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NOVEMBER 10, 1966

HANFORD ATOMIC PRODUCTS DEPARTMENT

✓ MONTHLY REPORT

OCTOBER 1966

GE-AEC BUSINESS RECORD

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HANFORD ATOMIC PRODUCTS DEPARTMENT

MONTHLY REPORT - OCTOBER 1966

R. L. Dickeman
General Manager

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

During October the N-Reactor operated with a time-operated efficiency of 39.0 percent, a production availability of 66.1 percent and a steam availability to WPPSS of 35.0 percent. Operating continuity was adversely affected by 9 shutdowns for charge-discharge, equipment maintenance and fuel failures. A total of 264 fuel columns were processed.

Analysis of recent failures of N-Reactor fuel, when combined with accumulated experience, yields a correlation between the probability of fuel failures and increased exposure or residence time in-reactor. Based upon these correlations and underlying technical analyses, the decision was made during the last week of October to discharge 178 columns of fuel.

All fuel discharged were at an average exposure of 2200 Mwd/t or above, which is in the range of 105 to 170 percent of nominal goal exposure of 2100 Mwd/t. Because the exposure residuals in the reactor remained high even after this discharge, it may be necessary to replace another 150 to 300 columns of fuel prior to engaging the coproduct demonstration program late this year.

The demonstration irradiation of Np-237 to produce Pu-238 proceeded smoothly. The first column of this material was discharged for examination, reprocessing, and for return of the residual neptunium to production channels.

The first cobalt fuel element irradiated in the graphite cooling system was discharged. The estimated specific activity is eight curies per gram.

Initial evaluations of thermal hydraulic, fuel design, and operating parameters show the feasibility of operating the N-Reactor at 6000 Mw if certain modifications are made. Additional confirmatory studies related to the criteria, technology, and economy of operating at this level are in progress.

The only aspect in which the technology for the coproduct fuel has not been essentially confirmed is the target support system. The buggy spring type of target support for the Mark II coproduct fuel has been dropped from further consideration. Effort now concentrates upon the spring-stop configuration using a W-type spring in place of the buggy spring. Some fuel assemblies equipped with this type of target support have successfully achieved more than 60 days in the high velocity test loop which provides exaggeratedly severe testing of the thermal-mechanical properties of the fuel assemblies. No indication of fretting or other damage has been noted indicating the spring-stop support may well solve the target support problem.

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In view of the delays caused by the strike in July and August, and by the emphasis being given to the target support program, charging of the first increment of the coproduct demonstration is now scheduled for December. Production test NR-66, which was interrupted due to target support problems, has been resumed, using prototypical W-type, spring-stop elements. It will thus provide a pilot group of assemblies which will have achieved goal exposure prior to exceeding about 400 columns of the coproduct demonstration load in the reactor. Forty-two tubes of Mark II fuel utilizing the spring-stop target supports were charged.

Administrative progress was made in the projects aimed at providing a package boiler installation to back up the existing Foster-Wheeler auxiliary boiler, and at retubing steam generators in cell 3. Firm schedules for these projects are being developed.

Steam was supplied, as available, to Washington Public Power Supply System for generation of electricity at a nominal 400 Mwe level. The necessary transformers for permitting generation at power levels above 400 Mwe will be available during December according to present information. Successful dual-turbine trips from 400 Mwe were performed on October 12 and 16. The transients caused by the export steam reduction caused no significant effects on reactor operation. Smooth system response attested to excellent tuning of system controls.

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OPERATING STATISTICS

In-reactor residue (Kmw)	494.31
Highest exposure tube (Mwd/t)	4,484
Input production, plutonium (Kmw)	31.8
Input production, coproduct (Kmw)	1.25
Maximum power level attained (Mw)	3,850
Average power level while operating (Mw)	2,626
Time operated efficiency (%)	39.0
Number of shutdowns	9
Scheduled	1
Unscheduled	8
Reactor process data	
Power at which computed (Mw)	3,850
Primary coolant inlet temperature (°F)	386
Primary coolant outlet temperature (°F)	508
Maximum moderator temperature (°F)	1,025
Maximum tube power (kw)	4,530
Heat loss	
Heat to graphite cooling system (Mw)	30.5
Heat to shield cooling system (Mw)	1.5
Heat to rod cooling system (Mw)	1.10
Steam generation temperature (°F)	355.7
Steam generation pressure (psig)	130
Dump condenser flow	325,000
Inlet temperature (°F)	57
Outlet temperature (°F)	107.2
Nuclear steam generating rate	
Maximum (1000's lb/hr)	8.829 x 10 ⁶
Water collected from gas atmosphere (gal/day)	Negligible
Demineralized water produced (gal/hr)	76,380
Helium losses (1000's ft ³)	591.9
Fuel oil usage (bbl)	21,217

