[redacted]

HW--80559-5

DE93 003422

DECLASSIFIED

N-REACTOR DEPARTMENT

MONTHLY REPORT

MAY

1964

[redacted]

HANFORD ATOMIC PRODUCTS OPERATION

RICHLAND, WASHINGTON

GENERAL ELECTRIC

DECLASSIFIED

[redacted]

[redacted]

HW-80559 5

This document consists of
53 pages, copy no. [redacted]
of [redacted] copies, Series [redacted]

June 8, 1964

HANFORD

88656

Classification Cancelled And Changed To
DECLASSIFIED
By Authority of CG-PR-2
R. Sten - 8-26-92
By J. Wells - 8-31-92
Verified By PM Eck 9-2-92

DECLASSIFIED

N-REACTOR DEPARTMENT
MONTHLY REPORT - MAY, 1964

R.L. Dickeman
General Manager

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

[redacted]

[redacted]

Classification reviewed for de
cla...ion but left unch...
by ...

[redacted]

[redacted]

-1-

DECLASSIFIED

DISTRIBUTION

Copy Number	
1	W. E. Johnson
2	R. L. Dickeman
3	A. B. Greninger
4	H. M. Parker
5 ✓	H. D. Tibbals
6, 7, 8, 9, 10, 11	J. E. Travis, RLOO-AEC
12, 13	E. P. Baranowski Director - Div. of Production - Washington, D.C.
14, 15	F. E. Crever Savannah River Operations Office
16	M. C. Leverett
17	L. M. Loeb
18	W. M. Mathis
19	J. S. McMahon
20	J. Milne
21	W. A. Shanks
22, 23, 24	Extra Copies
25	300 Files
26	Records Service Center

ROUTE

Name	Payroll Number	Location	Files Route Date	Signature & Date
J. E. Travis		703		

DECLASSIFIED

[redacted]

TABLE OF CONTENTS

[redacted]

[redacted]

OPERATING SUMMARY

Input Production Pu-KMWD	1.863				
Input Production Tritium Equivalent KMWD	0				
Power Level MW (Max.)	400				
Power Level MW (Avg.)	Not applicable - N-2 testing				
Steam Produced (Standby boiler)	350,000 #/hr. Avg. while operating (10 days)				
Steam Availability (steam dumped)	Not applicable - N-2 testing				
TOE (%)	Not applicable - N-2 testing				
Time Utilization (hours)	Not applicable - N-2 testing				
Scrams	6				
Fuel Balance	.094	1.25	Coprod	Other	Total
Fuel Charged - tons	0	0	0	0	0
Fuel Discharged	0	0	0	0	0
Net change	0	0	0	0	0
Total in reactor at month's end	361.5	0	0	0	361.5
Metallic Uranium Received	77.6 Tons				
Acceptable Fuel Elements Produced	1664				
Scrap Shipped				30,290 lbs.	
Initial Inventory			94.4 (Tons)	5,104 Assemblies	
Final Inventory			128.2 (Tons)	6,778 Assemblies	
He Consumed M Cu. Ft.	106.6				
Fuel Oil Consumed in M Gals	929				
Demineralized Water Produced Gals/Min. Average	1135				

[redacted]

SUMMARY

The principal development of the month is the successful completion of the N-2 low level testing at 400 MW. Testing started on April 23 and concluded three weeks later on May 10. Plant performance was excellent. Those systems that could be significantly tested at this level showed their ability to fulfill design criteria. Reactor power level was very steady, requiring very little adjustment of the control rods to hold constant power. During planned transients the reactor was easily controlled by movement of the rods. The primary and secondary systems performed well. The secondary is quite sensitive to fluctuations in level and pressure, particularly with respect to control of the surge tank level. Individual process tubes were satisfactorily put on diversion from the recirculation mode and then returned. Diversion flows were cooled adequately by the recuperative heat exchangers. The plant service boiler demonstrated sufficient capacity and has responded to acceleration ramps in excess of design values although none of these ramps has yet been attempted from a banked condition. The demineralization plant produced good quality water, with a minimum of control problems.

Although the plant is complex, it has proved satisfactorily controllable during low level testing. There were few unplanned failures or transfers of critical components, and those few that did occur did not result in upsets in systems. Control of system parameters was demonstrated in both automatic and manual operating modes.

The N-2 tests went smoothly and well within the time allotment. The system responded more slowly to disturbances than had been predicted. The slow response is probably due to the present good conductivity of the graphite and the relatively

large expansion volume in the secondary loop. After higher power operation for some time it is expected that the response to disturbances will be close to original predictions, since irradiation will decrease the graphite conductivity and high power will reduce the relative expansion volume. The plant was shut down as planned, for maintenance, modifications, repairs and preparation for N-3.

During startup a number of arrays of control rods--uniform, peripheral, central, end column and single column were studied. Two group diffusion theory of the space-energy distributions of neutrons is being applied in an attempt to find a single theoretical model that will represent all conditions of control, however complex they may be.

The first panel of tubes was placed in steam generator 4A during the week of May 18. Of the 110 panels required to complete the retubing, 27 have been received and 28 more are expected the first week in June. These 55 panels consist of 958 tubes.

The enrichment for the co-producer element irradiation test has been defined as 1.95 percent U^{235}, and the technical criteria have been determined. The test will determine the irradiation performance of 20 lb/ft prototype co-product elements. The enrichment is typical of that which would be used in a full production load.

The FLEX 2 computer study on temperature coefficients of N-Reactor spike fuel vs. the standard fuel shows that they are the same. Water and graphite coefficients of the spike fuel are more negative for exposure up to at least four months at full power, than those of the standard fuel. These two coefficients become more positive with exposure for both fuels, but the effect is smaller for the spike case than for the standard. The study also re-confirmed that either the rod system or the ball system could shut down the reactor in any credible case.

The behavior of thorium fuel spiked with highly enriched uranium is being studied for possible use in the power-only operation of N-Reactor. A modified MOFDA computer program plotting enrichment vs. reactivity indicates that an enrichment of 2.3 percent by weight of U^{235} would be required.

A forged "T" section outer fuel support shows improved stability over the current support with respect to deformation.

The initial 20 extrusions of 1.25 percent enriched thick walled outer tubes for the co-producer program were completed. The Li-Al cores being fabricated by Hanford Laboratories will be completed in June and the finished assemblies available for reactor charging by the first of August. Assembly of 35 percent of the second reactor load of 0.947 percent fuel elements has been completed.

A technical review has been made of the on-power charging machine developed for use on a CANDU-type reactor. Although it is technically feasible to add this on-power machine to N-Reactor, the capital investment does not appear economically justified in view of the long pay-off period.

A reactor concept for desalination or power generation, or both, has been developed that utilizes a modified NPR-type reactor with a higher specific power. Several improvements in economics are indicated over a recent previous design, with energy costs dropping from 16 to 10 cents per million BTU's.

PLANT STATUS AND BEHAVIOR

N-2 Tests

The plant was engaged for the first 10 days of May in performance of the N-2 tests, low level nuclear operation. The remainder of the month was occupied in performing subcritical tests, repairs, and modifications in preparation for power ascension operation.

Plant performance during the low level testing at 400 MW was excellent. Plant systems under test have largely demonstrated the ability to fulfill design criteria.

The nuclear reactor was operated up to 400 MW and performed very well. Heat flux was extremely stable at the low power levels and a check of power coefficients confirmed the values predicted in the Hazards Report. Heat transfer between primary coolant and graphite moderator was somewhat more rapid than calculated, resulting in slower cool down of the primary system following a scram. Control rod movement has been responsive enough to control any transients experienced during the testing. However, hydraulic control rod drives have shown some difficulty in the control of multiple-group rod movements at slow speeds in a smooth, uniform manner. The hydraulic drives have also required more than expected maintenance.

The primary system has given little difficulty. Some slight valve leakage has been experienced but was easily corrected. There has been some instability in the injection water system, attributable in part to vane movement in the hydraulic drive coupling of the injection water pumps. The system tends to hunt when the control signal is constant. It has been observed that, with no mixing header on the reactor inlet piping, there is some preferential distribution of flow through the inlet headers and process tubes.

N-2 Tests (cont'd)

Combined with equipment errors in process tube flow measurement, this has made setting of the process tube flow monitor safety trips difficult. The problem can be alleviated in part by adjustment of pump speeds.

