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N-REACTOR DEPARTMENT

MONTHLY REPORT

NOVEMBER 1965

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HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

GENERAL ELECTRIC

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N-REACTOR DEPARTMENT

MONTHLY REPORT - NOVEMBER 1965

R. L. Dickeman

R. L. Dickeman
General Manager

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

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OPERATING SUMMARY

In-reactor KMWD residue	426.26
Highest exposure tube (MW)/T	2,648
Input production Pu (KMWD)	72.545
Input production coproduct (KMWD)	1.28
Maximum power level attained (MW)	3,775
Average power level while operating (MW)	3,374
Time operated efficiency (%)	70.9
Number of shutdowns	7
Scheduled	0
Unscheduled	7
Reactor process data	
Power at which computed (MW)	3,775
Primary coolant inlet temp. (°F)	362
Primary coolant outlet temp. (°F)	496
Maximum moderator temperature (°F)	1,060
Maximum tube power (KW)	4,890
Heat loss	
Heat to graphite cooling system (MW)	39.4
Heat to shield cooling system (MW)	2.3
Heat to rod cooling system (MW)	1.0
Steam generation temperature (°F)	324
Steam generation pressure (psig)	80

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Dump condenser flow (gpm)	325,000
Inlet temperature ($^{\circ}\text{F}$)	60
Outlet temperature ($^{\circ}\text{F}$)	142
Nuclear steam generating rate (lb/hr)	
Maximum (1000's lb/hr)	11.7×10^3
Average while operating (1000's lb/hr)	10.6×10^3
Water collected from gas atmosphere (gal/day)	Negligible
Demineralized water produced (gal/hr)	66,010
Helium losses (1000's cu. ft.)	426
Fuel oil usage (bbls)	21,020
Reactor flattening efficiency (%)	78.7
Fuel conditions	
Heat flux maximum (BTU/hr/ft ²)	625,000
" " average " "	447,000
Clad temperature	
Maximum ($^{\circ}\text{F}$)	595
Average ($^{\circ}\text{F}$)	534
Uranium temperature	
Maximum ($^{\circ}\text{F}$)	905 (471 C)
Average ($^{\circ}\text{F}$)	763 (392 C)

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Fuel failure indications (unconfirmed) 2

Fuel balance	<u>0.95%</u>	<u>1.25%</u>	<u>1.25%</u> <u>Coproduct</u>	<u>1.96%</u> <u>Coproduct</u>	<u>Total</u>
Fuel charged (tons)	4.22	4.96	0	0	9.18
Fuel discharged (tons)	3.11	4.96	0	0	8.07
Net change (tons)	+1.11	0	0	0	+1.11
Total in reactor at month end (tons)	<u>318.59</u>	<u>42.31</u>	<u>.88</u>	<u>5.11</u>	<u>366.89</u>

Fuel element output (assemblies)	<u>November</u>	<u>October</u>	<u>CYTD</u>
0.95% Tube-in-tube	2036	1872	17,146
0.95% Single tube	0	0	75
1.25% - 0.95% Tube-in-tube	0	0	1,357
1.96% Coproduct	0	0	309
1.25% Coproduct	0	0	- 22
Total assemblies	2036	1872	18,865
Total tons	39.5	38.9	376.8
% of Forecast	96.3	92.6	88.5
Uranium utilization percent (est.)	70	67.9	71.2

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SUMMARY

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The reactor operated 21 days or 70.9 percent of the available time. p. 8

Extensive efforts to achieve improved time operating efficiency are in progress. A number of specific improvements have been made or are underway. p. 9

First blowdown of export steam lines to WPPSS generating station accomplished. p. 9

A second shipment of fuel (about 10 tons) has been sent to Nuclear Fuel Services. This will continue routinely. p. 9

Equipment performance was generally good and system response during several scrams was excellent. There were two shutdowns because of fuel failure indications. Neither has been confirmed as an actual failure. p. 10

Reasons for "dimpled" N outer extrusions have been identified as due to chips of burned lubricant. p. 18

Fuel element cap thickness tester now in service. p. 18

Ceramic target elements for a coproduct test fabricated.

Coextrusion backup capability established at the AEC, Ashtabula site. p. 19

Sufficient British alloy has been ordered to enable partial conversion of the Fuels Shop to this alloy - not only for coproduct drivers, but eventually for all N fuel. p. 19

Evaluation of graphite cooling tubes recently removed from the N-Reactor indicate that hydriding has not occurred. p. 25

Reactor stack oxidation is within predicted limits. p. 25

Coproduct driver fuels (1.25% enriched test) have been dissolved. Unirradiated targets have been sent to the Savannah River Laboratory for dissolution tests. p. 27

Studies indicate that thermal hydraulic operating margins for coproduct fuel are satisfactory. p. 30

A program to irradiate 5 kg of neptunium in the N-Reactor is awaiting release of the Np by AEC. p. 35

Pilot cobalt-60 irradiations are planned. p. 37

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REACTOR PLANT OPERATION

Operating Pattern

The N-Reactor Plant operated 21 days in November, for a plant availability factor of 70.9 percent. The reactor was shut down at the beginning of the month because of a prior fuel failure and boiler leaks. Operation was further interrupted during the month by two manual scrams to remove fuel elements which were giving (unconfirmed) rupture signals, two accidental scrams, two manual shutdowns forced by equipment failures and one shutdown for a spurious rupture indication. The reactor was in shutdown status at month end. A total of 22 fuel columns were discharged and replaced with fresh fuel.

Operating History

The reactor operated from October 25 to November 1 at power levels near 3800 Mw. On November 1 it was necessary to shut down the reactor because of fuel failure indications in tube 2844. The fuel in this tube (standard N-Fuel) was charged on February 11, 1964 and had accumulated an average exposure of 1643 MWD/T. During the shutdown, investigation disclosed a leak in the auxiliary plant boiler. The leak was repaired and operation was resumed on November 4. A shutdown occurred on November 6 when it was observed that a power supply to the rupture monitor system (gross gamma) had lost its voltage regulating ability. The power supply was repaired and the reactor was started up on November 6. On November 8 a scram occurred when an operator pulled a wrong switch, thereby removing all control power from No. 1 and No. 3 recirculating raw water pumps. During this shutdown the "Primary Drive Turbine Speed Reduction Test" (PT-NR-46) was successfully accomplished and operation was resumed on November 9. On November 12 the reactor was manually scrammed because of a fuel failure indication in channel 1562. The reactor was restarted on November 13 and operated without interruption until November 20. On this date gross gamma and gamma energy indications for process tube 2451 rose abruptly to very high values and the reactor was shut down. Measurements of radioactivity in the diversion header and in the waste management trend recorder did not indicate abnormal levels of radioactivity, however, a subsequent analysis of the iodine content of the water sample taken from this process tube showed no abnormal conditions. Moreover, the high readings on the gross gamma and gamma energy instruments disappeared immediately after shutdown. Interpretation of these facts suggests that a small piece of uranium-bearing debris lodged temporarily in the rupture monitor sample chamber, and shortly after shutdown was flushed out of the

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chamber, thus permitting the readings to drop back to normal levels. Operation was resumed on November 22 and continued without interruption until November 30 when it was shutdown to repair boiler leaks.

A discussion of the fuel element failure indications associated with the shutdowns noted above is given on page 22.

The first blowdown of export steam lines from the N-Reactor to the Washington Public Power Supply System generating station was accomplished on November 24. It was executed smoothly and without significant difficulty.

Ten tons of irradiated fuel were shipped to Nuclear Fuel Services during the month. This was the second such shipment. These shipments will continue routinely in the future.

Plant Performance

We reported last month a concerted drive to improve plant Time Operated Efficiency (TOE) and to decrease the frequency of unplanned shutdowns. The TOE experienced in November (about 71%), although close to prior prediction, must be improved if 1966 goals are to be attained. A study of outage time and causes in 1965 up to October 31 reveals the following:

<u>Maintenance</u>	<u>Hours Charged</u>	<u>% Of Total Time</u>
Primary System	1410	19.3
Secondary System	80	1.1
Electrical & Instruments	300	4.1
Boiler	390	5.3
Other	<u>490</u>	<u>6.7</u>
Maintenance Total	2670	36.5
Charge-Discharge	320	4.4
Rupture or Suspected Rupture Removal	280	3.8
Other	<u>180</u>	<u>2.5</u>
	3450	47.2

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It is apparent that mechanical maintenance is the major offender, accounting for about 70 percent of the total outage time. Fuel ruptures and suspected fuel ruptures have been relatively minor causes of outages - considerably less important than the auxiliary boiler. This is partly due to the fact that, although there have been a number of outages for suspected failed fuel removal, such outages have been short, averaging less than 24 hours each.

This information is being used as a guide in assigning effort to improve plant TOE.

Plant equipment performance was generally good. System responses during the several scrams were excellent and primary system pressurizer post-scram outsurge was consistently low indicating improvements which resulted from previous work on the control systems.

