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N-REACTOR DEPARTMENT
MONTHLY REPORT - OCTOBER, 1964

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General Manager

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OPERATING SUMMARY

Input production Pu (KMWD)	15.64
Input production T (KMWD)	0.
Maximum power level attained (MW)	2,800
Average power level while operating (MW)	2,456
Time operated efficiency (%)	23.3
Plant utilization (%)	23.3
Number of Shutdowns	1 *
Scheduled	0
Unscheduled	1 *
Reactor Process Data	
Power at which computed (MW)	2,800
Primary coolant inlet temperature (°F)	345
Primary coolant outlet temperature (°F)	452
Maximum graphite temperature (°F)	990
Heat Loss	
Graphite cooling (MW)	40
Shield cooling (MW)	1.8
Rod cooling (MW)	0.9
Steam temperature (°F)	320
Steam pressure (PSIG)	75
Dump condenser flow (gpm)	300,000
Inlet temperature (°F)	60
Outlet temperature (°F)	127

*(Continuation of September outage)

Nuclear Steam Generating Rate

Maximum (1000's lb/hr)	8,670
Average while operating (1000's lb/hr)	8,170
Water Collected from Gas Atmosphere (gal/day)	Negligible
Demineralized Water Produced (gal/hr)	59,280
Helium Losses (1000's cu. ft.)	260
Fuel Oil Usage (bbls.)	28,930
Flattening Efficiency (%)	78

Fuel Conditions

Heat flux	
Maximum (BTU/hr/ft. ²)	410,000
Average (BTU/hr/ft. ²)	330,000

Clad temperature .

Maximum (°F)	515
Average (°F)	480

Uranium temperature

Maximum (°F)	725
Average (°F)	660

Fuel Failures 0

Fuel Balance	<u>0.95</u>	<u>1.25</u>	<u>Co-Prod.</u>	<u>Other</u>	<u>Total</u>
Fuel charged (Tons)	2.23	3.31	0	0	5.54
Fuel discharged (Tons)	5.87	0	0	0	5.87
Net change (Tons)	-3.64	+3.31	0	0	-.33
Total in reactor at month end (Tons)	327.75	39.74	0	0	367.49

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Fuel Element Output

0.95%	1,584
1.25% (driver)	0
1.25% + 0.95% (spike)	360
Total assemblies	1,944
Total tons	39.4
% of forecast	112
Uranium Utilization (%)	72

SUMMARY

The reactor resumed operation on October 24, after an outage of a month for repair of the leaking dump condensers and for other scheduled maintenance and modification of equipment. The dump condensers were successfully repaired by the addition of steel fingers, girth bands and springs. A resulting decrease in vibration in the condensate steam lines has been observed.

The tie-in of steam generator No. 4 was completed and the reactor is now operating with all five steam generators in full service. Operation is stable, and steam distribution in the header is better balanced than it was with only four steam generators. Reactor power was raised to the previous maximum level of 70 percent.

The cold, xenon-free, excess reactivity is 38.1 mk, within 0.5 mk of the expected value. There are now 144 spike columns charged.

Studies have been completed on criticality of spike and co-product fuel elements under conditions expected in the fuel discharge storage basin. Storage will be more efficient in small shielded canisters, although materials handling is more efficient with the present large canisters.

Cracks identical in appearance to stress corrosion cracks have been confirmed as the cause of the previously reported failure of a 17-4 PH stem in a large primary system valve. Stems of Inconel X-750 have replaced this stem and another one removed for examination.

At month's end the reactor was prepared to go to higher power level pending appropriate authorization.

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REACTOR PLANT OPERATION

OPERATING PATTERN

N-Reactor resumed operation on October 24, after a month-long outage for repair of leaking dump-condenser tubes and for miscellaneous maintenance. On October 27, power was increased to the previous maximum level of 70 percent of design power level, and continued at this level the remainder of the month. Twelve columns of "spike" enrichment were charged to improve reactor flattening. Four crud monitor fuel columns were also charged during the shutdown.

Five-Cell Operation

Repairs on steam generator 4A and the construction work on Cell 4 equipment were finished and the cell components were placed into service and tested. Testing consisted of calibration of instruments, functional checks of valves, and performance of the Cell 4 Recirculating Test in which the primary system was filled and recirculated, utilizing the turbine, pump, and associated controls and instrumentation. Following satisfactory performance of the operational test, the cell was tied in with the primary system. The startup on October 24 marked the first plant operation with all five cells in service. During startup, data were collected at various power levels to evaluate operation with five pumps in service.

Operation of the plant with five cells in service is as stable as had been predicted from previous operating experience. Total primary system flow is slightly greater than design values. Operation with five pumps naturally creates an unbalanced flow, and there is observed about 10 percent more flow through the process tubes on the left side (the three-pump side) of the reactor than on the right side. Steam distribution in the 46" header is more equitably balanced than during four-cell operation. This balance results in a more favorable load distribution to the dump condensers.

There appears to be less vibration in the steam-condensate piping in the turbine bay since the modifications to the condensers were made.

Broken Valve Stem

During completion of startup checks, hydraulic vibrations were encountered in Cell 3 primary piping at pump speeds above 2300 RPM. This vibration was traced to a broken stem on the steam generator bypass valve in Cell 3. The valve stem was removed, the valve plug was blocked closed, and the valve reassembled. The failure was a clean fracture across the stem at a necked-down segment of the 304 stainless steel shaft and appeared to be similar to that experienced on a similar valve in July. The failed shaft is undergoing metallographic examination.

PROCESS PHYSICS

Reactivity Measurements

A measurement of the cold, xenon-free, excess reactivity was performed on October 24. The excess is 38.1 mk, which is within 0.5 mk of the expected value. The total reactivity contribution of the 144 spike columns is now estimated to be 6.3 mk in the green condition. The excess reactivity without the spike enrichment would be 31.8 mk and the predicted (FLEX-3) value normalized to the startup test results is 30 mk. The reactivity behavior of the reactor is therefore close to predicted values.

Equilibrium Status

Owing to the shutdown status of the reactor for most of October, no equilibrium calculations were made.

Natural Uranium Requirements

Requirements of spike enrichment columns and natural uranium columns were outlined for the remainder of the power ascension program and for subsequent operation at full power level.

