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HW.

N - REACTOR DEPARTMENT

MONTHLY REPORT

DECEMBER 1964

[redacted]

HANFORD ATOMIC PRODUCTS OPERATION

RICHLAND, WASHINGTON

GENERAL ELECTRIC

MASTER

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N-REACTOR DEPARTMENT
MONTHLY REPORT - DECEMBER, 1964

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 Residual	225
Highest Exposure Tube (MWD/T)	933
Input production Pu (KMWD)	82.72
Input production T (KMWD)	0
Maximum power level attained (MW)	4,000
Average power level while operating (MW)	3,476
Time operated efficiency (%)	77
Plant utilization (%)	77
Number of Shutdowns	3
Scheduled	0
Unscheduled	3
Reactor Process Data	
Power at which computed (MW)	4,000
Primary coolant inlet temperature (^{o}F)	354
Primary coolant outlet temperature (^{o}F)	495
Maximum graphite temperature (^{o}F)	1,150
Heat Loss	
Graphite cooling (MW)	47
Shield cooling (MW)	2.1
Rod cooling (MW)	1.5
Steam temperature (^{o}F)	313
Steam pressure (PSIG)	66
Dump condenser flow (gpm)	325,000
Inlet temperature (^{o}F)	47
Outlet temperature (^{o}F)	132

[redacted]

Nuclear Steam Generating Rate	
Maximum (1000's lb/hr)	12,500
Average while operating (1000's lb/hr)	12,000
Water Collected from Gas Atmosphere (gal/day)	Negligible
Demineralized Water Produced (gal/hr)	65,238
Helium Losses (1000's cu. ft.)	319
Fuel Oil Usage (bbls.)	38,390
Flattening Efficiency (%)	82
Fuel Conditions	
Heat flux	
Maximum (BTU/hr/ft.^{2})	585,000
Average (BTU/hr/ft.^{2})	470,000
Clad temperature	
Maximum (^oF)	575
Average (^oF)	535
Uranium temperature	
Maximum (^oF)	870
Average (^oF)	775
Fuel Failures	0

Fuel Balance	0.95 %	1.25 %	Coproduct	Other	Total
Fuel charged (Tons)	0	0	0	0	0
Fuel discharged (Tons)	0	0	0	0	0
Net change (Tons)	0	0	0	0	0
Total in reactor at month end (Tons)	327.75	39.74	0	0	367.49

Fuel Element Output (Assemblies)

0.95%	1997
1.25% (driver)	0
1.25% + 0.95% (spike)	0
Normal uranium	48
Total assemblies	2,045
Total tons	40.6
% of forecast	135.3
Uranium Utilization (%)	70

SUMMARY

The reactor reached its full design power level of 4000 MW early in the month, and, except for two temporary shutdowns, continued at this level. Operating conditions were as predicted and performance was excellent. There was no increased tendency toward flux cycling. p 8

Manual shutdowns or scrams were caused by broken valve stems, air in steam lines, a leaking steam generator tube, a leak in an oil line to the boiler, and a loose wire in an electrical circuit. p 9

Unusually cold weather froze some sensing lines and resulted in slow operation of hydraulic pilot valves. p 10

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At full power equilibrium operation the excess reactivity held in the control rods was 8.5 mk and the flattening efficiency was 0.81. p 11

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The use of ammonia instead of lithium hydroxide to control pH in the graphite cooling system gives at the same time a remarkable control of oxygen down to 7 ppb. p 38

Technical investigations are being made on the dysprosium and europium as poisons for control rods, on the allowable strain limits for process tubes under bending stresses, and on imbalances in temperatures in fuel element coolant channels. p 40

REACTOR PLANT OPERATION

N-Reactor reached full design power operation at 4000 MW on December 9, 1964, less than 12 months after initial criticality was obtained. The final ascension from 3600 took place without incident. Operation at full power continued for the rest of the month, with only two interruptions. The following table summarizes plant operating conditions with five pumps running and sixteen dump condensers on the line.

<u>Plant Operating Conditions at 4000 MW</u>

Parameter	Expected	Observed
Average Reactor Inlet Temperature, ^{o}F	354	353
Average Reactor Outlet Temperature, ^{o}F	497	494
Maximum Reactor Tube Outlet Temperature, ^{o}F	525	526*
Primary Coolant Flow, lb/hr	85.8×10^{6}	85.5×10^{6}
Max. Tube Power, kw	5000	4930**
Max. Power Density, kw/ft	230	215
Max. Fuel Cladding Temperature, ^{o}F	600	575
Excess Reactivity held in control rods, mk	8	8.5 ± 1.5
Flattening efficiency	0.8	0.81
Pressurizer pressure, psig	1425	1420
Max. Steam Pressure, psia	80	80

It will be observed that performance is almost exactly as had been predicted prior to the final ascension. Operation is well within the limits imposed by the Process Standards and the Operating Limits. Operation of the reactor is stable although a slow drift of power from front to rear has been noticed, which is perhaps indicative of moderate proximity to the cycling threshold. The minimum downtime resulting from a scram at these conditions would be 22 hours to critical.

* Maximum allowed by Process Standards is $530^{o}F$.

** Includes estimated 75 MW actually removed via graphite cooling system.

Continuity of Operation

The plant was shut down on the first day of the month because of a broken stem on one of the main steam pressure reducing valves from the oil-fired boiler. Operation was resumed on December 2. On December 4, the reactor was shut down to investigate a suspected leak in steam generator 5B. The leak was confirmed and found to be caused by a circumferential leak on one peripheral tube. It was plugged, as was an adjacent tube which was found to be damaged by vibration. An examination was made of steam generator 3A, which had previously shown leak indications. A similar broken tube was found in it. This tube was plugged off, and reactor operation was resumed on December 6.

On December 8, an increase in power to full design reactor power was authorized. The increase was completed and operation at full power established at 2245 hours on December 9, 1964. The plant remained at full design power for the next seven days.

On December 15, the reactor was manually shut down to investigate a suspected broken stem on the primary loop block valve to steam generator 1A. The valve stem was found to be broken in a similar manner to that experienced on a similar valve on November 5. A dummy valve stem was installed and operation was resumed on December 17.

On December 20, a reactor scram occurred from full power, caused by a momentary interruption of the rod scram safety circuit when an electrician moved a loose wire on the 125 V DC power to the safety circuit. The plant performance on the scram was extremely good. Required scram actions took place, except that drive turbine No. 4 failed to slow down automatically and had to be tripped off manually. Primary loop repressurization following the scram was only 110 psi. The loose wire, and the 125 V DC ground which was being corrected when the scram occurred, were repaired, and operation was resumed on December 21.

The data taken during this scram are being examined to determine whether or not they will suffice in lieu of the planned scram from 4000 megawatts which has been a part of the future program of operation.

Plant Performance

Plant performance continued to be excellent during reactor operation. The full power operation tube powers were more than 150 KW below maximum allowable and tube outlet temperatures more than 5 degrees below maximum allowed. No increased tendency towards flux cycling was evident.

Dump condenser performance continued to be satisfactory. At full power operation, indicated condensate levels in most of the dump condensers were zero. Additional testing determined that the actual level was about plus two feet for these conditions and sufficient condensation capacity was evidenced by continued low circulating raw water and condensate outlet temperatures.

Boiler response to system demands continued to be satisfactory. During most of the month, the boiler was operated with only six burners in service when the reactor was in operation. Automatic lightoff of four burners was tested during reactor shutdown, and during the scram of December 20, six-burner automatic lightoff was experienced with no difficulty. Authorization to reduce to four-burner operation was received on December 28. This reduction will proceed as soon as some repairs and adjustments can be made to the fuel oil supply system.

The boiler base load will be down from 180,000 lb/hr to 120,000 lb/hr.

Some temporary filters have been installed on chlorine analyzers and on the demineralized water supply lines to filter out fine iron oxide particles. It now appears that a more extensive installation of filters will be required.

Unusual Incidents

On December 5, during reactor shutdown to investigate a broken valve stem, extremely cold weather occurred. This resulted in a number of abnormal situations in the plant:

1. Abnormally slow closure times were experienced on both steam vent valves and ventilation confinement valves on the 109N building roof. These hydraulic mechanism and supply lines are exposed and the cold weather caused sluggish operation of the hydraulic pilot valves. Satisfactory performance was obtained by cycling the valves closed at four-hour intervals until the cold weather relented. One of the roof vents became non-operational and it was necessary to restrict maximum loop temperature to a level such that loop enthalpy was consistent with the relieving capacity of the remaining vents. This entailed a modest power restriction.

2. Sensing lines at exposed locations throughout the plant were frozen, including steam flow sensing lines on all of the steam generators. These were thawed out, but at the time of the scram on December 20, steam flow sensing lines for steam generators 5A and 5B were found to be frozen again, and the presence of an erroneous flow signal following the scram caused unnecessarily high feedwater flow rates into the steam generators.

3. The closing time of the discharge valve for the No. 1 circulating raw water pump was adversely affected. This condition was alleviated by building a heated shelter over each of the valve operating mechanisms.

4. On December 22, a hydraulic pressure line broke on the hydropack to the ventilation gates in the stack exhaust filter building. The gates were blocked off and the reactor power was reduced to a safe operating level such that no fuel melting would have occurred even if primary coolant had been lost and emergency raw water had failed to come on.

5. For a short time, one control rod was stuck in the 'in' position, thus making it impossible to achieve the degree of power flattening necessary for operation at 4000 Mw. This resulted in a power restriction.