Secondary system operation during the steam-blow tests was excellent. Only slight manual biasing of the dump condensers on the test quadrant was required to maintain constant steam pressure on the entire 46-inch steam export header and the resulting effects on the efficiency of nuclear operation were slight. Control of the modulating control valves on the steam export lines to WPPSS was smooth and the associated instrumentation operated satisfactorily.

Extensive testing of and control adjustments to the speed governors for the primary pump drive turbines has resulted in a reduction of the post-scram speed range from 1300-1500 gpm to 900-1200 rpm. Instances of tripping to pony motor following coastdown have also been reduced as a result of this emphasis.

A development test for dump condenser level control was successfully completed during the reactor startup following the October extended outage and the necessary system adjustments have been made. Report RL-NRD-615 covers the results of this program.

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Outage Experience

<u>Date</u>	<u>Outage Type</u>	<u>Duration (Hours)</u>	<u>Reason</u>
11- 1-65	Manual scram	94.4	Rupture indication. Remained shut-down to repair boiler leak. Other maintenance completed. 21 tubes discharged and recharged.
11- 6-65	Manual shutdown	19.2	Failure of gross-gamma monitor off-set power supply.
11- 8-65	Accidental scram	20.7	Pump loss caused by accidental opening of a DC breaker.
11- 9-65	Accidental scram	4.0	Spurious circulating raw water low flow trip from faulty relay protector.
11-12-65	Manual scram	24.8	Rupture indication. PT-NR-36 accomplished.
11-20-65	Manual scram	31.0	Spurious rupture indication.
11-30-65	Manual scram	15.3	Boiler leak repair.

Process Physics

On November 29 at 8:00 a.m. the reactor power level was 3700 Mw; the reactivity held in the rods was 4.89 mk; the hottest graphite temperature was 1040°F (547 C); the flattening efficiency was 78.7 percent; the average fuel exposure was 1146 MWD/T.

As expected, the large fuel discharge in October resulted in a low equilibrium excess reactivity, a reduction in flattening efficiency (to approximately 78 percent) and restriction of power (to approximately 96 percent). The excess reactivity is now increasing slightly, and the equilibrium rod pattern is being changed to rods closer to the center to control axial peaking. The reactivity gain is not expected to be very beneficial from a flattening standpoint, since the reactivity increase is occurring in the newly charged center columns. Power will probably be restricted by outlet temperature limits until after the fringe of the reactor is recharged and reaches 600-700 MWD/T exposure. This situation can be remedied by charging additional spike enrichment, but economic analyses do not support such a step.

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EQUIPMENT MAINTENANCE AND PLANT MODIFICATION

Switchgear Preventive Maintenance

Twelve motor control centers were checked and cleaned as part of the program to increase plant reliability. New procedures now allow the accomplishment of this work during plant operation. This type of work was previously confined to shutdown periods.

Process Tube Flow Monitor Improvement Program

The cause of past transfer problems in the bias voltage power supplies has been discovered and a method proposed to utilize the present system open-diode detection scanner to check the output of each of the 1,003 controller channels for bypassed and near-tripped conditions.

Transfer from the primary to the secondary units of the master bias voltage supply was found to occur during the insertion of a bypass plug into a controller channel bypass jack switch. Inspections disclosed that about 10 percent of the 1,003 bypass switches would cause a momentary short between the 120 volt discriminator bias and the 90 volt master bias circuits during the insertion or removal of a bypass plug. A short between these circuits trips the over-voltage protective device of the master bias supply and initiates a transfer to the secondary supply. Modification of the open-diode detection scanning circuit to scan each controller output for the presence of correct trip voltage and to determine the channel nearest to a trip condition, was found to be feasible. Another function being studied relates to the identification of any by-passed channels. This is dependent upon the development of proper switching and comparator circuits, however, and the acquisition of isolation and measurement equipment.

Twenty thyratrons were replaced in flow monitor controller modules. Eighty percent of the system thyratrons have been tested and replacements are being made as required to improve the system.

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Steam Generator Level Control

Evaluation of charts and data from steam generator narrow range uncompensated level recorder indicates that improvements in the steam generator level control system can be made to simplify operation and increase reliability. A production test has been prepared to place the level controller on an uncompensated narrow range (0-10-inch H₂O) signal. The zero level point will be at the top of the tube bundle. Chart recordings indicate that during normal operation the equivalent of about 3 to 4 inches of water exists above the tube bundle at 90 percent steam load. The actual steam-to-water interface, as determined by the floats, is about 16 inches above the tube bundle at this condition. If the test program demonstrates that satisfactory control can be obtained the complicated level compensation system may be eliminated. The present control system results in excessive operational attention during startup and shutdown and usually allows the uncovering of tube bundles during scram transients.

Resistance Temperature Detectors (RTD's)

Special emphasis is being given to the development of improved resistance temperature detectors for recording process tube temperature. The high failure rate of the installed clamp-on RTD's adversely affects operating efficiency. Several new models are under evaluation. A number are currently undergoing tests in the N-Reactor or in the Battelle Northwest Standards Laboratory.

F" and "R" Elevators

Construction and installation of these two freight elevators which serve the upper elevations in the reactor building were completed.

IN-PLANT TESTS

The status of production tests is as follows.

PT No.	Document No.	Title	STATUS			
			Approved	In Progress	Completed	Final Report
2	HW-80369	Routine Graphite Sample Irradiations		X		
3	HW-81478	Routine Monitoring of Graphite Oxidation in N-Reactor		X		
4	RL-NRD-218	Evaluation of Monitor Column Fuel Elements in N-Reactor		X		
6	HW-81339	Isotope Producing Rods in N		X		
8	HW-81327	Coproducer Demonstration Test (1.25)		X		
9	HW-81609	Low Goal Irradiation of Fuel Elements with Varying Amounts of White Oxide on Surface				X
10	HW-82136	N-Reactor Corrosion Monitoring	X			
12	HW-82241	Copper Base Alloy Corrosion Program		X		
13	RL-NRD-299	Coproducer Demonstration Test (1.96)		X		
14	HW-82385	Irradiation of Th-U Crud Monitor Elements in N-Reactor				X
14 SUP SUP A	HW-82385	Supp. to PT-14, Irradiation of 9.47" Long Fuel Element Assemblies				X
18	HW-82659	Primary Loop Pressure Relief Valves (RV-2-1 and 2) Relief and Reseating Pressure Test				Partially Complete
20	HW-82933	Hydraulic Actuators for RWSV-805 Valves				X
20 SUP SUP A	HW-82933	Supplement A to PT-NR-20				X

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PT No.	Document No.	Title	STATUS			
			Approved	In Progress	Completed	Final Report
24	HW-84095	Plant Service Boiler Base Load Reduction Test Program		X		
24	RL-NRD-528 SUP	PT Change Authorization No. 2		X		
27	HW-84238	Steam Generator Moisture Carryover Detection			X	
27	HW-84238 SUP SUP 1	Supplement 1 to PT-NR-27			X	
30	HW-84232	Eval. of Ammonia for Controlling Oxygen in the Graphite Cooling System			X	
30	RL-NRD-196 SUP	Supplement A to PT-NR-30			X	
32	HW-84367	Exposure of Corrosion Coupons in N-Reactor Process Tubes		X		
33	HW-84401	In-Reactor Enthalpy Imbalance Measurements		X		
33	HW-84401 ADD ADD 2	Addendum 2: Eccentricity Measurements		X		
36	RL-NRD-55	Simulated Driver-Target Elements		X		
41	RL-NRD-297	Eval. of Direct Addition of Ammonia to Graphite Cooling System		X		
43	RL-NRD-379	Testing of Converted Reactor Plant		X		
43	RL-NRD-637 SUP	PT-NR-43, PT Change Author. #1	X			
45	RL-NRD-390	Increase in Primary Coolant System Pressurizer Pressure	X			

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PT No.	Document No.	Title	Status			
			<u>Approved</u>	<u>In Progress</u>	<u>Completed</u>	<u>Final Report</u>
46	RL-NRD-427	Primary Drive Turbine Speed Reduction Tests			X	
47	RL-NRD-426	Horizontal Rod Coolant Channel Purging			X	
48	RL-NRD-466	Corrosion Test Samples in End Caps of Graphite Cooling System		X		
49	RL-NRD-492	100-N Area A Bus Power Supply Staged Transient Loading	X			
51	RL-NRD-508	Process Tube Strainer Clogging			X	
52	RL-NRD-511	Fast Indexing Drive Modification to Rupture Monitor Turrets		X		

FUEL MANUFACTURE AND TECHNOLOGY

Production Summary

The input production of 69.9 tons was 118.8 percent of forecast. The output of 39.5 tons was 96.3 percent of forecast. Output production was low due to mid-month problems with the support welder instrumentation.