An alternative to charging natural uranium to compensate for reactivity gains is to allow the reactivity absorbed in rods at equilibrium to increase. This increase would also be reflected in an increase in the reactivity at turnaround. The consequences of this alternative would be:

- 1) a reduction of roughly one percent in the total conversion ratio of the reactor,

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- 2) the possibility that lower power levels will be necessary during startup to maintain a positive shutdown margin, and
- 3) an increase of approximately 10 to 13 spike enrichment columns to compensate for the extra rods penetrating the spike enrichment ring.

These consequences may be preferable to discharging 50 columns of 0.95 percent fuel at low exposure in order to charge natural uranium. Thus, the natural uranium charging would be deferred until the 0.95 percent fuel is first scheduled for discharge near the end of December. This action would have the effect of delaying the charging of the first 20 columns of natural uranium for about 1.5 months. Total inventory requirements for natural uranium would not be altered.

Ball Safety System

The ball safety system appears to have a surplus of control. Consideration is being given to the possibility that some of these channels might be utilized for supplementary nuclear control.

MECHANICAL AND THERMAL EQUIPMENT PERFORMANCE

Dump Condenser Repair

All sixteen dump condensers were repaired and returned to service. Installation of radial steel fingers, steel girth bands, and springs is expected to eliminate leakage from damage caused by vibration of the tubes.

In order to evaluate success of the repair, the No. 1 condenser has been fitted with instrumentation to measure selected temperature, pressure, vibration and displacement of the tube bundle support. Also to permit inspection of the tube bundle and adjustments to instrumentation during plant operation, a shut-off valve was installed in the 30-inch steam admission line to dump condenser No. 1 so that this condenser may be isolated during plant operation.

Horizontal Rod Seals

The modified seal for the horizontal control rod cylinders, which controlled leakage during its initial tests, was tested again after several weeks of nonoperation. This time the seal leaked a considerable

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amount of oil at each stroke. When removed from the cylinder for inspection, the seal rings were found to have approximately 0.010" diametral clearance. A new set of seal rings fitting the rod without clearance will be assembled, installed and tested.

Rupture Monitor System

Checks have been made on the circuitry and trip ability of the differential alarm modules in the rupture monitor system. False indications of failure of the GM tubes had been caused by low voltage to the tubes, and this in turn had been the result of error in the high voltage meters. Spurious trips on the differential alarm modules were caused by bad silicon controlled rectifiers, shorted GM tubes, and low sample flows in the right side turret room. Increasing the differential pressure of the system from 164 psig to 195 psig, and readjusting sample flow transport time corrected the last item.

Control valves for the rupture monitor system have been subject to erosion from high velocity flow. Four manually-operated isolating valves, after having stellited needle point trim installed in them, can now be throttled so as to greatly reduce the pressure drop across the control valves. This allows the control valves to be operated in a more open position with lower velocity flow through them, thus removing the cause of the observed erosion.

Movement of Fuel Element Spacers

The fluoroscopy method of detecting movement of upstream spacers has proven unworkable and another method must be found. A concept of analyzing by sound has been proposed and will be investigated. To obtain some data on spacer movement with the upstream spacer removed, two process channels now have upstream spacer trains made up of two short spacers, plus one 23-1/2 inch and two 27-1/2 inch spacers. This arrangement will position all feet of long spacers on surfaces not touched previously. Follow-up inspection by borescope will reveal either some degree of movement or the absence of it.

Reactor Gas Leak

The faulty welds in the vicinity of the expansion joint gas seal at the right outlet corner of the reactor have been repaired and strain gauges have been installed around this weld to monitor strains during the coming operational period.

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Baffle plates were installed on the top inlet and outlet corners of the thermal barrier to divert hot air away from the corner and avoid another hardening of the hypalon seal because of excessive temperature.

A polyvinyl dispersion and a silicone rubber have been tested as sealants for reactor gas leaks. After nine days exposure at 450 F the silicone retained excellent strength and flexibility, but the polyvinyl dispersion became brittle and tended to flake off.

The silicone rubber was subsequently used to repair cracks in the omega seal, and has contributed significantly to the overall reduction in loss of reactor gas.

Primary Loop Operating Pressures and Flow Rates

A study to determine the operating pressures required in the primary loop for Phase I and Phase II 5-loop operation shows that with the relief setting at 1650 psig, and certain other conditions being met, it will be possible to meet the predicted steam pressures at full design power. In addition, flow rates have been predicted for all three phases of N-Reactor operation and for various loop combinations which take into account the apparent reduced hydraulic resistance of the primary loop. These predictions will facilitate correct resetting of flow instrumentation to cover a broader range of flow rates.

Pressurized Air Tubing

Metallurgical examination has disclosed evidence of stress corrosion cracking of the 304 stainless steel tubing providing air to two of the diversion valve actuators. The corrosive agent that contributed to the cracking has not been determined, but pieces of the failed tubing appear to have had some unknown substance smeared or dropped on them at the points where cracking occurred.

Dye penetrant checking of approximately 30 percent of the remainder of this tubing within the thermal barrier enclosure failed to reveal any further evidence of cracking. It is concluded that the two tubes that failed are isolated cases.

Afterheat Removal Tank Membrane

It has been decided not to repair and replace the existing afterheat removal tank membrane because the potential for another failure appears to be high. Increasing the tank temperature, and direct addition of hydrazine are being evaluated as means for providing interim control of dissolved oxygen.

Steam Generators

During a maintenance inspection of the secondary side of steam generator 5A, wet spots were observed on the tube sheet. Air was blown through the secondary side of units 1A, 3B and 5A to dry them prior to conducting a more thorough inspection. Pressurization of the primary side to 1000 psig resulted in the development of more wet spots on the tube sheet of unit 5A. The leakage was estimated at one pint per hour. Several wet spots also developed on the tube sheets of units 1A and 3B, but the leak rates were insufficient during three hours to cause dripping of water from the tube sheet to the shell. A multichannel gamma analyzer has been set up to measure radioactivity in the secondary coolant and assist in evaluating the magnitude of primary water leakage into the secondary side of the steam generators. On the basis of present observation, the leakage is considered to be very minor and does not imperil plant operation.

Because of continuing problems with steam generator level measurement, a vendor representative was called in to assist with the calibration problems. It was found necessary to install a fixed resistor in the middle correction factor computer of generator 1A and later in other generators to allow the circuit to be calibrated properly. Heretofore, an error has resulted from this lack of compensating resistance.

