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N - REACTOR DEPARTMENT

MONTHLY REPORT

SEPTEMBER

1964

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HANFORD ATOMIC PRODUCTS OPERATION

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N-REACTOR DEPARTMENT
MONTHLY REPORT - SEPTEMBER, 1964

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OPERATING SUMMARY

Input production Pu (KMWD)	34.09
Input production T (KMWD)	0.
Maximum power level attained (MW)	2,800
Average power level while operating (MW)	2,660
Time operated efficiency (%)	46.8
Plant utilization (%)	49.5
Number of Shutdowns	
Scheduled	1
Unsechduled	3
Reactor Process Data	
Power at which computed (MW)	2,800
Primary coolant inlet temperature (°F)	360
Primary coolant outlet temperature (°F)	480
Maximum graphite temperature (°F)	935
Heat Loss	
Graphite cooling (MW)	39
Shield cooling (MW)	1.8
Rod cooling (MW)	0.9
Steam temperature (°F)	324
Steam pressure (PSIG)	80
Dump condenser flow (gpm)	420,000
Inlet temperature (°F)	70
Outlet temperature (°F)	133
Nuclear Steam Generating Rate	
Maximum (1000's lb/hr)	8,300
Average while operating at (1000's lb/hr)	7,076
Water Collected from Gas Atmosphere (gal/day)	Negligible
Demineralized Water Produced (gal/hr)	79,980
Helium Losses (1000's cu. ft.)	828
Fuel Oil Usage (bbls.)	26,000
Flattening Efficiency (%)	82

Fuel Conditions

Heat flux

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

Clad temperature

Maximum (°F)	510
Average (°F)	480

Uranium temperature

Maximum (°F)	725
Average (°F)	660

Fuel Failures

0

Fuel Balance	<u>.094</u>	<u>1.25</u>	<u>Co-Prod.</u>	<u>Other</u>	<u>Total</u>
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)	331.38	36.43	0	0	367.82

Fuel Element Output

0.95%	792
1.25% (driver)	0
1.25% + 0.95% (spike)	576
Total assemblies	1,368
Total tons	27.4
% of forecast	78

Uranium Utilization

73.6%

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SUMMARY

The reactor reached the 70 percent power level and operated at this level for eight days without interruption. The samarium dip has been passed and long-term reactivity gains are being realized. A small amount of flux cycling appeared and was controlled by adjustment of the rods. Additional spike fuel was charged, and gave a slightly higher reactivity effect than predicted. The flattening efficiency was 82 percent.

Vibration from high velocity steam has caused tubes in the dump condensers to leak. The corrective measure being taken is the insertion of numerous steel fingers radially into the tube bundle to be held in place by spring loaded steel girth bands. Repairs are well under way.

The fuel element segregation system has been improved by changes in both the mechanical and electrical components. It is now operating on automatic control.

Carryover of moisture from the steam generators has been reduced to a satisfactory level by reducing the liquid level in the steam generators. Interim liquid-steam interface level controls in the steam generators were effective in preventing excess water carryover. Steam quality is about 98 percent.

REACTOR PLANT OPERATION

OPERATING PATTERN

N Reactor operated for twelve days during September at power levels of 60 percent and 70 percent of design power level. A successful scram test was carried out from the 70 percent level, with no major system malfunctions. Discovery of leaking tubes in the dump condensers is requiring an extended shutdown for repairs.

Power Ascension

On September 4 the power was raised to 60 percent of design power level. A front to rear neutron flux cycle with a period of about 30 hours was observed and readily controlled by adjusting the rod pattern. On September 8 power was increased to 70 percent and no flux cycling tendencies were encountered.

Test Scram from 70 Percent Power

The plant continued in efficient operation at 2800 MW until September 16, at which time a test scram was executed in accordance with earlier plans. The scram went smoothly. No excessive variations of pressures or liquid levels were noted. Secondary system transients dropped the surge tank level below the indicator zero point, but did not result in loss of condensate pump suction. The primary loop volume contracted somewhat more than previously, first because the No. 1 drive turbine slowed to only 50% of its normal scram speed, resulting in greater flow through the steam generators, and secondly because the steam generator bypass valve in cell 3 stuck partially open and had to be opened by manual initiation from the console controls. Both malfunctions are being corrected.

From the behavior of the plant during the previous eight days of uninterrupted operation at 2800 MW preceded by four days at 2400 MW, it is concluded that continuation of the power ascension program beyond 2800 MW could take place with safety.

Condensate Carryover in Steam

The reactor scrammed on August 30 from a circulating raw water low flow trip caused by a low-frequency tripoff of Nos. 2 and 4 circulating raw water pumps. Just prior to the scram, the feedwater control valve

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on 1A steam generator had increased to full-open when the reactor power was increased slightly above 50% of design power. The increase in feed-water caused high-moisture in the reactor-produced steam; this increased moisture fed into the station turbogenerator set and caused it to slow down and lose frequency.

At the time the scram occurred, the cause of the apparently high feedwater rate to 1A steam generator was not known. During the ensuing outage a section of the 12" feedwater line was cut out to determine if it was plugged. Nothing abnormal was found, and the pipe section was reinstalled. It was concluded that the high feedwater flow was due to water carryover in the steam from 1A steam generator, caused by excessively high liquid levels. The level instruments were recalibrated, the steam line from 1A was fitted with a steam sample tap to permit steam-calorimetric measurements of steam quality, and operation was resumed.

On September 3, at 50% power, a scram occurred from low flow of circulating raw water, which duplicated the scram of August 30. The turbine generator was slowed by a surge of high moisture steam caused by abnormally high flow of feed-water to 2B steam generator. Just prior to the scram, air was lost to the pressure-regulating valves in the diversion system, and they failed closed. This failure of the air line to the valve positioners was caused by high vibration at the valve station. The line was repaired, and operation resumed the next day.

It is believed that the condensate carryover and high feedwater flowrate may have been caused by too high a liquid level being maintained in the steam generators. Only indirect methods are available for determining the liquid levels in each of the steam generators except 1A, which is equipped with a special float. Even the use of a float is complicated by the necessity to evaluate the effects of the varying density of the steam-water mixture which leads to an indefinite "froth-zone" instead of a sharp line of demarkation between condensate and steam.

The liquid level in the specially instrumented steam generator 1A was compared with the float indications for various levels up to the liquid level for which condensate carryover was signalled by the steam quality meter installed in the steam outlet line. The data thus obtained indicate that the liquid in the generators had been maintained at too high a level. All steam generator liquid level sensing systems have been recalibrated, and the generators are now being operated without significant condensate carryover.

When the reactor was started up after the scram test from 70 percent power it was observed that an excessive amount of makeup water was being demanded by the secondary system. The reactor was shut down and investigation of the cause of the discrepancy was started. It was found that water was passing from the steam side of the dump condensers to the water side. This fact was verified by filling the dump condensers steam space with water, closing all drain valves and observing that the water level in the steam side of the dump condenser was falling, presumably due to leakage through leaky or broken tubes.

Later, the raw water boxes on 3 condensers were opened and in each approximately 7 tubes were found to have full flow leaks. Upon opening up the shell side of condenser No. 2, it was found that the tinned copper webbing straps placed on the tube bundle about a month ago had broken and the tubes were hence free to vibrate in the steam. Several tubes were found to be completely broken off at the topmost support baffle, presumably as a result of vibration. It is suspected that at least 10 of the dump condensers are leaking. Only one of the 11 so far checked appears to be intact. Repairs are proceeding as described on pp 11.

Thermal Shield Cooling

Three thermal shield cooling tubes are running hotter than expected. Partial plugging with debris from the raw water system is suspected. These tubes are being backflushed during the current outage.

Unusual Incidents

While the plant was operating at 60% power, a heat-dissipation-console operator accidentally opened the steam generator bypass valve on Cell No. 1 when he mistook the bypass switch for the similarly-shaped switch for starting the cell sump pump. Opening the bypass valve increased the reactor inlet temperature and reduced the reactor power. As soon as the operator realized what he had done, he actuated the valve to close it. When it closed, the resulting temperature transient caused a momentary power increase, actuating the power setback circuit and automatically reducing the power before high flux and high rate of rise scram trips were reached. Plant recovery after the setback was normal. Following this occurrence, both the incident and the proper corrective sequence were reviewed with each control operator. Distinctive handles have been ordered for the valve switches.