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FUEL MANUFACTURE AND TECHNOLOGY

PRODUCTION STATISTICS

November

October

CYTD

Input (Extrusions)

0.95% Mark A (outer), 8"			22
Regular	228	215	1495
0.95% Mark N (inner)	144	166	1269
0.95% Mark D (single tube)	0	0	87
1.25% Mark A (outer, spike)	0	0	127
Total Extrusions	372	381	3228
Total Tons	69.5	69.9	574.7
% of Forecast	118.8	124.8	101.5

Output (Finished Production)

0.95% Tube-in-tube	2036	1872	17146
0.95% Single tube	0	0	75
1.25% - 0.95% Tube-in-tube	0	0	1357
1.96% Coproduct	0	0	309
1.25% Coproduct	0	0	22
Total Assemblies	2036	1872	18865
Total Tons	39.5	38.9	376.8
% of Forecast	96.3	92.6	88.5
Uranium Utilization (Est.)	70%	67.9%	71.2%

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PROCESS DEVELOPMENT

Extruded Surface Defects

Dimple defects in the inner diameter of the coextruded fuels have presented a yield problem with N outer extrusions. During the October extrusion campaign, the defective rate averaged 6.5 percent. Investigation of this problem indicates that the principal cause of dimples was chips of burned lubricant in the bore of the billet assembly. Defects have now been produced deliberately on test coextrusions. Changes in lubrication and loading procedures have been instituted to eliminate the entrapment of the foreign material.

Through Transmission End Test

The cap clad through transmission end test was placed in service. In addition to improved detection of defects, this new test includes fail-safe circuitry and the ability to reject end caps of less than 0.125 inches thickness.

Length Inspection

The length inspection of all material held in storage has been completed. A total of 14,868 assemblies were inspected and eleven thin end caps were found. The thickness of these end caps ranged from less than 0.100 to 0.170 inch. A 100 percent length inspection was initiated in the Final Assembly Operation to prevent recurrence of this problem.

Ceramic Target Elements

Aluminum clad lithium-aluminate target cores were prepared by various processes in BNW facilities to evaluate potential fabrication routes and to obtain fuels for irradiation and separations studies. Two basic forms of core material were tested, cylindrical sintered pellets and sintered particles. The sintered particles were loaded by vibratory compaction. The bulk densities ranged from 78 to 91 percent depending on the compaction process used. Both types of ceramic targets were clad in aluminum and were then canned in 0.03-inch thick Zircaloy tubes, approximately 1.3 inches in diameter. The completed assemblies were autoclaved at 300°C and 1500 psi. All of the fuel elements showed some degree of Zircaloy collapse which was evidenced by longitudinal wrinkles. Cladding collapse was most severe on

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the vibratory compacted core materials indicating that further densification of the core material took place during autoclaving cycle. The fuels containing pellets showed a collapse about equal to the clearance in the original assemblies. Changes to eliminate this problem are being tested.

Alloyed Uranium Billets

The planned demonstration of coproduct in N-Reactor will utilize uranium of the British composition for the driver tubes. This alloy will be used because swelling data obtained at HAP0 indicates that this alloy will swell only about one-half as much as the swelling expected with the standard alloy (150 ppm iron, 100 ppm silicon), under coproduct exposure conditions.

Production test quantities of the British composition billets in N outer tube size have been ordered to serve as a basis for establishing a firm capability at the feed sites and in the N-Reactor coextrusion facility and to provide N-Reactor evaluation of a significant test quantity prior to coproduct conversion. Fifteen tons per month of the N outer tube requirement will be procured. In addition, a single quantity of ten tons will be procured in the N inner size to provide some experience with this geometry in the event that higher exposures in tube-tube alloy are planned. Receipt of these test quantities of the British alloy composition is scheduled to start in January.

Coextrusion Backup Capability Off-site

In the unlikely event of major damage to the Hanford coextrusion press, for example, a cracked cylinder, an outage of six to sixteen weeks could occur. Therefore, a program was established to provide a backup capability within the AEC complex on the extrusion press located at Ashtabula, Ohio. In the event of a major outage at Hanford, billets will be shipped to Ashtabula along with the minor coextrusion tooling and coextrusion returned to Hanford for finishing operations. To permit rapid startup of the Ashtabula facility in the event of a Hanford outage, liners and containers sized for N outer and N inner coextrusion have been obtained and stored at Ashtabula.. Four outer and four inner billet assemblies prepared at Hanford were coextruded at Ashtabula in September. These eight coextrusion constituted a pilot run to be followed with ten coextrusion in each geometry after evaluation of the initial eight.

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The fuel sections from the initial eight coextrusions have been processed through the balance of the fuel fabrication operations. The results of the fabrication evaluation were favorable and plans have been made to proceed with the additional twenty-coextrusions. Two of the N outer coextrusions were rejected at peel test so only two outer and four inner coextrusions were processed. The data obtained in wall thickness variation, fuel diameter, warp, heat treat length growth, and in-cladding and bonding were all comparable to the results obtained with coextrusions produced at Hanford with the exception of cladding thickness variation for outer fuels as shown in the table below. The cladding thickness variation obtained on the N outer extrusions fabricated at Ashtabula is somewhat better than normally obtained at Hanford, however, the sample size is small. This potential yield improvement will be investigated further when the additional billets are coextruded at Ashtabula. The reason that the two outer extrusions failed peel test is unknown, but it is believed that the problem was related to shipping damage and not to the coextrusion conditions.

Bond and Clad Test Results for Outers

<u>Pc. Number</u>	<u>Maximum Ext. Clad Variation</u>	<u>Minimum Ext. Clad Thickness</u>	<u>Maximum Int. Clad Variation</u>	<u>Minimum Int. Clad Thickness</u>
X309-1	.005"	.0255"	.007"	.0205"
X309-2-12	.004	.027	.007	.023
X309-13	.007	.026	.006	.019
X311-1	.004	.028	.004	.0235
X311-2-12	.004	.028	.006	.0225
X311-13	.008	.027	.005	.022

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CONVERSION

The two Cell 6 steam generators were received and have been unloaded from the rail car. The No. 6-B exchanger has been moved about halfway into the sixth cell through the opening in the west wall, No. 6-A is on cribbing adjacent to the rail siding.

Cell 6 drive turbine is now in place and is being positioned for the proper elevation and installation.

Central Data Logger

All equipment required to make the N-Reactor central data logger print out information from WPPSS (185 Building) and the sixth cell have been received, installed and tested. The computer program furnished with this equipment has been checked out and corrected as necessary. Wiring is complete between the central data logger input terminals and the signal sources in the Bailey System cabinets. Jumpers between terminal blocks in the central data logger to complete the circuit may be installed whenever input signals become available from WPPSS and the sixth cell.

Monitoring of the 109-N Steam Piping System

The 109-N steam piping system between the ten steam generators and the sixteen dump condensers is a complex piping array. The connection of eight additional, 36-inch diameter, WPPSS export steam lines to the 46-inch HP steam headers on the heat exchanger building (109-N) roof level further complicates this piping system. The actual force effect of this tie-in to the 109-N piping will be monitored by strain gages, dial indicators and pencil boards on two representative export steam lines, adjacent to where they join the 109-N 46-inch steam header. The installation of dial indicators, strain gages, and pencil boards was completed prior to the first export line blowdown.

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TECHNICAL ACTIVITIES

TECHNICAL SUPPORT OF CURRENT OPERATIONS

Evaluation of Fuel Element Failures in N-Reactor

Two indications of fuel element failures have occurred since the October report (Nos. 12 and 13). Neither has been confirmed by visual examination. Failure No. 11 (reported last month) has been confirmed. This failure occurred in a fuel column that had accumulated only 91 MWD/T tube average exposure, over 11 operating days. This element was an outer piece and the point of water entry was immediately adjacent to a suitcase handle support. A small piece of wire was lodged under the support and the failure is thought to have been caused by fretting of the outer cladding by the wire. The material of this wire has tentatively been identified as Zircaloy (by gamma-scan techniques).

Three indicated failures that remain unconfirmed after both visual examination and sniffing for evolved fission gases are summarized in Table I. Failure No. 11 is also included.

TABLE I

<u>Indicated Failure No.</u>	<u>Tube No.</u>	<u>Date of Failure</u>	<u>Tube Power (kw)</u>	<u>Tube (MWD/T)</u>	<u>Element (MWD/T)</u>	<u>Fuel Type</u>	<u>Type of Failure</u>	<u>Assigned Loss (MWD) (Days)</u>
10	2361	10-17-65	3795	595	--	SDT	Uncon- firmed	57.6 1.07
11	1864	10-25-65	4433	91	74	Outer	Fretting	40.7 1.07
12	2844	11-01-65	3046	1643	--	Tube-tube	Uncon- firmed	35.6 .98
13	1562	11-12-65	4242	806	--	SDT	"	39.1 1.03

In every case, the unconfirmed failures remain unconfirmed after both visual and sniffing for fission gases evolved from the elements after a 15-minute residence in argon gas. (The argon gas is displaced through a charcoal filter and the activity of the filter is checked with a G-M counter both before and after use).

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Sniffing tests on the confirmed rupture numbered 11 and on some accidentally broken fuel elements have shown this technique to be a sensitive method of detecting exposed core metal in contact with argon gas.