The calibration program was expanded to include all steam generator level instruments, including pressure, level and steam flow transmitters, computing circuits and indicating gauges. The level indication systems for steam generators 1A, 3A, 4A and 5A were completed and the systems for steam generators 1B, 2A, 2B, 3B, 4B, and 5B are 67 percent complete. The balance of this calibration is scheduled for the next outage.

Steam Valve Control Malfunctions

Following shutdown for the extended outage, the high pressure valves that admit steam from the header to the dump condensers were all closed, the block valves were closed, the hydraulic power packs were de-energized, the hydraulic pressure was bled to zero pressure, all heaters were turned off, and the Servo electronic control was turned off. During first attempts to activate the valves, only four of the sixteen operated. Control system malfunctions included two faulty Servo valves, one plugged solenoid control valve, one stuck check valve, and several relief valves releasing pressure below the operating setpoint.

In the steam valve controls, the hydraulic fluid recommended by the manufacturer has been replaced with Mil-H-5606 A oil, a lower viscosity fluid. Use of the new fluid has resulted in a significant improvement in reliability of the controls.

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Valve Stem Failures

Last July a 17-4 PH valve stem in a 24" primary system valve broke. It has now been replaced with a new stem manufactured from Inconel X-750. A report giving results of tests conducted on an eight-inch length of the broken stem has been issued. It states that the material is not in the specified 1025 F heat treated condition. Findings show a heat treating temperature below 975 F and probably as low as 925 F. This heat treatment is known to leave the metal susceptible to stress corrosion cracking. The report confirms that the cracks that caused failure were identical in appearance to stress corrosion cracks. However, no corrosive agent has been identified.

The stem from the corresponding valve in Cell 5 has been removed for inspection and has also been replaced with an Inconel X-750 stem. The replaced stem is also 17-4 PH stainless steel, apparently heat-treated the same as the broken stem. This particular stem was selected from among other candidates for inspection on the basis that its hardness of Rockwell 40-C was as high as that of the broken stem. It was inspected by immersion ultrasonic techniques and fluorescent dye penetrant. The fluorescent penetrant test revealed a very tight crack over an arc of approximately 160° in the same location where the other valve stem had failed. Also, one small area four to six inches above the crack contains surface pits which appear to be caused by corrosion.

Failure of a 304 stainless steel stem in a different type of valve was discovered on October 23. Severe water hammer was experienced in Cell No. 3 when the valve plug broke free from the stem in a steam generator bypass valve. This hammering was of such magnitude that it caused failure of the hydraulic snubbers on the steam generator 18" bypass line and of one snubber on the primary coolant return line above the primary pump.

The upper part of the broken stem was removed and a stainless steel dummy inserted to seal the line. The valve plug was blocked in the closed position in order to restore Cell No. 3 to service. The upper part of the stem has been taken to the laboratory for investigation of the cause of failure. The stem is 304 stainless steel, with a stellite overlay immediately above the break.

Preliminary examination indicates that the cause of failure was definitely not stress corrosion but some purely mechanical cause.

Supports for Back Pressure Control Valves

Modifications have been made to the back pressure control valves to improve reliability. Additional supports have been installed on the operators of the valves and a sway brace was added to the piping upstream of the valves. These additional supports are intended to reduce the vibration that has caused the operators to be frequently inoperable.

Steam Condensate Traps

An engineering review of steam lines, to locate the cause of a carry-over of condensate into the turbine-generator, brought to light a deficiency in the discharge capacity of the condensate traps. It was determined that this was because the six traps involved were sized for a maximum pressure differential of 500-600 psig. With the present low pressure differential of only 50 psig, the size of the existing valve seat orifice severely limits condensate discharge capacity. Larger valve seat orifices will provide for a 5-fold increase in the condensate discharge capacity of these traps under present operating conditions.

Circulating Raw Water System

Increased protection is needed against short-circuiting pump backflows that could unnecessarily actuate once-through emergency cooling upon the loss of one of the power sources and failure of a pump discharge valve to close properly. A proposed design modification calls for a spring-loaded, fast-acting, nonslam type check valve to be installed downstream of the existing butterfly valve in the discharge line of each of the four circulating raw water system pumps.

Funds in the amount of \$105,000 have been made available by the AEC to cover the cost of installing check valves in the raw water system. The design for this installation is in progress. In the meantime, hydraulic valve timers have been installed in the circulating raw water system and tested. Valve closing times were about 5 seconds, as required. The timers are to be used for monitoring reliability of the river pump discharge valves.

Confinement System

A number of modifications were made to improve reliability of confinement circuits. Additional diodes were installed and all diodes were checked. The counting circuit resistors which actuate the system were replaced with resistors of larger wattage rating.

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Reactor Nuclear Flux Monitor

The reactor subcritical monitor and fission counter preamplifier performed successfully during several startup tests. Minor transient electrical noise problems are being resolved. Calibration of two more preamplifiers for use in the subcritical monitor system is about 50 percent complete.

Traveling Wire Flux Mapping System

Installation of flux wires for the traveling wire flux mapping system was started during the last outage. One channel was operated successfully, and it appears that the balance of the system can be successfully installed.

Coolant Pressure Control

Extensive testing was done on the primary coolant pressure control system to determine under what conditions this system transfers from automatic to manual control. If a transfer to manual control were to occur during a reactor scram, and immediate action were not taken by an operator, backup and cold spray would not come on, and repressurization to an unacceptable limit could result.

The cause of the transfer from automatic to manual control was traced to the summing point monitors for the pressure controllers. Decreasing the sensitivity of the summing point monitor prevents the transfer to manual control for rapid pressurizer pressure setpoint changes. It will be effective for a wide range of controller settings and will improve overall operation significantly.

Pony Motor Controls

All pony motors and ventilation fans were modified and tested. The testing included a check of the automatic start of the pony motor when the turbine is shut off, and the automatic stop of the pony motor when it is overdriven by the turbine. The automatic start worked satisfactorily on all units, but, at first try, the automatic shutoff did not work on any unit. The limit switches which operate when the turbine stop and throttle valve is opened were the cause of failure to shut off the pony motor when the turbine begins to drive the load. These switches were modified and the controls now operate satisfactorily.

Limit switches on all five turbines were modified on the west side trip and throttle valves, and all turbines were tested and adjusted. Modification to the switches on the east side of each turbine has been scheduled for the next reactor outage.

125 Volt DC Ground System

Five critical circuits previously powered from the 109-N Building 125 V DC system have been relocated to the 125 V DC system of the reactor building. With this modification, unnecessary reactor scrams will seldom be caused by grounds on the 109-N system.