A trip-off of the 480 V feeder breaker to Motor Control Center No. 13 while the plant was at 70% power resulted in the loss of three condensate pumps in the secondary system. Immediate action by the heat-dissipation-console operator to start spare condensate pumps prevented a system upset. The trip-off was found to be due to derated breaker heaters. New heaters of the correct capacity have been installed.

Process Standards and Process Change Authorizations

Revision of process standards to reflect increased plant operating experience and the results of engineering studies based on operating data is carried out on a continuing basis. The following standards were recently revised or reissued.

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Total Control
Secondary Cooling System
Ball Safety System
Safety Circuits
Fissile Material Control

A total of 25 Process Change Authorizations (PCA's) were issued since the last report. Thirty-two PCA's are in force as of this date.

PROCESS PHYSICS

Equilibrium Conditions

Reactor Power Level	2800 MW
Reactivity in Rods	6.4 mk
Flattening Efficiency	82%
Rod System Worth	67 $\pm$ 4 mk
Spike Enrichment Inventory	132 columns in the 3rd and 4th lattice unit
Reactivity Contribution of Spike Enrichment	5.6 mk
Inferred Reactivity held in Rods without Enrichment	0.8 mk

The flattening efficiency of the reactor was increased to approximately 82% as a result of charging the last 68 spike columns. The reactivity effect and flattening efficiency are slightly higher than the theoretical predicted values, 5.25 mk and 78.2% with the 132 spike columns.

Flux Cycling

A small amount of flux cycling in the front-to-rear direction was noticed during the month at an equilibrium power level of 2400 MW. This was combated by changing the equilibrium rod pattern to include more front and rear rods instead of central rods. This enabled better control and the flux cycling was stopped. Since the tendency to cycle will be enhanced with higher fuel exposure and power level, a watchful eye will be placed on this aspect of reactor operation in the future.

Reactivity Gains

During equilibrium operation at 2800 MW, it was apparent that the "Samarium-dip" has been passed and the reactor is now realizing some long-term reactivity gains. This increasing reactivity will aid in providing more reactivity for control; however, it will also have the effect of de-flattening the reactor since most of the reactivity gain will occur in the central zone of the reactor.

MECHANICAL AND THERMAL EQUIPMENT PERFORMANCE

Dump Condensers

The problem of the leaking dump condenser tubes described earlier (pp8-9) is one of fatigue failure of the tubes at the top support baffle, induced by tube vibration caused by aerodynamic forces in the steam. The corrective measure being taken is the insertion of numerous steel fingers radially into the tube bundle in a plane normal to the tubes at the top of the inverted U formed by them. The fingers are held in place by a pair of steel girth bands. Springs provide the needed latitude for thermal expansion. The effect of the fingers is to sharply limit the possible amplitude of vibration of the tubes in the plane of the U bend. Initially such vibration was controlled by steel bands encircling the tube bundle at several points along the U. When these proved to cut into the tubes they were replaced with copper wire webbing straps (tin plated). These straps proved too weak and failed in less than a month.

One dump condenser has been fitted with the fingers as above described, and work is starting on others.

Tests made on a model of a dump condenser showed that installation of additional baffles of the right design would probably reduce turbulence markedly, and presumably reduce the forces exciting the tube vibration. However, it was felt that gains from this expedient were somewhat uncertain, and in no case likely to eliminate the need for restraining the tubes mechanically. It was therefore decided not to alter the baffling at present.

Grayloc Couplings

Repair of nine leaking Grayloc couplings in the right rear connector space was initiated at the start of the present outage. Severe wire drawing of two Graylocs necessitated weld-buildup of the seal surface before machine repair was possible. More compact tooling is being designed to greatly reduce the setup time required for machining a Grayloc flange.

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Control Rod Seals

Several instances were encountered when, following a scram, horizontal rods stuck in the full-in position because of binding at the drive cylinder rod seal.

To study the problem of binding, cylinder rod seals made of various materials are being tested. Modifications have been made to a recommended seal by removing brass spacers that separated the V-rings and by replacing one of the Teflon V-rings with one made of buna-N material. This seal has been installed in the horizontal test drive, and 50 scram test runs made with it, without any indication of binding. Test results also show the leakage rate of this seal assembly to be comparable with that of the original vendor-installed seal. Tests will continue to determine whether the seal will last a satisfactory lifetime and still not bind.

Rupture Monitor System

Leaks in the right and left side differential pressure control valves were repaired by installing Hastalloy liners in the bodies. Stellite needle point plugs and seats were installed in the maintenance valves of the rupture monitor differential pressure system for the purpose of taking a set amount of the necessary pressure drop across them. Sample flow lines that were causing trips on the gross gamma system were backflushed and the flow rates and hole-up times were checked.

Reactor Gas Seal

To reduce the helium loss from the reactor during operation, all static and dynamic gas seals on the outlet and inlet faces of the reactor were tightened. Several of the blanking flanges covering the spare tube holes on the unit were also tightened.

The upper right corner section of the outlet face gas seal was found to have lost its elasticity, probably from overheating from air leaks through the thermal barrier at that point. A splice was found to have begun to open. The parts are being replaced. Baffles will be installed at the four corners on each face to eliminate the undue heating of the seals.

Afterheat Removal Tank Membrane

The membrane has once again been removed from the afterheat removal tank because of a large rip in it. Once again there are a number of theories as to what caused the rip. A scope study is being made of floating roofs to replace the membrane. There are five known manufacturers that build floating roofs, primarily for petroleum storage. Two of these are of all-aluminum construction and are therefore not suitable for our service. Two are of all-steel construction and could be coated or fabricated of stainless steel. The fifth style of lid is made of foamed plastic sandwiched between two layers of aluminum; it could be made with stainless steel. All designs use a peripheral wiper type seal. To accommodate a floating roof, the interior of the tank will

require some rework. The membrane attachment flange must be cut off and ground smooth; then the top half of the tank must be sandblasted and lined. In the meantime, discussions are being held to determine the desirability of repairing and replacing the membrane.

Irradiated Metal Handling

All irradiated fuel elements except those in three loaded dummy baskets have been packed in canisters. The segregation table and turrets are now operable on automatic control. On the discharge conveyer all graphalloy bushings are being replaced and a remotely operated chain tightener and a new air-operated clutch installed. The fuel diversion and gaging stations are being removed, the dummy diversion station relocated, and a new conveyer wiping station mounted. Electrical design work was initiated to eliminate a source of fuel element jamming by providing a better interlock between the segregation table, storage chamber feed, and turret feed.

Emergency Dump System Valves

An analysis of the primary coolant emergency dump system has disclosed that there is a possibility that leakage of the pressurized nitrogen past the metal piston rings in the valve operating cylinder could cause the valve disc to creep from its standby Position I, which directs the emergency dump effluent to the quench basin, toward its Position II, which discharges to crib. This undesired creep of the valve disc is possible because of the difference in areas of the two piston faces, created by the piston rod, and the continuous pressurization of the operating cylinder. Continuous pressurization of the valve operating cylinder, however, is not a basic requirement for the satisfactory operation of these valves because they have a comparatively simple operating linkage, and the large over-center type spring will hold the valve disc in its set position. By a simple modification to the wiring of the solenoid valves that control the supply and exhaust of the pressurized nitrogen, it will be possible to keep the valve operating cylinder at atmospheric pressure except when the valve is being operated.

ELECTRICAL AND INSTRUMENT EQUIPMENT PERFORMANCE

Central Data Logger

Troubleshooting of the central data logging system is continuing. Several cross wires were found. Adaptor plugs, new typewriter keys and some solenoids are needed. The installation of an eight channel key punch was completed. The rod data logger typewriter can be made compatible with the central data logger.

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Secondary Coolant Pressure Control

The pressurizer level control circuit was revised to include rate action.