Reactor flattening efficiency (%) 85

Fuel conditions

Heat flux maximum (Btu/hr-ft²) 535,000

Heat flux average (Btu/hr-ft²) 455,000

Cladding temperature

Maximum (°F) 590

Average (°F) 561

Uranium temperature

Maximum (°F) 859

Average (°F) 791

Fuel failures 5

	Natural			Coproduct	Coproduct	
<u>Fuel Balance</u> (tons)	<u>0.71%</u>	<u>0.95%</u>	<u>1.25%</u>	<u>1.96%</u>	<u>2.1%</u>	<u>Total</u>
Fuel charged	0	65.80	18.32	0	8.60	92.72
Fuel discharged	0.73	91.42	4.78	1.36	0.20	98.49
Net change	-0.73	-25.62	+13.54	-1.36	+8.40	- 5.77
Total In Reactor	<u>0.36</u>	<u>266.49</u>	<u>82.34</u>	<u>0.65</u>	<u>9.83</u>	<u>359.67</u>

<u>Fuel Element Output</u> (Assemblies)	<u>October</u>	<u>September</u>	<u>CYTD Total</u>
0.95% tube-in-tube	0	0	9,094
0.95% single tube	0	0	33
1.25%-0.95% tube-in-tube	0	0	2,991
1.96% coproduct	0	0	0
1.25% coproduct	0	0	-10*
2.1% Mark E (driver)	130	481	1,013
Total assemblies	130	481	13,121
Total tons	2.0	7.6	254.3
Percent of forecast	8.5	13.4	92.2
Uranium utilization, percent			
Mark E, 2.1%	78.5	84.8	75.7
Mark F, 2.1%	70.4	80.0	76.9

*Fuels returned for rework include some accepted in previous months.

REACTOR FRONT FACE LOADING MAP

		2A,3A										1A,4A										5A,6A											
		75										60										45											
		80										55										40											
		85										50										35											
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Date: 10/31/66

- A - PT-33 Thermocouples
B - Blank
C - PT-57 Ceramic Target
D - PT-32, 1 Coupon
E - PT-32, 2 Coupons
F - PT-66, 2.1% Coproduct (13 pcs.)
G - PT-66, 2.1% Coproduct (11 pcs.)
H - PT-68 Np Target
K - PT-76
M - Monitor Column
N - 0.71% Natural fuel
P - PT-64
R - PT-13, 1.96% Coproduct
S - Spike, 1.25% Enriched
T - Indicates tube has special thermocouples
☐ - Not filled in - Mark I Standard

REACTOR PLANT OPERATION

OPERATING HISTORY

N-Reactor Plant operated with a time-operated efficiency of 39.0 percent, and a production availability of 66.1 percent. Steam availability (defined as the percent of time that the reactor operated at greater than 800 Mwt power level) to WPPSS was 35.0 percent.

The plant, which was operating at the beginning of the month, was shut down for a scheduled outage to charge spike enrichment on October 3. Startup was effected on October 8. There were three unscheduled outages during the month resulting from equipment failures. The first equipment failure outage occurred October 8 when a control failure on the secondary loop makeup valve resulted in the loss of the boiler. A boiler tube failure occurred during the incident and was repaired prior to startup. The second equipment failure came during startup on October 11 when a temperature monitor failure occurred on a newly charged coproduct tube, and the third came on October 23 when a rear face confinement fog spray supply valve failed which permitted water to enter electrical junction boxes and resulted in numerous 125-V DC grounds.

Five outages resulted from fuel failures which occurred on October 13, 16, 18, 20 and 25. See Technical Activities Section, topic 1.1 for detailed information. Following the fifth failure on October 25, 174 fuel columns were discharged to reduce failure incidence in high exposure fuel. Analysis of recent failures of N-Reactor fuel when combined with accumulated experience, yields a correlation between the probability of fuel failures and increased exposure or residence time in-reactor. Based upon these correlations and underlying technical analyses, the decision was made to discharge these 174 columns of fuel.