SPECIAL TESTS PERFORMED

The status of Production Tests is as follows:

PT No.	HW No.	Title	Status		
			Approved	In Progress	Completed
2	80369	Routine Graphite Sample Irradiations	X	X	
3	81478	Routine Monitoring of Graphite Oxidation in N-Reactor	X	X	
6	81339	The Use of Isotope Producing Rods in N-Reactor	X	X	
7	81029	NPR Shield Evaluation; Shield Plug Measurement During Start-up Test N-2			X
8	81327	Co-Producer Demonstration Test (1.25)	X		
9	81609	Low Goal Irradiation of Fuel Elements with Varying Amounts of White Oxide on Surface	X	X	
10	82136	N-Reactor Corrosion Monitoring	X		
11	82074	105-N Fog Spray Control Valves			X
12	82241	Copper Base Alloy Corrosion Program	X	X	
14	82385	Irradiation of Thorium-Uranium Crud Monitor Elements in N-Reactor	X	X	
15	83664	N-Reactor Scram from Loss of BPA 13.8 KV Bus			X

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PT No.	HW No.	Title	Status		
			Approved	In Progress	Completed
16	82468	Horizontal Rod Scram from 400 MW, N-Reactor			X
17	82619	Horizontal Rod Scram No. 1, Power Ascension Program, N-3 Test			X
18	82659	Primary Loop Pressure Relief Valves (RV-2-1 and 2) Relief and Reseating Pressure Test	X	X	
20	82933	Hydraulic Actuators for RWSV-805 Valves			X
21	83200	Optimization of Secondary Coolant Supply Operational Modes	X	X	
21	83216 SUP.	Supplemental Testing Procedure, PT-NR-21, Surge Tank Pressure Pressure Control Test			X
21	83442 SUP.	Supplemental Testing Procedure, 184-N Turbine Generator Tests			X
21	83857 SUP.	Supplemental Testing Procedure, PT-NR-21, Pressurizer Level Control Test			X
23	83389	Horizontal Rod Scram No. 2, Power Ascension Program, N-3			X
24	84095	Plant Service Boiler Base Load Reduction Test Program			X
25	83966	Horizontal Rod Scram No. 4			X

OUTAGES

There was only one outage, that extending from the end of September through October 24, for repair of dump condensers and for other maintenance.

FUEL MANUFACTURE AND TECHNOLOGY

PRODUCTION STATISTICS

	<u>Current Month</u>	<u>CYTD</u>
1. <u>Input (Extrusions)</u>		
Outer Billets-0.95%	189	879
0.95% Single	8	8
1.25% Driver	0	85
1.25% Spike	0	378
Inner Billets-0.95%	<u>99</u>	<u>925</u>
Total Extrusions	296	2275
Total Tons	56.5	421.4
% of Forecast	113	92
2. <u>Output (Finished Production)</u>		
Assemblies-0.95%	1584	10,628
1.25% Driver	0	655
1.25% Spike	<u>360</u>	<u>3,077</u>
Total Assemblies	1944	14,360
Total Tons	39.4	281.4
% of Forecast	112	89
3. <u>Uranium Receipts</u>		
Total Billets	74	--
Total Tons	14	--
4. <u>Uranium Utilization</u>	72%	73%
5. <u>Shop Yield</u>		
Outers	77.6%	81.7%
Inners	69.1%	78.3%
Total	73.1%	80.5%

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PRODUCTION SUMMARY

The input production during October was 13 percent above forecast. This was the highest production, from the standpoint of extrusions (297) and tonnage (56.5) since October, 1963. Special scheduling of production flow enabled this higher extrusion rate to be effectively handled by fuel finishing operations even though heat treating equipment was out of service for several days.

Included in the input production were eight billets of special alloy uranium of 0.95 percent enrichment which were extruded into single tube (heavy wall) fuel elements. This material was produced to evaluate the processing characteristics of the new alloys and to determine equipment processing capabilities for this particular geometry, which is the same as that to be used for the 1.95 percent enriched driver fuel elements scheduled for extrusion in November.

Special extrusions were made of zircaloy tubing, lithium-aluminum ingot, and uranium sleeves in support of the 1.95 percent enriched co-producer fuel program. Output production of 39.4 tons during October exceeded forecast by 12 percent. This was the highest production achieved since April, 1964. Processing of the first batch of 1.25 percent spike enrichment fuel assemblies, started five months ago, is essentially complete. In addition to material for approximately 145 reactor charges, there are spike enrichment assemblies in finished storage for an additional 110 reactor charges.

PROCESS DEVELOPMENT

Unbonded End Closure Development

Tungsten-inert-gas welded and brazed (TIG-Braze) end closures have successfully been produced on outer fuel elements. The cap-wire design and basic welding parameters have been established. The design results in a closure that has a total penetration of about two and a half times the cladding thickness. Approximately 45 end closures have been made and destructively examined metallurgically. No deficiencies in the closure design were found.

Direct Extrusion of Components to Size

Extrusion of uranium and zircaloy components directly to size and elimination of the usual machining operations would significantly reduce material cost. To be successful, it is necessary to obtain very close dimensional control on the as-extruded stock.

A zircaloy billet was extruded directly to the size required for driver tube inner sleeves. Both inner and outer diameter dimensions were excellent, but the outer surface was rough owing to die pickup. It appears that it will be necessary to can the components in copper until better zircaloy extrusion lubricants are developed. Wall thickness variation was approximately 0.020 inch. Additional development on wall thickness control may be necessary.

Chemical Milling of Mechanically Recessed Fuels

Pilot plant evaluation has indicated that the chemical milling process for mechanically recessed fuels can be simplified by processing the fuels in a horizontal position, thus allowing both ends to be milled simultaneously without use of complicated baskets or agitation of the bath. It has also been shown that the copper sulfate buffer can be eliminated from the solution without resulting in undue surface variation or pitting.

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SOME ASPECTS OF FUEL ASSEMBLY DESIGN FOR CO-PRODUCT OPERATION

INTRODUCTION

Engineering studies have consistently shown N-Reactor to be a versatile production reactor. This versatility in production performance is the result of the flexibility of the reactor to accommodate a range of fuel designs and types, the corrosion resistance of the Zircaloy clad fuel, and the consistently favorable results from the supporting research and development efforts. Economical performance is aided by the dual-purpose feature of the plant in which process steam becomes a major product value.