Operations Staffing and Certification

Nine supervisory and twenty-one operator candidates successfully completed the fourth or oral examination phase of their certification testing, and are, therefore, certified for their assignment in the Plant Control Center.

PROCESS PHYSICS

Equilibrium Conditions

Reactor Power Level (MW)	4000
Reactivity in Rods (mk)	8.5
Flattening Efficiency	83 percent
Axial Power Distribution	Maximum peak/average ratio as measured by in-core flux monitoring chambers is 1.51, which is about 3 percent higher than a cosine distribution
Control Rod System Worth	67 ± 4
Spike Column Loading	144 columns (no change)
Reactivity contribution of spike enrichment (mk)	6

The physics assessment for equilibrium operation was essentially the same as reported in the table on page 8, except for a slight (0.5 mk) increase in operating reactivity resulting from increased fuel exposure. Maximum axial peaking indicated by in-core monitors was 1.51 peak/average.

A zero power critical check on December 2 indicated a cold reactivity gain of 0.6 mk. The cold, xenon-free reactivity was 38.7 mk. Turnaround after the startup of December 6 was achieved at a power level of fifty percent with a shutdown margin of greater than 11 mk, as computed for the hypothetical cold water injection accident. All startups are so conducted that, should a cold water injection accident occur, the rod system would be able to shut down the reactor with 10 mk of reactivity to spare.

MECHANICAL AND THERMAL EQUIPMENT PERFORMANCE

Grayloc Repair

Drawings are now complete for the new three-in-one tool to speed redressing of Grayloc flanges in the front and rear face connector piping. The three machining operations (seat cut, face cut, and chamfer), previously requiring three separate tools, are now combined into one tool. The physical dimensions of the tool have been significantly reduced to permit installation in the outlet pipe space without alteration to structural framing.

Testing was completed on the use of hollow metallic O-rings for sealing leaks in the Grayloc couplings. After more than 500 combined thermal and bend cycles (at temperature limits of 395° and 555°F, and 11,000 inch pounds bending moment), four fittings were pressure tested to 2100 psig. One of the four O-ring assemblies had a slight leak, which upon disassembly was found to be due to a radial scratch across the face of the Grayloc. Further evaluation of the O-rings will continue.

Control Rod Seals

Test and development work on cylinder rod seals continues on an intermittent basis. Four seal materials in various arrangements have been tested. All have shown excessive leakage or have failed by extrusion of the material between the cylinder rod and the male adapter ring on the pressure side of the packing. Other materials and seal arrangements will be tested.

Fuel Element Spacers

Some fuel element spacers have been chattering from hydraulic forces in the process tubes and causing some marking of the tube inner surfaces. Inspection of upstream spacers from four process tubes on-reactor revealed evidence of movement, although nowhere near the magnitude evidenced prior to removal of the first spacer. Boroscoping of four process tubes has been scheduled for the next fuel discharge outage. Work was initiated on the special ultrasonic device to be used with the full-tube mockup in detecting sounds of fuel spacer movement beyond the audible range. Special transducers and a coupling carrier medium are being investigated for durability in the 550°F temperature range.

Rupture Monitor System Control Valves

The existing valves have been operating satisfactorily with the Hastalloy sleeves installed to arrest the erosion of the valve bodies downstream of the valve trim. Needle point trim was installed in the maintenance valve upstream of the control valve and in the bypass valve. The needle point trim permits fixed throttling and reduces the pressure drop across the control valve. This allows the control valve to operate in a more open position and reduces the erosion problem. The flow range was reduced slightly based on operating information obtained to date. This will permit a smaller valve trim to be used which in turn will reduce the erosion problem. The present adjustments are temporary measures, and the control valves need to be replaced.

Regulated Air Supply

During routine observation of the regulated air supply for the diversion valves and inlet butterfly valves, it was noted that the refrigeration system was introducing moist air into the regulated air supply system. Several leaks were located in the refrigerant piping. The leaks were repaired and the refrigerant charge restored to normal. Calibration of pressure and temperature gages on the refrigeration unit will permit closer monitoring of the unit.

Injection Water Valves

During an investigation of the slow response of the injection water valves to open, it was found that the valves had a 3 to 15 psig diaphragm and that the air supply to these valves was set at 40 psig. This meant that 25 psig of air had to be bled from the diaphragm before the valve would start to open. The supply air was set at 20 psig and the trapping valves and pressure switches associated with the air failure detection system recalibrated. In addition, the pneumatic counting circuit was set up to allow both valves to operate in a full open position when four pumps are running rather than with one valve less than 80 percent open.

Water in Lube Oil System

Evidence points to the north oil cooler as the possible source of water in the No. 2 primary drive turbine lubrication system. No water contamination in the oil has been experienced since this cooler was isolated and the south cooler put in service in November. Turbine speed transients similar to those experienced during the initial water contamination incidents have been encountered, with no apparent problem resulting. The north oil cooler will be tested with air or water in an attempt to locate a leak.

Drive Turbine Speed Control

The cleanup of snubbers in the controller appears to have improved the control system. The turbines now correctly reduce speed on discontinuous action. Data on turbine loads versus speeds were taken at several intervals as the drive turbines increased speed up to full load. These data were compiled and graphed to provide the manufacturer's representative with information to assist in correction of the existing 2400 RPM speed range of the electrohydraulic transducer to a range from 900 RPM to 3600 RPM. It is expected that this correction of the speed range will eliminate the additional control problems which appear from time to time.

Afterheat Removal Tank

Recently the membrane in the afterheat removal tank was removed for the third time with large rips in it. Interim methods of operating the tank were adopted to minimize the amount of oxygen entering the process systems prior to final correction of the problem. Subsequently, an engineering study concluded that a full-length membrane should be suspended from the top of the tank rather than from the midpoint.

Steam Generators

On December 3 an increase of about 200 gal/min occurred in the makeup flow to the primary coolant system. This was equivalent to the leak rate for a complete circumferential break of a steam generator tube. Analysis of condensate return flows and steam flows showed that leakage was occurring in unit 5B. The following day the reactor was shut down because of high ammonia content and activity levels detected in the secondary system. A broken tube and a damaged tube in unit 5B were located and plugged. The breakage was apparently caused by fatigue, presumably from vibration of the tube in the steam-water mixture.

There is little or no indication that corrosion was partly responsible. Rubbed marks were found indicating that the tube had been rubbing on its neighbors and it is believed that additional support against vibration may be required for this tube and for some others because of the long unsupported lengths.

One tube in steam generator 3A which had given previous indications of gradually increased leakage also was plugged off. It is situated in a position similar to that of the ruptured tube in 5B.

Multichannel analyzer tests of blowdown samples subsequently taken from all steam generators did not indicate detectable leakage from the primary to the secondary side of any unit. However, later samples gave some indication of leakage in unit 1A. Blowdown samples from all units are being taken regularly for analysis, to detect any new leaks that may occur.

It is necessary to control the liquid level in the steam generators so that there is no carryover to the turbines. In addition to the use of instruments to indicate the liquid level, it has been advisable to develop empirical relationships and some understanding of how they change as the power level changes. These relationships have been studied at 2400 MW and at each of the plateaus above this power level. These data have made possible successful control of the liquid level at various operating power levels.

Cell Valve Stem Failure

On December 15 a third 17-4 PH valve stem failed in a primary loop block valve in the 18" return line to steam generator 1A. This is the second failure of an 18" valve stem. The break occurred in the same place and the same manner as the previous failure. The reactor was shut down and the broken stem replaced with a dummy stem.

A contingency maintenance job authorization for the replacement of stems in

25 valves now installed in the primary loop was approved by the AEC. It was further agreed that, due to the urgency of this problem, the Commission would take action to release sufficient funds to cover the cost of raw materials and forging operations. This would permit production to start while contractual negotiations were still in progress, and thereby expedite delivery by four to six weeks.

Loss of Vacuum in Condensers

A simultaneous loss of vacuum in all drive-turbine condensers occurred recently when the source of steam for the drive turbines was inadvertently shifted from boiler steam to steam-generator steam early during the reactor startup. The loss of vacuum was caused by air that had not been adequately vented from the 28" and 46" steam-generator lines. Operational procedures have been modified to lessen the possibility of recurrence. Permanent changes are being studied for additional assurance that the venting system will operate as intended.

Dump Condensers

The shell side of dump condenser No. 1 was entered and the condition of the tube bundle and tube bundle support mechanism inspected. The tubes and support mechanism were found to be in good condition, with very slight wear (approximately one to two mills deep) on the tubes where they are in contact with the support spacers. The support mechanism showed no sign of deterioration and there was no visible corrosion. The instrumentation leads were broken off the transducers and a sheathed strain gage lead was broken loose, indicating the presence of substantial turbulence during operation.

Dump condensers No. 5 and No. 6 were opened and inspected from the water side. No leakage was found in No. 5, but one leaking tube in No. 16 was repaired.

Two tubes in dump condenser No. 1 were recently found to be leaking when entry was made to the raw water side to remove a temporary baffle. Chemical analyses of dump condenser condensate had not indicated any increase in hardness or chloride content, and some procedural changes in sample taking will be made to make the chemical analysis a more certain method for detection of tubing failures.

Figure 1 shows the tube bundle support, with the carbon steel U-shaped spacers. The spacers are used to prevent side movement of the 3/4" OD tubing as well as to allow a more even shell-side distribution of the steam. The entire assembly is spring-mounted to allow for thermal expansion of the tubes relative to the heat exchanger shell. The assembly provides support for the tubes under all operating conditions. Approximate spring forces and stresses for the 302 stainless and for the Inconel X springs as initially installed and at operating temperature are:

Material	Temperature	Compression	Force	Calculated Stress*
302	Cold	1/2 inch	1960	119,000
302	Hot	3/8 inch	1470	85,000
Inconel X	Cold	7/16 inch	1400	123,000
Inconel X	Hot	5/16 inch	1000	92,000

* The calculated stress includes the correction factor for curvature and vertical shear stress.