All fuel discharged were at an average exposure of 2200 Mwd/t or above which is in the range of 105 to 170 percent of nominal goal exposure of 2100 Mwd/t. Because the exposure residuals in the reactor remain high even after this discharge, it may be necessary to replace another 150 to 300 columns of fuel prior to engaging the coproduct demonstration program late this year.

A total of 264 fuel columns were processed during the month. The plant was operating at month end.

Processing of irradiated N-fuel through the Isochem Redox plant began on October 19. At month end 8.2 tons of 1.25 percent enriched coproduct drivers, 15.46 tons of 1.25 percent enriched spike and 2.22 tons of 0.95 percent standard fuel had been shipped to Redox.

PLANT PERFORMANCE

Cell Removal Procedure

Cell 3 was successfully removed from service just prior to the October 3 outage. The only difficulty that was encountered was in cooling the cell down after it was isolated. It was presumed that there was a leak in the primary loop cell isolation valves, but this was not verified. An initial power level cut to 3000 Mw was made before the cell was removed. Because the removal went so smoothly, it is anticipated that a smaller power cut will be required in the future. However, this will depend upon the status of tube loadings at the time and the status of orificing in the fringe tubes.

Resistance Temperature Detectors

During the scheduled shutdown of October 3, 28 strap-on and 6 zone temperature resistance temperature detectors (RTD's) were repaired. Nine unreadable strap-on RTD's were repaired during the October 26 outage.

Plant Service Boiler

The outage on October 8 occurred when the secondary system coolant makeup was initiated during startup. One of the makeup control valves was inoperative in the open position. When the feedwater valve that supplies the makeup control station was opened, full-line flow was established to the secondary coolant system. The control valve would not close and the result was a loss of adequate feedwater supply to the standby boiler. The boiler fire was dumped, but a boiler tube failure occurred which prevented continued boiler operation. The boiler tube was repaired and the Plant returned to operation on October 11.

Primary System Vent Valves

The reactor was shut down shortly after startup on October 11 to repair a failed thermocouple on a newly charged coproduct tube and to determine the source of a primary coolant system leak which was causing excessive moisture carry-over into the 105 Zone I exhaust air system. The leak was traced to two open primary system vent valves in cell 3. Closure of the valves established loop makeup at normal levels and corrected the moisture carry-over problem.

Electrical Grounds

Multiple grounds in the reactor building 125 V-DC electrical system occurred on October 23, following the failure of a valve seat in the confinement fog spray raw water supply which covers the right rear pipe space and the D elevator shaft. The rubber valve seat had split, allowing enough leak through the 6-inch butterfly valve to wet down electrical junction boxes causing numerous grounds. The cause of the failure was attributed to over tightening the valve flange nut.

Rear Face Crate Weld Repair

Temporary repairs were again made to the cracked rear face crate weld to reduce helium loss. The effectiveness of the repairs is not yet known.

Charge-Discharge Equipment

Two hundred sixty-four fuel columns were charged during the October charge-discharge outage without any major charging machine problems. Installation of a replacement set of drive rollers on the machine is about 50 percent complete. The drive mechanism has been fabricated and is ready to be installed.

Horizontal Control Rod Tip

Horizontal control rod tip 47, which previously failed from overheating, was removed from the reactor to shielded storage for metallurgical evaluation. A new rod tip was installed.

Fuel Enrichment Tester

Installation of the fuel enrichment tester on the monotube loader was completed. Training on its use is being provided by representatives of Battelle Northwest, who developed the unit. Operational use is planned for early November. The function of the tester is to check the enrichment of the fuel elements as they are loaded onto the monotube loader.

Module Tester

The rupture monitor module tester now has procedures established and specifications written to the point of being able to test, troubleshoot and repair 8 of the 14 types of modules. Procurement requests have been issued for the necessary additional and replacement equipment.