The secondary system has been only partly utilized because insufficient quantities of steam have been produced for full-system testing. The system's ability to accept steam and condense it in a controlled manner has been satisfactorily demonstrated; and tests on individual units have confirmed designed capabilities for heat dissipation. The system is quite sensitive to fluctuations in level and pressure, particularly with respect to surge tank control. Surge tank capacity appears to be less than optimum.

The plant service boiler has demonstrated sufficient capacity, and once the light-off of the main burners has been established, the boiler has responded satisfactorily to acceleration ramps considerably in excess of designed values. There are still some adjustments to be made to assure automatic light-off and eliminate flaming out as additional light-off burners are lighted. Boiler efficiency is not yet as high as predicted. Additional load tests are being conducted.

The demineralization plant has produced good quality water and has demonstrated an adequate capacity with a minimum of control problems. There is, however, a weakness in the mechanical design of the resin strainer supports which permits them to buckle and lose the ability to retain resin.

One point should be emphasized. Whereas there has been some opinion that the plant might be too complex to be controlled, operation during the low level testing has revealed otherwise.

N-2 Tests (cont'd)

There were few unplanned failures or transfers of critical components, and those that did occur did not result in system upsets. Although the operating crews were augmented by additional engineering support for the plant startup, performance of both supervisors and operators exceeded expectations. Control of system parameters was demonstrated in automatic and manual operating modes and both operator and control responses were adequate to manage the plant under the test conditions.

The operating experience and principal test steps during the N-2 test execution are portrayed graphically in Figure 1. There were 12 unplanned scrams during the test period, including two full ball safety system drops. In every instance the safety circuits and auxiliary systems functioned correctly. There were no significant upsets in the secondary steam-condensate system as the result of these scrams. A separate report is being issued summarizing the N-2 test and results.

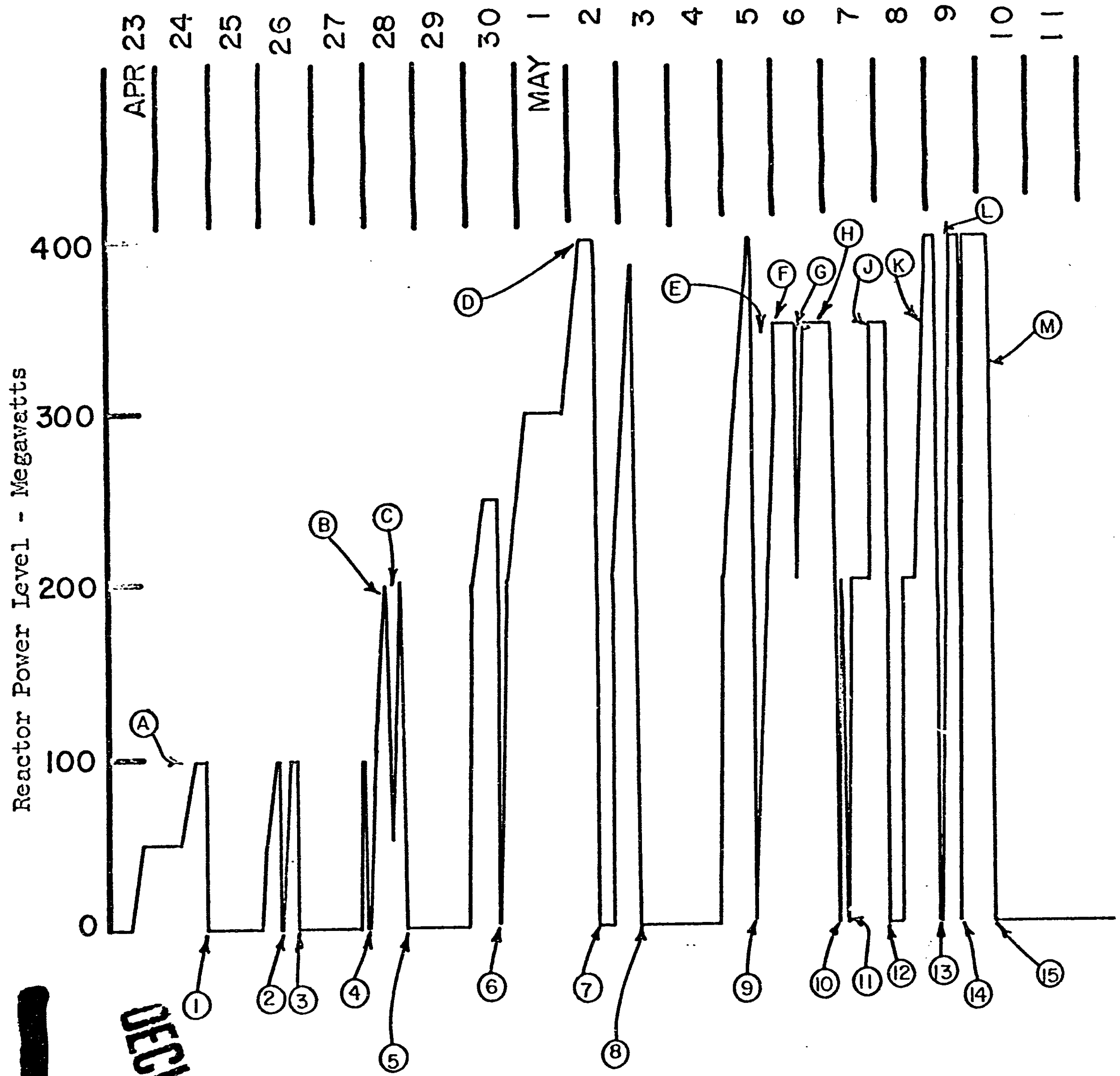
Both the gross gamma and gamma energy fuel rupture monitors were placed in service and tested during the nuclear operating period. Sample flow and pressure controls were adjusted and both of the systems gave indications of being able to detect fuel ruptures.

The process tube diversion capability was tested during the nuclear operating period; the ability to remove individual process tubes from the recirculation mode and to return was satisfactorily demonstrated. Diversion flows were cooled adequately by the recuperative heat exchangers.

At the conclusion of the N-2 tests the plant was shut down and a broad program of repair and modification initiated to prepare for the power ascension operation. This program will include the

CONFIDENTIAL

[redacted]



SEQUENCE OF TEST

A. STARTUP TO 100 & DATA Step 9 & 10
B. INCREASE TO 200 & DATA Step 11 & 12
C. SETBACK TO 50 Step 13
D. INCREASE TO 400 & DATA Step 14 & 15
E. SCRAM FROM B BUS TRIP Step 20
F. STEAM PRESSURE INCREASE Step 22
G. SETBACK TO 200 AND CELL ISOLATION Step 18 & 24
H. CONSTANT STEAM PRESS. CONTROL - 3 CELLS Step 25
FULL LOAD DUMP CONDENSER Step 31
J. DIVERSION SYSTEM Step 32
K. SKEWING & POWER VARIATION Step 35
L. DATA AT 400 Step 37
M. BALL SAFETY SCRAM Step 38

CONTINUITY OF TEST

1 SCRAM HI AVG. FLUX CALIBRATION ERROR.
2 SCRAM LO STM. GEN. LEVEL OPER. ERROR
3 SCRAM LO FLOW TRIP CLOSE SETTING
4 SCRAM FAULTY FLOW MONITOR SELF TEST (manual)
5 BALL DROP FAULTY WIRING TO FLUX MONITORS
6 SCRAM HI FLOW TRIP. CLOSE SETTING. Step 14 DATA
7 SCRAM LO FLOW TRIP CLOSE SETTING. Step 19 DATA
8 BALL DROP FROM DC POWER SWITCHING ERROR
9 SCRAM B BUS TRIP OFF Step 20
10 SCRAM SURGE TANK LOW LEVEL — VALVE FAILURE
11 SCRAM LOSS OF TG VACUUM (manual)
12 SCRAM (manual) PRE-STEP 38 MODIFICATIONS
13 SCRAM BOILER FLAMEOUT (manual)
14 SCRAM LO FLOW FROM TG STM PRESS. LIMITER
15 BALL DROP Step 38

STARTUP TEST N-2 - LOW POWER TESTING PROGRAM

5/11/64
E. A. Grimm

[redacted]

DECLASSIFIED

[redacted]

N-2 Tests (cont'd)

repair or calibration of numerous valves and instruments, replacement of defective components, correction of flow irregularities, cleaning of condensers, tanks and strainers, correction of system leaks, and modifications to plant equipment and controls.

Operational Testing

Operational testing has been resumed, with the principal effort being applied toward testing of the confinement system, the circulating raw water system, the standby boiler, the 184 turbine-generator, and the water quality sampling system. In addition, corrections are being made to exceptions on completed tests.