The Zircaloy clad fuel costs are high relative to the costs of the aluminum clad fuel employed in low temperature production reactors. Therefore, product conversion costs also tend to be on the high side for those operating modes requiring a very short fuel residence time in reactor; these modes do not utilize the irradiation lifetime engineered and manufactured into the fuel. These higher product costs are especially evident when the product requirement is weapon grade plutonium containing 6 percent plutonium-240.

Previous studies of N-Reactor production capabilities have led to the conclusion that production of weapon grade plutonium is economically attractive when tritium is produced as a co-product in N-Reactor.

The higher uranium enrichment required for the production of tritium extends the fuel exposure required to reach the 6 percent plutonium-240 limit, thus reducing the amount of fuel required per year. An added benefit of this type of fuel loading is the increased product yield resulting from an increased nuclear cross section of the fuel and target.

Table I presents production and cost comparisons of plutonium-only and co-product operations of N-Reactor during a typical year when credits for steam delivered approach an equilibrium value. Analysis of these data leads to the conclusion that the co-producer program provides a method of producing weapon grade products up to \$7 per gram cheaper than through the plutonium-only route. When the reactor reaches equilibrium operation as a dual-purpose reactor producing process steam for credit, it will be a competitive producer of plutonium and tritium in the AEC complex.

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TABLE I

COST COMPARISONS OF PLUTONIUM-ONLY
AND CO-PRODUCT OPERATIONS

<u>Operating Mode</u>	<u>Plutonium-only</u>		<u>Co-Product</u>
Goal Exposure	800 MWD/T	2100 MWD/T	1700 MWD/T
Products (Plutonium equivalents)	1242 kg	1088 kg	1411 kg
Percent Pu-240	6.0*	12.1	6.0*

*Weapon grade plutonium

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FUEL DESIGN

The present production fuel in N-Reactor is a tube-in-tube design containing uranium enriched to 0.94 percent uranium-235. Each tube is clad with Zircaloy-2 by a co-extrusion process. With the aid of a computer code (FLEX-3), new fuel geometries can be designed. To convert from the present plutonium-only operation to co-product operation, the fuel design will be modified by thickening the wall of the outer fuel tube and substituting a rod containing an aluminum-lithium alloy for the inner tube (Figure 1). Supports and locking devices for the assembly will be similar to those for the tube-in-tube fuel.

Thus, the co-product fuel charge will differ in that a portion of the uranium-238 fertile material will be replaced with lithium-6 in the aluminum-lithium alloy target material. The overall cross section of the fuel-target assembly will be increased, total production of product materials will increase, and fuel exposure to attain a given plutonium-240 content will increase.

Once the quantity of tritium to be produced by N-Reactor has been established, the corresponding enrichment level of the fuel to produce this quantity becomes fixed. After the specific uranium enrichment has been chosen, increasing the weight of fuel per foot decreases the unit product cost. The maximum fuel weight is limited by the maximum allowable core temperature, by heat dissipation requirements, by control strength available, by driver-target effective cross sections, by nuclear safety requirements, by burnout heat flux, by boiling limits and by reactivity behavior.

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Two of these factors - core temperature and driver-target effective cross sections - are discussed below.

CORE TEMPERATURE

At constant reactor power level the peak core temperature will be directly related to the weight of fuel per foot.

Figure 1
Co-Product Fuel Design

Figure 2 shows the peak core temperature in a maximum power tube (5000 kw) as a function of fuel weight.

The peak uranium temperature is limited because of the large and abrupt volume change that occurs at the α - β transformation temperature (660 C). Cycling of uranium fuel through this transformation creates excessive stresses in the cladding and has resulted in cladding failures in test fuels. A 50 C margin of safety was chosen, thus fixing the peak uranium temperature at 610 C.

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10 X 10 TO THE CM. 358-14
KEUFEL & LESSER CO. MADE IN U.S.A.

Figure 2
Effects of Reactor Power Level on Peak Core Temperatures at various Fuel Weights

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From Figure 2 it may be observed that the limiting core temperature (610 C) will be reached with a fuel weight of 21 pounds per foot at a reactor power of 4000 Mw. The nominal clad thicknesses assumed for these calculations are 35 mils on the outer surface of the driver tube and 25 mils on the inner surface. At 4800 Mw the fuel weight must be reduced to 15 pounds per foot. Thus, for the same peak core temperature the fuel weight at a power of 4000 Mw would be about 6 pounds per foot more. For the same fuel weights, the core temperature at a power of 4800 Mw is about 80 to 90 C higher than at a power of 4000 Mw. The peak core temperature for a 20 pound per foot fuel element at a reactor power level of 4000 Mw will be about 590 C, or some 120 C higher than the maximum temperature in the present tube-in-tube fuel. This 20 lb/ft design is being considered for the first full reactor load of co-producer fuel.

The use of a 20 lb/ft driver operating at a reactor power level of 4000 Mw is also dependent on the development of a swelling resistant uranium alloy. To operate at 4800 Mw using a 20 lb/ft driver requires that both the swelling resistant alloy and a method of lowering the peak uranium temperature be developed. Lower peak uranium temperatures may be achieved by using fuel with thinner cladding.

Based on present technology, the cladding thicknesses on the outer and inner surfaces of the driver tube have been established at 35 and 25 mils, respectively. The rate of heat removal from the uranium is sensitive to the cladding thickness because the cladding acts as a relatively good insulating barrier between the uranium and the cooling water. A reduction in cladding thickness to about 15 mils on both surfaces would make it possible to lower the peak core temperature by about 60 C (Figure 3). Figure 4 shows that the core temperature is lowered about 4 C per mil of cladding removed. Therefore, an aggressive research and development effort to demonstrate that the present cladding thickness could be reduced is essential.

DRIVER-TARGET EFFECTIVE NEUTRON CROSS SECTION

For the production of tritium, lithium-6 is used as the fertile material in a target. The lithium-6 added to the fuel-target complex must be compensated for by increasing the uranium-235 in the driver.

Once the fuel driver weight has been chosen, the geometry of the driver and target is set within a very narrow range to achieve a balanced heat distribution to the two coolant flow channels. Changes in enrichment level of the driver fuel can result in only minor changes in the geometry.