OUTAGE EXPERIENCE

<u>Date</u>	<u>Outage Type</u>	<u>Duration</u>	<u>Reason</u>
10-3	Scheduled	137.5 hrs.	Charge spike tubes for reactivity and maintenance
10-8	Scram	53.7 "	Malfunction of CONDV-243 valve caused low flow trip
10-11	Unscheduled	6.4 "	Failure of temperature monitor
10-13	Manual scram	32.4 "	Fuel failure - tube 0252
10-16	Manual scram	22.3 "	Fuel failure - tube 0647
10-18	Manual scram	13.9 "	Fuel failure - tube 1456
10-20	Manual scram	19.1 "	Fuel failure - tube 1356
10-23	Unscheduled	17.2 "	Failure of confinement fog spray valve, permitting water to enter electrical junction boxes, resulting in ground of 125-V DC electrical system
10-25	Manual scram	<u>151.8</u> "	Fuel failure - tube 0957
Total Outage Hours		454.3	

Production losses were ascribed to:

	<u>Hours</u>
Fuel failure	86.4
Charge-discharge	165.8
Process equipment failures	70.9
Maintenance	81.7
Production tests	30.0
Startup preparation	<u>19.5</u>
Total	454.3

Figure 1 compares production availability and time operated efficiency over the past several months.

Figure 1. Production Availability, Time Operating Efficiency and Steam Availability

Production availability measures the extent to which the N-Reactor approaches full-time production within internally controllable limitations. It excludes external conditions such as BPA failure and shutdowns required by the Washington Public Power Supply System.

Time operated efficiency (TOE) measures the extent to which the N-Reactor approaches full-time production regardless of shutdown causes.

PROCESS PHYSICS

Maximum authorized power level was not attained during October. Power level was limited by reactivity considerations at the beginning of the month. Equilibrium status was not attained after the scheduled outage of October 3 because of the frequency of subsequent outages.

IN-PLANT TESTS

The status of production tests is as follows:

<u>Production Test</u>	<u>Status</u>
PT-NR-2, Routine Graphite Sample Irradiations (HW-80369)	Continuing. Four graphite boats being irradiated in rod channel 74. Ball channel 60 also contains samples.
PT-NR-3, Routine Monitoring of Graphite Oxidation in N-Reactor (HW-81478)	Continuing. Approximately 20 graphite oxidation samples will be charged when Zr-2 samples (PT-NR-69) are discharged (estimated in November).
PT-NR-4, Evaluation of Monitor Column Fuel Elements in N-Reactor (RL-NRD-218)	Continuing. Supplementation of this PT necessary prior to initiation of PT-NR-75 to provide possible equipment changes to accommodate Mark I fuel. 17 columns discharged for examination during October.
PT-NR-6, Isotope Producing Rods in N-Reactor (HW-81339)	Continuing. Discharge and recharge of targets from rod 23 was accomplished on October 6. Target inspection is scheduled to evaluate recommended residence time for future charges.
PT-NR-8, Coproducer Demonstration Test, 1.25 (HW-81327)	Final report in preparation

<u>Production Test</u>	<u>Status</u>
PT-NR-10, N-Reactor Corrosion Monitoring (HW-82136)	Continuing. Samples are being exposed in 2 secondary system facilities; additional facilities are planned.
PT-NR-12, Copper Alloys in the N-Reactor Primary System (HW-82241)	Continuing
PT-NR-13, Coproducer Demonstration Test, 1.96 (RL-NRD-299)	Continuing. 4 columns discharged and examined in October. Irradiation is continuing on 2 columns containing British alloy fuel.
PT-NR-14, Irradiation of Th-U Crud Monitor Elements in N-Reactor (HW-82385)	Final report being prepared.
PT-NR-24, Plant Service Boiler Base Load Reduction Test Program (HW-84095)	Continuing. Current testing phase is complete with interim results issued. Next phase of testing awaits development of 1/4 sized burner tips.
PT-NR-27, Steam Generator Moisture Carryover Detection (HW-84238)	Test results being analyzed.
PT-NR-30, Evaluation of Ammonia for Controlling Oxygen in the Graphite Cooling System (HW-84232)	Final report being prepared
PT-NR-32, Exposure of Corrosion Coupons in N-Reactor Process Tubes (HW-84367)	Continuing. Corrosion coupons on 10 columns were re-installed following charge-discharge during October.
PT-NR-33, In-Reactor Enthalpy Imbalance Measurements (HW-84401)	Continuing. 2 thermocouple trains were charged to monitor coproduct columns on October 6. 12 thermocouple trains now in-reactor -- 2 trains available for charging.
PT-NR-36, Simulated Driver-Target Element (RL-NRD-55)	Final report being prepared
PT-NR-41, Evaluation of Direct Addition of Ammonia to Graphite Cooling System (RL-NRD-297)	Final report being prepared