Overall status of testing progress may be summarized as follows:

Total number of tests scheduled 39(1)

	This Month	Total
Scoped	-	39
Issued for Comment	-	39
Issued as Approved	(2)	39
Tests Started	-	39
Tests Completed	2	30 (3)
Final Approval of Test Data and Results	13	37

	Preparation		Performance	
	Scheduled	Actual	Scheduled	Actual
This Month	100.0%	100.0%	100.0%	95.0%
Last Month	100.0%	99.0%	97.8%	82.0%
Net Change	--	1.0%	2.2%	13.0%

(1) Total includes N-1 and N-2 tests.
(2) N-2 test consists of 38 steps which were prepared and approved individually. All of these steps were approved by month's end.
(3) Eleven of the 37 total are interim approvals for N-2.

[redacted]

Primary Loop Components

Steady state running of the NPR-PCE loop in testing of primary loop components has attained a total of 199 days. Two weeks of loop time was spent in thermal cycle testing of "H" and other special type fittings. Testing of the pressure relief valve arrangement for the primary loop was resumed following delayed delivery of special components.

Relay Testing

The life testing of relays in search of an optimum for application in the horizontal rod control system has continued, with a total accumulation of 385,000 operations to date. Two Allen Bradley, two General Electric, three Ebert mercury type and one Durakool mercury type are under cyclic test at maximum inductive loads.

EQUIPMENT MAINTENANCE AND REPAIR

The number of concurrent efforts undertaken during this shutdown make it impractical to cover all of them in detail. In total, some 2300 individual scheduled and 900 unscheduled maintenance jobs were undertaken while daily support was given to programs such as final annunciator check out and vendor craft assistance. Specific items worthy of particular note follow:

Revisions to the remaining twelve dump condensers to preclude water hammer have been completed.

The temporary chamber in use in one of the three intermediate range flux monitor locations has been replaced by a new permanent chamber.

The vendor repairs to the governor of the turbo-generator turbine have corrected the previous problems with this equipment.

The 66-inch discharge valves of the large river pumps would not close in a consistent time during the early test period. Valve repairs have been made and an engineered modification to obtain uniform valve actuation time required for high level operation is in progress. Preliminary tests on two of the four units completed gave consistent and acceptable closing times.

Complete functional tests have been run on the horizontal control rods and all irregularities corrected.

Previous tests of the control power relays revealed that the secondary system control cabinets would not sustain a transfer to an alternate power source without tripping some of the control segments. Breakers in the cabinets have been changed to thermal overload type devices and satisfactory transfer tests performed.

The throttle and trip valves have been modified on all four primary drive turbines to improve their proper slow-down response upon a scram initiated signal.

The derated overload heaters for the main lube pump motors of the primary drive units have been replaced to preclude their intermittent trip out behavior.

The steam pressure control valves immediately down stream from the boiler drum, which again malfunctioned and sustained mechanical damage to their internals, are being repaired.

The automatic analysis features of the water quality control facilities are being actuated. They will be tested, concurrent with plant startup. Manual analysis will be continued until automatic equipment is completely functional.

Control Valves

There are 168 control valves in the secondary system of the same type as the HPV-209's. It has been found that approximately 40 of these valves have become damaged as a result of ruptured thimble-housings over the position indication magnets. The subsequent leakage water soaked the position indication switches and wiring and necessitated repairs and replacements. The leaky thimble tubes were repaired by silver brazing.

Dump Condenser Fill Valves

Tests show that successful operation of the HPV-209 dump condenser fill valves requires external application of about 25 pounds of steam pressure; all valves will be fitted with a supplemental steam supply that will provide this necessary operating pressure.

Primary Loop Vent Valves

The numerous vent valves used for filling the primary loop are closed by action of a float actuated magnet when the loop is full. Failures have occurred in this mechanism in recent months because the small diameter rod that connects the float to the operating linkage apparently breaks under impact loading. On damaged units, mechanisms have been modified by installing larger diameter rods by a specially developed welding process.

Primary Coolant Valves

Controls for the nitrogen powered actuators of the emergency dump valves have been modified to eliminate gas leakage. To-date 80 out of 88 solenoid control valves have been reworked at the factory to provide leak-tight seats and have been re-installed.

Valves and Pipe Supports - General

Vibration problems have been encountered with the pressure reducing valves HPV-107 and -108 in the 184 boiler house. The valves have been dismantled, repaired--including some body welds--and are again in service.

The two diversion system back pressure control valves have been found to vibrate during operation. The cause is two-fold: inadequate pipe bracing, and operation at flow too close to shut-off. The vibration has caused some damage to the control linkage. Additional braces have been installed to stabilize the valves.

Emergency Cooling Water Flow Switches

Replacement flow indicating switches have been ordered to provide a faster trip response time if a loss of primary coolant flow requires initiation of emergency cooling.

Shield Doors

Control circuits for the nine rear face shield doors which are interlocked in the charging circuit were recently modified to provide fail-safe control. New contacts for the latch-in switches involved have been installed on eight of the nine units involved.

Expansion Joint EJ-5 FL (LP) Line

Repairs have been almost completed on the failed expansion joint and its anchor. A newly designed anchor has been fabricated and is being installed. A new bellows is being welded into position.

FUEL MANUFACTURE

PRODUCTION SUMMARY

Statistics

1. Input (extrusions)

		Current Month	CYTD
Outer billets -	.947% Enriched	47	465
	1.25% Enriched (driver)	37	37
	1.25% Enriched (spike)	50	50
Inner billets -	.947% Enriched	126	515
Total Extrusions		260	1067
Total Tons		46.2	191.7
% of Forecast		89	88

2. Output (finished production)

		Current Month	CYTD
Assemblies -	.947% Enriched	1664	7748
	1.25% Enriched (driver)	0	0
	1.25% Enriched (spike)	0	0
Total Assemblies		1664	7748
Total Tons		30.3	145.1
% of Forecast		87	39

3. Month-end Inventory

Finished Inventory

		300 Area	100-N Area	Total
Assemblies -	.947% Enriched	6624	154	6778
	1.25% Enriched (driver)	0	0	0
	1.25% Enriched (spike)	0	0	0
Total Assemblies		6624	154	6778
Total Tons		122.6	2.1	124.7

Bare Inventory

Outer billets -	.947% Enriched	415
	1.25% Enriched	61
Inner billets -	.947% Enriched	188
Tons -	.947% Enriched	115.9
	1.25% Enriched	12.3
Total Tons		128.2

DECLASSIFIED

[redacted]

		Current Month	CYTD
4.	**Scrap Shipments**		
	Pounds shipped	30,290	149,033
5.	**Uranium Utilization**		
	Outers		67.4
	Inners		76.8
	Total	70.9	71.3
6.	**Shop Yield (April)**		
	Outers	89.8%	82.6%
	Inners	83.3%	85.2%
	Total	87.2%	83.5%

Shop activity is being directed almost entirely to the production of a 45-49 reactor tube co-producer element test and a 160 tube spike enrichment charge. Inner billet extrusions were increased during May to provide the twenty-six inch length inner tubes for the 1.25 percent enriched outer tubes of the spike enrichment program. The outer billets of 1.25 percent enrichment are also being extruded. As a result of the increase in inner extrusions, total tonnage for the month is approximately 10 percent below forecast.

The initial 20 extrusions of 1.25 percent enriched thick walled outer driver tubes for the co-producer program were completed. The lithium-aluminum cores being fabricated by Hanford Laboratories will be completed early in June. Finished assemblies will be available for reactor charging by August 1.

The overall billet inventory is at a satisfactory level, but a significant imbalance exists between inner and outer billets. The high priority assigned to the production of 1.25 percent outer billets at the feed site will not permit correction of this imbalance for at least three months. This situation reduces the flexibility of scheduling.

All off-site inspections of essential materials are being discontinued at a saving of about $21,000 per year. Local receiving inspection will continue.

The billet enrichment test has been running satisfactorily since the first of April. All uranium used in production extrusions, plus the special components being used for the driver tubes has been tested. All uranium has shown the proper count differences for its enrichment. A design test is presently being run to further evaluate the test capability using natural, .947, and 1.25 percent enrichments. At the conclusion of these tests, the tester will be put into production use.

Physical Tests of Uranium Billet Materials

A specification for hot hardness testing of uranium billets at the Feed Materials Production Center has been prepared and transmitted to the AEC. Hot hardness measurements will be obtained at 300, 400, 500, and 630 degress C, on samples from 50 production ingots after primary extrusion. If meaningful differences can be found in the hot hardness between and within ingots, it is intended to incorporate hot hardness testing as part of the routine specifications on a sampling basis.