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Fuel Weight - lb/ft

Figure 3

Effect of Clad Thickness on Peak Core Temperature at Various Fuel Weights, and Reactor Power Level of 4000 Mw

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Figure 4

Effects of Clad Thickness on Core Temperature

The maximum lithium content of the alloy is limited to 3 weight percent lithium. Above this concentration the metallurgical structure of the alloy may permit an undesirable amount of swelling from gas formation in the target during operation. Thus, for any given driver weight the maximum amount of lithium-6 is limited by both geometrical and metallurgical reasons.

At high driver weights the maximum amount of lithium-6 limits the uranium-235 level and thus limits the tritium production. Figure 5 shows the maximum amount of lithium-6 that can be placed in the target for each fuel weight. For example, the maximum quantity of lithium-6 that can be placed in a 21.3 lb/ft fuel design is 4.9 gram lithium-6 per foot of target. Correspondingly, the uranium-235 enrichment level is limited to 2.17 percent, which would produce 4.8 kg/hr. of tritium. Any further increase in enrichment beyond the 2.17 percent uranium-235 would require that the excess reactivity be taken up by the control rods, and consequently would provide no increase in tritium produced.

A second effect is the increased relative blackness of the driver as enrichment increases. As the enrichment of the driver fuel is increased, the ratio of neutrons arriving at the target to neutrons entering the fuel becomes smaller. The uranium-235 content could be increased until the driver becomes so black to neutrons that there would be no increase in tritium produced. Figure 6 illustrates this relationship.

In the design of co-product fuel assembly there are many other factors that are also still being considered such as heat dissipation requirements, control strength available, nuclear safety, burnout flux, boiling limits and reactivity behavior.

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"B"
Enrichment - 5.0%

Figure 6

The Zones of Neutron Absorption Illustrating
Effects of Target and Driver "Richness"

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KODAK SAFETY FILM
KODAK SAFETY FILM

TECHNICAL ACTIVITIES

NUCLEAR SAFETY

Irradiated Fuel Storage Criticality Studies

This work has been completed. The results include storage limits in three basic modules for regular 0.95 percent U²³⁵ tube-in-tube fuel, for 1.25 percent U²³⁵ outer-0.95 percent U²³⁵ inner spike fuel, and for 1.95 percent U²³⁵ co-product fuel. The modules differ primarily in the method of isolating arrays to prevent neutron interaction. The center-to-center spacings of the modules are:

23" x 30", providing at least 8" of water
between arrays

13" x 22", using canisters with neutron
poison in their outer walls

13" x 22", with 2" concrete walls containing
a neutron poison.

The 23" x 30" module is inefficient for the spike and co-product fuel but is the type of storage arrangement which is presently in place. The 13" x 22" modules are more efficient and fuel storage is limited primarily by the height to which it can be stacked and still maintain a safe depth of water shielding for handling operations. However, the type of canister used to contain and transport the fuel influences the storage height since freshly discharged fuel must be transported over stored fuel in some instances.

From the point of view of nuclear safety and maximum storage density the most efficient canister would be about 26" high and would hold a maximum of seven fuel elements. These canisters would be stacked in the modules at least four high, and possibly higher, depending on the radiation shielding requirement. Fuel spillage could be tolerated, since seven co-product drivers represent only about 42 percent of a minimum critical mass. Such a small canister load would, however, require many more fuel handling operations than are presently necessary.

Any canister which could be used in the 13" x 22" module and which would not increase the number of handling operations above the present level would have to be guaranteed against failure. Such a guarantee would not be necessary with the smaller 26" high canister. Also, the fuel load per module with taller canisters would be reduced at least 25 percent over the 26" high canisters due to the radiation shielding requirement during handling operations.

Critical Mass Experiments

Measurements are continuing on 1.25 percent enriched spike outer fuel elements and 1.25-1.95 percent spike assemblies. Measurements have been completed on the 2.81" and 3.1" lattice pitch arrays. The 3.4" lattice experiments will be completed by mid-November.

Off-Site Shipping Cask

An analysis was made of the heat transfer characteristics of a particular design of cask loaded with N-Reactor fuel. Seven different cases were investigated, involving fuel with 1300 MWD/T exposure aged 150 days, and fuel with 300 MWD/T exposure aged 200 days. Nothing was found in the analysis which would prevent the use of this cask for N-fuel transportation, considering existing criteria. The limiting considerations would be structural and heat dissipation integrity rather than the radiological consequences of an accident.

Ball Safety Backup

The slow rod counting circuit which provides ball system backup was revised to trigger a partial ball drop upon indication of two slow rods. The maximum number of rods which can be either removed from service or fail to scram without backup is therefore one.

Effluent Control

No significant release of radioactive material occurred via stack effluent during the month of October. No significant activity was discharged to the Columbia River.

Process Standards

Fifteen Process Change Authorizations were issued during the month. Of these, six were required to provide for equipment modifications, three to cover specific test requirements, and four to update Standards requirements. There are currently 37 PCA's in force.

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NEW EQUIPMENT

Clad Thickness Tester

The cost estimate for the clad thickness tester to be used in the Irradiated Fuel Examination Facility was recently reduced from \$15,500 to \$6,800 by incorporating this tester into the existing nondestructive tester. Performance requirements for the end cap profilometer were rewritten to simplify the device and reduce the cost estimate from \$20,000 to \$11,000.

Graphite Monitoring Facilities

The design and modifications of the ball, rod, and process channels to convert them to graphite monitoring test holes have been completed. Modifications to the tool for retrieving graphite samples from the rod channel test hole have been completed and it was satisfactorily tested.

NEW TECHNOLOGIC DEVELOPMENTS

Strength Tests on Connectors

A mixture of dry ice and acetone has been used successfully to create an ice block and shut off flow through connector piping while repairs were being made on the Grayloc fittings. Possible effects of the cold on the strength of the piping were investigated. No brittleness was found. Three samples were run with the following results:

	Sample		
	No.1	No.2	No.3
Temperature	Ambient	(-) 100 F	(-) 100 F
Strain Rate (in/min)	0.005	0.005	0.5
Yield Strength (psi)	45,100	56,500	71,700
Ultimate Strength (psi)	70,800	83,000	90,000
Elongation (%)	30	37	33
Reduction in diameter (%)	62	60	61

Half-rods

Design study continues on the possible installation of six half-rods for reactor control. The recommended design at present has tubular target elements loaded directly into the existing titanium tip shells, and has

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tubular dummies at each end of the element column to hold elements in proper location. The control strength is equal to 70 percent of a full-length boron carbide filled rod. Two alternate designs require less fabrication and installation time, and it may be desirable to utilize one of them during the interim period required to procure and fabricate the recommended rods. Disadvantages of the alternate designs are that one of them results in reduced control strength, and the other in reduced isotope production.