<u>Production Test</u>	<u>Status</u>
PT-NR-46, Primary Drive Turbine Speed Reduction Tests (RL-NRD-427)	Final report being prepared
PT-NR-55, Evaluation of Flow Nozzle Tap Relocation in Steam Generator Steam and Feedwater Lines (RL-NRD-581)	Final report being prepared
PT-NR-57, Coproducer Target Element Evaluation (RL-NRD-657)	Final report being prepared
PT-NR-62, Evaluation of Single-Pass Filtered Water Cooling (RL-NRD-668)	Approved. Test to start anytime reactor transfers to single-pass filtered water cooling
PT-NR-63, On-Reactor Tests for New Resistance Temperature Detectors (RL-NRD-695)	Continuing. 7 integral connector RTD's are presently being evaluated on the reactor.
PT-NR-64, Exposure of Zircaloy Coupons in N-Reactor Process Tubes (RL-NRD-716)	Continuing. One column of fuel was charged on October 4. This column to be discharged during PT-NR-75 fuel transition
PT-NR-65, A Test to Demonstrate the Operability of Rupture Monitor System with the Helical Sample Chamber Design (RL-NRD-726)	Approved. Test is awaiting arrival of tubing.
PT-NR-66, Coproducer Demonstration Test, 2.1 (RL-NRD-830)	Continuing. Supplements 7,8 and 9 to this PT issued during month. Also, AEC approved expansion of test from 45 to 50 columns (RL-GEN-1216) on October 28. Initial columns of fuel fitted with spring-stop supports and charged on October 6. Examination of a column of fuel fitted with targets with tandem interference fit supports did not disclose any fretting corrosion.
PT-NR-67, Cobalt-60 Production in N-Reactor (RL-GEN-948)	Continuing. Discharged cobalt target on October 5. Examination of target, spacers and tube for possible fretting is continuing.

<u>Production Test</u>	<u>Status</u>
PT-NR-68, Pu-238 Demonstration Program (RL-NRD-793)	Continuing. Initial column of test fuel and its supporting spike discharged during October.
PT-NR-69, Irradiation of Tubular Zr-2 Samples in N-Reactor Gas Atmosphere	Continuing. Zr samples experienced higher than anticipated temperatures and are scheduled for convenient discharge.
PT-NR-70, Measurement of Confinement System Filter Efficiency by Use of Condensation Nuclei Detector (RL-NRD-847)	Final report being prepared.
PT-NR-74, Experimental Determination of Effects Produced by Increased Fuel Support Diameter (RL-GEN-979)	Final report being prepared.
PT-NR-76, Irradiation of 8-Inch Fuel Elements for Subsequent Fuel Failure Testing (RL-GEN-1018)	Continuing. 3 columns of fuel charged in September. Additional columns to be charged.

EQUIPMENT MAINTENANCE AND PLANT MODIFICATIONS

A summary of the major equipment and construction programs in progress or scheduled follows:

<u>Title</u>	<u>Cost</u>	<u>Funds Assigned to GE</u>	<u>Percent Funds Expended</u>	<u>Status</u>
• CGN-182 - Replacement of 5 KV Cable, 100-N	\$165,000	\$165,000	75	All cable received. 16 circuits of 25 have been tested and replaced as necessary.
• CGN-184 - Phase I, Coproduct Fuel-Target Fab. Equipment	203,000	203,000	99.5	Physical Completion Notice issued 7-25. All exceptions completed 9-16; project complete at less than authorized funds.
• CGN-185 - Steam Gen. Isolation Valves (16 valves)	215,000	215,000	0	Installation of 16 valves authorized by RL00. Design work started 8-24. Bids on valves opened 9-13. Darling Valve & Mfg. Co. is low bidder at \$6,224/valve. Delivery of first 4 valves scheduled for January 1967.
• GCP-400 - Backup Boiler Facility	865,000	0	0	Funding to be by WPPSS; Burns & Roe to perform A-E work. Technical requirements issued by RL00.
• GAP-401 - Upgrading Fire Protection, 100-N	150,000 (revised)	2,600	0	Total project authorized October 6; to be managed by AEC.
• GCE-402 - Coproduct Dem. Prog., Phase II, Irrad. Fuel-Target Hand., Stor. & Shipping Equipment	609,000	165,000	68 on auth. basis; 18 on total est. basis	Orders for Mark II stripper, 500 canisters and 2 tritium monitors placed. Project proposal revision for total funds submitted to AEC on October 3, routed for authorization.
• Fuel Spacer & Refuse Disposal Facilities	95,000	0	0	Design criteria approved by Design Council on 8-30. Project proposal completed and being reviewed.
• ERW Tank By-Pass and System Improvements	165,000	0	0	Project Proposal approved by Design Council on 8-30. Reliability review in progress.
• General Isotope Prod. Facility	485,000	0	0	Feasibility and justification for preliminary project proposal is being evaluated.
• Corrosion Test Facility	40,000 (revised)	40,000	73	Loops 4 and 5 completed. Work in progress on loops 1, 2 and 3.