Melting Times for 1.25 Spike Fuel

Temperature transients for isolated spike-enriched fuel columns operating adjacent to standard fuel, have been calculated for the range of adjacent tube powers from 4200 through 5200 kw. The results shown in Figure 1 and in Table II indicate that the specific power limit currently in application (231 kw/ft) is overly restrictive when applied to spike columns. Extrapolation of the curve in Figure 1 will show that, if a 220 second melting time is considered the minimum tolerable on the basis of confinement requirements, a 250 kw/ft peak specific power could be allowed in the spike fuel (equivalent to a tube power of about 5600 kw assuming a smooth cosine and a full 35-foot fuel column).

TABLE II

MELTING TIMES FOR SPIKE FUEL LOADED ADJACENT
TO STANDARD FUEL

Adjacent Phase II Fuel Column Power		Spike Fuel	
<u>Tube Power</u>	<u>Max. Spec. Power</u>	<u>Max. Spec. Power</u>	<u>Spike Fuel</u>
(kw)	(kw/ft)	(kw/ft)	<u>Melting Time</u>
			(sec)
4200	177	198.0	396
4400	186	208.0	345
4600	194	217.0	304
4800	203	226.5	273
5000	211	236.0	248

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Figure 1

TIME FOLLOWING COOLING LOSS FOR
MELTING OF SPIKE ENRICHED FUEL

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High Flux Graphite Irradiations at Savannah River

Evaluation of the use of C-Reactor at Savannah River for irradiation of graphite samples has been completed. Experimental confirmation indicates that SRP neutron flux will be very little higher than in the Engineering Test Reactor (ETR) and because of complications of capsule design and hardware modifications no further action at SRP is planned.

Graphite Cooling System

Following removal of graphite coolant tube 0788 and 0899, unirradiated end sections of each tube were sent to Battelle Northwest for hydrogen analysis. In addition, an irradiated center section of tube 0788 was sent to Radiometallurgy Laboratory for hydride evaluation to assess the effect of the graphite coolant system dump incident. Results obtained to date show no hydriding on the end sections of either tube (10 to 18 ppm as opposed to 10 to 20 ppm in "as received" material). Although metallography on the irradiated section is not complete, no hydride "case" has been observed on the sample. From these results, it is concluded that hydride formation similar to that of K-Reactors is not a problem in the N-Reactor graphite cooling system, and that the previously reported dump incident (in which it is believed boiling occurred for a short time in the graphite cooling tubes) had no serious deleterious effect on the zircaloy coolant tubes.

Reactor Stack Oxidation

Samples removed from channel 1648 during the October outage showed a maximum oxidation rate of 0.2 percent/1000 days on one sample. All other samples showed weight losses equivalent to a rate less than 0.1 percent/1000 days. It is concluded from this that little or no oxygen is entering the reactor stack when stack temperature is in excess of 300 C. These oxidation values are well below the criteria for 35-year reactor life.

Simulated Driver-Target (SDT)

Two of the four columns of this test have been discharged because of fuel failure indications. No actual fuel failures have yet been identified in either case. The exposure of the two remaining columns of U-800 ppm, Al-250 ppm, Fe-350 ppm Si alloy elements has reached 900 MWD/T. These

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elements are operating at maximum uranium temperatures of about 570 C, and are providing high temperature fuel performance data of direct interest to the coproducer program.

Ten to fifteen more columns of SDT fuel made with several different uranium alloys are scheduled for irradiation charging during December and January. Fabrication of a somewhat larger number of elements is scheduled for completion during April.

In-Core Chambers and Flux Distribution

Five of the nine available carbon-steel traversing wires have been placed in service. Activity buildup on these wires indicates that 20 traverses each can be obtained before handling problems force a changeout. These wires are therefore a temporary substitute for titanium wires in the flux monitoring application.

Sufficient traversing data has been obtained with the carbon-steel wires, however, to establish calibration on most of the in-core ion chambers. The chamber readings have been translated in terms of local specific power and limits applied to insure compliance with the confinement criterion.

Rod Cooling System Review

In view of the proposed increase in operating level of N-Reactor and the possible use of a mixture of nitrogen and helium as the pile gas, a design review was made of the horizontal rod cooling system to determine if the system will maintain acceptable rod temperatures under various combinations of anticipated conditions. The calculations were performed with a FORTRAN coded digital computer program. The mathematical model was composed on the basis that the rod is centered at the bottom of the graphite channel. This geometry was considered to be the worst possible case with respect to the heat load transported from the graphite.

Although the use of nitrogen as a pile gas constituent will result in increased graphite temperatures, which in turn will affect the rod, the graphite contributes only 30 to 40 percent of the total heat load to the rod coolant stream. Thus the graphite temperature only partially determines the coolant outlet temperature and is not primarily significant in determining the rod sheath temperature. For example, a 29 percent change in graphite temperature results in only a 6 to 10 percent change in the rod sheath temperature at the bottom section of the rod. The outlet coolant temperature, as expected, has a one-to-one proportional bearing on the rod sheath temperature.

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The results of the study indicate that for an inlet coolant temperature of 78°F (the upper limit expected for the temperature of the river) and a power level corresponding to the design goal increase, a nominal rod coolant flow of 11 gpm will maintain rod coolant outlet temperatures below the limit presently specified in the Process Standards. The calculations were based on maximum rod insertion in the maximum power region of the reactor and graphite temperatures about 100°F greater than that which would be expected with a 50-50 mixture of nitrogen and helium as the pile gas. The calculations also indicate that the temperature gradient of the rod proper (poison and sheath) will be maintained well within an acceptable range under the conditions examined.

COPRODUCT PROGRAM

Coproduct Test - 1.25 Percent Enriched

The discharged drivers from the 1.25 percent coproduct test block have been identified, sorted, combined according to the process tube where they were irradiated and shipped to Redox for determination of the Pu and Pu-240 content. Dissolving of these samples has commenced in the Redox plant. As previously reported the targets from this test have been shipped to Savannah River Laboratory for tritium yield analyses.

Three of the special weighed and measured columns of the 1.25 coproduct test remain in the reactor. The exposure is now 2100 MWD/T (3000 peak). The water channel thermocouples remain operative and have not indicated development of an enthalpy unbalance. Columns were discharged at 1100 and 1700 MWD/T (1400 and 2500 maximum). At 1100 MWD/T the swelling values ranged up to 0.8 percent and at 1700 to 2.6 percent. Early in December the exposure for the remaining columns will be about 2300 MWD/T (3400 maximum), and assumption of a linear swelling rate from 1100 MWD/T gives a value of 4.4 percent for the swelling. Thus the occurrence of inner clad buckling by the next outage is possible. Another possibility, one that would be a breakthrough for tubular metallic fuel, is that swelling will saturate at about 3 to 4 percent. The mechanism that is used to explain cavitation swelling would permit this to happen if the external restraint is high enough. Further, this is the best explanation for the behavior of the K sized enriched N fuel elements previously irradiated in the K-Reactor test loop. These incurred no apparent swelling between 1700 and 2700 MWD/T.

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Coproduct Test - 1.96 Percent Enriched

The 15 columns of this test have now reached an exposure level of 800 MWD/T average (1200 maximum). The power level remains lower than planned, maximum fuel temperatures are only slightly greater than 500 C - probably less than that of an earlier test (PT-8) of the driver elements. The test elements were loaded, however, in such a way as to give good comparative data on standard uranium and British alloy performance no matter what the fuel temperatures turned out to be. Goal exposure of approximately 1700 MWD/T for this test block should be achieved in March or April 1966.

Coproducer Ceramic Target Element Evaluation (PT-57)

Assembly of the fuel elements for this evaluation test has been completed. The purpose of the test is the production of a quantity of irradiated targets for tritium recovery tests, both at Hanford and at Savannah River. The targets should reach a desirable tritium content in about two months.

Ceramic Coproduct Target Extraction Studies

The third and fourth sodium tetraborate extraction runs were completed. Analytical data for the fourth run are not yet available. The temperature of the third run was only 760 C instead of 800 C. This was the maximum temperature attainable. New heavier heating elements were installed in the furnace for the fourth run.

The low temperature for run number three (20 C above the melting point) did not provide for the complete release of the tritium and about ten percent of the product remained in the residue after 8-1/2 hours at temperature. The fourth run went smoothly and is expected to provide good data.

A suggestion has been received that borane might be produced during the sodium tetraborate extraction step. A review of available literature and consultation with several authorities indicates that borane should form in only very small amounts, if at all, in the extraction vessel. Precautions have been taken, however, to prevent hydrogen or tritium explosions and these are believed adequate to prevent borane explosions if any borane should form. Nevertheless, experimental confirmation of these conclusions is in progress.

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Graphite and Process Tube Temperatures During Coproduct Operation

The use of a 14.6 lb/ft driver element for coproduct operation is estimated to result in an increase in the heat generation rate in the graphite. This increase coupled with a planned column shortening from 35 to 31 feet plus a 30°F increase in coolant temperature for Phase II will cause increases in maximum graphite temperature and process tube surface temperatures. Computer runs are now in progress to clarify and refine graphite and tube temperature calculations for these conditions and to more precisely define the severity of these temperatures in relation to hydriding of the process tubes. A test is planned which will place sections of Zr-2 tubes in the rear portion of channel 1648 where samples are exposed to the reactor atmosphere. The samples will be equipped with thermocouples and positioned to incur irradiation temperatures from 350 C to 450 C.