Dump Condenser Capacity

A study of the dump condensers concludes that the present heat transfer capacity is adequate for increases in power level up to 120 percent of design, with parallel increases in steam generator pressure to 150 psia. It was noted that any limitations in the heat dissipation system would be evidenced first in the circulating raw water system and not in the dump condenser heat transfer capacity.

Irradiated Metal Handling

Sixteen columns of fuel were discharged and processed through the packaging equipment. The experience obtained indicates that a sustained rate of at least four columns per hour can be achieved. The dummy diversion station appeared to work better in its new location, with no serious jams occurring. The piggy-back eliminator (for preventing one fuel element from riding partially atop the next element on the conveyor) worked satisfactorily with individual piggy-back elements; however, when more than one piggy-back condition occurred in a row the wiping station raised up and allowed all elements to pass.

Post-Irradiation Fuel Examination

Seventy-one inner fuel elements that were irradiated in the test loops have been retested nondestructively to evaluate the latest non-destructive testing techniques. Six of these fuel elements showed indications that are interpreted to be longitudinal cracks in the uranium near the end caps. Radiometallurgical examination to confirm the non-destructive test results has begun.

One element was examined destructively and a crack was indeed found. The crack extended from the cap interface to the area of the supports. Several much smaller cracks were also found. This element performed satisfactorily, with only 0.85 percent swelling after 2460 MWD/T exposure. Since all of the six pieces that appeared to have cracks experienced high exposures, on the order of twice goal exposure, there may be a relationship between exposure and cracking. In any case, the nondestructive technique appears adequate to detect this kind of flaw.

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ADMINISTRATIVE SUMMARY

OCTOBER, 1964

EMPLOYMENT

Section	<u>Permanent</u>		<u>Temporary</u>		<u>Total</u>		<u>Change</u>	
	<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	1	1	-0-	-0-	1	1	-0-	-0-
Finance	16	13	-0-	-0-	16	13	+1	-0-
Fuels	44	98	5	-0-	49	98	-0-	-1
Plant	64	218	3	-0-	67	218	-3	+1
Project	68	21	-0-	-0-	68	21	+1	+1
Research & Engineering	46	8	-0-	-0-	46	8	-1	-0-
Employee Relations	-2	2	-0-	-0-	2	2	-0-	-0-
Total Employees	244	361	8	--	252	361	-2	+1
Net Changes						613		
								-1

SAFETY

Days without a disabling injury	701
Hours worked without a disabling injury	2,254,762

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VISITORS

<u>Name of Component and Location</u>	<u>Name</u>	<u>Date(s)</u>	<u>Contact(s)</u>	<u>Purpose</u>
Atomic Energy Commission Washington, D. C.	E.F. Miller	10/29/64	M.C. Leverett	Discuss confinement
Allison Campbell, Seattle, Washington	E. Scanlon	9/29/64	W.W. Arnold J.E. Hansen	Consult on Campbell cut-off saw modifications and instructions
AMF Atomics York, Pennsylvania	R. Shustrin W. Pitman	10/15-20/64	R.C. Walker	Discuss automatic irradiated handling equipment and review the fuel handling system
Automation Industries Seattle, Washington	J.T. McElroy	10/6-7/64	R.W. Gilmore	Discuss lithium-sulphate ultrasonic transducers
Bailey Meter Company Seattle, Washington	J.M. Henderson	10/6-9/64 and 10/21-23/64	R.D. Crosier G.R. Klinger	Assist in the steam generator level calibration; review current state of the HPV-201 valves
Baldwin-Lima-Hamilton Ohio	D.G. Selma	10/23/64	R.G. Curran	Discuss extrusion press problems and cost vs. features
Beckman Instruments Fullerton, California	R.W. Awalt R.P. Connor R. Villalobos	10/7/64		Discuss application of gas chromatographs
C.F. Braun Co. Alhambra, California	J.E. Soehrens, Consultant	10/21/64		Consult on dump condenser system
Brush Beryllium Co. Cleveland, Ohio	K.G. Wikle	10/12/64	R.R. Studer L.H. Steves	Study Braze components
Chemical Proof Corp., Seattle, Washington	P.K. Schoening	10/5/64	R.C. Walker	Inspect the failed AHR tank membrane and provide estimates for repair

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VISITORS

Foxboro Company Seattle, Washington	C. Liakos	10/5-6/64	Discuss steam generator level measurement and power calculator modifications.
General Electric Co. Installation & Service Engineering Dept. Portland, Oregon	D. R. Resner	10/23/64	Consult on turbine problems
General Electric Co., Schenectady, New York	J.W. Healy Consultant	10/20/64	Consult on environmental hazards evaluations
General Electric Co., APED, San Jose, California	T.M. Snyder Consultant	10/22-23/64	Consult on physics prob- lems
Hallidie Machine Co. Seattle, Washington	J.R. Mohondro	9/29/64	Consult on Campbell cut-off saw modifications and in- structions
O.J. Harper Company Renton, Washington	O.J. Harper	10/9/64	Discuss mechanical fittings
Minneapolis-Honeywell Seattle, Washington	E. Haugen	10/13/64	Discuss high speed recording equipment
Nuclear Metals, Inc. Boston, Mass.	J.L. Kelin	10/26-30/64	Discuss N-fuel end closures
Sciaky Bros. Chicago, Illinois	W. Kelly	10/26-30/64	Discuss welding equipment
Wah Chang, Albany, Oregon	R.A. Graham	10/9/64	Discuss zircaloy strip and wire
Wallace & Tiernan Seattle, Washington	W.J. Larson	9/16/64	Deliver parts for the hot water quality laboratory oxygen analyzers