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FUEL MANUFACTURE

PRODUCTION STATISTICS

<u>Input (Extrusions)</u>	<u>October</u>	<u>September</u>	<u>CYTD Total</u>
0.95% Mark A (outer)	0	0	451
0.95% Mark N (inner)	0	0	526
0.95% Mark D (single tube)	0	0	19
1.25% Mark A (outer, spike)	0	0	258
2.1% Mark E (driver)	271	39	789
2.1% Mark F (driver)	44	316	360
Total Extrusions (Uranium)	315	355	2,403
Total Tons	65.8	75.9	473.2
Percent of Forecast	93.1	84.4	99.0

Output (Finished Production)

0.95% Tube-in-Tube	0	0	9,094
0.95% Single Tube	0	0	33
1.25%-0.95% Tube-in-Tube	0	0	2,991
1.96% Coproduct	0	0	0
1.25% Coproduct	0	0	-10*
2.1% Mark E (driver)	130	481	1,013
Total Assemblies	130	481	13,121
Total Tons	2.0	7.6	254.3
Percent of Forecast	8.5	13.4	92.2

Uranium Utilization, Percent

Mark E, 2.1%	78.5	84.8	75.7
Mark F, 2.1%	70.4	80.0	76.9

*Fuels returned for rework include some accepted in previous months.

N-FUEL PRODUCTION SUMMARY

Production Operation - Coextruded Fuels

Output production was limited to the completion of fuel assemblies for production test No. 66. Final processing (support welding and autoclaving) of fuels for production test No. 75 was initiated on October 24, but no material was completed through final assembly before month end.

In-process inventory status of production test No. 75 driver fuels is as follows:

Completed through process, awaiting support welding	8,213
Completed through autoclave, awaiting assembly	499

NLO Strike Assessment

Production operations were resumed at National Lead of Ohio on October 24 after a 3-week work stoppage. The work stoppage will have no effect on HAPD ability to meet current coproduct schedules.

Yields

Process yields in the coextrusion shop achieved record highs for October. Two major factors are 1) a continuing low defective rate for all end-associated causes, and 2) continued improvement in all defective categories, especially in the "human error" type of category. It is apparent that continued operation on a single-product model is providing some relief from set-up, changeover and recalibration types of errors.

Production Operation - Target Fuels

Production of sintered pellets for target elements of coextruded fuel proceeded on schedule throughout the month. However, throughput at subsequent stations was limited to less than 50 percent of schedule because of problems with the pellet sizing operations. Additional equipment (cutoff saw and centerless grinder) required to attain desired production rates is being installed.

Status of in-process material inventories for production test No. 75 are:

Total assembled in Zircaloy	2,559
Completed through autoclave	317

PROCESS DEVELOPMENT

Uranium Billet Quality

Investigation is continuing of uranium billets containing unusual amounts of large carbide particles. All fuels in which the carbon content exceeds 1000 ppm anywhere in the fuel cross section will be segregated. Examination of the metallographic samples from each coextrusion and supplementary analysis for carbon were made of any suspicious areas in the cross section. These analyses show that the maximum carbon concentrations occur at the outer diameter and near the end of the billet or at the start of the billet. To date, nine coextrusions have been found with carbon content exceeding the limit. The problem was caused by high temperature casting and recycle casting of top crops. Casting temperatures were lowered from 2775°F to 2725°F, and the recycle of top crops was suspended. However the effect of these changes has not been fully evaluated because of the interruption in feed site activities by the recent strike at the Fernald Plant of National Lead of Ohio.