Tipped Mandrel Primary Extrusions

A billet fabrication process based on primary extrusion of large outside and inside diameter ingots over tipped mandrels is being developed. The thin wall ingot design should result in a more effective beta heat treatment before primary extrusion and the extrusion over the tipped mandrel should result in more alpha working of the ingot structure, particularly on the I.D. surface. The purpose of this program is to more effectively obliterate the original uranium grain domains in the cast ingots that are believed to be the principal cause of clad thickness variation, particularly on the I.D. of the N-outer tube.

Surface Repairs on Extrusion Billets

Pits in billets have been drilled to a maximum depth of only 0.087 inch and with a wide angle of 120°. After extrusion there were no unbonds and the cladding thickness variation was below the detectable limit of about 0.003 inch. A specification change will be initiated to incorporate reconditioning of billets at NLO. To prevent acceptance of billets produced during out-of-control casting periods, the specification will limit the number of pits that may be reconditioned on a billet and the number of billets that may be reconditioned in any one time period to the numbers that are encountered during normal control periods.

TIG Brazed Closures

The TIG-braze process for unbonded end closures requires a beryllium-zircaloy wire preform. Hanford Laboratories is working on improved processes for fabricating this preform. One process involves canning the beryllium-zircaloy alloy in steel, then swaging and drawing at elevated temperatures to produce 25 mil diameter wire. Approximately 70 feet of acceptable wire has been fabricated by this process, and it is estimated that an eventual fabrication cost of about \$.75-1.00 per foot is attainable. Another process involves inserting several small beryllium-zircaloy alloy rods in steel or copper billet and fabricating the wire by the honeycomb extrusion process. This should cost less than \$.75 a foot.

Extrusion Bonding

Recent evaluations of unbond have shown that failure of the copper weld during the preheat cycle and/or during the extrusion upset is a primary contributor to unbonding. Several methods of increasing copper weld strength are being evaluated.

Results of weld strength tests indicate that modifications in mechanical design can be used to strengthen the weld in both tension and compression. The testing techniques and fixtures were designed to approximate extrusion conditions.

Reduction of End Defect Losses

A revised schedule for cutting after extrusion has been established. With this new cutting schedule, an average of only ten inches of good inner tube and six inches of good outer tube will be lost from each extrusion. The revised outer tube cutting schedule represents a 4% increase in material utilization. Further study will be made of the economics of alternate cutting patterns to improve utilization of this material.

Outer Support Improvement

Production was converted to the use of 50 mil thick outer supports. Charge testing will be completed soon and a summary report will be published. Preliminary analyses of the performance of the 50 mil thick support have shown it to be capable of withstanding charging forces up to the limit of charging machine capability.

Driver Tube and Target Fabrication Development

Twenty driver tube billets have been canned and extruded. Bond testing for both clad-core and uranium-uranium interfaces has been completed. All ten extrusions passed the peel test and the uranium-uranium bond on metallographic samples was satisfactory. No unusual extrusion problems were encountered.

Development of the final assembly process is progressing satisfactorily. Detailed assembly and measurement techniques necessary to guarantee an interference fit between the target and driver fuels have been developed. These techniques will be debugged and documented prior to startup of final assembly operations.

Process specifications for fabrication of the driver tubes were completed and issued. Process specifications for fabrication of the target fuels have been issued for comment. Driver tube fabrication is on schedule. Target fuel fabrication is behind schedule but the completion dates for the program have not been jeopardized. Delays in the target program have resulted from (1) late delivery of C64 aluminum cladding components, and (2) delays in establishing the variability of lithium and core material and in controlling the content by coextrusion billet design to provide a uniform lithium content in the finished product.

Outer Tube Inner Clad Dimpling Defects

Although improved billet lubrication techniques continue to control the dimpling reject problem, four extrusions were affected by dimples early in May. Investigation revealed that the dimpling was related to performance of an extrusion mandrel. A check of the mandrel showed that its hardness had decreased significantly and it had necked down about 2 mils out of tolerance. This further confirms that the dimpling problem is closely associated with extrusion lubrication conditions. Since present operating procedures require change-out of mandrels after five consecutive extrusions, it is conceivable that more careful checking procedures can be developed to eliminate this cause of dimple defects.

Equipment

Procurement, fabrication and modification of all parts required in the fabrication of the co-product assembly is 90% complete. Design of parts and modification of equipment for the 1.25 percent spike enrichment program is complete. Counterbore lathe tooling was successfully tested.

Criticality Studies on 1.95 Enriched Scrap

The philosophy for control of enriched uranium during shipment, even in the event of the maximum credible accident, is to limit the number of so-called safe masses in close proximity at any time. Safe masses are usually defined as about 45% of critical mass. An example of the manner in which these controls are implemented is that no more than two safe masses are ever loaded in a railrod car for return of scrap to FMPC. At the present enrichment levels these restrictions cause no particular problem, but if the N-Reactor is converted to 1.95% enrichment many safe masses of scrap would be generated in any calendar year. Some form of critically safe shipping container will be needed that will permit the return of more than two safe masses in any one railroad car.

Criticality Alarm from Power Outage

A false criticality alarm sounded in the 333 and the 303-E Buildings. The criticality alarm and the fire alarm were both set off about 15 seconds after a complete power outage had occurred.

Emergency power was provided within 15 seconds after the initial outage. The 333 Building was completely evacuated within one minute, and further, as a result of the alarm, the total 300 Area was evacuated. After approximately forty-five minutes, all employees had returned to their work stations. It was determined that the power outage caused the false alarms.

Nuclear Safety

A nuclear safety violation was experienced in the 333 Building when a chemical operator loaded four baskets, each containing eleven twenty-six inch 1.25 percent enriched outer fuels, into the chem mill tank. Nuclear specifications limit this to three baskets of not more than twelve fuels per basket. No criticality occurred or was expected to occur. Appropriate administrative actions to prevent recurrence have been taken.

Deionized Water

More than one full production day was lost this month from lack of deionized water. Adjustments and repairs of valves were undertaken, and one unit was recharged with new resin. It has since been providing water of exceptional quality.

Peel Rejects Down; Warp Rate Up

The reject rate for the month indicates: (1) A partial return of the extrusion related losses with external cladding variation on outers averaging about three percent. Internal variations, internal minimum clad, and internal unbonds were all in evidence. Peel test rejects, however, were at an all time low. (2) Warp rates on the twenty-six inch inner pieces were significantly higher than the previous twenty-four inch runs, even when specifications were changed to allow for the additional length. (3) Nearly 20 percent of the total production run was set aside for alpha counting examination, with only one and one-half percent rejected for uranium in the braze. Inner material has shown good control on uranium in the braze, with reject rates dropping to less than one percent.

PLANT MODIFICATIONS

Repair of Steam Generator 4A

Following the removal of all 100,000 feet of tubing affected by extensive intergranular corrosion, steam generator 4A was thoroughly cleaned preparatory to installation of new Inconel tubes. The first panel of tubes was placed in the steam generator during the week of May 18. A total of 27 panels, consisting of 486 tubes, has been received at Hanford. Another 28 panels, consisting of 472 tubes, have been prepared at Chattanooga, Tennessee; they are expected to arrive at Hanford during the first week in June. A total of 110 panels is required to complete the retubing of the steam generator. Manufacture of 100,000 feet of the tubing by Sawhill Tubular Products has been completed.

Much of the Combustion Engineering field effort consisted of the qualification of welders, and the optimization and modification of various fixtures and lifting devices for moving panels of tubes from the outside of the building, through the elevator shaft, into the cell, and into the steam generator shell.

Special Fuel Handling and Examination Facilities

Design criteria have been completed for the means to remove irradiated R&D and regular fuel from the reactor and deliver it to the storage area without significant damage to the fuel element fixing member or self-supports. Basic equipment is needed in the third quarter of FY-1965 for such handling and for examination of R&D fuel.

CRWS Pump Discharge Valves

As a result of a visit and recommendation by the vendor's representative, an equipment modification procedure has been prepared to improve the response and reliability of the hydraulic operators of the pump discharge valves in the circulating raw water system.

The modification will provide for filtration of the hydraulic system and a change from a sliding spool-type pilot valve to a non-sticking, poppet-type. This will eliminate the erratic time delays previously encountered in operation of the pump discharge valves.