By increasing the gain on the secondary system pressure controllers the butterfly valves between the main steam header and the dump condensers can be made to close more quickly following a scram. This will maintain the pressure setpoint and thus reduce the rate at which the standby boiler picks up its required steaming demand.

Work on the surge tank level control system was directed toward adjustment of the controller and time lag settings. Specifically, the controller gain was changed from 4.0 to 0.5, and then back to 1.0. This action reduced significantly the oscillation of the normal makeup feedwater flow into the secondary system. Also, it was demonstrated that the time lag setting could be reduced from 3.8 minutes to 0.9 minutes with no apparent increase in feedwater flow or surge tank level oscillations. Reduction of the time lag setting will diminish the rapid overshoot of feedwater flow into the secondary system following an increased demand in auxiliary steam flow or upon a decrease in steam demand. It will therefore prevent the makeup flow from changing so rapidly that it causes a significant change in surge tank level. Better makeup flow control will also lead to better surge tank pressure control.

Gas System Instrumentation

The effect on the gas system instrumentation of changing the reactor gas from He to CO₂ was investigated. The chromatograph which measures CO₂, CO, O₂, N₂ and H₂O will require column changes, interior piping changes, and programming changes as well as a period of testing to verify that the modifications produce the desired results. It is expected that the chromatograph measuring H₂ will require only minor modifications.

Steam Generator Level Instrumentation

Difficulty in obtaining reliable measurement of liquid level in the steam generators from the existing differential pressure instrumentation has resulted in a study of the problems and investigation of other means to measure this level. Displacement type level transmitters are being considered because they appear to offer improved accuracy and sensitivity of control. Float-operated level transmitters, although perhaps not as sturdy as the displacement type, would have many of the same advantages and are also being considered. The float-operated level transmitter now installed in steam generator 1A for calibration purposes has shown good promise. Both types of level transmitters will be installed in one steam generator for comparison and further evaluation.

The range of the wide range steam generator level transmitters is being increased to cover all possible level conditions during operation in the water-to-water mode. In addition, both alarm meters and annunciation will signal high water level in the steam headers. The new range will be from 5'6" below the center line of the steam generator to 44'6" above it.

Steam Generator Instrumentation System

Cleaning and calibration of instrument systems is continuing. The steam and feedwater flow transmitters for steam generator 1A level control system have been cleaned. The steam flow transmitters were found to be clean, but the feedwater flow transmitters contained a black scum, which was removed.

The steam and feedwater flow transmitters for all steam generators have now been calibrated. The uncompensated level transmitters have also been calibrated. Work is continuing to improve the accuracy of calibration of compensating circuits.

Pony Motor Controls

Tests were conducted on four pony motors to determine their ability to start automatically. Normal performance was verified. The reverse power trip for each of the pony motors was found to function incorrectly due to improper adjustment of the throttle and trip valve limit switches. Investigation of an accidental start of pony motor No. 5 has shown that a combination of sticking of the speed sensing relay, manual tripping of the interlock bypass switch, and failure of the trip and throttle valve limit switches to be the contributing factors causing the start. Calibration of the speed sensing relays, installation of surge suppressors across certain relay coils, and readjustment of the trip and throttle valve limit switches will prevent a recurrence.

Cell 4 Checkout

Calibration and functional checking of the instrumentation systems of Cell No. 4 were initiated. Calibration was completed on the primary coolant flow system, the primary temperature system, the drive turbine seal water control system, and the steam generator wide-wide level instrumentation.

Other Maintenance

Maintenance included also the following:

- a) Replacement of one of the subcritical chambers of the reactor flux monitoring system and shipment of the displaced chamber to the vendor for repair.
- b) Installation of shut-off valves in the air supply lines to the modulating damper positioners of the ventilation controls for the reactor building.
- c) Modification and repair of one of the two platform bridge cranes serving the spent fuel storage basin. Similar work is proceeding on the second crane.
- d) Primary cooling system installation of servo valves, following factory reconditioning, on all sixteen high pressure valves.
- e) Replacement of the underrated resistors in the confinement and fog spray counting circuits.

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SPECIAL TESTS PERFORMED

The status of Production Tests in the N-Reactor is as follows:

PT No.	HW No.	Title, Date	Status		
			Approved	In Progress	Completed
2	80369	Routine Graphite Sample Irradiations 1/3/64	X		
3	81478	Routine Monitoring of Graphite Oxidation in N-Reactor, 3/26/64	X	X	
6	81339	The Use of Isotope Producing Rods In N-Reactor, 3/17/64	X	X	
7	81029	NPR Shield Evaluation: Shield Plug Measurement During Startup Test N-2, 3/2/64	X		X
8	81327	Co-Producer Demonstration Test (1.25) 5/4/64	X		
9	81609	Low Goal Irradiation of Fuel Elements with Varying Amounts of White Oxide on Surface, 3/30/64	X	X	
10	82136	N-Reactor Corrosion Monitoring, 5/25/64	X		
11	82074	105-N. Fog Spray Control Valves, 4/30/64	X		X
12	82241	Copper Base Alloy Corrosion Program, 6/4/64	X	X	
14	82385	Irradiation of Thorium-Uranium Crud Monitor Elements in N-Reactor, 5/21/64	X		
18	82659	Primary Loop Pressure Relief Valves (RV-2-1 and 2) Relief and Reseating Pressure Test, 6/11/64	X	X	
19	82800	Surge Tank Inert Gas Evaluation, 6/12/64	X		X

<u>PT</u> <u>No.</u>	<u>HW</u> <u>No.</u>	<u>Title, Date</u>	<u>Status</u>		
			<u>Approved</u>	<u>In Progress</u>	<u>Completed</u>
20	82933	Hydraulic Actuators for RWSV-805 Valves, 6/23/64	X		X
21	83200	Optimization of Secondary Coolant Supply Operational Modes, 7/8/64	X	X	
24	84095	Plant Service Boiler Base Load Reduction Test Program, 8/15/64	X		
25	83966	Horizontal Rod Scram No. 4, 9/9/64	X		X

OUTAGES

<u>Date</u>	<u>Outage Type</u>	<u>Duration</u>	<u>Cause</u>
8/30/64	Scram	99.1* hrs.	Circulating raw water low flow from CRW pump trip-off on low B bus frequency caused by water carryover into the turbogenerator set.
9/3/64	Scram	13.5 hrs.	Same as above.
9/16/64	Scram	34.8 hrs.	Performance of production test for evaluation of plant performance.
9/18/64	Scram (Manual)	3.0 hrs.	Boiler burner failure, requiring scram because of loss of boiler and impending loss of electric al bus "B".
9/18/64	Manual shutdown	286.5 hrs. at month end.	Multiple leaks in dump condenser tubes.

*Including 45.6 hours in August.

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FUEL MANUFACTURE AND TECHNOLOGY

PRODUCTION STATISTICS

	<u>Current Month</u>	<u>CYTD</u>
1. <u>Input (Extrusions)</u>		
Outer Billets-0.95%	163	690
1.25% Driver	0	85
1.25% Spike	0	378
Inner Billets-0.95%	<u>70</u>	<u>826</u>
Total Extrusions	233	1979
Total Tons	44.9	364.9
% of Forecast	90	89.7
2. <u>Output (Finished Production)</u>		
Assemblies-0.95%	792	9044
1.25% Driver	0	655
1.25% + 0.95% Spike	<u>576</u>	<u>2717</u>
Total Assemblies	1368	12,416
Total Tons	27.4	242
% of Forecast	78	86.4
3. <u>Uranium Receipts</u>		
Total Billets	147	--
Total Tons	14.5	--
4. <u>Uranium Utilization</u>	73.6%	70.6%
5. <u>Shop Yield</u>	<u>Aug.</u>	<u>CYTD</u>
Outers	82.3%	82.0%
Inners	67.5%	80.2%
Total	80.0%	81.4%

PRODUCTION SUMMARY

Input production for September was 90 per cent of forecast. Major equipment failures in the chemical bay exhaust system and in the heat treating facility were responsible for the drop.