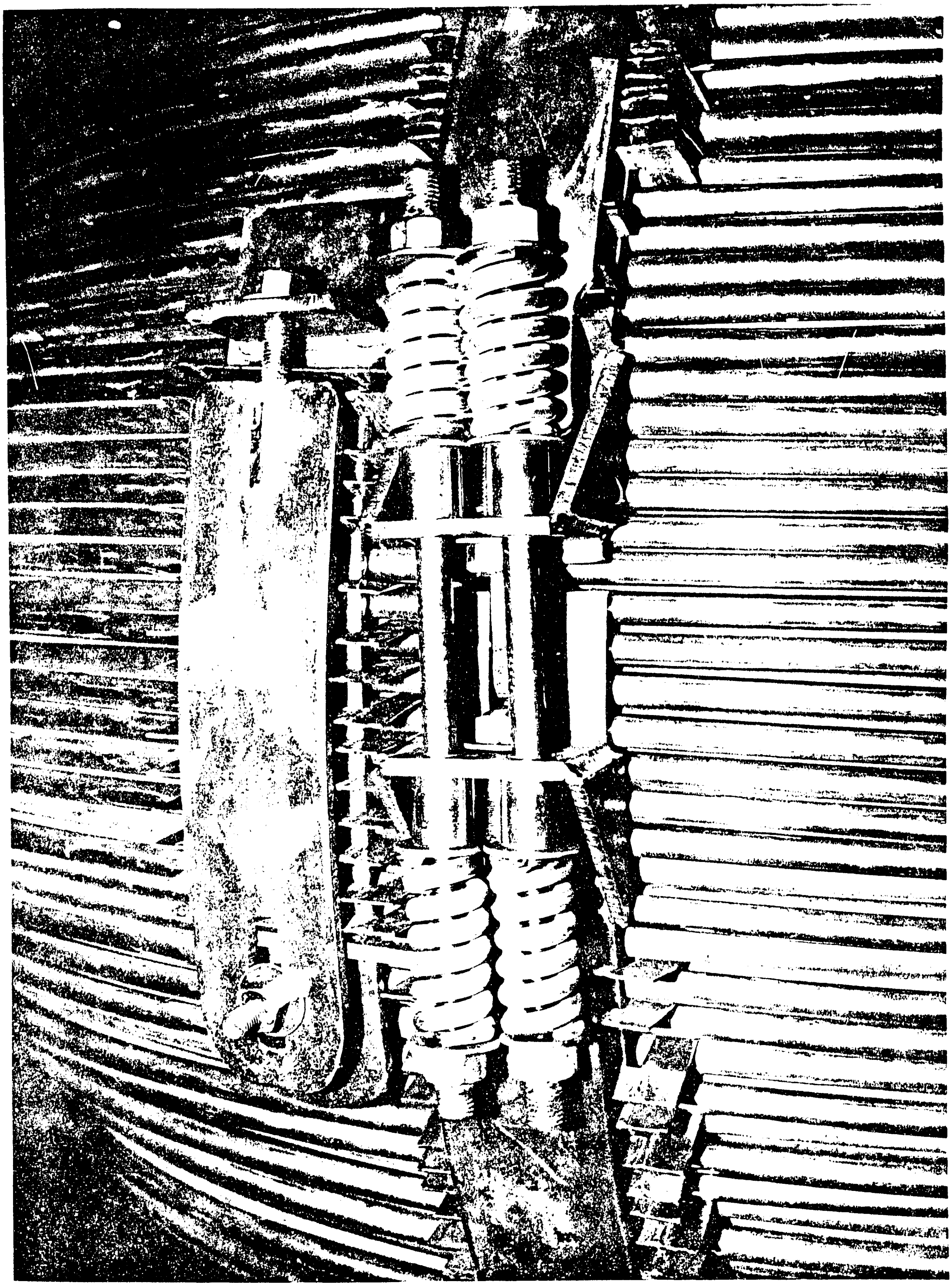

Figure 1

Cell Sump Pumps

The sump pumps have been subject to mechanical problems and to blowing of fuses from overloads. A recording ammeter was installed on motor No. 5 and is working well. This pump apparently does not have the mechanical problems experienced with the other four. Time lag fuses have been recommended for use on the motors. Fuses previously used were not always the time delay variety, and have blown on several occasions. An auxiliary 10 hp motor and sump pump will be added to each cell, and will be operated by the existing motor controller.

Storage Basin Cranes

A hazard study on the storage basin crane platforms is in progress. The platforms on cranes No. 1 and No. 2 are in sections and are suspended with pipe bolted to the crane superstructure. There is considerable vibration in the suspended sections while the cranes are being operated, and a study is underway of means of preventing a section of the platform from suffering fatigue failure and breaking loose.

Air Pumping in Outfall Line

The arrangement of the outfall seal well is such that air is pumped down the outfall line by the surging action of the water being returned from the dump condensers to the river. This effect was anticipated in design and vents were provided for relieving the air pressure. In mid-November, it was found that the pressure of the air in the line was high enough to cause a blowback through the lift station drain line, resulting in some spread of low-level radioactive contamination in the vicinity of the lift station. The condition was corrected by the partial removal of the cover on a manhole in the outfall line. This gave sufficient additional venting capacity to eliminate blowbacks.

Gravity Drain Line

A gravity drain line has been designed to prevent flooding of the sumps. Excavation for this line continued until several cave-ins occurred from excessive water drainage. The source of the water was found to be a leak in the 66" raw water return line at the penetration to the outfall seal well. The leak is about 50-70 gpm, and installation of the gravity drain line was deferred pending repair of the 66" line during the next plant outage, when the line can be taken out of service.

Confinement System Valves

The automatic transmission oil in the 72" steam vent confinement valve hydraulic system was replaced with Aircraft MIL-5606-A fluid. This less viscous fluid was added to decrease the valve operating time at low ambient temperatures.

Hydraulic fluid in other confinement valves was not replaced because of limited outage time; however, the valves were operated and timed on a cold day at +2°F to determine the effect of low temperatures. The valves that did not initially operate within the time specified by the Process Standard were cycled at predetermined intervals to reduce operating time.

Steam Pressure Reducing Valves

The main steam pressure reducing valves in the boiler plant have been operating satisfactorily since they were repaired last month. The failure of these valves occurred when the threads on the lower valve stem stripped out of the plug and the operator could no longer control the valve plug. The valves were repaired by installing new diaphragms in the operators, new valve stems, new bushings machined from 17-4 stainless steel, and by cleanup machining on the plug guiding surfaces. The hydraulic snubbers were serviced and adjusted. It is planned that these valves be replaced with a new pressure reducing valve station during the Washington Public Power Supply System tie-in outage.

Rupture Monitor System

In late November, the reactor was shut down because of large signals in both the gross gamma monitor and the gamma energy monitor for a particular process tube. These rupture indications were later shown to be the result of corrosion products trapped in the sample chamber.

Modifications of the rupture monitor system are being evaluated in an effort to desensitize it to corrosion products. It has been determined that the signal channel energy levels of the gamma energy monitor must be raised beyond the photopeaks of corrosion products. A multichannel analyzer spectrum will be taken of the coolant to determine the optimum setting of the signal channels.

A GM tube test facility has been fabricated, and is being used to test GM tubes before they are put into service, and also to verify that GM tubes removed from service are actually bad. A testing and development program has been iniated to investigate the GM tube failures which have been occurring at a rate of approximately one per day. The goals of this program are: (1) to determine cause of failure and methods of repair, and (2) to develop an improved GM tube if necessary.

Grounds in DC Power System

A series of 125 V DC grounding problems made necessary a continuing program of circuit checkout. The scram of December 20 was attributable to this problem. A few intermittent grounds are still being encountered, but not in critical shutdown circuits.

Dump Condenser Level Instrumentation

Temperature switches were installed on the condensate lines of the 16 dump condensers to serve as a detection device for low condensate level by signaling an alarm on a temperature increase in the dump condenser condensate.