Rupture Monitor Room Shielding

During initial low power level testing it was noted that a rod coolant drain line passing through the right rupture monitor room would require shielding for protection of personnel and for proper operation of the gamma energy monitor at higher power levels. A design modification provides lead brick around the lower 24 inches and lead shot encased in a 16-inch sleeve around the upper 72 inches of the rod coolant pipe. Sufficient shielding is provided to reduce radiation levels at the surface of the sleeve by about 99 percent.

Process Tube Monitoring Equipment

Response from vendors for the radiation resistant T.V. system has been poor. AEC purchasing has terminated the invitation to bid on this equipment and plans to reissue the package for negotiated bidding. The depth gauge probe is being detailed and fabrication will start in the near future. Detailed drawings of the control panel and console are complete and ready to build.

In order to estimate how long the 40-inch T.V. camera probe could be used in increasingly bent process tubes, calculations of minimum estimated radii of the process tubes after 10 and 15 years were made. On the basis of these calculations it was determined that the 40-inch T.V. camera probe could easily pass through process tubes after 10 to 15 years of operation.

105-N Nitrogen System

Difficulty has been experienced in obtaining satisfactory leaktightness in the bottled nitrogen system that actuates the 105-N emergency cooling valves. Two corrective proposals have been developed to reduce the loss of nitrogen. Both recommend the retention of the existing dual-manifold nitrogen system, but with the addition of two 5-cfm air compressors to compensate for leakage from the system.

One proposal calls for the nitrogen bottles of each manifold station to be isolated from the respective line until automatically brought into service by the signal for valve actuation. In that arrangement, the air compressors would maintain line pressure and compensate for leakage from the system external to the gas bottles. The other proposal calls for the nitrogen bottles of each manifold station to float on the respective line and also to function as a common receiver for the two compressors. For either proposal, suitable compressors are readily available at a relatively low cost.

Heat Dissipation System Control

During recent testing activities it was found that minor power fluctuations produced circuit breaker trips to the branch circuits of the control system. This has been changed by replacing the extremely fast tripping circuit breakers with those which will ride through the power fluctuations and yet provide adequate protection to the circuits involved.

Dump Condenser Modification

Modifications will be made to the remaining 12 dump condensers in a manner similar to that performed on the first four dump condensers by an interim procedure in March. The purpose is to avoid severe steam hammering when the condensers are drained down from water-to-water to steaming operation. This has been accomplished by introducing an auxiliary steam admission line directly into the top of the dump condenser dome to permit the water to be drained out of the dome before the main steam enters the sides of the condenser vessel. Otherwise, the steam reacts with the water trapped in the dome and causes severe hammering.

Dump Condenser Drain Valve

A study is in progress to determine feasibility and cost of converting the ON-OFF type condensate outlet valves for the sixteen dump condensers to throttling type valves so that dump condenser level can be automatically controlled. The valves can be converted by installing valve positioners for proportional positioning of the valves and by changing the present full line-size, on-off quick opening trim to a reduced area linearly characterized trim.

Pony Motor Controls

A review of the pony motor controls showed that one of the control relays for starting each of the pony motors was powered from a relatively unreliable 24 volt power source. Reliability has been improved by replacement of the relays, and transfer of the function to the reliable 125 volt DC battery system.

Additional 230 KV Power Supply Connection

A scope study has been completed on the cost and feasibility of providing a second 230 kv power supply connection for 100-N Area. (Ref. HW-82286)

Feasibility Study of On-Power Charging Machine

A technical review of the on-power charging machine being developed by AMF Atomics for use on a CANDU-type reactor was completed and a report (HW-82186) issued. The report states that it is technically feasible to add the on-power refueling capability to N-Reactor, while retaining the high-speed charging ability during shutdown. However, the anticipated capital investment would require many years in economic write-off. On the basis of current information, an affirmative recommendation could not be made for acquisition of the equipment.

TECHNICAL ACTIVITIES

PLANT CHARACTERISTICS

Primary Coolant Flow Distributions

Flow maps taken during initial low power level testing show a tendency toward preferential coolant flow in various portions of the primary loop. With four loops in operation, the maximum deviation of process tube flow from the average was plus 6.3 percent and minus 6.5 percent. At the time the readings were taken, there was some imbalance in pump speeds, and in resultant loop flows. However, considering the tubes on a given riser, the deviation from the average flow was still as much as 6 percent, plus or minus. The problem of preferential coolant flow will continue to be an area of investigation during the power ascension program.

Primary Loop Relief Valves

Functional tests at 189-D, with water at 310F and 560F, were performed on the relief valve complex comprising the factory rebuilt spare pilot valve, the prototype control valve, and the spare primary valve with the heavier closure spring. The results indicated that the primary valve operates satisfactorily with the new spring, but that the factory rebuilt pilot valve is still unacceptable because excessive leakage past the valve seat prematurely triggers the control valve. Consequently, the actuation pressure is erratic and generally lower than the pilot valve set-point. The rebuilt spare pilot valve will be returned to the manufacturer for examination and correction. The two comparable pilot valves installed on the reactor perform acceptably.

Analysis of Accidental Primary Loop Dump

An analysis was made of the temperature transients following an accidental dump of the primary loop in which the emergency cooling system operated as intended. The fuel element temperature transients were calculated for the following three cases: 1) cooling by nucleate boiling, 2) cooling by film boiling, and 3) cooling by forced conduction to steam. The first and second cases represent the probable limits of the actual case. However, the third case was examined in the event that total steam blanketing might occur in a few tubes. In all three cases, the fuel element temperatures were calculated to remain below the alpha-beta transition temperature.

Process Tube Flow Monitor

The process tube flow monitor was put into service during initial low power level testing when it was required for performance of the reactor safety circuit system. No malfunctions were experienced in the primary safety application of the flow monitor. However, several scrams resulted from lack of operating experience with the self-check circuits. The operation of the self-check circuit, which monitors for open diodes in the trip logic circuits, was not completely satisfactory for some conditions between actual flows and trip points. The probability of such operation had been recognized during the original system checkout. New open diode detector boards have been fabricated to a modified design and these circuits are being tested for correct operation under all flow conditions.

Technical manuals for the flow monitor system were received this month from the vendor. These manuals are of great assistance for system maintenance and provide information which will aid in the selection of spare parts and for the self-check circuits and the DC power supplies.

Shielding of Process Unit Steam Vents

During initial low-power level testing it was observed by radiation monitoring personnel that high levels of neutron and gamma radiation existed beneath the reactor at the minus 16-foot level and in adjacent rooms. An investigation revealed that most of this radiation was coming from the two 20-inch steam vents which penetrate the foundation block vertically into the reactor core. This condition is being alleviated by installing shield plugs in the vertical sections of the steam vents, encasing the elbows beneath the vertical sections in shielding concrete loaded with boron frit, and providing removable shielding blocks around the four expansion joints. An analysis of the pressure-drop characteristics of the shield plugs assured that the steam vent system pressure-drop will not be affected by the insertion of these plugs.

Rod Scram Surge Problems

Simulated service tests on the rod drive at 189-D have produced shock pressures in the vent line greater than three times the scram accumulator pressure when gas is present in the scram piping. When the gas is completely bled, no severe shock pressures occur. Tests have shown also that the shock pressures can be eliminated by installing an orifice 0.0145-inch in diameter in the vent fitting of the 1-inch scram piping at the beginning of the vent line. A modification has been approved and a work order has been issued for installing a special orifice fitting in the vent line connection of each rod drive to eliminate the damaging shock pressures.

Water-In-Reactor

Additional information has been gathered pertaining to the source of the water collected in Drip Cell No. 6 of the reactor exhaust gas plenum. A gas sample taken from the inlet face shield vent was analyzed and compared with a sample taken previously during the Hot Dump Test of the unit; the comparison shows the drying trend being experienced by the shield concrete and provides further assurance that the water collected in the gas plenum drip cell originated with the shield concrete rather than with the process tubes.

Rupture Monitor System

The gamma energy monitor has operated successfully without significant equipment problems. Transistor failures have occurred in the position identification circuits about one each week, and efforts are underway to determine the cause of failure. No delay in reactor operation will result from the faulty transistors.

The gross gamma monitor has operated satisfactorily, with the exception of failures to the differential alarm modulus. This problem has been traced to temperature sensitive silicon controlled rectifiers.