Production of finished fuel element assemblies was 78 per cent of forecast. Unusually high rejects from warp and from weld defects on the 26-inch inner fuel tubes for the spike enrichment program decreased the total production of finished fuel elements by about 400 assemblies. Inventory of completed fuel element assemblies for the spike enrichment program is sufficient to satisfy reactor requirements through October.

Accelerated uranium billet deliveries from the feedsite during the past several months have resulted in total deliveries and inventories being considerably ahead of schedule. The shipping schedule is being adjusted.

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PROCESS DEVELOPMENT

Bonded Chevron End Closure Development

The design for chevron end caps for the standard N-outer fuel has been developed. Several hundred fuel elements will be processed with this cap design to determine its economic aspects. Although the initial cost of the chevron end cap is higher than that of the present cap, it is anticipated that there would be sufficient gains in yield to offset this. There is another advantage in that the chevron recess provides an added margin of safety against cap-core shear failure, should this problem threaten the integrity of fuel element closures.

Unbonded End Closure Development

Development work has continued on the unbonded tungsten inert gas braze end closure. Welder modification, cap designs, recess designs, and sizing processes for cladding are essentially complete. The critical item at this point is the fabrication of acceptable beryllium zircaloy wire to be used in the braze preform.

Two processes for wire fabrication are being actively pursued in the Hanford Laboratories facilities. One involves the canning of a beryllium zircaloy casting in stainless steel and successive swaging at about 750°C. But the repeated swaging is expensive, and there is difficulty in removing the stainless steel with selective etchants that will not attack the beryllium zircaloy wire. Attempts to use other cladding materials have been unsuccessful.

The other fabrication process involves a two-step coextrusion with interim swaging steps. It appears possible by this process to fabricate over 3000 feet of wire with only one coextrusion assembly. The major difficulty at the present time is control of heating during swaging.

To provide a parallel and backup development program to the studies on tungsten inert gas braze, work has been initiated to develop a high power density electron beam weld. An offsite contract has been placed to demonstrate the feasibility of this process. Prototypic fuel ends will be supplied to the contractor.

Cracked Welds

Three fuel elements at final inspection were discovered to contain radial cracks through the outer weld bead. These cracks appeared to originate from a stamp mark and terminate in the outer clad area. The stamp mark had been placed so that it contacted the weld bead and introduced a stress in it. Sectioning of these cracks has shown that they extend to the bottom of the recast zone of the weld bead. The fabrication history of each piece indicates

no common characteristic of braze lot, processing time, or welding time that would associate the cracks with a process parameter. Because there appears to be a strong correlation between the stamp marks and the cracks, investigation is under way to determine if the stamping operation should be performed prior to welding. Examination of these fuel elements will continue in a search for other clues to the cause of the cracks.

Warp

The warp of inner fuel element tubes has increased coincidentally with the start of production of 26-inch lengths, the change to bailless hangers in heat treating, and a change in type of extrusion lubricant. Each of these changes has been thoroughly analyzed but no significant effect on inner fuel element warp has yet been discovered.

Inspection for Cracked End Caps

Following the in-reactor fuel element failure in July, an analysis of input data was made to determine the probability of sustaining additional reactor failures as a result of cracked fuel element end caps. The results of this analysis have been documented.

Identification of a cracked end cap as the cause of failure has resulted in a requirement for 100 per cent dye penetrant inspection (Zyglo) of all end caps prior to process use. Another preventive step is revaluation of the criteria for visual inspection of end closure welds. Currently, all weld beads are receiving rigid examination with the aid of optics. This type of inspection will be revaluated in order to establish more meaningful visual standards and written definitions. Training sessions will be conducted with this new information to assure effective use of the inspection criteria.

Recanning Eliminated

A new procedure has been developed that eliminates the recanning of a uranium billet that is not extruded due to loss of vacuum. The new method provides for removal of the faulty crimp, and soldering on a copper sleeve and an additional short section of an evacuation tube. The billet is then re-evacuated and crimped just below the copper sleeve, leaving sufficient tube length for the vacuum check. The procedure saves the cost of the copper sleeve and the labor of decanning and recanning the billet.

Connector Bolts

Two incidents of failure of a rod connector bolt occurred at the beta heat treating facility. These are the first failures of this type on this equipment. The first resulted from stress failure. The second resulted when a bearing failure caused a secondary stress failure of the connector bolt. Preventive maintenance procedures have been revised to preclude a recurrence of this kind of incident.

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Exhaust Fan Impellers

The impeller on one of four chemical bay exhaust fans broke. The fragments from this impeller knocked out another fan, destroyed two exhaust blowers and cracked the impeller of a third blower. The fourth blower was not damaged. Emergency crews were called in to repair the damage and return the plant to partial operation. They were able to use spare parts and get two exhaust fans back in operation within 24 hours. The third fan was restored later.

Total estimated cost of damage is \$8,800. Details of this incident have been formally documented.

Corrective measures being taken are, (1) replacement of polyvinyl chloride impellers with stainless steel; (2) procurement and installation of blast mats between fans; (3) reduction of the RPM of impellers; and (4) inspection of impellers every 30 days.

CONVERSION

ELECTRICAL INTERCONNECTIONS

A study is being made of the electrical interconnections between the reactor plant and the power generating plant in order 1) to define possible transient electrical loads that might be imposed on the reactor electrical buses, 2) to review adequacy of design, and 3) to determine necessary testing to show functional isolation.

ADDITIONAL SURGE CAPACITY

The functional requirements have been defined for a three-vessel surge tank system necessary to obtain stable response and control during all modes of Phase II operation. The General Electric portion of the work has been completed to the extent of modifying the existing surge tank by adding internal baffles, enlarging the steam admittance line, and installing an orifice in the condensate line.

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HIGH TEMPERATURE TESTING OF ALLOY FUELS

Background of the Program

By 1962, the development of the tube-in-tube element for the first load of fuel in N Reactor had been largely completed. Interest turned to seeking ways to make significant cost reductions by developing new fuel element designs. Use of a single tube fuel element was one of the most promising approaches since it appeared that the cost of cladding could be reduced significantly by this expedient. The greatest uncertainty in the approach arose from the higher fuel operating temperature. The calculated maximum uranium temperature in the single tube element is about 575 C, with 600 C being attainable under peaking conditions. This is substantially higher than the 450 C maximum temperature calculated for the tube-in-tube element. Data from previous tests had indicated that although fuel volume instability was not expected to be a problem in the tube-in-tube element operating under normal conditions, the swelling rate increased rather rapidly at temperatures above about 450 C. Thus the first important step in determining the feasibility of the single tube element was seen to be a determination of the behavior of tubular uranium fuel during irradiation at temperatures up to 600 C in a high pressure system.

Experimental loops in the KE reactor (KER-1 to KER-4) provided the only fuel testing space available at that time. The two large loops (KER-3 and 4) had recently been modified and equipped with process tubes large enough to accept N fuel. However, the proof-testing of full-size N elements in these loops prior to reactor startup was a high priority effort so tests aimed at providing performance data on new fuel designs had to be carried out in the smaller loops (KER-1 and 2), with 2.1" ID process tubes. A new test element was designed, using 1.6 percent enriched uranium, and sized to be compatible with the tube size and the thermal hydraulic capability of the small loops, and to operate at metal temperatures up to 600 C. This element was designated as the KSE-5, using the nomenclatural code in use at the time. Procurement of the enriched uranium and other components was initiated and irradiation scheduled for early 1963.

During this same period the cost advantages of a fuel-target element for N reactor which would produce both weapons-grade plutonium and tritium were becoming clear. The conceptual design which used a tubular driver element and a coaxial target rod was the most attractive approach. Design studies indicated that the driver element would be very similar, if not identical, to the single tube element in size, and the fuel operating temperature range would be the same. However, the goal exposure for Co-Producer driver element was about 30 percent higher (1700 MWD/T) than that of the single tube element (1300 MWD/T). The volume instability of uranium fuel elements was known to increase with metal temperature and fuel exposure, so there was greater uncertainty regarding the feasibility of this design.