It appears that process tube temperature may limit raising of the reactor power from 4000 to 4800 Mw in the coproduct mode. The process tube apparently will reach its maximum allowable temperature at power levels above 4000 Mw but below 4800 Mw unless some relief can be gained through refinement of calculations.

Coproduct Physics

A technique to more accurately predict the relative tube power of coproduct and other charges in a field of standard charges has been derived. The method has been verified and proves to be somewhat conservative by comparison with observed data as tabulated below:

	<u>Relative Powers</u>	
	<u>Calculated</u>	<u>Observed</u>
Spike Fuel	1.09	1.07
1.25 Coproduct	1.03	1.01
1.96 Coproduct	1.09	1.06

The calculated relative power of coproduct fuel in a field of .95 percent fuel is now 1.06, however, since the coproduct charges are 10 percent shorter than the .95 percent charges, the relative tube power is .99. This will ease the transition to coproduct fuel, assure full power operation during the transition, and permit charging this fuel into the central zone without concern over limiting the total reactor power.

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A 10-20 column coproduct test is planned as a metallurgical and thermal hydraulic proof test. While this test will not be large enough to detect significant physics effects, by charging the columns in a block surrounding a graphite thermocouple train, we can expect to determine the effect on graphite and process tube surface temperatures of changing to coproduct fuel. The recommended arrangement would center the block in the reactor. The only apparent method of detecting process tube effects is by direct inspection; the test may have to be preceded and followed by tube removals.

Physics calculations have been completed to determine the reactivity effect of variations in target isotope concentration in the coproduct fuel. The reactivity sensitivity is .9 mk per percent change in target isotope content. Reactor operational requirements will require that the operating reactivity not vary more than plus or minus 2 mk, thus setting a tolerance of about plus or minus 2 percent on target isotope content. These requirements will be incorporated in the coproduct fuel design.

Operating Margins - Coproduct Program

Hydraulic characteristics of the coproduct fuel were developed by extrapolation of experimental data obtained in the BNW Thermal Hydraulics Laboratory for the tube and tube fuel (shown in Figure 2). Since the coproducer driver has nearly the same dimensions as the outer cylinder of the present fuel, the characteristics shown are considered valid for a first approximation. It will be noted that the two-phase demand curve of Figure 2 is only moderately "humped", indicating that although the fuel column resistance of the coproducer has a significantly greater hydraulic resistance than does current fuel, the pressure drop in outlet fittings still is the predominant factor in determining the over-all resistance characteristics.

The curves of Figure 2, together with data obtained from earlier studies of present fuel, were used in developing data shown in Table III, which compares operating margins for the different fuels under various reactor operating conditions. A significant conclusion evident from Table III indicates that operating margins for coproduct at 4800 Mw appear to be considerably more favorable than those initially estimated for the tube and tube from design operating parameters and calculated two-phase pressure drop characteristics. Data tabulated for the other Hanford reactors is included in Table III only for purposes of general interest; comparison of magnitudes of the margins is not meaningful because of dissimilarities between reactors and methods of their operation.

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Figure 2

TUBE SUPPLY AND DEMAND CURVES VS. FLOW FOR COPRODUCT FUEL

TABLE III

COMPARISONS OF OPERATING MARGINS FOR MAXIMUM
POWER TUBES

Reactor Fuel	N-REACTOR						Small Reactors	K Reactors
	Present	(Tube and Tube)	Coprodut	(Tube and Rod)				
Reactor Power, Mw	4000	4000	4000 ^(c)	4000	4000	4800	HW-74095	SUPP3
Max. Tube Power, kw	5000	5000	5000	5000	5600 ^(d)	6000		
Steam Pressure, psia ₃	110	150	87	150	150	150		
Normal Tube Flow, 10 lb/hr	90	100.7	80	98.4	98.4	98.4		
Rear Riser Pressure, psig	1406	1406	1200	1450	1450	1450		
Margins with Variations of Single Parameter								
Instability								
Trip Flow/Instability Flow ^(a)	1.29	1.41	1.04	1.30	1.18	1.10	1.19	1.31
Trip Pressure/Instab. Press.	1.50	1.48	1.09	1.47	1.27	1.14	----	----
Instability Power/Norm. Power	1.43	1.44	1.13	1.44	1.28	1.18	1.32	1.48
Film Boiling (inner annulus)								
Trip Flow/Film Boiling Flow ^(b)	1.76	1.91	1.04 ^(a)	>1.76	>1.56	1.43	2.14	2.57
Trip Flow/Film Boiling Flow ^(a)	1.29	1.41	1.04	1.30	1.18	1.10	----	----

Notes: (N-Reactor Only)

- (a) Due to system flow reduction; further reduction to film boiling due to instability.
- (b) Plugging upstream from fuel column
- (c) Calculated from original design operating conditions reported in Hazards Summary Report (HW-79630 Vol 1) and margins reported in HW-83655
- (d) Maximum tube power expected during transition to demonstration load

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Melting Calculation

In support of the coproduct hazards review, a study was made to determine the fraction of drivers melted as a function of time following a complete loss of coolant, and results are shown in Figure 3. It is apparent from these results that although the assumed power level is 20 percent higher than tube and tube design power, the total fraction of uranium melted is not significantly greater than that initially estimated in original design studies of current fuel.

Figure 3 indicates a shorter time to initial melting than was earlier reported (185 seconds as opposed to the 220 seconds in the October report). This reduction is primarily due to an assumption of higher initial graphite temperatures based on a greater fraction of heat generated in the graphite than expected at the time the first coproduct melting times were calculated. Similar calculations are in progress for coproduct operation at the 4800 Mw level. Results of this study are expected to be considerably more favorable than those given in Figure 3 primarily because the maximum tube power now expected during transition loading is about 5000 kw (as opposed to the 6000 kw initially estimated).

Accidental Dump During Coproduct Operation

The consequences of an accidental dump during coproduct operation at goal power level were examined. The analysis indicated that if the primary coolant is assumed to behave as a homogeneous mixture when flashing occurs in the cold leg, the maximum fuel element temperature does not at any time in the transient exceed the normal operating temperature. If complete separation of the steam-water mixture is assumed, the maximum driver element temperature may increase to about 1120°F. But even in this worst case the driver temperature is only slightly greater than the maximum allowable uranium temperature. It is concluded from the study that neither the driver nor the target elements during coproduct operation will experience excessive temperature excursions following an accidental dump.

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Figure 3

DEGREE OF MELTING AND CLADDING FAILURE VS.TIME

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OTHER ADVANCED PROGRAMS

Neptunium Irradiation

A program to irradiate approximately 5 kg of neptunium has been proposed for N-Reactor. This program includes the withdrawing and converting of $\text{Np}(\text{NO}_3)_2$ to NpO_2 , the casting of neptunium oxide target elements, the irradiating of three 3/4-length process tube quantities of material, and the chemical processing of the irradiated target elements.

The proposed irradiation would be completed in two test loads - the first being 1850 grams of neptunium charged in one tube, and the second being 2400 grams of neptunium charged in two tubes. The target elements for the first and second test loads will contain 70 grams Np/ft and 45 grams Np/ft, respectively.

A programmatic schedule has been designed to withdraw Np-237 from the AEC production stream, to complete the Np-237 irradiation program and to return the neptunium nitrate material to the Savannah River Plant for further irradiation in as short a time as possible to avoid a loss of Pu-238 production.

The anticipated procedure to complete each facet of the proposed program is discussed as follows:

a) Withdrawal and Conversion of $\text{Np}(\text{NO}_3)_2$

The neptunium nitrate solution would be withdrawn from the normal production stream after purification in the Purex Q-cell and delivered to the Battelle Northwest Plutonium Chemistry Laboratory for oxalate precipitation and conversion to the oxide. Total time allocated for the withdrawal, conversion and delivery for fabrication of the first 1850 grams Np is 24 days. For the second withdrawal, consisting of 2400 grams Np, the scheduled time is 36 days.

b) Fabrication

The anticipated target element design would be NpO_2 and aluminum powder cast in a .250-inch annular ring, clad in aluminum, and canned in a 2.4-inch OD Zr-2 can. The ID of the annular target ring will be approximately 1.75 inches and the central core will contain graphite. The target elements for the one tube test will contain 70 grams Np/ft and the elements for the two tube test will contain 45 grams Np/ft.

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c) Irradiation

The one tube test will involve irradiating target elements 142 days assuming a 70 percent plant factor. The two tube test will be irradiated for 135 days assuming a 70 percent plant factor to achieve the same exposure.

After discharge, the target elements will be stripped of the Zr-2 clad and the graphite core will be removed. The elements will be cooled to approximately 500 to 700 curies I-131/kg Np and transferred to BNW. Efforts will be directed to attain a total elapsed time, including cooling, of approximately 160 days for each test load.

d) Separations

The elements will be dissolved in nitric acid, processed through a solvent extraction column and sent through an anion exchange column to separate Np from Pu.