VISITS

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<u>Name of Component and Location</u>	<u>Date(s)</u>	<u>Name</u>	<u>Contact(s)</u>	<u>Purpose</u>
American Material Handling Society Region IX, Technical Conference, Seattle Washington	9/23/64	H.P. Kraemer D.L. Cornell		Attend show
Anchor Equipment Company San Francisco, Calif.	10/2-3/64	J.P. Sluka		Perform final inspection on replacement stems for primary coolant system valves
Advisory Committee on Reactor Safe- guards (ACRS) Washington, D. C.	10/8/64	M.C. Leverett W.S. Nechodom R.E. Trumble		Make presentations on N-Reactor
AEC Division of Reactor Development, Washington, D. C.	10/6-9/64	H.R. Kosmata	W. Williams J. Vallance	Discuss status of Desalinization and Reactor Development programs
Burns & Roe Hempstead, New York	10/16/64	R.A. Rohrbacher		Discuss Conversion instrumentation
General Electric Company San Francisco, Calif.	10/14-16/64	M.C. Leverett N.R. Miller R.E. Trumble		Attend General Electric Technological Hazards Council meeting
General Electric Company Advanced Technology Lab., Schenectady, New York	10/15-16/64	D.O. Allred	P.C. Bogiages	Discuss instrument developments and techniques
ISA International Show New York City, N. Y.	10/12-14/64	D.O. Allred		Present paper, "A Continuous Gamma Monitoring System for Detecting Fuel Element Failures in Hanford's New Production Reactor"

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VISITS

<u>Name of Component and Location</u>	<u>Date(s)</u>	<u>Name</u>	<u>Contact(s)</u>	<u>Purpose</u>
ISA International Show New York City, N. Y.	10/12/64	R.A. Rohrbacher		Present paper, "Sample and Control for Fast Detection of Fuel Failure in a Pressurized Water Reactor"
Montek Salt Lake City, Utah	9/28/64	W.D. Bainard		Approve detail drawings - Conversion
Vacu-Blast Co., Abilene, Kansas	10/26-28/64	D.E. Blahnik	D. Knox S. Kizowski	Perform and witness final inspection and test of new abrasive cleaning machine
AEC-Division of Production, Washington, D. C.	9/13-16/64	D.R. Stenquist	W. Devine	Production Studies Conference

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SIGNIFICANT REPORTS

Report Number	Title	Author	Date
HW-84244	Annulus Gauge Description and Operation	K.O. Creek	September 25, 1964
RL-NRD-10	Coextrusion Billet Design for NRD Co-Product Target Elements	R.G. Curran	October 14, 1964
RL-NRD-20	Comments: Target Element Thermal Stress Tests	A.E. Guay	October 20, 1964
HW-83918	Construction Completion Report - CAI-816 - 100-N Reactor Plant	C.L. Buckner	October 15, 1964
HW-84405	Design Criteria for Target Alloy Production Facility	R.R. Studer	October 1, 1964
RL-NRD-2	Design Study of Half Rods for N-Reactor	G.A. Newell	October 12, 1964
RL-NRD-6	Differential Pressure Test Dump Condenser #1	C.A. Munro	October 11, 1964
RL-NRD-15	Dump Condenser Repair Completion Report	J.W. Helton	October 30, 1964
HW-84336	Dump Condenser Report - N-Reactor	C.L. Buckner	October 2, 1964
RL-NRD-13	The Effect of Goal Exposure on Phase III Electrical Power Generation	R.K. Robinson	October 19, 1964
HW-84076	Failure Probability - End Cap Defects	J.P. Keenan	September, 1964
HW-83893	Gravity Drain Line from 109N Turbine Bay Basement		October 19, 1964
HW-84395	Grayloc Repair Procedures	J.H. Fastabend	October 2, 1964
RL-NRD-30	Grayloc Repair Program	J.H. Fastabend	October 28, 1964
HW-83794 Part 2	100-N Equipment Experience Report 0-20 Percent Design Power Level	J.W. Helton	September 23, 1964
HW-84406	N Fuels End Closure Brazing Review	W.M. Scott	October 15, 1964

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Report Number	Title	Author	Date
HW-84177	Conf N-Reactor Plant Heat Balance at 70% Power	A.J. Ebens V.J. DeJong L.D. Smith	September 24, 1964
HW-82100 PTI	Conf N-Reactor Startup Physics Test Results - Approach to Critical and Critical Reactor Parameters	L.L. Grumme J.W. Hagan	July 24, 1964
HW-84423	Uncl NPR Fuel Element Spacer Mark I-A Pressure Drop Tests	M. Pociluyko	September, 1964
RL-NRD-28	Conf Need for Hydrogen Addition to N-Reactor Primary System Coolant	W.K. Kratzer	October 26, 1964
RL-NRD-7	Secret Official Production Forecast, N-Reactor Fuels Operation, October 1964 through June 1966	L.M. Loeb	October 12, 1964
HW-84024	Uncl Phase II Export Power - Reactor Power Relationships	H.R. Kosmata	October 12, 1964
RL-NRD-12	Uncl A Preliminary Consideration of the Testing Procedures for the Control, Instrumentation and Systems Operation of the Conversion to Phase II Operation of the N-Reactor Plant	D.W. Leiby	October 15, 1964
HW-84280	Uncl Preliminary Scope of a Technical Review of the Control Functions Required for Conversion to Phase II Operation of the N-Reactor Plant Heat Dissipation System	D.W. Leiby	September 30, 1964
HW-84196	Uncl Reclaiming of Inner Fuels with Cracked Supports	W.G. Hudson	September 18, 1964
RL-NRD-23	Uncl Secondary System Instrumentation Modifications for Conversion	A.A. Maupin, Jr. D.W. Leiby	October 21, 1964

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SIGNIFICANT REPORTS (Continued)

Report Number	Title	Author	Date
RL-NRD-14 Conf	A Study of N-Reactor's Required Primary Loop Operating Pressures and Limits for Phase I and Phase II Operations with Five Heat Exchanger Cells	H.G. Johnson	October 19, 1964
RL-NRD-34 Conf	Summary of Presentations to the Advisory Committee on Reactor Safeguards	M.C. Leverett	October 29, 1964
HW-84178 Secret	Target Element Specifications, Production Test NR-13	D.H. Walker	October 1, 1964
HW-84422 Secret	Technical Basis for Target Element Fabrication	D.H. Walker	October 8, 1964

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 October, 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.

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SUGGESTION PLAN PARTICIPATION

	<u>OCTOBER</u>	<u>OF 1964</u> <u>TO DATE</u>
Number of Eligible Employees	361	363
Number of Suggestions Received	51	499
Number of Suggestions Acted Upon	32	470
Number of Suggestions Adopted	12	101
Net Annual Savings	\$ 417	\$134,220
Amount of Awards	\$ 105	\$ 7,655
Percent of Awards to Savings	44.4	5.7
Average Amount of Awards	\$15.42	\$ 42.20

SUGGESTIONS ADOPTED/1000 EMPLOYEES/YEAR

1964

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