SPECIAL TESTS

Status of Production Tests:

PT No.	Doc. No. HW-	Title	Status Approved	In Progress	Completed	Final Report
2	80369	Routine Graphite Sample Irradiations	X	X		
3	81478	Routine Monitoring of Graphite Oxidation in N-Reactor	X	X		
6	81339	The Use of Isotope Producing Rods in N-Reactor	X	X		
8	81327	Coproducer Demonstration Test (1.25)	X			
9	81609	Low Goal Irradiation of Fuel Elements with Varying Amounts of White Oxide on Surface	X	X		
10	82136	N-Reactor Corrosion Monitoring	X			
12	82241	Copper Bas Alloy Corrosion Program	X	X		
14	82385	Irradiation of Thorium-Uranium Crud Monitor Elements in N-Reactor	X	X		
18	82659	Primary Loop Pressure Relief Valves (RV-2-1 & 2) Relief and Reseating Pressure Test	X	X		HW-83175 (Interim Report)
21	83200	Optimization of Secondary Coolant Supply Operational Modes	X	X		

PT. No.	Doc. No. HW-	Title	Status: Approved	In Progress	Completed	Final Report
26	84094	Calibration of an Operational Rod Withdrawal Sequence	X			
30	84232	Evaluation of Ammonia for Controlling Oxygen in the Graphite Cooling System	X	X		
32	84367	Exposure of Corrosion Coupons in N-Reactor Process Tubes	X			
33	84401	In-Reactor Enthalpy Imbalance Measurements	X			
34	RL-NRD-26	Horizontal Rod Scram No. 5, Power Ascension Program, N-3	X	X		

OUTAGES

Date	Type	Duration (hours)	Reason
12/1/64	Manual Scram (continued from 11/30)	28.2 (In December)	Failure of HPV-108 valve stem in the steam line to the dump condensers.
12/2/64	Manual Scram	2.6	Low vacuum tripoff during startup of drive turbines and TG set from air in steam lines.
12/4/64	Controlled Shutdown	58.2	Leaking tube, steam generator 5B.
12/6/64	Controlled Shutdown	3.5	Boiler oil leak during startup.
12/15/64	Controlled Shutdown	50.7	Failure of PCRV-203-1 valve stem in primary system
12/20/64	Accidental Scram	29.1	Loose wire on 125 V DC power to safety circuit

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FUEL MANUFACTURE AND TECHNOLOGY

PRODUCTION STATISTICS

	Current Month	CYTD
Input (Extrusions)		
0.95% Outer	211	1098
1.25% Driver	0	85
1.25% Spike	0	378
0.95% Inner	54	1140
1.96% Driver	22	65
Total Extrusions	287	2766
Total Tons	57.8	512
% of Forecast	103.2	93.8
Output (Finished Production)		
Assemblies - 0.95%	1997	13,597
1.25% Driver	0	655
1.25% Spike	0	3,125
Normal Uranium	48	588
Total Assemblies	2045	17,965
Total Tons	40.6	351.6
% of Forecast	135.3	92.5
Uranium Utilization (0.95% Enriched)	70%	73%
Shop Yield		
Outers	86.3	81.0
Inners	74.1	77.6
Total	78.3	79.8

PRODUCTION SUMMARY

Production input exceeded forecast by three percent, and was the highest since October 1963.

The extrusion of 1.95 percent enriched driver fuels was completed early in December, approximately two weeks behind original schedule. This delay was the result of brazing problems encountered in the end closure process. Normal production flow of 0.95 percent enriched fuels was adversely affected by this delay. By month end, the initial production phase of the 1.95 percent enriched drive tube was completed and the bulk of the fuels were delivered to the exponential pile for reactivity testing.

Despite the problems associated with moving 1.95 percent enriched fuel elements through the shop, finished production of 0.95 percent material was 35 percent above forecast. This excellent performance was a result of effective scheduling which permitted the makeup of a six-ton deficit from November, plus the finishing of fuel elements previously held because of questionable quality. This material represents those fuel elements having minor deviations from specifications, but judged to be of sufficient integrity to perform satisfactorily if irradiated in selected zones of the reactor. In addition, inner tubes for these fuel assemblies were obtained by reprocessing fuel elements which had defective zircaloy supports. These fuel elements were reclaimed by removing the old supports, recleaning, and attaching new supports. There are approximately 500 fuel elements of this category, mostly in the 24-inch lengths.

PROCESS DEVELOPMENT

Unbonded End Closure Development

Development and evaluation of the tungsten inert gas (TIG) braze end closure continued. The current process developed for simulated driver tubes and N-outer fuel elements involves machining and chemical milling the V-shaped recess in the uranium fuel after beta heat treatment. The cladding is sized by cold rolling to obtain an ID that is concentric with the OD and to obtain a good fit between the cap and the clads to assure precision control of the TIG weld. Some unfavorable orientation of hydride platelet precipitates on the compression side of the cold sized bend has been noted after autoclaving. Formation of these oriented hydride platelets occurs for all values of offset in the cladding but does not occur in all specimens. Based on the evidence obtained from examination of these specimens, fuel element wall thickness variations greater than 0.030 inch should probably not be sized by the cold rolling technique. It should be noted that these hydride platelets will not exist at reactor operating temperatures because the amount of hydrogen involved is less than 100 ppm and will be dissolved by the zircaloy at temperatures above 300° C.

Horizontal Chemical Milling

Development of chemical milling for counterbored fuel elements can result in improved handling and in processing efficiencies. About 50 outer fuel elements have been horizontally milled, with no problems in excessive surface variation or uranium carryover on the zircaloy cladding. Additional quantities will be processed to define the process parameters and equipment changes necessary to convert to horizontal milling.

Inner Fuel Element Rewelding

A rewelding process for inner fuel elements is in the final stages of development. The process is designed to recover inner fuel element weld defects except excessive OD and ID overhang. The process differs from the reweld technique used for outer fuel elements in that no machining of the weld bead is required before rewelding.

A total of 51 inner fuel elements containing 61 defective welds have been rewelded to date, with a recovery rate of 82 percent. Data on penetration and depth of clad melt indicate that these parameters are well within process specifications.

Direct Cast Billets

Evaluation of direct cast coextrusion billets is well underway. Metallographic work to date indicates that a fine grain structure has been achieved by heat treatment of the as-cast structure. Extrustion pressures were not as high as anticipated, and the billet will be redesigned to obtain a higher reduction ratio, to improve clad-core bonding. The direct cast fabrication route offers considerable economic incentive since it would eliminate the requirement for primary extrusion of large diameter ingots.

Extrusion Press Alignment

Some problems were experienced with alignment of the extrusion press, and resulted in the inability to properly center the stem in the coextrusion liner closely enough to use traveling mandrels. Also, the main ram on the press was found to be rolling and raising as it moved forward during the extrusion cycle. These problems resulted from a buildup of graphite in the stem (contributing to the mandrel problem), and from a ram shoe not being properly positioned in the ram slide.

Target Fuel Fabrication

Sufficient target rods have been completed to supply material for assembly into the 1.95 percent enriched uranium driver fuel elements for the exponential pile tests. Fabrication of the remaining material on hand and reworking of target rods currently in reject categories will proceed at a reduced rate.

Rejects occurred principally in the extrusion and aluminum welding processes. The extrusion problem was caused by die pickup on the first co-extrusion; it has been corrected by improved fabrication techniques. The weld rejects were caused by cleaning difficulties; the majority of these rejects can be reclaimed.

Uranium Waste Recovery

The uranium recovery and disposal problem in the N-Fuels production plant has been reanalyzed. The original recommendations were to construct a separate facility to process waste solutions, based on the acid dissolution of uranium saw fines and lathe turnings, and recovery of these uranium waste solutions by neutralization with caustic and filter pressing of diuranate precipitate to a semi-solid oxide cake. The restudy proposes downgrading of the 1.95 percent enriched material to a 0.95 percent enrichment that could be recovered in the facility currently used. Several waste product materials of different enrichment levels have been considered.

PLANT MODIFICATION

Emergency Raw Water System Improvements

If cold water were to be introduced into the emergency raw water (ERW) storage tank, there would be a possibility of failure by brittle fracture. A proposed method of preventing such an incident consists in filling the tank with warm demineralized water, and allowing this water to be the initial source of emergency cooling supply to the high-lift diesels.

An optional second source of tempered water would be provided by a valved cross-tie line between the demineralized water storage tank and the ERW tank. The combined arrangement would provide over 30 minutes of once-through emergency cooling with warm demineralized water. (In many instances this would allow recovery from an inadvertent or a transient primary loop dump without requiring the introduction of raw river water into the reactor process tubes and primary piping.)

After the approximately 30-minute drawdown of demineralized water in the storage tanks, the low liquid level would automatically unseat a buoyant ball valve in a newly constructed standpipe (of good notch toughness properties) and divert the discharge of the low-lift river pumps from the standpipe overflow and direct it into the high-lift suction header to provide a continuing supply of emergency cooling water to the reactor.

Metal Examination Facility

A revision of the project proposal was approved by the AEC and a directive was received authorizing the total project ($465,000 plus $7900 Transferred Capital Property) with an interim authorization of $365,000 plus $7900 TCP. Construction status charts showing the revised Design and Construction schedules for the total project were also approved by the AEC. The contractor has begun installation of structural work for the facility.

Ammonium Hydroxide Facility

Design criteria have been established for a new ammonium hydroxide storage facility, and for the necessary modifications to existing chemical injection equipment for the safe handling of ammonium hydroxide for injection into the primary and graphite cooling systems for pH and oxygen control.

CONVERSION

Conversion to Phase II

The secondary steam system model for pressure and flow distribution analysis has been revised and updated to include recent information received from the architect engineer. Computer runs are being made to predict the expected pressure differences and flow distributions which will exist under the possible inlet and outlet steam flow conditions. A mathematical model of the modifications to the heat dissipation plant due to conversion is being developed. The analog computer simulation of the N-Reactor plant will be modified to include the effects of the alterations and additions resulting from conversion.

STATUS OF THE FISSION PRODUCT CONFINEMENT SYSTEM

Background

At equilibrium fuel exposures, the N-Reactor fission product inventory reaches about 10^{10} curies. To prevent or minimize release of these fission products to the plant environs was a key goal in the design and construction of the reactor plant. It is also an over-riding factor in determining limits to plant operation.

Conceptually, there are two ways by which major fission product release from the clad fuel could occur:

1. Occurrence of a power excursion in which heat generation rates exceed available cooling capacity.

2. Loss of cooling leading to fuel melting, even though the reactor were brought to a sub-critical state by the nuclear control and safety systems.

One of the prime goals of reactor design was to make accidents of the first class incredible.* Redundant nuclear safety systems were provided. The control rods and the safety poison ball system have each sufficient control strength and insertion speed to control any identified credible means of reactivity increase by driving the reactor sub-critical prior to fuel melting and maintaining the reactor sub-critical thereafter.