The purified $\text{Np}(\text{NO}_3)_2$ will then be shipped to the Savannah River Plant. The $\text{Pu}(\text{NO}_3)_2$ will be converted to PuO_2 and then shipped to Mound Laboratory. The estimated time for the separation and shipment of the neptunium nitrate is 70 days. The conversion of $\text{Pu}(\text{NO}_3)_2$ to PuO_2 and the shipment of PuO_2 will follow the shipment of the $\text{Np}(\text{NO}_3)_2$ solution.

e) Analytical Results

Analysis of selected samples taken from the 13th, 19th and 25th positions in the process channel will be needed. Shipment of $\text{Np}(\text{NO}_3)_2$ to SRP would not however be held up pending analytical results. Consequently, no delay time in shipping $\text{Np}(\text{NO}_3)_2$ to SRP is anticipated.

Total elapsed time for chemical analysis is estimated to be approximately 30 days for each test load.

f) Np-237 Irradiation Physics

From a physics standpoint, this test should consist of one column of targets containing 70 grams Np-237 per foot, and two columns containing 45 grams per foot; the poison effect of these columns will be compensated by charging four and three spike columns per target column respectively. These three columns will consume the 5 kg of Np-237 requested.

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Cobalt-60 Program

The Cobalt-60 Program has progressed at a rapid pace since the basic technology of fabricating and irradiating cobalt targets to produce cobalt-60 is well established. Test irradiations in up to three graphite cooling tubes are planned to start in about three months. Specific activities up to 100 curies/gm are expected.

TECHNICAL ACTIVITIES

NUCLEAR SAFETY AND ENVIRONMENTAL CONTROL

Decontamination Waste Disposal

Although technically feasible, filtration of the presently stored decontamination waste solutions has been found to have a number of drawbacks. Continued testing is required for optimization of filter media. The resultant process of filtration would require about \$4,000 worth of cartridges plus at least 100 manhours for changing of cartridges. In addition, weather protection would be required for the work, thereby further increasing the capital costs.

A proposal for disposal of such waste has also been received from a vendor who is building a truck-mounted radioactive waste evaporation system. The proposal costs were \$0.25 per gallon of raw waste, with a minimum of \$50,000 per batch and a minimum guaranteed contract price of \$175,000 over about 18 months.

Investigations have indicated the liquid wastes could be trucked to the 200 Area for disposal in existing cribs that are no longer being used and that one of the three trucks used in moving 300 Area wastes could be made available for this purpose. About 40 such trips would be required to move the 185,000 gallons of waste solution that are in the 100-N storage tank. The total estimated cost is \$5,000. The transport of the waste is planned for early 1966.

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Modification of the Steam Vent Monitoring System

Modifications are in progress to provide for radiation monitoring of the steam vent system. The system consists of two subsystems:

- 1) Gross gamma monitor for all roof steam vents in the heat exchanger (109-N Bldg.) and the Reactor (105-N Bldg.).
- 2) Halogen gas and particulate gas monitor for any release through a specially controlled vent on the 105-N roof.

The modification and acceptance testing has been 90 percent completed on the gross gamma monitor subsystem, and the halogen gas and particulate monitoring subsystem is ready for testing.

Use of Boron in the Fuel Storage Basin

As previously reported the use of boron (as sodium tetraborate) in the fuel element basin had been proposed to enhance nuclear safety during co-product operation if high enrichment fuel were to be required. Engineering studies have now shown that for the fuel enrichments considered boration of the basin water will not be required. Work on this program has therefore been suspended.

Radiation Protection

No exposure of personnel above AEC or N-Reactor limits occurred during the report period.

Environmental Effects

No significant release of radionuclides from N-Reactor to the environs occurred.

The N-Reactor crib is operating well, reducing the total quantity of radionuclides routinely released to the Columbia River in coolant water by retaining many nuclides and allowing time for decay of nuclides which are not retained.

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**EFFECT OF ACCIDENTAL DUMP OF
PRIMARY LOOP DURING OPERATION**

The primary loop is equipped with eight valves which open automatically in response to various combinations of signals which signify the necessity to establish "last ditch" cooling. These valves permit the contents of the primary loop to escape into the quench basin, thus reducing loop pressure to a level low enough to permit the diesel-driven emergency raw water pumps to inject raw water into the loop, and providing also a path of egress for the raw water so injected. The dumping of the primary loop subjects the loop and the fuel to unusual thermal and mechanical stresses. The ability of the loop and fuel to withstand these stresses (or stresses considerably more severe than could arise in the present mode of operation) is not in question (this ability has been both designed-in and verified by actually dumping the loop), but as a matter of prudent procedure the dump transient has now been re-analyzed for the proposed coproduct mode of operation. The coproduct conditions differ somewhat from those of the plutonium-only mode, for which the reactor was originally designed and analyzed.

The work done, the results obtained and the conclusions drawn are summarized in the following pages. First, however, it should be emphasized that an accidental or spurious loop dump is regarded as a highly improbable event, and also that the objective of plant design and operation is that at no time will failure of a single component (primary loop piping excepted) necessitate dumping the loop. The accidental dump of the primary system is improbable since the logic control for the dump valves was designed with special backup and redundancy features, the valves were designed to fail shut in event of control failure, the trips are set high to prevent spurious trips and maintenance of the system received priority consideration. Operating experience to date has shown no instance where the primary system has unintentionally approached a dump condition.

Work Done and Results Obtained

The operating parameters selected for the most demanding conditions of coproduct operation are summarized in Table IV and they are compared also with the corresponding conditions of Phase II operation wherein only plutonium (and steam for electrical power operation) would be produced.

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TABLE IV

Coproduct Operating Parameters

	<u>Max. Coproduct</u>	<u>Phase II, 4000 MW</u>
1. Reactor Power	4800 MW	4000
2. Total Reactor Flow	92.9×10^6 lb/hr	96.0×10^6 lb/hr
3. Bulk Coolant Temperatures		
a. Inlet	389.5°F	386.4°F
b. Outlet	566.5°F	532.7°F
4. Pressure Drop @ Normal Flow		
a. Riser-to-Riser	256.7 psi	200 psi
b. Total Loop	381.9 psi	334 psi
5. Maximum Tube Power	6000 Kw	5000 Kw
6. Maximum Tube Flow	27.3 lb/sec	28.0 lb/sec
7. Maximum Specific Power	281.1 Kw/ft.	211.7 Kw/ft
8. Secondary Steam Pressure	150 psia	150 psia
9. Fuel Assembly		
a. Driver Element Dimensions		
Outer cylinder		Outer cylinder
OD = 2.40"		OD = 2.406
ID = 1.72"		ID = 1.763
Zr-2 cladding = .025" thick (min.)		Zr-2 cladding .0250 (min.)
b. Target Element Dimensions		
Rod		Inner cylinder
OD = 1.34"		OD = 1.250
Zr-2 cladding = .030" (min.)		ID = 0.438
		0.0275" (min.)
10. Core Length = 31 ft.		

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Coproduct operation at 4800 MW was chosen for analysis since, if it can be established that the fuel and the loop can withstand a dump from these conditions, they can also withstand even more easily a dump from the lower-power coproduct conditions which are more probable in the coproduct demonstration.

The calculations are reported in detail in Reference 1*. The mathematical model which was used in these calculations lumps together the distributed flow resistances and volumes of the actual physical system into a finite number of increments or nodes. The blowdown transient is approximated by a series of steady-state solutions of the flow and volume balance equations.

The assumptions used in this analysis were also consistent with those used in Reference 1. Namely, a scram occurs immediately after the dump valves open and the normal scram sequence of primary pump coastdown to 25 percent speed and one-third heat exchanger flow bypass occurs. Unlike the reference study, however, the make-up injection flow was assumed to be 700 lbs/sec and constant during the entire transient. Latest pump data indicate that this flow is more realistic than the 1000 lbs/sec assumed in the earlier study.

Two-phase pressure drops were calculated using the Martinelli-Nelson multiplier with critical flow defined by Fauske's model. These correlations were found to simulate two-phase flow in the dump line quite satisfactorily based on data obtained in the primary loop dump test.

From the previous study, it was determined that the critical period in the transient arises when the cold leg coolant at the inlet to the process tubes flashes and exists in two phases. This occurs when the Emergency Raw Water Supply (ERWS) coolant begins to flow into the reactor. The ERWS coolant in the hot leg is, in general, at a considerably lower enthalpy than even the coolant remaining in the cold leg and, therefore, does not flash. Consequently as the hot fluid remaining in the hot leg which has been pressurizing the system is lost through the dump line and pumped to the cold leg, the pressure falls rapidly until the saturation pressure of the hottest fluid in the cold leg is reached. At this time the cold leg fluid flashes.

The blowdown calculations show that cold leg flashing begins after about 200 seconds. ERWS flow begins at about 190 seconds. The cold leg does not become subcooled again until about 430 seconds. During this period the worst case fuel temperature transient is calculated assuming the steam-water mixture is completely separated into the two phases. As a result a

*REF. 1 - HW-82239 "NPR Accidental Dump Transient Analysis" J. Muroaka, May 15, 1965 Confidential.