During the same period, British investigators published fuel test results

from their power reactor program showing, for the first time, that small alloying additions could significantly decrease the swelling rate of metallic uranium fuel. (Reference: Bellamy, R. G., "The Swelling of Alpha-Uranium under Neutron Irradiation to 0.7% Burnup", Institute of Metals Symposium on "Uranium and Graphite", 1962, Paper No. 8). The particular alloy used by the British contained, nominally, 800 ppm aluminum and 400 ppm iron.

These results were very encouraging as it now appeared that metallic uranium fuel could be used to much higher exposures at high temperature than had previously been believed. Some reservations were necessary, however, regarding the possibility of duplicating this improved performance with material fabricated by U.S. processes. British fuel rods are cast to size and beta-heat-treated. Rapid cooling is achieved in the casting of small diameter fuel elements in contrast with the necessarily slow cooling rate in large diameter ingots. The basic metallurgical structure (i.e. grain size and second phase particle size in particular) of materials prepared in these two ways is different. The size of uranium carbide particles is much finer in the chill-cast material. The iron and aluminum intermetallic compounds can be at least partially redissolved by heat treatment and reprecipitated. The mechanism by which improved performance is obtained with the British composition is not known, so it was not clear whether iron and aluminum second phase particles, or uranium carbide particles, or material in solid solution, or combinations of these effects were responsible. The uncertainty could not be resolved by analysis, so comparative performance tests of standard N uranium (150 ppm iron, 100 ppm silicon) and uranium of the British composition were planned, using the KSE-5 fuel element design as the test vehicle.

The mechanism of volume instability in uranium operating under N reactor temperature and exposure conditions has been uncertain. For a number of years it has been assumed that fission gas bubbles are the cause but experimental proof has been slow in coming, largely because fuel has not been irradiated to produce large volume increases. In addition, metallographic techniques for study of porosity at low swelling values have had to be developed. Several years ago, when a new fuel design was introduced in the Savannah River reactors, large volume increases were observed in portions of the fuel operating at 350-400 C or higher. Large pores of very irregular shape were found in the metal. This new phenomenon was called cavitation or grain boundary tearing and is believed to be caused by the grain boundary shearing that results from anisotropic irradiation growth of the individual uranium grains. Metallographic study of the fuel indicated that fission gas bubbles contributed less than a 1 percent volume increase while grain boundary tearing caused increases up to 10 percent. Because of the different operating conditions in N reactor, it was not known which mechanism would be the cause of volume instability in N fuel. Thus another important objective of the KSE-5 tests was the determination of the mechanism of volume instability.

Test Element Description

The KSE-5 test element was clad in Zircaloy-2 coextrusion. The nominal dimensions of the finished element were 1.790" outer diameter, 0.975" inner diameter, with 30 mil outer cladding and 22 mil inner cladding. The elements

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were cut to 12-inch lengths and closed by brazing and welding Zr-2 end caps into the chemically milled end recesses in the usual manner. Beta-heat-treatment was carried out with the normal temperature cycle--12 minutes at 730 C in a salt bath, quenched in a 300 C salt bath and air-cooled. "Suitcase handle" supports were attached by spot-welding and a 0.592" diameter Zr-2 rod, equipped with "buggy-spring" supports, was inserted into the fuel tube with an interference fit. The purpose of the inert rod was to provide the right water flow split between the two coolant channels.

Three different fuel compositions have been tested: (a) nominally pure uranium (300-700 ppm carbon, no intentional additions), (b) standard N Reactor uranium (150 ppm iron, 100 ppm silicon) and (c) the British composition (800 ppm aluminum, 400 ppm iron). The microstructure of the US-made alloy of the British composition has been characterized as having the usual large uranium carbide particles and a fine dispersion of UAl_2 particles. Although a few intermediate size U_6Fe particles were present, most of the iron and some of the aluminum are still in solution or as particles too small to be resolved by optical microscopy. The hold at high alpha-phase temperatures during irradiation is expected to cause precipitation of U_6Fe and UAl_2 to varying degrees depending on the local temperature. The microstructure of the British-made alloy of the same composition has been studied and, in the beta-heat-treated, alpha-annealed condition, it has been found to contain a uniform dispersion of 5 micron uranium carbide particles, a grain boundary precipitate of U_6Fe , and UAl_2 particles in a fine network structure. The grain size is somewhat smaller than that of standard N uranium. The effect of heat treatment on the KSE-5 elements of British composition is somewhat complicated by the gamma-heat-treatment experienced by the ends of the element during the brazing operation. This treatment leads to formation of a UAl_3 precipitate network that is not altered particularly by the subsequent beta-heat-treatment.

Test Description

Test charges consisted of eight one-foot elements loaded in the center of the process tube. Four charges have been irradiated but one was discharged very early because of loop operating difficulties which prevented operation at high temperature. The aborted test will not be discussed further since no useful data were obtained.

Eight elements made with nominally pure uranium were irradiated in KER-2 during the period March-May, 1963. The fuel power was approximately 30 MW/T, the maximum fuel temperature was about 600 C and the exposure levels ranged from 1070 to 1350 MWD/T. The system pressure was 1600 psi. Fuel element volume changes were measured by the immersion density method using whole elements. One element has been sectioned and examined metallographically.

The other two tests each consisted of eight one-foot elements made with standard N uranium and with uranium of the British composition, charged alternately in the column to provide a direct comparison of the performance of the two fuel compositions. One test ran from April to June 1964, in KER-2, at a power of 27-28 MW/T to an average exposure of 1530 MWD/T. The

system pressure was 1350 psi and the calculated fuel temperatures ranged up to 585 C. Fuel volume changes were determined by immersion weighing (with the equipment and techniques in use at HAP0, accuracies of about 0.1% are believed to be routinely attainable). One element of each composition is currently being examined in the Radiometallurgy Laboratory. The third test was of a mixed charge of eight elements, identical to that of the second test. The test was run in KER-1 from about May 1 to mid-August 1964. The fuel power was 24-25 MW/T, somewhat lower than that of the two previous tests. The average exposure level was 2500 MWD/T and the system pressure, again, was 1350 psi. The fuel temperatures have not yet been calculated but are estimated to be somewhat lower than those of the second test since the tube power values were 5-10% lower. The water flow rate was lower in the third test, and will compensate somewhat for the lower power. The fuel element volume changes have been measured and one element of each composition will be sent to the Radiometallurgy Laboratory shortly for detailed examination.

Results

The objectives of the KSE-5 tests were: (1) to determine the rate of volume increase for standard N uranium and U-800 ppm Al-400 ppm Fe during irradiation at fuel temperatures up to 600 C in a high temperature, pressurized water environment, (2) to determine the mechanism of volume instability in these materials, and (3) to determine what fuel dimensional changes accompany the volume increases. The first objective was to be accomplished by measuring the fuel volumes before and after irradiation as a function of exposure and fuel temperature. The second objective was to be met by metallographic study of the irradiated fuel. The third objective was to be accomplished by visual examination and dimensional measurement of the fuel before and after the irradiation.

The volume change data obtained in these tests, along with individual element exposures and fuel temperatures, are given in the Table and are plotted on the accompanying graph. Fuel element numbers are assigned, starting from the upstream end of the column. The average tube exposure is determined by water flow, temperature rise data and the fuel weight. "Weasel" data are then used to determine the relative exposures (and specific power levels) of individual element power values. Power levels and fuel temperatures are still being calculated for the third test. The current best estimate is that the individual element temperatures are as much as 30 C lower than those for the second test.

The visual appearance of all the elements was excellent. No sign of striations in the outer cladding, buckling of the inner cladding or sharp curvature of the cladding at the cap-to-fuel juncture was noted. Quantitative dimensional change data are not yet available.

The metallographic data are still quite limited; work is still in progress. In the first test, it was observed that grain boundary tearing had occurred near the ends and near the cylindrical surfaces where fuel temperatures were approximately 400 C, in agreement with DuPont observations that grain boundary

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tearing occurs at temperatures of 375-450 C. In the hotter regions of the fuel, fine, crystallographically aligned, spherical porosity was observed. No random spherical pores of the type usually ascribed to fission gas were seen. A quantitative estimate of the relative contribution of each mode was not made.