* In this discussion, the word "incredible" is taken to mean "of such a low probability of occurrence that providing additional protection is not justified."

Similarly, efforts were made to maximize adequacy and reliability of the coolant system. Though loss of coolant is an event of extremely low probability, it remains within the bounds of credibility. For example, the primary coolant line is fabricated to stringent quality standards, but the possibility of failure cannot be altogether discounted. Should this occur, a coincidental failure of the emergency coolant system, which does not depend on an intact primary loop, could lead to partial core meltdown.

For such an eventuality, a third nuclear safety system - the fission product "confinement" system - was provided to minimize the release of radioactive materials.

At the time of N-Reactor scoping and design, the conventional means for minimizing the release of radioactive materials was "containment," that is, the reactor and primary loop were located within a pressure vessel designed to withstand the pressures resulting from a broken primary coolant line and subsequent energy release from the reactor core. While such an approach was considered technically feasible for the N-Reactor, it had undesirable features. The reactor is physically large. The primary coolant system required for so large a reactor was neither compact nor small in volume. Hence:

1. The containment vessel would have had to be able to contain an enormous amount of energy.

2. The vessel itself would have had to be physically very large.

Fortunately, other avenues for protection were available in addition to the brute-force approach of containment. This resulted from the fact that power excursion, resulting in fuel melting, could be considered incredible; and, at the beginning of an accident involving core meltdown, levels of contamination in the primary coolant would be quite low. Excluding the short-lived, few second, half-life nuclides, the total contamination in the primary coolant would about to about 10 to 100 curies.

Therefore, it was perfectly feasible to vent steam directly from the confiner to atmosphere, thus avoiding the necessity for constraining a major part of the total energy release during the accident, and permitting the confiner to be designed to withstand relatively low internal pressures.

With this approach of fission product "confinement" instead of "containment", there are two important issues.

1. Steam venting capacity must be sufficient to prevent overpressurizing the confiner.

2. The direct venting of steam to atmosphere must be terminated prior to release of fission products into the atmosphere.

Implementation of Confinement Concept

While the concept is straightforward, its implementation can follow more than one course. Briefly, the system works as follows:

Should a primary loop line fail, steam would be released to the atmosphere through a number of six-foot diameter steam vents in the roofs of the reactor and 109N buildings. (There are 13 vents now and two more are being added to accommodate changes associated with Phase II operation.). The steam vents consist of a stand-pipe, a butterfly valve held open during reactor operation, a weather cap held in place by a shear pin, and a back-up closure. Should an accident occur involving release of steam and pressurization of the confiner, steam is automatically vented when the shear pin fails and releases the weather cap. The total venting area supplied would suffice to prevent overpressurizing of the confiner following complete failure of the largest line in the primary loop.

While this establishment of steam venting is the first critical requirement of the system and utilizes only passive equipment, the accident would be sensed by pressure switches located in confiner spaces. A small pressurization, two-inches water column, would signal the advent of a confiner accident. This signal would sound annunciators, close butterfly valves in ventilation ducts to prevent destruction of equipment and shut off potentially large leaks that would be undesirable in later phases of the transient, shut off ventilation fans to prevent their overheating, isolate the plant control center ventilation system minimizing the chance for contamination to enter this vital nerve center, actuate interlocks to prevent opening of doors into confiner spaces, and start timers that would provide information to operating crews and automatically trigger other system reactions. At a slightly higher pressure (eight inches water column) confinement fog spray systems would be actuated. The fog spray system serves first to rainout a fraction of the fission products which could be released to the confiner; and second as a heat sink for further energy releases following termination of steam venting. A higher pressure setpoint is used for the fog spray than for the other confiner actions because:

1. Fog spray initiation is not critical during the first one or two minutes of the transient. The other actions described above must be taken earlier.

2. The use of the higher setpoint reduces the chance for spurious fog spray actuation which could cause unnecessary damage to piping insulation and equipment.

Different alternatives were available for the handling of the accident following initial steam release. Some thought was given to establishing low-pressure "containment" during the development of the present design. This would involve closing the steam vents following near completion of the primary loop blowdown and leaving the confiner "bottled-up". This approach had some attractive features:

1. Maximum hold-up of fission products would result, reducing the total dose to the environs.

2. System control would be relatively simplified.

Balanced against these points were several disadvantages:

1. Following vent reclosure, all subsequent steam releases would have to be accommodated by the fog spray system, otherwise the confiner could be overpressurized by relatively small continuing releases of energy from the primary loop.

2. The concentration of fission products in confiner vapors would be high. This, together with above atmospheric pressures, would result in making confiner leaktightness a very critical issue; i.e., with slight pressurization of the confiner, a driving force would be available to expel vapors from any leaks. Since the vapors would be highly contaminated, the leaks would be quite serious. The confiner is large, is penetrated by many doors, pipes, ventilation ducts and other openings, so perfect leaktightness would be very difficult to obtain and maintain.

3. In a closed confiner filled with steam, continued fog spray operation would in time tend to cause a vacuum in the confiner. Again, the confiner could be damaged if vacuum relief devices failed to function.

Because of these drawbacks, the filtered release concept was selected. In this approach, ducts to ventilation exhaust filters and stack are opened just prior to the termination of atmospheric steam venting, and the roof vents are closed. This is accomplished in the following way:

1. As soon as steam release rates decay sufficiently, the butterfly valves in all but one vent close. A signal for this is decay of primary loop pressure to 275 psig (pressure is a measure of remaining energy in the loop and hence, the potential release rate), and a time delay of 200 seconds after the initial confiner pressurization to two-inches water column. The time delay signal prevents a sooner-than-necessary establishment of the filtered release.

2. With all vents but one closed, the resulting confiner pressure becomes a measure of steam venting rate. That is, the flow through an open vent as a function of driving pressure has been previously measured. With this calibration and a knowledge of the confiner pressure, one could determine confiner steam venting rates through one open vent. Thus, as soon as the steam venting rate decays to within the capability of exhaust filter and fog spray system (as signaled by a

confiner pressure of three-inches, water column, or less), and after a 260-second time delay after the initial pressurization, the valves in the ventilation ducts to the filters would open. The time delay is included to ensure that all steam vents but one would be closed and the confiner pressure would have reached a corresponding value before further actions are taken.

3. When any one of the three exhaust valves opens, the last steam vent butterfly valve would be signaled to close and the filtered release would be established.

This filtered release approach to confinement has the following advantages:

1. Filter flow capacity is added to fog spray condensation capacity to prevent buildup of pressure in the confiner.

2. Confiner pressures following steam release remain at or near atmospheric, and thus reduces the importance of leaktightness.

3. Fission products, other than noble gases, which are not rained out, would be immobilized on the exhaust filters rather than remaining for long periods of time in a state capable of release to the environs.

4. Since filters can "breath" in either direction, vacuum relief becomes of less importance.

The filtered release approach to confinement has been used in an exhaustive study of the events following an accident. Significant features are:

1. Particulate and halogen fission products are quite efficiently retained on filters and thus withheld from being released outside the confiner. This is desirable because release of these materials could cause contamination of food supplies and long-term ground contamination.

2. Noble gases are held up and diluted with confiner vapors, but are almost completely released to the environs. While this results in a "cloud" of high radioactive intensity, studies show that effects are tolerable. Ground deposition would not be expected and no permanent land contamination would result. Meteorological studies show that exclusion zone and low population boundaries (according to 10CFR100 criteria) would be 3 and 9 miles, respectively. Such boundary zones are quite acceptable at the relatively isolated Hanford site.

Maintenance of System

Maintaining the adequacy of the system requires a continuing effort. Proposed changes in the process must be carefully evaluated to ensure that potential accidents can be accommodated by the system and that any required modifications are provided. For example, extra venting capacity was required by the changed primary loop parameters resulting from conversion to Phase II operation.

Secondly, as should be the case with all "ultimate" safety systems, the system will probably not be "required" at any time during the life of the plant. But, it must be kept in a state of readiness so that it could respond to an accident whenever it might occur. Maintaining this state of readiness requires, as has been provided, a set of mandatory, documented, and monitored procedures covering testing, equipment, performance of preventive maintenance, and continual operator training.

Figure 2 shows estimated pressurization transients after a manifold failure in the pipe gallery.