Safety Circuit Trip Devices

In December, 1963 it was discovered that the circuit breakers employed as master trip devices in both the safety circuits and ball 3X system failed to trip on demand. The cause of the fail-negative events was attributed to a faulty linkage in the under-voltage trip mechanism associated with each breaker. The linkages were repaired, and over a period of four months, both the safety circuit breakers and the ball 3X breakers have been tripped many times without failure. The vendor has also reviewed the application and has developed improvements to the trip mechanism. After factory tests are completed, Hanford tests will be made with the trip mechanism applied to a spare circuit breaker. Preliminary engineering review of the proposed modifications indicates that the improved device will be acceptable. In the meantime the four circuit breakers involved will be tested ten times each on a monthly basis to determine any undesirable trends.

Thermal Hydraulic Studies

Transient experiments with the electrically heated half-column test section have been completed. As previously reported, these experiments were made to simulate the transient conditions following a complete failure of an inlet connector upstream from the butterfly valve, and involved blowdowns from various initial test section powers, supply tank pressures, and coolant temperatures equivalent to N-Reactor operating conditions. Heat generation decay simulating a scram was programmed into test runs made with the inlet butterfly valve both in open position and at its low flow stop.

The principal objective of the experiments was to determine flow and "fuel" surface temperature transients during the blowdowns. Preliminary study of the data revealed the following results:

1) With a scram programmed to begin 1.5 seconds after the start of each of the blowdowns, there was no significant increase in temperature of the heated sections representing the fuel column. Such was the case for runs with the inlet butterfly valve in either of the two positions.

2) The duration of the flow transient was extremely short. During the transient some oscillation was observed in pressure differential across the inlet venture (a production model) and there were short periods in which the differential was negative. After the transient, which lasted only a few seconds, the flow stabilized to a steady state level. The magnitude of the flow indicated that some upstream flashing to two-phase mixtures had occurred early in the test and persisted after the flow had become steady.

The results indicate that reverse flow would be adequate for cooling the fuel subsequent to any complete failure of an inlet connector, even in the extreme case in which the inlet butterfly valve is forced to its low flow stop at the onset of the accident.

<u>Charging Machine</u>

During initial loading of the reactor, one bearing on the cross-travel drive truck of the charging machine failed several times. The design of the axle bearing and bearing housing has been changed on the No. 3 truck only, fabrication completed and new parts installed. In addition, strain gauges were installed on the tie rods to provide a more accurate means than the torque method used before of placing an initial tension on the tie rods. The tie rods have been properly tensioned, and the charging machine is now ready for service.

Exhaust Filter Cells

Following the issuance of the formal design criteria and design drawings, work was started during the outage to install a system to provide water cooling in the filter cells for the Zone I exhaust system. Installation has consisted of additional spray headers and nozzles in three of the four cells in the 117-N Building, an automatic control system and modifications to the sump pumping system. All work required within the 117-N Building has been completed during the current outage. The remaining work outside the building necessary to complete the system can be performed during reactor operation.

Other Equipment

Heating coils have been added to the ERW tank to increase the heating capacity, and vent piping has been installed in the demineralized water tank. Installation of the water quality monitoring instrumentation is essentially complete, and the facility has been turned over to Operations for startup.

It was discovered that certain secondary system piping in the pipe cells did not have penetration seals. These seals have been installed and confinement testing of the 109 and the 105 Building confiner has been in progress. Installation of the radiation monitor for the primary loop diversion header was completed and tested. Balancing of the heating and ventilation system has been completed for all areas in the 105-N Building. All work required to increase the dc supply voltage for the rupture monitor system from 24 volts to 28 volts has been completed and tested.

ACRS Presentation

On May 7, presentations were made before the AEC advisory committee for Reactor Safety at the Argonne National Laboratories. The results of physics testing, operational testing and the results of the N-2 low level shakedown tests were reviewed. Consequences of fission product releases and associated waste management concepts and programs were also discussed.

N-3 Tests

Document HW-81580, entitled "The Startup Test - N-3, N-Reactor Power Ascension Program", edited by E.E. Leitz, was transmitted to the RLOO-AEC. All requisite documentation for beginning N-3 has been delivered to RLOO-AEC.

INSTRUMENTS, EQUIPMENT AND FUEL TESTS

Freeze Plug In Header

A concentrated effort has been applied to establishment of a freezing method for blocking water flow from the inlet, outlet, and diversion header arms during inlet and diversion valve replacement. Experiments on the mockup have resulted in the use of a dry ice acetone mixture, which forms a four inch freeze plug from initial water temperature of 120 to 130F in 45 minutes. The freeze plug has retained water at header pressures to 80 psig.

A screw feed type cutter for removing irradiated process tubes was demonstrated. The tool cut a sample piece of process tube four times at an average of 15 seconds per cut, but some difficulty was experienced with the cutter wheels freezing to their shafts. A modification is being made to the cutter wherein the screw feed mechanism will be replaced by a cam-feed. This will accomplish the cutting operation in one minute thereby reducing the load on the cutter wheels and prolonging their life.

Status of Fuel Testing

Table I summarizes the current status of tests in the KER loops. Since Loop I operated at a low temperature for a significant period of time because of a leak, the K sized enriched (KSE-5) elements in this loop were replaced by a new charge of similar elements. These elements are now being exposed at N-Reactor conditions.

CURRENT TEST

KER LOOP #	PURPOSE	ELEMENTS	Exposure-MWD/T Current	Exposure-MWD/T Goal	Elements	Goal (Exposure)
1	Compare high temperature behavior of standard uranium and of "British" uranium	8-12" KSE-5 elements, 4 of each composition	500	2500	8-12" KSE-5 elements, 4 of each composition	3500
2	Compare high temperature behavior of standard uranium and of "British" uranium	8-12" KSE-5 elements, 4 of each composition	1000	1500	6-9" prototype target rods, 5 WIE-1 heater elements	6 GVR 400 MWD/T
3	Evaluate behavior of co-producer test elements	13 standard co-producer test elements; 26" long	1000 (1500)	1800		
4	Evaluate crud monitor element	1 crud monitor thermocouple train, 10 heater NAE-1 element	375	1000		

TABLE I

Fuel Charging Tests

Fuel charging tests were carried out to determine the relative performance of various types of fuel element supports. A forged "T" section outer fuel support showed improved stability over the current model with respect to deformation and arrangements are underway to procure supports of this type. An outer support 10 mills thicker than the current model will be produced in the Fuel Element Pilot Plant for use during the interim.

Zirconium Technology

The first sample of the Zr-2 pressure tube from KER Loop 3 in which a slot had been milled was burst tested at room temperature. The sample, with a 3/4" long, 1/8" wide slot milled 80% through the wall, began to leak at an internal pressure of about 2500 psi. The leak was sealed, and the tube again pressurized to failure. Failure occurred at about 10,000 psi. The latter pressure is about the same as for unirradiated tubing with the same size slot in it. The burst pressure for irradiated sample (undefected) was considerably higher than for a comparable unirradiated sample. The pertinent conclusion is, that although the effect of the presence of this crack was proportionately greater for the irradiated tube, the ultimate pressure at failure was not significantly different than a similarly defected unirradiated tube.

Spike Fuel Behavior with Temperature

The FLEX 2 computer study regarding the behavior with temperature of the N-Reactor spike fuel vs. the standard fuel has been completed and shows that the metal temperature coefficient is the same for both cases. The water and graphite coefficients of the spike fuel are more negative, for exposure up to at least four months at full power, than those of the standard fuel. Although the water and graphite coefficients become more positive with exposure for both fuels, this effect is smaller for the spike case than for the standard. For the cold, green fuel, k_{eff} of the spike fuel is $\sim$ 76.6 mk higher than that of the standard fuel. This difference for the hot, green case is $\sim$73.3 mk.

General

Prototypes of the crate spot facing tool and of the Grayloc seal ring retainer were completed. Design of the prototype of the special process tube pusher was completed and placed in the shops for fabrication.

Materials Manual

The N-Reactor Materials Manual (HW-79050) was issued to provide a ready source of information regarding materials, drawing numbers, equipment functions and vendors for the primary, secondary and graphite cooling systems.

PLUTONIUM R&D AND REACTOR PHYSICS

Control Strengths in N-Reactor

The large number of control rods in N-Reactor will allow a rather flexible and detailed control of the chain reaction. It will also produce a varied and somewhat complex set of control conditions. Some rod arrays were studied experimentally during startup; uniform, peripheral, central, end column and single column. A single theoretical model of the controlled reactor that describes all of these experimental situations is desired. Two-group diffusion theory of the space-energy distributions of neutrons is being applied at present to the determination of a model of the lattice cell that will allow the determination of reasonable spectra and hence reasonable group constants for the diffusion model.