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few process tubes near the top of the reactor are cooled by saturated steam at the inlet which becomes superheated in passing over the fuel column. On this basis, which is extremely conservative, the maximum driver element temperature increases to about 1120°F (600 C). Since the maximum driver temperature during normal equilibrium operation is 1060°F, the worst case excursion temperature is acceptable. The maximum driver temperature occurs at the end of the active zone and is attributed to the superheated steam, therefore, the target element temperature will also approach 1100°F. This also is acceptable.

Later in the transient, about 580 seconds, the cold leg will flash again and steam cooling may occur again until about 700 seconds. Calculations indicate that during this transient the driver and target temperatures will not exceed 1060°F or less than the maximum temperature experienced earlier. Beyond 700 seconds the cold leg will be subcooled. At about 925 seconds the entire loop will again be filled and the dump transient will be complete. Beyond 925 seconds some boiling may occur in the active zone but will not cause a temperature excursion in either the driver or the target elements.

The complete blowdown transient is shown in Figures 4 and 5. Figure 4 contains the flow transients and Figure 5 contains the outlet riser pressure and bulk outlet enthalpy transient. Figure 6 contains the complete temperature transients.

The solid line in Figure 6 shows the calculated variation of maximum fuel temperature with time on the assumption that some tubes will receive only saturated steam as coolant. The dashed lines show the more probable but less precisely calculable case of cooling with water in a nucleate boiling regime, i.e. no steam-water separation in the cold leg. The true temperature of the fuel during the transient will lie between these extremes. The temperature shown will apply equally to both the driver and target a short time after the scram. The initial temperatures apply to the driver element operating at the maximum specific power.

Although there are several significant differences in the operating conditions, most notably the power level, the maximum uranium fuel temperatures following an accidental dump for the Phase I case and the coproduct case are calculated by this model to be nearly identical. The maximum temperature for the Phase I case is 1060°F as compared to 1120°F for the coproduct case. This seemingly abnormal consequence has a simple explanation. The steam cooling transient begins about the time that ERWS

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flow begins. ERWS flow begins when the system pressure decays to the dead head of the high lift pumps (320 psia). Because of the higher power level in the coproduct case, the hot leg fluid is at a higher enthalpy before and during the transient than in the Phase I case. Consequently the pressure decay rate is much slower. As a result steam cooling does not begin until about 200 seconds in the coproduct case as compared to 80 seconds in the Phase I case. The greater power decay following scram offsets the initially higher specific power.

For comparison with the calculated fuel temperatures attained it is of interest to cite experimental evidence regarding the capability of the fuel to withstand temperature transients.

Tests of N fuel under severe temperature transient conditions have been performed over the past two years, on both unirradiated and irradiated fuel.

The unirradiated N fuel tests involved the heating of fuel elements to 1482°F (800 C) and quenching in cold water. No significant fuel damage was observed. The tests of irradiated fuels were completed in two series. The first series used two N inner, enriched, fuel elements and two N outer enriched fuel elements which had been exposed to 2200 MWD/T. These were heated in 300 seconds to 1800°F (982 C) and quenched in 68°F (20 C) water. No significant damage was observed. The second series involved five N fuel elements (both inners and outers) which were heated to, and held at, elevated temperatures. This was done to determine how long the fuel element would withstand extreme conditions and to define what these failure conditions were. These data confirm the integrity of the fuel for long periods of time at 1700°F (926 C) or above. Table V summarizes test conditions and failure times.

TABLE V

<u>#1</u>	<u>Type</u>	<u>Exposure</u>	<u>Test Temperature</u>	<u>Failure Time</u>
1	NIE	1600 MWD/T	1945°F (1070°C)	75 minutes
2	NIE	1600 MWD/T	1945°F (1070°C)	105 "
3	NOE	1800 MWD/T	1857°F (1025°C)	54 "
4	NOE	1800 MWD/T	1700°F (926°C)	87 "
5	NIE	1600 MWD/T	1920°F (1055°C)	3.2 "

It is clear that the fuel should easily withstand temperature transients of the size computed in the present study.

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Conclusions

In an accidental dump during coproduct operation at 4800 MW, if the coolant behaves as a homogenous mixture whenever boiling occurs, then the maximum fuel element temperature during the transient will not exceed the normal fuel operating temperature. If a complete or near complete steam-water separation of the flashing cold leg coolant at the inlet to the process tubes occurs, then the maximum fuel temperature can increase to about 1120°F during the transient. But even in this worst case the fuel temperature is considerably less than the alpha-beta transformation temperature for uranium (1230°F - 660 C), and several hundred degrees below temperature which the fuel has withstood in laboratory tests.

The maximum driver element temperature occurs in a tube cooled by steam. The peak temperature occurs at the tube outlet and approaches the local steam temperature. Therefore, the non-heat generating target element also approaches 1100°F or the exit steam temperature in the worst case. The lowest melting material in the target is aluminum, m.p. 1230°F (660 C) thus no target damage is likely.

The results of this worst case study show that neither the coproduct driver nor the target elements would be damaged by the temperature transients resulting from a primary loop dump.

The mechanical aspects of a loop dump in the coproduct mode do not differ sufficiently from those of the present mode of operation to permit drawing different conclusions about their effects, i.e., in neither case is loop damage to be expected.

Figure 4
Flow Transient

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Figure 5
Pressure and Enthalpy Transients

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Figure 6
Temperature Transient

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ADMINISTRATIVE SUMMARY

NOVEMBER 1965

EMPLOYMENT

SECTION	Permanent		Temporary		Total		Change		Total
	<u>E</u>	<u>NE</u>	<u>E</u>	<u>NE</u>	<u>E</u>	<u>NE</u>	<u>E</u>	<u>NE</u>	
General	3	2	-	-	3	2	-	+1	5
Finance	23	15	-	-	23	15	+2	-	38
Fuels	38	115	6*	-	44	115	+4	-	159
Plant	59	210	2*	-	61	210	+1	+6	271
Project	68	16	1*	-	69	16	+1	+2	85
Research & Engineering	54	22	1*	-	55	22	+2	+1	77
Employee Relations	<u>7</u>	<u>3</u>	<u>6**</u>	<u>1</u>	<u>13</u>	<u>4</u>	<u>-2</u>	<u>+1</u>	<u>17</u>
Total	<u>252</u>	<u>383</u>	<u>16</u>	<u>1</u>	<u>268</u>	<u>384</u>	<u>+8</u>	<u>+11</u>	<u>652</u>

652

Department Total

+19

Net Change

*MIP

**Tech Grads

Safety

Days without a disabling injury 180

Hours worked without a disabling injury 593,368

VISITORS

<u>Name</u>	<u>Component & Location</u>	<u>Contact</u>	<u>Date</u>	<u>Purpose</u>
A. B. Carson P. R. Hill, Jr.	GE. Santa Clara, California	W. J. Dowis	11- 8 and 11- 9-65	Division Safeguards Evaluation Function - Review Nuclear Safety Criteria, Guides and Standards.
D. D. Cannon	Union Carbide Corp. Oak Ridge, Tenn.	R. K. Robinson	10-26 and 10-27-65	Discussed desalination reactor studies.
R. Hubbard	GE - APED, San Jose, California	G. T. Haugland	11- 4-65	Discussed drive system for supplementary control studies.
N. H. Meyers	GE. Pasco, Washington		11- 4-65	Discussed drive system for supplementary control studies.
D. Guy	Alcoa Aluminum, Seattle, Washington	D. Marinos	11- 4-65	Discussed canister fabrication.
J. Segall	Burns & Roe, Hempstead, N.Y.	R. C. Hoffman	11- 4-65	Discussed Conversion electrical design.
L. M. Nelson D. Covey	National Automatic Sprinkler Co. Seattle, Wn.	R. C. Hoffman	11-11-65	Discussed fire protection for 100-N.
L. M. Nelson D. Covey	" "	D. B. Janes	11-11-65	Discuss scoping requirement problems.
P. B. Smith	Bailey Meter Co. Seattle, Washington	G. L. Swezea	11- 9-65	Discussed Bailey instrumentation.
J. P. Blackwell	Engelhard Industries Newark, New Jersey	J. L. Rand	11- 4-65	Discuss improvements of the Engelhard strapon RTD design.
R. L. Meyer	Betz Laboratories, Inc. Seattle, Washington	P. K. Telford	11-18-65	Discuss chemical cleaning agents for the HCR coolant channels.
		J. H. Fastabend	11-18-65	Discuss fuel oil treatment for 184-N.

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VISITORS

<u>Name</u>	<u>Component & Location</u>	<u>Contact</u>	<u>Date</u>	<u>Purpose</u>
R. G. Hayes	Tektronics Inc. Beaverton, Oregon	A. C. Rediske	11-11-65	Demonstrate the use and application of three Tektronics scopes. Provide training for instrument and electrical maintenance personnel.
J. R. Mathews	Marotta Valve Corp. Santa Ana, California	N. J. Roy	11-18-65	Discuss solenoid valve O-ring applications.
		J. H. Fastabend	11-18-65	Discuss high pressure, high temperature pipe couplings and high temperature O-ring status.
Roger Graham	Wah Chang Corp. Albany, Oregon	D. W. Darsow	11- 9-65	Discuss zirconium purchase practices.
I. G. Rice	AEC Headquarters,	L. C. Clossey	11-12-65	Discuss N-Reactor classification and
D. J. Pflaum	Washington, D. C.			tour of plant.