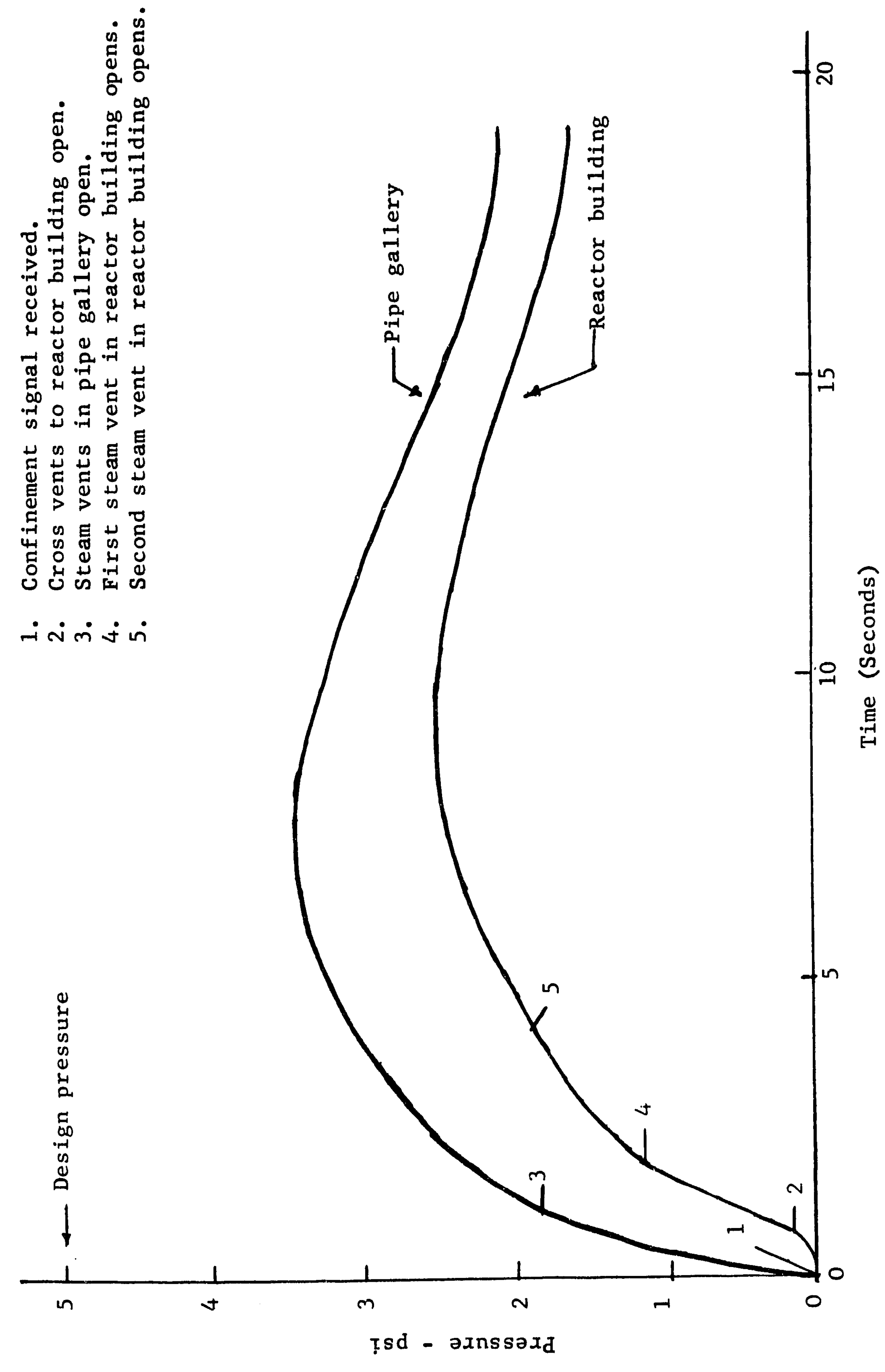


Fig. 2

PRESSURIZATION TRANSIENT FOLLOWING MANIFOLD FAILURE IN PIPE GALLERY

TECHNICAL ACTIVITIES

NUCLEAR SAFETY

Fuel Shipping Cask

The SCRUP cask was evaluated for possible shipment of low-exposure fuel to the NFS plant. The margins between allowable limits and calculated results were found to be generally smaller than those of the Stanray cask evaluation. In addition, there were questions about cask performance that will require testing to demonstrate acceptability. One such test would be to demonstrate the satisfactory convection coefficients that will develop with the closely packed array of fuel elements proposed for the SCRUP cask. The Stanray cask was given further evaluation for the effects of solar heat, and it was found that temperatures in excess of allowable limits would not be produced. The added heat load of uranium corrosion reactions was evaluated and found to be negligible.

Both casks were found suitable for shipment of fuels with low exposure, low enrichment and long cooling time.

Standards and PCS's

No new or revised standards were issued. A total of 31 Process Change Authorizations were issued, and forty-eight are now in force.

NEW EQUIPMENT

On-Reactor Process Tube Rolling Equipment

The special on-reactor tube rolling equipment has been successfully demonstrated on the mock-up assembly. This tube rolling unit contains the nozzle backing clamp, torque arm, expander with duo-thrust collar and rolling motor.

Process Tube Cutter

Irradiated process tubes being replaced must be cut into sections as they are pushed from the reactor. The special mechanism for doing this cutting has been completed and tested. It was determined that the cutter had to be changed from a roller to a machine cutter to increase tool life and reliability. Included in the modifications will be a method of controlling contaminated cuttings.

Graphite Mock-up

The graphite mock-up has been equipped with an inlet assembly duplicating that installed in reactor process channel 1648. This facility was used to successfully demonstrate the graphite core sampling equipment prior to on-reactor use.

COPRODUCT

Coproduct Block Test - 1.95% Enrichment

Experimental verification of coproduct production rates is desired as an integral part of the preparation for conversion to a coproduct fuel load. An in-reactor irradiation in N-Reactor is ideal for this purpose. From an economics point of view the nuber of fuel columns involved in the test should be minimized; while from a physics point of view the minimum size has been set at 21 columns of coproduct fuel, five for actual separation and analysis and 16 for neutron spectral adjustment.

Two simple geometric arrangements considered are a 5 x 5 array with the corners missing, and a 3 x 7 array. The square array is ideally suited for an axial reactor position, while the 3 x 7 is preferred for a peripheral position. In either array the five columns that are positioned one lattice cell in from the edge of the array are the test cells. The surrounding single ring of cells is a buffer zone required to establish a very early characteristic equilibrium neutron spectrum in the five test cells.

The power density within these cells is a maximum and a naked array would definitely produce an undesirable power peak for the axial loading, while the peripheral load would at worst produce only a percent or so power peak. Furthermore, the very flexible control rod system of the N-Reactor would allow control of the peripheral block by placing only tips in the reactor. The axially loaded block would require nearly complete insertion of control rods for control, and would poison a larger portion of the surrounding fuel columns.

Based on these arguments, it has been proposed that a 3 x 7 array of 1.95 percent coproduct fuel columns be loaded one lattice unit above or below the midplane of the core. This loading will displace 14 columns of spike fuel, and will nearly maintain the present power density distribution, while making only an mk or so reduction in reactivity.

Theoretical investigation of this problem and all other N-Reactor studies to date have been based upon diffusion models of the space-energy distribution of neutrons. The inadequacy of this approximation is emphasized in the black target of the coproduct fuel element. In an attempt to improve the theoretical model of N-Reactor, the transport IBM code THERMOS is being used to obtain average thermal absorption and fission cross sections for homogenized multizoned reactor calculations. To date a ten percent reduction has been found for the average thermal values for the coproduct cell. Values for the 0.95 and spike cells are currently being calculated.

1.95 Percent Enriched Uranium Driver Tubes

Seventeen of the 65 billets extruded for the fabrication of the 1.95 percent enriched coproduct demonstration load were made from British composition uranium (350 ppm Fe, 800 ppm Al). This was the first extensive production experience with coextrusion of these alloys at Hanford. No fabrication difficulties were noted except that about 10 percent higher upset and running pressures were needed at coextrusion because of the stiffer alloy.

Bond strength losses occurred in the extrusion of the standard alloy billets. These weak bonds were attributed to the lower reduction ratio used because of the thick wall of the coextruded driver element. Future coextrusion of coproduct driver tubes will utilize a larger extrusion liner (8.1 inch OD instead of 7.5 inch OD) with a higher reduction ratio. The weak bond strength problem should be eliminated.

Another problem encountered during fabrication of the thick wall driver fuel elements was achieving the desired conditions at end closure brazing. Braze metal flow and bonding between the ID face of the chevron cap and the uranium were not consistent. The problem was alleviated by increasing the heat input to the uranium, and holding the assembly at temperature for 100 seconds instead of 90 seconds. More development on optimum brazing conditions will be necessary before resuming production of this fuel design.

Coproduct Fuel Meltdown Studies

Additional tests have been run on heating of individual targets. They confirm the preliminary results reported previously, and give further detail on the mechanism involved. In the first seven tests, unirradiated alloy target material canned in aluminum and jacketed in Zircaloy was heated in a helium atmosphere to 1100 C in approximately five minutes, held at 1100 C for times up to one hour and either slowly or rapidly cooled back to room temperature. In the eighth and ninth tests, the atmosphere was steam rather than helium. A thermocouple malfunction caused the ninth test to be terminated after reacing 900 C.

Four of the nine targets survived with no damage other than a pronounced swelling at the base of the Zircaloy end caps adjacent to the internal aluminum end caps. This swelling may have been caused by hydrostatic pressure of the Li-Al target material as it melted and was superheated. The cladding that incurred preferential strain was that which had been annealed during the welding of the end caps.

The five targets that failed did so either immediately upon reaching 1100 C or during cooling when the temperature reached about 850 C and an exothermic reaction was noted. However, the targets that failed did not release fluid Li-Al alloy. Rather, the targets seemed to be consumed by a reaction of molten aluminum and solid zirconium to form Al_3Zr. Cross sections of the failed targets showed that the Al_3Zr product formed wherever molten aluminum contacted zirconium.

It is tentatively concluded that the autoclaved oxide layer on the interior of the Zircaloy jacket prevents direct contact of the molten aluminum with the zirconium. If so, then the formation of Al_3Zr would be limited by the rate of diffusion of aluminum through ZrO_2. At the beta to alpha transformation of the Zircaloy (at 850 C) during cooling, ZrO_2 and Al_3Zr may develop gross defects from differential expansion relative to Zircaloy-2. The exothermic reaction may be simply the formation of Al_3Zr from the elements.