N-Reactor Cell Analysis

In an attempt to obtain in more detail the distribution of neutrons in energy and space in the N-Reactor lattice cell, an investigation of THERMOS, a transport theory code, has been started. The code computes the space-energy distribution of neutrons at as many as 20 space points (in one dimension) using as many as 30 neutron velocity groups. In addition, the spatial distributions of thermal reaction rates of various radioactivants are calculated on demand. The latter can have applications in the interpretations of the hot and cold lattice parameter and spectral index experiments conducted during startup.

The possibility that impurities in the N-Reactor graphite account for the observed 9 mk reactivity deficit is under investigation. To date, a statistical analysis of 24 diffusion lengths measured in the as-stacked graphite has been completed. The average diffusion length is 85.7 $\pm$ 0.2 cm rather than 84.7 cm reported earlier. The average diffusion length for directions normal to the process holes (no process tubes) is different from that for directions at 45° with respect to the process holes. The difference is, however, only 1% and the value is greater in the 40° direction as one might expect, since the major portion of the void structure is in the process tube direction. However, the uncertainties in the Behren's correction (for streaming of neutrons in the open channels in the stack) are so large relative to the effects being sought that no conclusive result is yet apparent.

CO-PRODUCT

Determination of Conversion Ratio for Co-Product
(1.25% enriched uranium test load)

A comprehensive technical review of the 1.25% enriched co-product test load was carried out. The production test document is being circulated for approval and the associated hazards analysis is underway. KER loop tests on co-product assemblies are proceeding on schedule.

Co-Producer Element Design Test
(1.95% enrichment)

The enrichment for this N-Reactor irradiation test has been defined as 1.95% uranium-235 and the technical criteria for the demonstration loading have been determined. This test is designed to establish the irradiation performance of 20 lb/ft prototype co-product elements. The uranium enrichment was selected as typical of the enrichment that would be used in a full production load of co-product elements. Detail design of the fuel driver and target was completed for the 1.95% enriched uranium co-product elements to be used in the Physical Constants Test Reactor (PCTR) test.

ADVANCED TECHNOLOGY

Advanced Reactor Study

A reactor concept for desalination or power generation, or both, has been developed that utilizes a modified NPR-type reactor with a higher specific power. This is a modification of the reactor proposed for inclusion in the Bechtel study sponsored by the AEC during the latter part of 1963. An economic evaluation of this reactor concept indicates considerable improvements over the one included in the Bechtel study. Energy costs are estimated to be 10 cents per million BTUs, compared to 16 cents per million BTUs for the earlier case. These costs are based on a value of 3 mills per KWH for the electric power generated and assume the use of municipal financing.

CONVERSION

Water Quality Instrumentation

Specification HWS-6890, "Automatic Condensate Quality Monitoring Equipment, 182-N and 109-N," has been issued for comments and transmittal to the AEC. The specification covers the design, fabrication and delivery of instrumentation to monitor the quality of condensate being returned from the 185-N turbine condensers.

Electrical Drawings

Conversion criteria HW-76667, "Primary and Secondary System Instrumentation", requires revision of about 35 existing General Electric drawings and three new electrical drawings. The drafting required for this work is in progress.

Annunciator Requirements

The annunciator equipment requirements for conversion have been reviewed and listed for transmittal to RLOO-AEC.

Construction

Miscellaneous tie-in work for the sixth cell has been performed during the outage. The electrical contractor is adding electrical components in the existing electrical and terminal cabinets and has pulled cable from the 109-N Building to this location. The siding has been removed from the south side of the 105-N Building transfer area to pour the wall for the sixth cell and pipe gallery extension. Form work and re-steel installation is continuing on the pipe gallery extension and west wall of the sixth cell.

PRODUCTION STUDIES

An earlier study disclosed that for the plutonium-only case, unit product costs continuously decrease as fuel exposures increase, up to 3000 MWD/T. A preliminary examination of costs associated with higher fuel exposures indicates that only minimal plutonium unit cost incentive exists to increase average fuel exposures above 3000 MWD/T. Total costs, however, do continue to decrease with increasing exposure for the range of cases studied. This becomes an important factor in evaluating incentives for metallic fuel for power-only operation. Costs examined in this study were for fuel exposures ranging from 2100 to 6000 MWD/T.

A new series of production studies is expected to commence in early June in order to refine the 1963 studies and to take a closer look at tritium production. In response to a Commission request that we look for computer applications to aid in this study, the LP-90 and PULP computer programs in use at HAPO have been suggested for determination of optimum complex operation. A demonstration of this program is planned for presentation to the AEC in late June.

ADMINISTRATIVE SUMMARY

EMPLOYMENT

	FORCE 5-31-64			CHANGE		
	E	NE	Total	E	NE	Total
General	1	1	2	-1	0	-1
Finance	16	6	22	0	-1	-1
Fuels	48	105	153	-3	-1	-4
Plant	65	220	285	0	+ 4	+4
Project	71	22	93	.8	0	-8
Research & Engineering	48	10	58	-1	0	-1
Total Department	249	364	613	-13	+2	-11

[redacted]

VISITORS

Name	Company	Contact(s)	Date(s)	Purpose
E.F. Miller	AEC Washington, D.C.			Discuss N-2 Startup Test results and N-3 Test plans.
G.A. Rost & J. Porter	Beckman Instruments Fullerton, Calif.	W.D. Bainard	5/20-21	Assist in diagnosing start-up difficulties on water quality monitoring instrumentation.
H.R. Killian	Henry Pratt Company	L.G. Henke & C.L. Goss	5/19-20	Consultation concerning the modifications required to improve reliability of the CRW 66" butterfly valve actuators and the major repair of the No. 1 valve.
J.A. Fellows	Mallinckrodt Chemical Works St. Charles, Mo.		5/27	
J.A. Cox & F.T. Binford	Oak Ridge Nat'l. Lab. Oak Ridge, Tenn.	J.W. Riches	5/11	Discuss isotope production - NPR.

VISITS

Name	Company	Contact(s)	Date(s)	Purpose
P.F. Nichols, R.E. Trumble & M.C. Leverett	Argonne National Lab. Chicago, Illinois		5/6-8	Presentations to ACRS.
J.W. Riches	Bonneville Power Admin. Portland, Oregon		5/14-15	Phase II Fuel Development discussion.
H.A. Carlberg	BPA Portland, Oregon		5/14-15	Consultation with BPA regarding NPR conversion.
N.O. Strand	BPA Portland, Oregon		5/15	To meet with Bonneville representatives to discuss a research and development proposal for Phase III elements.
W.R. Krehbiel	Bridgeport Brass Riverside, Calif.	L. Manns C. Hoehne J. Herroth	4/30	Discuss Quality System.

[redacted]

DECLASSIFIED

VISITS (continued)

Name	Company	Contact(s)	Date(s)	Purpose
W.R. Krehbiel	Buffalo, New York		5/4	ASQC Convention
W.E. Stavig	Buffalo, New York		5/4-6	ASQC Convention
R.R. Studer	DeCota Bros. Yakima, Washington	H. Boy, & J. Miller	5/4	Driver tube clad machining.
T.B. Correy	General Electric San Jose, Calif.	W.R. Smith	5/8	Discuss fuel element welding.
G.C. Hopkins	General Electric San Jose, Calif.	J.M. Clark	5/18-19	Audit of APED charges to N-RD for reactor startup.
W.R. Krehbiel	General Electric San Jose, Calif.	C. Saums, J. Spirakis D. Long K.W. Stave	4/28	Discuss quality system.
W.R. Krehbiel	General Electric New York, New York	W. Masser, S.T. White, H.L. Freeman	5/1	Discuss quality system.
J. Milne & D.R. Stenquist	General Electric New York, New York	R.F. Osborne R.I. Begyr R.J. Canning C.P. Moss	4/28-29	Discuss business planning and business training activities.
J. Milne	General Electric Schenectady, N.Y.	J.P. Fuller F.D. Kineke	4/30-5/1	Review N-RD financial practice.
W.R. Krehbiel	Harvey Aluminum Torrence, Calif.	G. Moudry & G. Christenson	4/29	Discuss quality system.
T.B. Correy	Lawrence Radiation Livermore, Calif.	R.E. Armstrong	5/8	Discuss unbonded end closure.
M.C. Leverett	Nat'l. Lead of Ohio Cincinnati, Ohio		5/8	Discuss fuel program.

