolled Zr strip.
L.H. Steves R.R. Studer	Wah Chang Albany, Oregon	R. A. Graham	7/15/64	Turk's head and coil rolled Zr strip.

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SIGNIFICANT REPORTS

HW-80559 7

Number	Title	Author	Date
HW-83091 Conf.	Bibliography - Coextrusion Process	Bonnie Thomas	July 1, 1964
HW-82116 Conf.	Design for Mark II Driver Tube	C. H. Shaw R. G. Curran	June 27, 1964
HW-83407 Uncl.	Engineering Analysis of the 105-N, 109-N 125 Volt DC Ground Problem	J. R. Bolliger	July 24, 1964
HW-83175 Uncl.	Evaluation Report, PT-NR-18, Primary Loop Pressure Relief Valves (RV-2-1 and 2), Relief and Reseat Pressure Test	E. A. Grimm K. L. Berrett	July 2, 1964
HW-81845 D Uncl.	Failure Analysis of 109-N Instrumentation System - Part 4 - Surge Tank Level and Pressure	A. A. Maupin, Jr.	July 14, 1964
HW-82162 REV Secret	Hazards Review, N-Reactor, 1.25% Co-Producer Fuel Element Test	N. R. Miller	July 27, 1964
HW-82100 PT 2 Conf.	Migration Area Test	R. A. Bennett	July 26, 1964
HW-83033 Conf.	Modified Cutoff Lengths for N-Fuels	W. M. Scott C. B. Ruhter	July 1, 1964
HW-82931 Uncl.	N-Reactor Isotope Production Capabilities	J. W. Riches	June 23, 1964
HW-83027 Conf.	N-Reactor Safety Circuit Modifications, Low Pressure Trips	J. R. Bolliger	July 14, 1964
HW-82100 PT 6 Conf.	N-Reactor Startup Physics Test Results, Part 6 - Temperature Coefficient Tests	J. W. Hagan L. L. Grumme	July 16, 1964
HW-83315 Secret	Official Production Forecast-N-Reactor Fuels Operation July, 1964 through June, 1965	L. M. Loeb	July 16, 1964
HW-82293 Secret	Prospective Materials for a Target Element for the Co-Producer Program	J. W. Carter	May 15, 1964

SIGNIFICANT REPORTS - continued

Number	Title	Author	Date
HW-83362 Secret	Revisions to Specifications for Uranium Metal Billets for N-Reactor Fuel Elements	L. M. Loeb	July 21, 1964
HW-83213 Secret	Specification for Aluminum-Lithium Alloy Billets	R. R. Studer	July, 1964
HW-82820 Conf.	Tipped Mandrel Coextrusions	W. A. Jahnke	June 22, 1964
HW-83217 Secret	Uniform Versus Spike Enrichment	J. W. Riches	July 8, 1964

INVENTIONS OR DISCOVERIES

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 July, 1964, except as noted below. Such persons further advise that for the period therein covered by this report, notebook records, if any, kept in the course of their work have been examined for possible inventions or discoveries.

None

DECLASSIFIED

UNCLASSIFIED

HW-80559 7

SUGGESTION PLAN PARTICIPATION

	<u>JULY</u>	<u>CY 1964 TO DATE</u>
Number of Eligible Employees	365	364
Number of Suggestions Received	51	333
Number of Suggestions Acted Upon	57	340
Number of Suggestions Adopted	19	131
Net Annual Savings	\$892	\$127,461
Amount of Awards	\$270	\$6,490
Per cent of Awards to Savings	30.3	5.1
Average Amount of Awards	\$ 14.21	\$49.54

SUGGESTIONS ADOPTED/1000 EMPLOYEES/YEAR
1964

UNCLASSIFIED

INDEX

Afterheat removal pump controls	16
AK-2 breakers	16
Boiler operation	9, 39
Braze rings	23
Cell 3 valve	14
Circulating raw water discharge valves	13
Condensate feedwater flow oscillation	9
Condensate quality monitoring	34
Confinement fog spray control	9
Confinement solenoid valves	15
Control rod evaluations	10
Control rod worth	36
Control rods - half rods	42
Conversion	34
Coolant loss - power decay study	45
Co-product billets	24
Co-product demonstration program	44
Desalination	46
Discontinuous actions test scram	8
Driver tube status (1.95% fuel)	25
Dummy fuel rattling	8
Dummy fuel spacing	18
Dynamic gas seal	41
Equipment development	43

Filter cooling system	43
Flattening efficiency	36
Flux monitor chambers	17
Fog spray accumulator relief valve	9
Fog spray control valve replacement	13
Fuel element failure	7, 25
Fuel element handling equipment	14
Fuel element supports	23
Graphite monitoring facilities	42
Graphite sample irradiations	17
Graphite temperature evaluation	11
Grayloc valve leaks	14
Grounds on 125V DC system	16
Half rods	42
Heat generation in loss-of-water accident	45
Helium loss	15, 41
High pressure reducing valves	14
Horizontal rod scram test	17
Horizontal rod seals	41
Horizontal rod suction lines	12
Hydraulic actuator test	17
Instrument air failure	15
Lattice cell model	10
Metal examination facility	33
N-3 testing	7

Nuclear safety specifications (1.95% fuel)	41
Outages	19
Oxygen scavenging	39
Physics parameters	35
Pressure reducing valves	43
Primary pump shaft seal confinement	13
Primary system pressure drop	38
Production statistics	21
Production summary	22
Production tests	17
Reactivity change with temperature	35
Reactivity effects - table	37
Rewelding of driver fuel elements	44
Reports published	50
Rupture monitor differential pressure control	14
Rupture monitor system	12
Safety circuit improvement	17
Secondary system controls	9
Secondary system operation modes	18
Secondary system stability	39
Spike charges	8
Startup testing - summary	35
Steam generator repair	11
Suggestion plan	52

Sump pumps	15
Surge tank addition	34
Surge tank level	9
Thermal hydraulics	45
Thermocouple trains	42
Unbonded end closure	23
Uranium oxide burner	23
Uranium specifications	24
Valve actuators for dump condenser steam supply	13
Vibration snubbers	15
Water quality sampling	16

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