Spectrographic analysis of a target that failed showed that lithium was not "lost" from the system. The material that appears to be unreacted Li-Al alloy showed a slight (though not significant) loss of lithium and the Al_3Zr showed no loss of lithium. However, a thin metallic-appearing film on the Vycor tube (that contained the target during heating) was rich in lithium and tends to indicate that the LiAl matrix suffered some lithium depletion. A complete material balance cannot be made until chemical analyses are completed.

The target element that was heated to 900 C in the ninth test was reheated several times in the course of investigating the cause of the thermocouple failure. When this repeatedly reheated target reached 1100 C, cooling water was intentionally introduced.

This last test seems to indicate that the heat of formation of the oxides of the metals tends to prolong a thermal excursion if any failed targets come in contact with water.

An irradiated target element was heated from 220 C to 935 C in 330 seconds to simulate the thermal cycle that would result if that target had been positioned in a driver tube operating at 210 KW/ft and primary loop coolant was lost. A flow detector device indicated that the Zircaloy cladding of the target failed at a temperature below 835 C and molten aluminum plus lithium was released. Heating was continued to 935 C after which cooling was effected by turning off the induction power source and maintaining a low flow of steam that simulated the residual atmosphere expected in a process tube if the coolant were flashed to steam.

The failed target appeared to have released most of the core material (Al + Li) by gravity flow but it is outstanding to note there was no lateral flow of the molten material past the Zircaloy end caps of the target. As was seen in the tests on unirradiated target elements, this irradiated target again shows evidence that any molten core alloy that is released will react with Zircaloy metal to form the intermetallic compound $ZrAl_3$; this reaction apparently occurs as peritectic reaction, viz:

$$3\ Al\cdot Li_{(l)} + Zr_{(s)} \rightleftharpoons ZrAl_{3(s)}$$

Coproduct Conversion Study

Two methods to convert from a plutonium-only operation to a coproduct operation have been studied: (1) incremental replacement of the present production

production fuel with coproducer fuel over about a six month period; and (2) a block discharge of present production fuel with a block charge of coproducer fuel. The second method allows for some physics tests that are desirable but not mandatory.

The incremental transition appears to be the most desirable from the operating and cost viewpoint because it would not require early increases in plant capacity and billet orders.

NEW TECHNOLOGIC DEVELOPMENTS

Oxygen Control in the Graphite Cooling System

Radiolytic decomposition of water in the graphite cooling system causes a great increase in the oxygen content of the water. If ammonia instead of LiOH is used to control the pH of the system, it also controls the oxygen content. A seven-day test, with five days at full reactor power, demonstrated clearly that oxygen could be maintained at the low desired level of 7 ppb when ammonia is used for control of pH. Typical coolant conditions before, during, and after the change to ammonia are given below.

	Before Change to NH_4OH	During NH_4OH Test	24 Hours After Termination of NH_4OH Test
pH Control Material	LiOH	NH_4OH	LiOH
pH	9.8	9.8	9.8
NH_3 conc. ppm	.02 - .05	8 - 15	0.1
N_2H_4 conc. (ppb)	<3	110	<3
O_2 conc. (ppb)	30,000-40,000*	<3	>26,000
H_2O_2 conc. (ppb)	100,000-150,000	<25	150,000
H_2 conc. (cc/liter)	>50*	>50*	>50*

* Estimated. Concentration greater than saturation in sample taken for analysis.

Because permanent use of ammonium hydroxide is being considered, a test program is being prepared to determine the effects of continuous operation on corrosion of copper, especially on the more than one thousand copper sensing lines.

Primary System Pressure Limit

Original design calculations estimated the amount of lift in the graphite stack from operation at 1700 lb. pressure. From the estimated resulting deflection of the process tubes, an allowable strain rate for the process tubes was calculated. All these calculations have been reviewed in an attempt to determine to what extent the primary system pressure can be raised. On the basis of data from conventional tensile samples irradiated in the G-7 loop of the Engineering Test Reactor, it has been concluded that a strain limit of 0.01 rather than the current 0.0055 can be used with negligible decrease in conservatism. Additional fracture studies of crack propagation, bending, and variable loading rates indicate that even the limit of 0.01 is very conservative.

Thermal Hydraulic Experiments

A study is being made of the thermal behavior of simulated fuel elements under boiling conditions. The modified cosine heated test section, representing the tube-in-tube loading, has been used to obtain

1) steady-state two-phase pressure drop data at rear riser pressures of 1200 and 1300 psi

2) equivalent tube powers ranging from average to hot channel conditions

3) exit steam qualities from zero to 30 percent by weight.

Observations made during the 1200 and 1300 psi runs indicate that inter-channel mixing may have a significantly beneficial effect upon flow reductions required to initiate film boiling in the fuel element. Thus, it seems certain that the data obtained with the previous model are conservative with respect to this factor.

Thermocouple Train in Process Tube

The thermocouple train program is designed to measure the temperatures of individual fuel element cooling channels at the outlets of the fuel columns, particularly to determine the extent of imbalance in the exit water temperatures. The train consists of a number of fuel element spacers end to end, with thermocouples suitably mounted on one of them, and with the lead wires extending down the row and out the nozzle cap.

The experiment has been delayed by a leaking brazed joint at one of the metal-sheathed thermocouples. High temperature brazing flux apparently plugged the hole temporarily and kept it from being detected during inspection.

Interpretation of N-1 Startup Spectral Indices Experiments

Nuclear reaction rates of Cu, Au, Eu, Lu, Pu, U-238 and U-235 were determined in N-1 startup tests. These rates have been corrected for radioactive decay and for other minor experimental variations. Interpretation of the corrected results has begun. Unavoidable and unmeasured control rod perturbation

[redacted]

of the neutron spectrum along the test fuel pieces has been calculated to be only a few tenths of a percent and can be neglected in the cold experiments.

The application of the concept of only two spectral indices -- neutron temperature and Westcott's r (fast to slow flux ratio) has been shown in transport calculations to be inadequate to describe the detailed neutron spectral shifts in the fuel annuli. Attempts are now being made to correlate the observed thermal reaction rate distributions with transport calculations. The relative magnitudes of the Cu-64 activity data fit quite well for both annuli. The precise distribution in each annulus is not well predicted because the cell model used in the THERMOS code is not adequate. Work to improve the cell model is underway.

Poison Materials in Control Rods

An investigation of the comparative behaviors of various poison elements under long-term irradiations is being made. Data for lithium, dysprosium and europium have been compared with the control behavior of boron. Dysprosium and europium are of considerable interest for long-term behavior since they do not show the marked decrease of control strength that boron shows in the period of 14 to 16 years. This rapid falloff of strength in boron rods requires a limit of 90% of initial control strength as one of the criteria for removing a boron rod. The characteristics of dysprosium and europium are such that control strengths of 70 to 80% of initial strength would probably be reasonable criteria in determining the acceptability of the rod, thus increasing the service life of the rod by a number of years.

[redacted]

ADMINISTRATIVE SUMMARY

DECEMBER 1964

EMPLOYMENT

Section	Permanent		Temporary		Total		Change	
	E	NE	E	NE	E	NE	E	NE
General	1	1	-0-	-0-	1	1	--	--
Finance	15	12	-0-	-0-	15	12	+1	-1
Fuels	45	96	4*	-0-	49	96	+1	-2
Plant	62	208	2*	6 T	64	214	-3	--
Project	70	21	-0-	-0-	70	21	+1	--
Research & Engineering	49	8	-0-	-0-	49	8	+2	--
Employee Relations	5	2	8xx	-0-	13	2	-1	--
	247	348	14	6	261	354	+1	-3
Total Employees						615		
Net Change								-2

* - MTP
xx - Tech.Grads.
T - Training (nonexempt)

SAFETY

Days without a disabling injury	762
Hours worked without a disabling injury	2,458,455

VISITORS

Name	Component & Location	Contact	Date	Purpose
Harold Steinberg	Theta Instrument Corp. Los Angeles, Calif.	--	12/3/64	To demonstrate a rod position transducer
Doug Jessup	Scientific Data Systems Bellevue, Washington	C.F. Poor	12/7/64	Discuss capabilities of their small computers
Dr.Guiseppii Volta	EURATOM	JH Fastabend	12/7/64	300 Area Meeting to present information on the N-Reactor rolled joint and the developed co-extruded carbon steel-Zircaloy-2 connection
R. Graham	Wah Chang Albany, Oregon	LH Steves HP Kraemer	12/7/64	Deliver quotation to buyer
Wm. Hubbert	Minneapolis Honeywell	--	12/9/64	Discuss recorder problems
J.L. Klein	Nuclear Metals Boston, Mass.	LM Loeb TD Naylor	12/14 - 12/18/64	Discuss final conclusions on N-fuel failure
G.J. Haysom	Computer Control Co. Kent, Wash.	CF Poor	12/17/64	Discuss capabilities of their small computers
J.A. Fellows	Mallinckrodt Chem. St. Charles, Mo.	JW Nickolaus TD Naylor	12/21 - 12/22/64	Discuss N-fuel element fabrication
Ralph Blair Jeffrey Gorman Geo. Wyss	Bettis Atomic Power Lab.,Naval Reactors Div. AEC	RH Shoemaker	12/15/64	Discuss primary pump seal experience