VISITS (continued)

Name	Company	Contact(s)	Date(s)	Purpose
R.H. Scanlon	Nat'l. Lead of Ohio Cincinnati, Ohio	J. Blaisdell	5/20-22	Phys. test discussion.
W.E. Stavig	Nuclear Metals, Inc. Concord, Mass.	A. Gilman	5/7-8	Vendor certification.
W.J. Dowis	ORNL, Oak Ridge, Tenn. AEC, Germantown, Md.		5/18-23	Discuss desalination research and development program.
R.H. Scanlon	RMI Astabula, Ohio	J. Puterbaugh	5/19	Tipped mandrel test.
T.B. Correy	Skiaky Bros. Chicago, Illinois	J. King	5/7	Discuss unbonded end closure.

SIGNIFICANT REPORTS

Number	Title	Author	Date
HW-82408 Undoc.	Analysis of Shoeing Plier for NPR Fuel Outer Supports	D.E. Blahnik	5/22/64
HWS-6890 Uncl.	Automatic Condensate Quality Monitoring Equipment - 182-N	W.D. Bainard	5/14/64
HW-82431 Sec.	Budget Study Report, Polonium Production Facility, N-Reactor Plant, 100-N Area	G.F. Bailey	
Undoc-Conf.	Co-Producer Fuel Element Costs	J.W. Nickolaus	5/7/64

SIGNIFICANT REPORTS (continued)

Number	Title	Author	Date
---Undoc.	Criticality False Alarm	S.G. Forbes	5/25/64
HW-82364 Conf.	Deviations from Chemical Specifications for Iron - NPR Ingots	L.M. Loeb	5/20/64
HW-82381 Conf.	Design, Development and Research Contract DDR-194-Electron Beam Welding Fuel Elements	T.B. Correy	5/21/64
HW-82337 Sec.	Effect of Enrichment Level on Handling of Uranium Scrap	J.E. Ruffin	5/20/64
HW-81724 Conf.	Evaluation of NOT Inner Clad Thickness Specification	R.H. Scanlon	5/1/64
HW-82236 Uncl.	Facilities Design Criteria, Zone I Ventilation Exhaust Filter Cooling System	D.D. Stepnewski	5/11/64
HW-81845 A Uncl.	Failure Analysis of 109-N Instrumentation System - Part I - Primary Coolant Temperature-Secondary Coolant Pressure Control System,	A.A. Maupin, Jr.	5/12/64
HW-81845 B Uncl.	Failure Analysis of 109-N Instrumentation System - Part II - Primary Coolant Flow Control System	A.A. Maupin, Jr.	5/14/64
HW-82283 Conf.	NPR Accidental Dump Transient Analysis	J. Muraoka	5/18/64
HW-82331 Conf.	NPR - Total Loss of Coolant at 10% Design Power	J. Muraoka	5/20/64

DECLASSIFIED

SIGNIFICANT REPORTS (continued)

Number	Title	Author	Date
HW-80334	N-Reactor Department Disaster Plan for 100-N Area	J.R. Bolliger & F. Cox, Jr.	4/30/64
HW-79050	N-Reactor Materials Manual	B.S. Kosut	5/64
HW-81989 Conf.	Physics Presentation to ACRS Subcommittee - April 20, 1964.	P.F. Nichols & W.S. Nechodom	4/24/64
HW-81936 Conf.	Process Specifications for 1.25% Enriched Co-producer Fuel Elements	W.A. Hendrickson	4/30/64
HW-82395 RD Conf.	Process Specification for Target Elements - Production Test NR-8 Co-Producer Demonstration Test (1.25)	W.A. Hendrickson	5/22/64
HW-82203 Sec.	Production Forecast	L.M. Loeb	5/11/64
HW-82431 Sec.	Reactor Design Analysis Monthly Report, April, 1964	H.R. Kosmata	5/25/64
HW-81087 Conf.	Reactor Engineering Data System - Input Data Requirements	R.G. Lauer	5/8/64
Undoc-Conf.	Reconditioning of Billet Surface Defects	R.H. Scanlon	5/12/64
HW-82186 Uncl.	Review of Technical Feasibility for Applying On-Power Refueling Equipment Developed by AMF Atomics to N-Reactor	H.D. Lenkersdorfer	5/1/64

DECLASSIFIED

SIGNIFICANT REPORTS (continued)

Number	Title	Author	Date
HW-81962 Conf.	Review-Process Development 1.25% Enriched Co-product Driver Fuel	C.H. Shaw	5/1/64
HW-82078 Uncl.	Scope Study - Additional Safety Devices for W, C, and D Hydraulic Elevators	H.D. Lenkersdorfer	5/1/64
HW-82363 Undoc.	Specifications for Physical Tests of Uranium Billets	N-Fuels Eng'g.	5/64
HW-82085 Sec.	Uranium Delivery Schedule NRD HAPO	L.M. Loeb	5/1/64

PROPOSALS

Number	Title	Author	Date
HW-81942 Uncl. Appendix A	Alternate Fuel Cycle for Power-Only Operation of N-Reactor		4/24/64

SECURITY VIOLATIONS

NONE.

INVENTIONS OR DISCOVERIES

NONE.

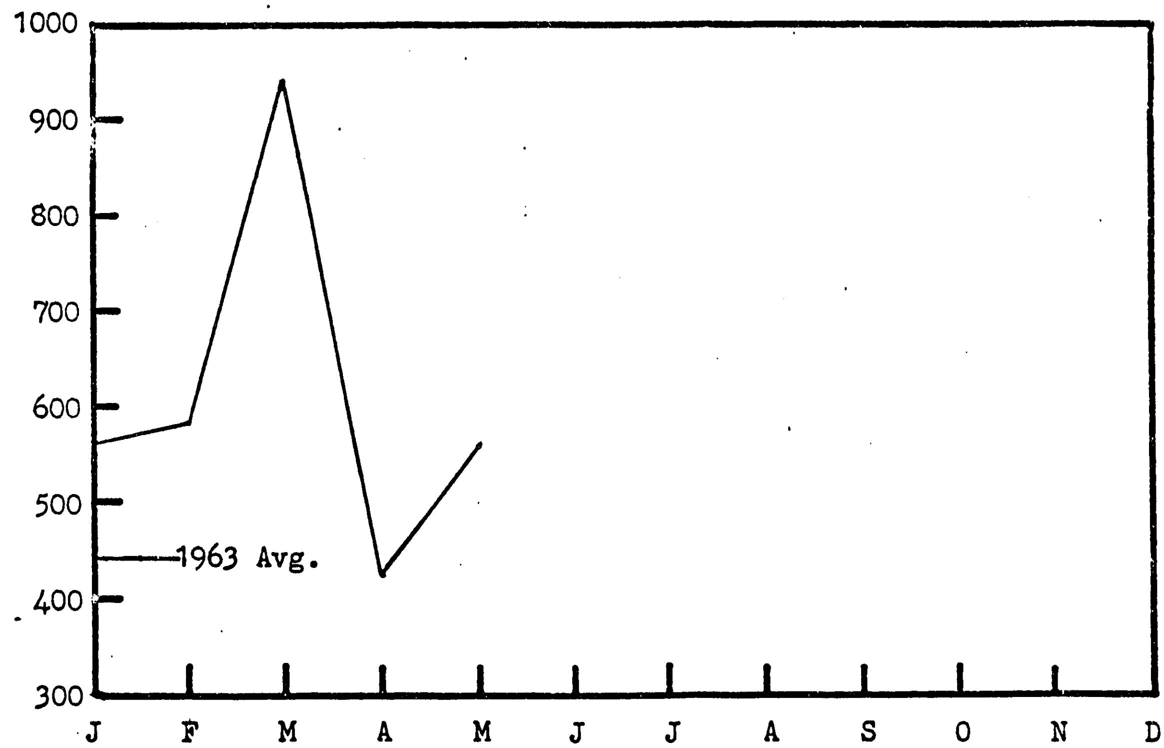
gb
6-8-64

Safety and Security

Days without a disabling injury	548
Hours worked without a disabling injury	1,784,211
Medical treatment injuries (May)	26

Suggestion Plan Participation

	May	CY 1964 to-Date
Number of eligible employees	364	365
Number of suggestions received	58	228
Number of suggestions acted upon	48	236
Number of suggestions adopted	17	94
Net annual savings	$713	$120,815
Amount of awards	$275	$ 5,490
Percent of awards to savings	38.6	4.5
Average amount of awards	$ 16.18	$ 58.40



SUGGESTIONS ADOPTED/1000 EMPLOYEES/YEAR

1964

INDEX

DATE
FILMED
01/20/93