Elements from the second test are currently being examined metallographically. Observations on the standard N uranium are essentially the same as for the "pure" uranium as described above. That is, both types of porosity were seen but quantitative estimates of the relative amounts have not been made. No porosity of either type has yet been seen in the element of British composition. Fuel elements from the third test have not yet been sectioned.

TEST #1 - PURE URANIUM

<u>Element No.</u>	<u>Exposure</u>	<u>Max. Fuel Temperature</u>	<u>% Volume Increase</u>
1	1068 MWD/T	490 C	0.78
2	1162	525	1.07
3	1256	563	1.72
4	1318	589	2.12
5	1350	606	2.28
6	1318	606	1.95
7	1287	604	1.97
8	1162	573	2.06

TEST #2 - N URANIUM (Odd Nos.) and
BRITISH COMPOSITION (Even Nos.)

1	1240	454	0.42
2	1380	485	0.86
3	1460	507	1.40
4	1565	535	1.18
5	1640	555	2.12
6	1660	568	1.15
7	1670	580	1.95
8	1670	585	1.05

TEST #3 - N URANIUM (Even Nos.) and
BRITISH COMPOSITION (Odd Nos.)

<u>Element</u>	<u>Exposure</u>	<u>Max. Fuel Temperature</u>	<u>% Volume Increase</u>
1	2200	-	0.58
2	2200	-	0.81
3	2375	-	0.75
4	2525	-	1.59
5	2650	-	1.13
6	2675	-	2.22
7	2750	-	1.17
8	2650	-	2.02

Discussion

Because all of the operational data are not yet available and because the examination is still in progress, the conclusions to be reached are still somewhat tentative. An important observation to be made first of all is that the internal consistency of the data is very good. One conclusion is unmistakable. Under the temperature conditions seen by the downstream four elements ($T_{\max}$ 550 C) at both exposure levels in test #2 and #3, the fuel made of the British composition experiences only one-half the volume increase that occurs in the standard N fuel and, similarly, its volume increase was only one-half that of the downstream six elements ($T_{\max}$ 550 C) in the nominally pure uranium test. Under the conditions of the upstream two elements ($T_{\max}$ 500 C) there appears to be little difference in performance between N uranium and the Al-Fe alloy, in confirmation of earlier data that predict similar performance for various uranium alloys at a given temperature to a threshold exposure that is characteristic of each alloy. It is more difficult to make direct comparisons with the "pure" uranium because the test conditions are not the same. That is, the pure uranium fuel ran at higher temperatures to lower exposure but at higher system pressure than the fuel of test #2. (The advantage of using a "striped" charge, as in tests #2 and #3, as an experimental device is obvious. It is, actually, nothing more than the use of an internal control.) Under these conditions the pure uranium suffered roughly 10-30% greater volume increase than did the N uranium. This difference may not be statistically significant considering the limited amount of data available and the differences in operating conditions.

A puzzling aspect of the results is the slight difference in volume increase between fuel elements in the second and third tests. The third test was run to 65% higher exposure than the second and at apparently only slightly lower fuel

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Figure 1

temperatures. Yet, the volume increases are essentially the same for both exposure levels. It is possible that the volume increase versus exposure curve is flat between 1500 and 2500 MWD/T although there is little support for this conclusion from other data. It is also possible that on closer analysis the fuel temperatures will be found to differ by more than is indicated by the preliminary values.

How did the U.S.-made Al-Fe alloy performance compare with that made by the British? This is difficult to answer because the results reported by the British are for small, isothermal samples subjected to little or no system pressure while the KSE5 elements are massive elements in a 1350 psi environment with Zr-2 cladding and with a temperature gradient of about 200 C over a length of 0.2 in. (one-half the tube wall). Bellamy (see paper referred to earlier) reports a volume increase of 2% at an exposure of 2500 MWD/T (0.3% burnup) for fuel temperatures of 550-575 C, and, at 1500 MWD/T, a volume increase of 0.5%. The observed values in the KSE5 tests, however, are about 1% for both exposure levels. With at least two mechanisms for volume instability, a different temperature and exposure dependence for both, a steep temperature gradient in the fuel samples, and probably a few more conditions not now recognized as significant, it should, perhaps, not be surprising that the differences in observed swelling are not readily explainable. It is clear that more experimentation will be required before the importance of the various factors can be assessed.

The limited metallographic data available indicate that both grain boundary tearing and crystallographically-aligned clusters of pores are contributing to the volume increases, but no estimate of the relative importance of each has been made. The former mechanism operates in the cooler surface regions of the uranium and the latter in the hot central region. The minor role played by what are usually taken to be fission gas pores is noteworthy. This is in agreement with existing data, especially that obtained in recent years in the controlled temperature capsule testing program conducted by Hanford Laboratories for the Division of Reactor Development.

Dimensional change data are not yet available so that little can be said about the mode of fuel deformation (i.e. diameter change) that occurs to accommodate the fuel volume increase. However, it is clear that volume increases up to 2.3% lead to no striking localized changes in dimensions. This is supported by other fuel test data for tubular elements of comparable dimensions which have shown that noticeable local deformation does not occur until 3+% volume increase, with severe deformation being present at a 5% volume increase.

There are several interesting implications in these data for the N-Reactor fuel development program. First, if the volume increase plateau between 1500 and 2500 MWD/T is real, and if the maximum fuel temperatures are nearly 600 C for both tests, then the standard fuel should be adequate for the Co-Producer driver application. However, it is equally clear that the Al-Fe addition leads to a significant reduction in fuel instability and therefore provides a greatly increased margin of safety. Of course, conclusions such as these must be considered tentative until all of the data are available.

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The presence of both grain boundary tearing and crystallographically aligned clusters of pores in irradiated fuel indicates that alloys that suppress either mechanism will provide better fuel performance and alloys that suppress both will provide the best performance. Low alloys that performed well in the Savannah River development program should probably behave better than N standard uranium if their ability to reduce grain boundary tearing is also operative under the somewhat different environmental conditions existing in N fuel. If the mechanism causing the clusters of pores is also suppressed by the same alloys, then further improvement is possible.

Summarizing briefly, the significant improvement in performance observed here for the Al-Fe alloy is impressive. The amount of testing performed to date is not enough to constitute proof that this alloy will be adequate for use in the Co-producer driver, but these results do offer great encouragement for further testing of this composition.

TECHNICAL ACTIVITIES

NUCLEAR SAFETY

Uranium Wastes

Management of the waste products from the N-Fuels fabrication process is becoming an increasing problem due to the higher enrichment level of current production. These waste products consist of uranium residues of solid scrap, saw fines and chips, and acid solution. Presently, the uranium solutions are recovered to a shippable form in the AlSi Fuel Production Plant, which is limited by design to uranium enrichments of less than 1 per cent. Special procedures have been developed and downgrading with normal uranium has been proposed to allow processing of limited quantities of 1.25 per cent enriched materials in these facilities. However, this is considered only a temporary expedient.

Nuclear Safety of Spike Fuel

Hanford Laboratories are currently conducting critical mass measurements on 1.25 per cent outer fuel elements. The preliminary results indicate that future relaxation of the nuclear safety specification for this spike enrichment fuel element may be possible. The major advantage of this change would be the possibility of increasing the autoclave loading to a full load.

Zone I Contamination

Low level smearable radioactive contamination is becoming evident in most Zone I locations and protective apparel is now routinely specified for entry into Zone I. Contamination was also expected, and found, on the spent fuel and dummy handling equipment located in the segregation pit and storage areas. Over-all, contamination control has been adequate to avoid personnel contamination and there were no cases reported for September.

Dose rates at work locations have increased as power levels increased, as was expected. Entry into the 109-N cells while the reactor was operating has been necessary on several occasions with dose rates of 1R/hr reported. With the reactor shutdown, dose rates at the rear face have been 3-10 mr/hr, with the front face charging elevator dose rates being 1 mr/hr. Maximum personnel dose rates during September were 1R/hr. No exposures to personnel exceeding control limits have occurred (control is 1 rem per 4 weeks as measured by film badge dosimeter).