VISITS

Name	Component & Location	Contact(s)	Date	Purpose
H.S. Davis	Stanray Corporation Hammond, Indiana		11/17/64	Consult on Stanray S1 Fuel Cask
D.L. Condotta	ANS-AIF Mtg. & Seminar San Francisco, Calif. Bechtel Engrg. Co. San Francisco, Calif.		11/30/64 thru 12/2/64	Attend ANS Meeting and Desalination Seminars, AIF, and to discuss desalination and methods of connecting to nuclear reactors with Bechtel Engrg. Co.
H.R. Kosmata	Bechtel Engineering Co. San Francisco, Calif.		12/1/64	To discuss desalination
	USAEC, San Francisco		12/1/64	To discuss desalination
	GE-APED, San Jose, Calif.		12/2/64	To discuss pipe rupture program
D.L. Condotta	GE-APED, San Jose, Calif.		12/4/64	Consult on reactor design problems
A.E. Guay J.E. Hammond	EI duPont Co., SRL Aiken, S. Carolina	R.T. Huntoon	12/7/64 thru 12/9/64	Discuss irradiation performance of alloyed fuel elements
R.E. Hall	Burns & Roe, Inc. Hempstead, L.I., N.Y.	I. Gabel	12/7/64 thru 12/9/64	Discuss conversion
R.C. Hoffman	Burns & Roe, Inc. Hempstead, L.I., N.Y.		12/8/64 12/9/64	Attend Conversion Project electrical conferences
R.C. Hoffman	Ther Elec. & Machine Works, Chicago, Ill.		12/10/64	Review 125-volt battery charger use and application at 100-N Area

VISITS

Name	Component & Location	Contact(s)	Date	Purpose
L.G. Henke F.J. Mollerus, Jr.	Weyerhaeuser Company Steam Plant Longview, Washington		12/10/64	Observe installation and operation of new steam pressure reducing valve (angle valve)
J.W. Nickolaus R.H. Scanlon	National Lead Co. Cincinnati, Ohio	P.N. McCreery J.F. Schiltz	12/14/64 thru 12/18/64	Discuss N-uranium fabrication
	Reactive Metals Ashtabula, Ohio	R.L. Bean J.F. Puterbaugh		" " " "
J.P. Sluka R.C. Walker	Anchor Equip. Co. San Francisco, Calif.		12/29/64 12/30/64	Review and discuss technical specifications for replacement primary coolant valve stems

INVENTIONS AND DISCOVERIES

All personnel 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 December 1964. Such persons further advise that for the period herein covered by this report, notebook records, if any, kept in the course of their work have been examined for possible inventions or discoveries.

SIGNIFICANT REPORTS

Report Number		Title	Author	Date
RL-NRD-105	Uncl	Control Aspects of a Large, Dual-Purpose Nuclear Reactor Plant	D.W. Leiby	December 10, 1964
HW-83892	Uncl	Copper-Base Alloys in the N-Reactor Graphite Cooling System	W.E. Gurwell B.S. Kosut	November 30, 1964
RL-NRD-87RD	Uncl	Design Criteria - A Second Light-Off System for the 184N Boiler	F.J. Mollerus,Jr.	December 1, 1964
RL-NRD-122	Uncl	Engineering Study - Separation of Stored Condensate from Air in AHR Tank	W.D. Bainard	December 14, 1964
RL-NRD-135	Uncl	Evaluation and Remedial Considerations of N-Reactor Rupture Monitor System False Indications	D.O. Allred	December 22, 1964
RL-NRD-112	Uncl	Export Power - Reactor Power Relationships	H.R. Kosmata	December 9, 1964
RL-NRD-103	Conf	Half-Rods for N-Reactor	W.S. Nechodom	December 2, 1964
RL-NRD-104	Uncl	Interim Test Report No. 2 - Plant Service Boiler Base Load Reduction Program (PT-NR-24)	D.L. Renberger	November 24, 1964
RL-NRD-107	Uncl	Interim Test Report No. 3 - Plant Service Boiler Base Load Reduction Program (PT-NR-24)	L.D. Smith	December 16, 1964
RL-NRD-147	Uncl	Nondestructive Analysis of Pu-240 Content of Post-Irradiated N-Reactor Fuel Elements	D.O. Allred	December 31, 1964

SIGNIFICANT REPORTS (Continued)

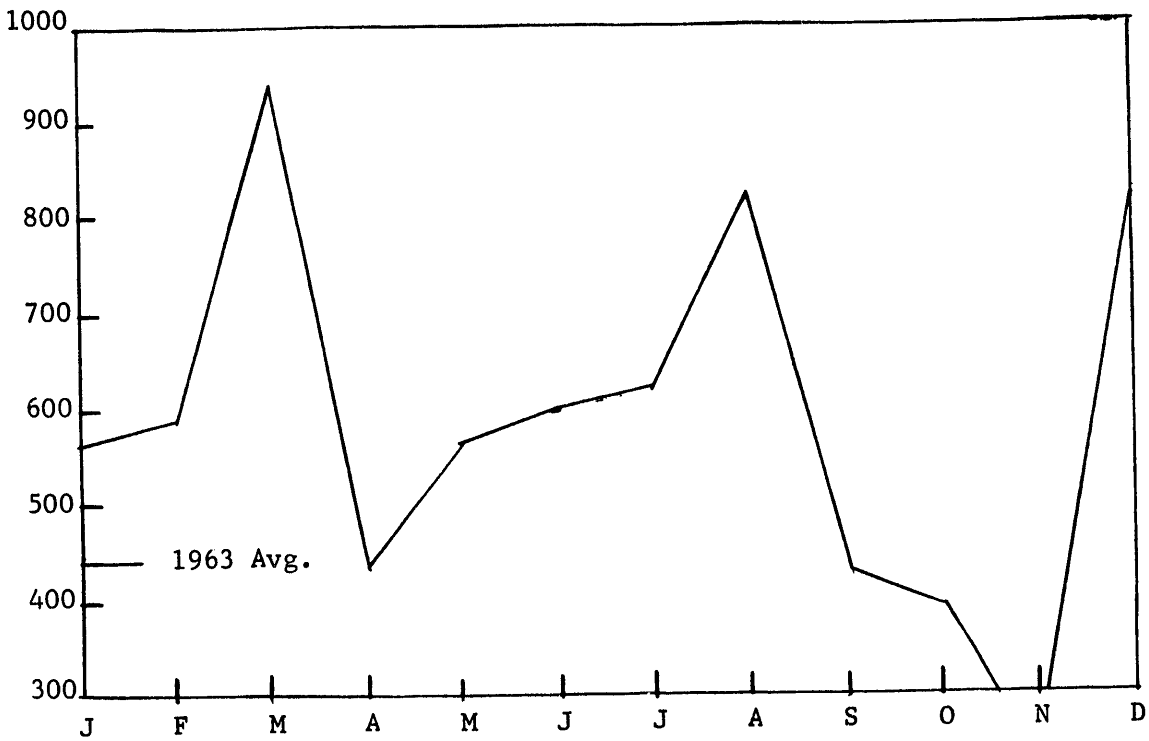
Report Number		Title	Author	Date
RL-NRD-79	Conf	NPR Fuel Melting Times Following Loss of Coolant	J Muraoka	December 7, 1964
HW-83828 SUP1	Conf	N-Reactor Summary Scram Reports	E.E. Leitz	September 30, 1964
RL-NRD-88	Conf	NR Circulating River Water System Flow Distribution	C.L. Goss F.J. Mollerus, Jr.	December 16, 1964
RL-NRD-120	Uncl	N-Reactor Design Criteria - Ammonium Hydroxide Facilities	W.D. Bainard	December 11, 1964
RL-NRD-106 RD	Conf	N-Reactor Phase II Operation Hazards Summary Report (Comment issue)	R.E. Trumble N.R. Miller J.R. Bolliger	December 18, 1964
RL-NRD-119	Secret	N-Reactor Department Semiannual Report of Research & Development	Research and Engineering Staff	December 15, 1964
RL-NRD-86	Uncl	100-N Conversion, Construction - Plant Forces Work Definition - Electrical and Instrumentation Systems	J.D. Fogelquist	December 1, 1964
HW-82794 Part 3	Conf	100-N Equipment Experience Report - 50-80 Percent Design Power Level	J.W. Helton	November 11, 1964
RL-NRD-59	Conf	Predicted N-Reactor Plant Heat Balances	V.J. DeJong A.J. Ebens L.D. Smith	November 6, 1964
RL-NRD-94	Conf-Und	Predicted Primary Loop Flow Rates	H.G. Johnson	November 30, 1964

SIGNIFICANT REPORTS (CONTINUED)

Report Number		Title	Author	Date
RL-NRD-127	Uncl	Preliminary Specifications for a N-Reactor Fuel Element Pre-Irradiation Enrichment Analyzer	D.O. Allred	December 16, 1964
RL-NRD-130	Uncl	Preparation, Process Tube Replacement	J.H. Fastabend	December 18, 1964
RL-NRD-65	Secret	Projected Costs for Fuel Ruptures	D.H. Curtiss	December 28, 1964
RL-NRD-124	Uncl	Some Possible Small Digital Computers for Application to N-Reactor Process Tube Power Calculations	O.F. Poor	December 14, 1964
HWS-6944	Uncl	Specification for Corrosion Resistant Replacement Valve Stems for Use in N-Reactor Primary Coolant System	J.P. Sluka	December 18, 1964

SUGGESTION PLAN PARTICIPATION

	DECEMBER	CY 1964 TO DATE
Number of Eligible Employees	354	362
Number of Suggestions Received	36	593
Number of Suggestions Acted Upon	66	567
Number of Suggestions Adopted	24	212
Net Annual Savings	$ 955	$135,501
Amount of Awards	$ 390	$ 8,150
Percent of Awards to Savings	40.8	6.0
Average Amount of Awards	$16.25	$38.63



SUGGESTIONS ADOPTED/1000 EMPLOYEES/YEAR

1964

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DATE
FILMED
01/22/93