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VISITS

R. L. Dickeman	AEC Headquarters,	F. P. Baranowski	11- 4-65	Meeting on confinement.
M. C. Leverett	Washington, D. C.			
R. L. Dickeman	AEC Headquarters,	" "	11- 5-65	Meeting on coproduct program.
M. C. Leverett	Washington, D. C.			
W. S. Nechodom				
J. W. Riches				
T. W. Evans				
J. W. Riches	ASTM Committee Meeting	ASTM Subcommittee	11-3-65	Proposed specifications for Zr products for non-nuclear applications.
R. L. Dickeman	AEC Headquarters,	F. P. Baranowski	11-16-65	Coproduct program.
M. C. Leverett	Washington, D. C.			
J. W. Riches				
W. S. Nechodom				

VISITS

<u>Name</u>	<u>Component & Location</u>	<u>Contact</u>	<u>Date</u>	<u>Purpose</u>
R. L. Dickeman M. C. Leverett	ANS-AIF Meeting Washington, D. C.		11-15 to 11-19-65	Attend meetings.
M. C. Leverett N. R. Miller	GE Technological Hazards Council Meeting Washington, D. C.		11-17-65	Meeting of GE-THC
N. R. Miller	Savannah River Laboratory Augusta, Georgia	E. Nelson	11-18-65	Discuss filter testing at SRL.
	Oak Ridge National Lab. Oak Ridge, Tenn.		11-19-65	Discuss filter testing at ORNL.
K. O. Creek F. M. Newby	Granger Associates Palo Alto, California	G. Riser D. Meadows	11- 9 and 11-10-65	Acceptance of TV system.
D. W. Darsow J. J. Wick	Brush Beryllium Elmore, Ohio Cleveland, Ohio	K. G. Wykle D. E. Evans R. L. Harnden	10-25 and 10-26-65	Vendor liaison pertaining to zircaloy cladding hardware and strip fabrication.
	Reactive Metals Niles, Ohio	W. G. Fetter	10-26-65	" "
	Nuclear Metals & Equip- ment, Apollo, Pennsylvania	A. Blum W. Behahn J. Dickey	10-27-65	" "
	Carborundum Metals Climax Akron, New York	W. W. Stephens J. V. Houston H. A. Andersen	10-28-65	" "
P. W. Hood	K-F Products Denver, Colorado	M. Keller J. J. Farrell	11-8-65	Routine vendor liaison in regard to quality, production and delivery problems.

VISITS(cont)

<u>Name</u>	<u>Component & Location</u>	<u>Contact</u>	<u>Date</u>	<u>Purpose</u>
P. W. Hood	Great Lakes Carbon Morganton, No. Carolina	R. Johnson	11-10-65	Routine vendor liaison in regard to quality, production and delivery problems.
		J. Meletioui		
		E. Denson		
		J. Mitchell		
		K. A. Welder, Jr.		
	Anaconda American Brass Ansonia, Connecticut	D. C. Gager	11-11-65	" "
		D. S. Chamberlain		
		E. K. Begg		
	Anaconda American Brass Buffalo, New York	G. Wyckoff	11-12-65	" "
		D. Henderson		
		W. Bell		
		B. Gustoven		
		N. Senf		
W. J. Dowis	Atomic Industrial Forum Washington, D. C.		11-15 to 11-17-65	Attend AIF Meetings.
	AEC, Germantown, Maryland	W. A. Williams, Jr.	11-17-65	Discuss Desalination Program.
J. D. McCullough	U. S. Court House New York City, N.Y.	P. H. Gantt, Chairman	10-28-65	Attend hearing before US AEC Board of Contract Appeals on the Avien claim.
M. G. Patrick	Boise Cascade Corp. Wallula, Washington	B. Lowe	11-19-65	Examine a package boiler instal- lation.
W. D. Bainard	Artisan Industries Waltham, Mass.	C. Angell	11- 1-65	Discuss equipment for the operation of radioactive wastes.
	Barnstead Still & Ster- ilizer, Boston, Mass	V. Smith	11- 1-65	" "

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VISITS (cont.)

<u>Name</u>	<u>Component & Location</u>	<u>Contact</u>	<u>Date</u>	<u>Purpose</u>
W. D. Bainard	Electric Boat, Groton, Conn.	E. Holt	11- 1-65	Discuss equipment for the operation of radioactive wastes.
	AMF Atomics, York, Pa.	P. Karter	11- 2-65	" "
	F. J. Stokes, Philadelphia	M. Fuentevilla	11- 3-65	" "
	Chicago Bridge & Drain, Oakbrook, Ill.	D. Madden	11- 4-65	" "
	Aqua-Chem, Waukesha, Wisc.	R. Ahlgren	11- 4-65	" "
	Swenson Evaporator, Harvey, Ill.	S. Macek	11- 5-65	" "
	Liquid Carbonics, Chicago, Ill.	T. D. Muhala	11- 5-65	" "
D. L. Condotta	Wickes Boiler Co.	R. Herrick	11- 2-65	Investigate package boilers and discuss application to supplement existing steam capacity at 100-N.
R. K. Robinson	Saginaw, Mich.			
H. L. Pringle	Babcock & Wilcox, Ohio Union Iron Works, Erie, Pa.	G. Boatright R. Sevin	11- 3-65	
H. L. Pringle	Combustion Engineering Windsor, Conn.	E. Barber	11- 5-65	" "
D. L. Condotta	Large Steam Turbine Dept.	D. H. Brown	11- 5-65	Discuss 185-N T-G sets.
R. K. Robinson	GE - Schenectady, N.Y.	D. L. Lustenader		
		F. L. Lamprey		

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SIGNIFICANT REPORTS

Report Number

RL-NRD-164	U	N-Reactor Conversion Testing Program Start-up Test N-4.	D. L. Renberger	November 16, 1965
RL-NRD-427 SUP 1	U	PT-NR-46, Primary Drive Turbine Speed Reduction Test, PT Change No. 1.	D. L. Renberger	November 8, 1965
RL-NRD-505 REV	C	N-Reactor Heat Balance and System Parameters.	V. J. DeJong et al.	August 12, 1965
RL-NRD-507	U	Startup Testing of Turbine Generator No. 2, Conversion Test 5.2.	D. L. Renberger	November 16, 1965
RL-NRD-521	U	Conversion Test 5.1, Steam Blow Test.	A. J. Ebens	November 16, 1965
RL-NRD-585 RD2	U	N-Reactor Design Bases, Effluent Control Facilities.	D. D. Stepnewski	November 8, 1965
RL-NRD-593	U	Evaluation Report, PT-24, Plant Service Boiler Base Load Reduction Program, Test Procedure No. 5.	A. J. Ebens	October 25, 1965
RL-NRD-606	U	Flow Restrictions from Misaligned Spacer Elements.	K. R. Ellison	October 26, 1965
RL-NRD-610	U	Use of Computer Code RIBD for Fission Product Analysis.	J. L. Rash	November 19, 1965
RL-NRD-614	U	N-Reactor Radioactive Effluent Control Program.	R. E. Trumble	November 24, 1965
RL-NRD-615	U	Development Test #4 - Dump Condenser No. 1 Flow and Level Tests.	A. C. Rediske J. T. Long	November 2, 1965

Report Number

RL-NRD-621	U	Project Proposal - Replacement of 5 KV Cable, 100-N Area.	A. A. Janos	November 5, 1965
RL-NRD-622	U	IRACS Report - Boiler-Turbine-Generator Deaerator Reliability Improvement Program.	J. R. Bolliger	November 5, 1965
RL-NRD-625 - 10	U	Radiological Status of the N-Reactor for October, 1965.	M. M. Hendrickson	November 5, 1965
RL-NRD-626	C	Accidental Dump During Coproduct Operation.	J. Muraoka	November 8, 1965
RL-NRD-630	U	109-N Steam Generators - Plugged Tube Record.	G. T. Haugland	November 11, 1965
RL-NRD-642	U	Backup Steam Supply Study - 100-N.	D. L. Condotta	November 19, 1965

SUGGESTION PLAN PARTICIPATION

	<u>NOVEMBER</u>	<u>YEAR TO DATE</u>
Number of Eligible Employees	382	375
Number of Suggestions Received	47	499
Number of Suggestions Acted Upon	55	513
Number of Suggestions Adopted	18	218
Net Annual Savings	\$4,956	\$33,166
Amount of Awards	\$ 735	\$ 5,715
Percent of Awards to Savings	14.8	17.2
Average Amount of Awards	\$ 40.83	\$ 26.22
Rejection Rate per 1000 Employees	97	797
Adoption Rate per 1000 Employees	47	581

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**DATE
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8/22/94

END