Reactor Safety Appraisal

The September 24 presentation to the Hanford Subcommittee of ACRS in Washington, D.C. produced no significant safety questions which would be expected to form a basis for delaying ascension to power. The subcommittee asked that we discuss with the full committee on October 8, 1964 the following subjects:

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1. Resume of operating experience to date.
2. Stuck rod criteria and reactivity margins.
3. Dump transient fuel temperatures, particularly as affected by the primary system injection pumps.
4. The filter cooling system and its possible effect on filtration efficiency.
5. Our proposed criteria for production tests.
(Production tests meeting these criteria would not be expected to be reviewed by ACRS.)
6. The co-producer program and the production tests planned as a part of it.

NEW PRODUCTS

Po-210 Study

A study on Po-210 indicates that 3 kg of polonium could be produced annually at Hanford at a unit cost of \$815/g.

Two schemes for producing Po-210 in N-Reactor were considered. Using the in-reactor volatilization process, N-Reactor could contribute 400 grams of Po-210 at a cost of about \$360/g at time of use. A new case, using thin walled Zr-2 process tubes, aluminum clad solid bismuth targets and a separate water supply in the 68 peripheral channels would permit N-Reactor to produce 535 grams of Po-210 at a cost of \$815/g at time of use (these costs contain no allocated overheads).

Flexibility for Varying Pu-240 Content

The flexibility of N-Reactor to produce plutonium with varying Pu-240 content was reported to the Atomic Energy Commission. Segregating fuel by position in-reactor and inner fuel tubes from outers will permit a wide variety of product mixes, ranging from 1751 kg of weapon grade plutonium, to 745 kg of weapon grade and 985 kg containing 12 per cent Pu-240, to 445 kg of weapon grade and 1130 kg containing 12 per cent Pu-240 during the campaign period to January 1967. Estimates of equipment requirements to achieve segregation of fuel amount to about \$100,000.

Instrumentation for Co-product Production Test

Sketches have been completed showing a method to sample the effluent from any process tube for analysis and detection of coproduct fuel target ruptures. Samples will be obtained from the fuel rupture monitor sample system in the rupture monitor rooms by manipulation of existing and proposed valves.

Thermocouples will be installed in special thermal trains to be inserted in selected process tubes downstream of co-product and tube-in-tube fuel elements.

Target Fuel Development

Two coextrusions of lithium aluminum alloy clad with aluminum were made, in addition to the five coextrusions produced during August. The desired unbonding was assured in the earlier extrusions by spraying the machined lithium aluminum billet with lubricant. This method, however, permitted hydriding of the lithium. In the two September coextrusions a different process provided for spraying lubricant on the inside of the heated aluminum billet prior to insertion of the lithium aluminum billet. From this process, the surface finish and general appearance of the coextrusions were excellent and both of them were fully unbonded. Hydrogen analyses are being performed to determine if there was any hydrogen pickup.

NEW TECHNOLOGIC DEVELOPMENTS

Primary Coolant Water Chemistry

Hydrogen was added to the primary system for a short time at the start of the current 70 percent power level operation to see what effect it would have on oxygen concentration in the coolant. Hydrogen addition was discontinued when it became apparent that at this power level more hydrogen was being formed in the system by radiolysis of the coolant water than was to have been added. During equilibrium 70 percent power level operation the primary system contains 10 to 20 ppb oxygen, 30 to 40 cc/l hydrogen, less than 10 ppb hydrogen peroxide, and 20 to 30 ppb hydrazine. The low oxygen and hydrogen peroxide concentrations probably result from the small, but significant, amount of hydrazine derived from the ammonia used for pH control.

Graphite Cooling Water Chemistry

The graphite cooling system coolant currently contains 30,000 to 40,000 ppb oxygen, 300 to 400 cc/l hydrogen, and 100,000 to 120,000 ppb hydrogen peroxide. Lithium hydroxide is used for pH control. The excessively high concentrations of oxygen and hydrogen peroxide are potentially hazardous because of localized corrosion (pitting or crevice corrosion) of the carbon steel components in the system and because of the collection of a mixture of hydrogen and oxygen in the graphite cooling system surge tank. The surge tank is being modified to reduce those concentrations of hydrogen and oxygen.

Corrosion Testing

A three-month test program has been completed on the effects of ammonia on copper-base and silver alloys. Test results on three valves containing copper alloys externally in the form of bushings and flanges, silver plated retaining rings internally, and Teflon packing around the valve stems, indicate that the copper alloys did not crack in the presence of ammonia vapors. The Teflon packing did not cause a pH increase nor were fluorides leached from this material. The valves will be tested for three more months under simulated primary loop operating conditions of the seats, packing and rings.

Graphite Temperature

Routine graphite temperature maps have been obtained at equilibrium power level for the various steps in the power ascension program. Extrapolation of these temperatures to design power level indicates that the maximum graphite temperature may be as much as 150 to 200 C below that predicted during design of N-Reactor. Because the thermal conductivity of graphite is known to decrease with accumulated neutron exposure, this extrapolation may be optimistic, since the total N-Reactor exposure experience to date is rather limited. These analyses will be continued as the power level is raised, however, to verify the temperature values and to explore the effects of a lower graphite temperature on process tubes on nuclear safety, on possible increases in power level above design, and on replacement of helium in the reactor atmosphere with less costly nitrogen.

Fuel Element Spacers

Pressure drop tests were performed on fuel element spacers, using two types of loading. The first was made up of two short and three long spacers and the second of one long and one short spacer and a newly designed Mark I spacer rod. The pressure drop across both types of loadings was essentially the same.

Control Rod Enhancement

Future fuel loadings for N-Reactor may require an increase in the worth of the control system. The present control rod system is being reviewed to determine a method of enhancement that does not require supplemental control.

Increasing the optical thickness of the present rods by increasing their physical thickness or macroscopic cross section by a factor of 1.5 or 5.0 would yield approximately 6 or 20 percent increases in reactivity worth, respectively. The smaller increase could be approached by maximizing the BCl_4 density, the larger increase would require 100 percent enriched $B^{10}Cl_4$, or, an equivalent such as B^{10} -steel.

Further increases in reactivity strength would require increases in "rubber band" circumference of the control element. Another 15 to 20 percent increase may be achieved by going to a wider rod (flat cylinder). Thus the total strength might be increased to 1.4 times the present value.

Scarce materials such as hafnium could be used, but boron mixtures can provide equivalent strength.

N-Reactor Data System

The installation of the data transmission and reception equipment at N-Area and near the computer has been delayed by the unavailability of the data phones. It was recently found that the phones would not be needed if the transmitters were connected directly by telephone lines used exclusively for data transmission. Consequently, a line was made available and the equipment was connected. Two test-tapes were successfully transmitted and the same data returned to N-Area with no errors of transmission.

The 7090 computer program to calculate each of the process tube powers and tube factors with paper tape input has been completely debugged and will be used for further check-out of the data transmission equipment.

The date for full use of this system for the transmission of operating data depends on NRD success in the acquaintance of "scramblers" which will encode the information for security protection during transmission.

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Filter Cooling Tests

Testing of the stack filter cooling system has been completed. Well-defined and fairly consistent heating and cooling cycles were measured. Temperature and heat-flow data were obtained which will permit setting the control ranges at the proper levels to prevent damage to the particulate and halogen filters from heating caused by entrapped radioactive material.

ADMINISTRATIVE SUMMARY
SEPTEMBER, 1964

EMPLOYMENT

Section	Permanent		Temporary		Total		Change	
	E	NE	E	NE	E	NE	E	NE
General	1	1	-0-	-0-	1	1	-0-	-0-
Finance	15	13	-0-	-0-	15	13	-0-	-0-
Fuels	44	99	5	-0-	49	99	-3	+1
Plant	67	217	3	-0-	70	217	-1	-2
Project	67	20	-0-	-0-	67	20	-2	-2
Research & Engineering	47	8	-0-	-0-	47	8	-1	-1
Employee Relations	<u>5</u>	<u>2</u>	<u>-0-</u>	<u>-0-</u>	<u>5</u>	<u>2</u>	<u>--</u>	<u>--</u>
	246	360	8	--	254	360	-7	-4
Total employees						<u>614</u>		
Net change								<u>-11</u>

SAFETY

Days without a disabling injury	670
Hours worked without a disabling injury	2,154,159

VISITORS

<u>Name</u>	<u>Company</u>	<u>Contact(s)</u>	<u>Date(s)</u>	<u>Purpose</u>
J. Heald	Heald Machining Co. Seattle, Wn.	D.C. Lehfeldd N.W. McKay	8/25/64	Review engineering equipment specification
J. Mohandice	Hallidie Machining Co. Seattle, Wn.	D.C. Lehfeldd N.W. McKay	8/25/64	Review engineering equipment specification
J.M. Henderson	Bailey Meter Co. Cleveland, Ohio	R.D. Crosier G.R. Klingler	9/3-4/64	Discuss steam generator level calibration problems
O. Wienola J.B. Bloomfield	AMF Atomics		9/25/64	Discuss problems related to AMF supplied irradiated fuel handling equipment
T. Ferris	U.S. Public Health Service	J.W. Vanderbeek	9/9/64	Discuss sanitary waste disposal practices
J.A. Ghormley	ORNL Oak Ridge, Tenn.	W.K. Kratzer	9/28-29/ 64	Discuss radiolytic decomposition of water
E.L. Knoedler	Sheppard T. Powell and Associates Baltimore, Md.		9/5-6/64	Consult on water treatment and corrosion problems
J.F. Schiltz G.K. Stearns	National Lead of Ohio Cincinnati, Ohio	J.W. Nickolaus T.D. Naylor R.H. Scanlon	9/8-9/64	N-fuels uranium, primary extrusion
R.L. Bean J.F. Puterbaugh E.J. Carozza	Reactive Metals, Inc. Ashtabula, Ohio	J.W. Nickolaus T.D. Naylor R.H. Scanlon	9/8-9/64	N-fuels uranium, primary extrusion
T. Bryant	Brush Beryllium Cleveland, Ohio	H.P. Kraemer L.H. Steves	9/29/64	Discuss quality problems regarding braze rings
C. Hoehne J. Driver	Bridgeport Brass Co. Riverside, California	L.H. Steves H.P. Kraemer	9/29/64	Discuss quality problems

VISITS

Name	Company	Contact(s)	Date(s)	Purpose
J.W. Vanderbeek	Vallecitos Atomic Lab., GE, Pleasanton, Calif.	D.L. Hessard	8/26/64	Discuss employment
M.C. Leverett	Space Sciences Lab., GE Valley Forge, Penn.	Dr. Leo Steg	9/14/64	Plant orientation
M.C. Leverett	Engineering Managers Mtg. GE, Haddonfield, N.J.	F.K. McCune	9/15-16/64	Meeting
R.L. Dickeman M.C. Leverett R.E. Hall W.S. Nechodom J.W. Riches R.H. Shoemaker R.E. Trumble	Advisory Committee on Reactor Safeguards (ACRS) Hanford Subcommittee Meeting, Washington, D.C.	E.F. Miller	9/24/64	N-Reactor program
M.C. Leverett	Nuclear Standards Board, ASA, New York City, N.Y.	F.K. McCune	9/25/64	Board meeting
D.H. Curtiss	Canadian GE, Peterboro, Ontario	Doug Boxall	9/29/64	Discuss Zr technology
D.H. Curtiss	AECL, Chalk River, Ontario, Canada	Winsor Evans	9/30/64 10/1/64	Discuss Zr technology
R.C. Hoffman	General Electric Seattle, Wash.		9/14/64	Re: Progress in DC Application of Power Semiconductors
G.L. Swezea	Convention-Wescon Los Angeles, Calif.		9/25-27/64	Attend IEEE Section Chairmen and West Coast Committee Representative
G.L. Swezea	Barton Instrument Co. Los Angeles, Calif.		9/28/64	Review pressure and flow transducer design
J.P. Sluka	Anchor Equipment Co. San Francisco, Calif.		9/11/64	Re: Replacement stems for PCSV-203 valves

SIGNIFICANT REPORTS

HW Number	Title	Author	Date
HW-83716 Uncl	Analysis of Primary Loop Emergency Cooling Test Data Obtained During Startup Testing of N-Reactor	H.G. Johnson	September 1, 1964
HW-83237 Conf	Ball Safety System Reliability	E.E. Leitz	July 13, 1964
HW-84154 Secret	Co-Product Operation of N-Reactor	E.G. Pierick J.W. Riches	September 21, 1964
HW-83397 Conf	Design Analysis of 117-N Building Filter Cooling System	D.D. Stepnewski	September 24, 1964
HW-84194 Uncl	Engineering Audit of N-Reactor's Emergency Cooling Water System	H.G. Johnson	September 22, 1964
HW-83219 Conf	Evaluation of Plant Performance, Horizontal Rod Scram	E.E. Leitz	July 17, 1964
HW-84119 Conf	Fabrication of Co-Producer Driver Fuel Elements (1.25 Enrichment) - Production Test NR-8	W.A. Hendricksen	September 14, 1964
HW-83894 Conf	Summary, Hazards Analyses Requiring Emergency Raw Water System	J. Muraoka	September 25, 1964
HW-83790 Uncl	Hazards Associated with Ion Exchange Resins in N-Reactor	W.K. Kratzer	August 25, 1964
HW-83887 Conf	Hazards Review - N-Reactor 1.95% Co-Producer Fuel Element Test	J.L. Benson W.S. Nechodom	September 8, 1964
HW-84195 Conf	N-Fuels Engineering Report on Fuel Failure 3048-16	R.M. Horton	September 18, 1964
HW-83943 Secret	N-Reactor Flexibility with Respect to Pu-240 Production	R.L. Dickeman	September 4, 1964
HW-83935 Secret	N-Reactor Graphite Program for FY-1965	D.H. Curtiss	August 21, 1964

SIGNIFICANT REPORTS (Continued)

HW Number	Title	Author	Date
HW-83367 Conf	N-Reactor Shutdown for Fuel Rupture Indications, Tube 3361	D.L. Renberger	July 31, 1964
HW-83922 Secret	NPR Fuel Element Spacer Mark I Pressure Drop	M. Pociluyko	September 9, 1964
HW-83200 SUP1 Conf.	Optimization of Primary Coolant System Operational Modes	R.E. Hall, et al	August 28, 1964
HW-83726 Conf	Processing Data for Co-Producer (1.25) Fuel Elements Charged in KER 3, PT-645-D	W.A. Hendricksen	September 15, 1964
HW-84196 Conf	Reclaiming of Inner Fuels with Cracked Supports	W.G. Hudson	September 18, 1964
HW-84106 Uncl	The Technology of Thoria and Thoria-Uranium Compositions	T.W. Evans	September 18, 1964
HW-83825 Conf	Temperature Distribution in the NPR Graphite Moderator	D.W. Pashniak	August 27, 1964
HW-83940 Secret	Three Kilogram Polonium Production Study	L.E. Kusler H.C. Rathvon J.M. Bird	September 4, 1964

INVENTIONS AND DISCOVERIES

HW-84188, Invention Report - Ground Detector for Underground Electric Power Systems, by J. L. Rand and J. R. Bolliger.

SUGGESTION PLAN PARTICIPATION

	<u>SEPTEMBER</u>	<u>CY 1964 TO DATE</u>
Number of Eligible Employees	362	364
Number of Suggestions Received	65	448
Number of Suggestions Acted Upon	51	438
Number of Suggestions Adopted	13	169
Net Annual Savings	\$6,113	\$133,803
Amount of Awards	\$ 700	\$ 7,470
Per cent of Awards to Savings	11.5	5.6
Average Amount of Awards	\$ 58.85	44.20

SUGGESTIONS ADOPTED/1000 EMPLOYEES/YEAR

1964

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**DATE
FILMED
01/20/93**