To ensure control, a specification has been transmitted to NLO through the local office of the AEC. This specification requires that an additional sample be taken between the B and C billets, and that the outer diameter of the sample be analyzed for carbon. Billets will not be acceptable unless an analysis shows less than 1000 ppm of carbon. This sampling and analysis is in addition to the regular sample taken just below the top of the ingot crop. The carbon specification at this location is 300 ppm minimum and 750 ppm maximum.

Support Characteristics

The locking clip and stop have been moved farther into the bore of the driver fuel to permit centering the solid support 3.75 inches from the end of the assembly at the mid-point of the longer (1.2-inch) leg of the W-spring. Fixtures have been designed and built for stress relaxation tests on the W-springs. The tests will be conducted on a control basis after autoclaving. The fixtures are undergoing evaluation prior to establishing the quality control material standards for incoming W-springs.

A tensile testing program to determine the mechanical properties of Zircaloy strip has been established with Battelle Northwest Laboratories and is tabulated below. The testing is scheduled for completion by mid-November.

W-SPRING VARIABLES TO BE TESTEDA. Thickness

50-mil
60-mil
75-mil

C. Material Origin

PRTR tubing
Thin-clad shell
Thick-clad shell
Sheet bar
Wah Chang strip
(if obtainable)

B. Material Condition

As-received
Cold-worked
Fully annealed

D. Test Temperature

Room temperature
300°C

Inner Weld Overhang Removal

The diameter of the fuels is non-uniform in the area of the end cap and below the cap where grain structure transformation from braze heating has occurred. Figure 2 is a typical profile of the cross section of a coproduct driver fuel showing the differences in inside diameter caused by the brazing operation. Figure 3 shows the same profile after end welding and the characteristic overhang of the weld bead which increases the end restriction more than is normally caused from brazing.

In assembling the target and driver fuel this diameter restriction causes over-deflection of the W-spring and reduces the force exerted by the W-spring after assembly. The function of the W-spring type support hardware is to exert sufficient force against the internal surface of the driver fuel to maintain rigidity of the target during in-reactor service.

Evaluations established that the end restriction on the approximately 4000 unautoclaved fuels processed past end welding could be removed without reducing end closure integrity. The weld overhang was ground from the inside diameter weld at the points where the supports enter the fuel inside diameter. The dashed lines in Figure 3 show the approximate amount of the restriction removed by grinding. Process requirements were established to remove sufficient overhang by grinding so that dimension B, Figure 3, would not exceed dimension A by more than 0.01-inch. Prior to welding the difference in these dimensions is about 0.006-inch. The operation is controlled by mechanical devices that prevent excessive grinding or removal of weld metal over the braze joint between cap and cladding. Concurrent with the weld grinding process development work, end-welding equipment and techniques are being investigated to eliminate weld overhang.

Figure 2. Inside Diameter End Restriction from Brazing

Figure 3. Inside Diameter End Restriction from Welding

Braze Ring Quality

As previously reported, a small (two to three percent) number of the braze rings do not melt and flow uniformly, but leave a lumpy appearance on top of the cap. This condition sometimes results in rejected fuel because insufficient braze material flows down under the cap. Correlations have previously been found between this melting condition and the hardness of the braze rings. Additional data have now been obtained showing that the hardest rings occur at the bottom of the cast cylinders from which the braze rings are parted. Since the bottom of the casting freezes first, it is postulated that the non-uniform melting of the braze rings will correlate with non-equilibrium alloying of the Zircaloy and beryllium from insufficient time in the molten condition during the fabrication process.

Hardness data for rings from three heats were obtained. Four rings from the top, center and bottom of the eight individual cast cylinders were tested. Apparently the metal in the bottom of the cylindrical molds freezes quite rapidly producing the hardest rings and a fine grain structure. The metal in the upper portions of the molds cools rather slowly producing a softer ring with a larger grain structure. The center portions of the cylinder evidently cool faster than the tops since the hardness of the rings originating from this area is similar in hardness to those from the bottom of the tube. The average (Rockwell) hardness values obtained from three tests on each of twelve rings from the top, middle and bottom portions of eight tubes contained in three heats were:

Top	30.5 R _c
Middle	36.0 R _c
Bottom	37.3 R _c

There is an indication that the rings from the top portions of cylinders seven and eight are producing the softest rings. These are the last cylinders filled and possibly these molds have been warmed sufficiently to retain liquid metal for a longer period of time.

Several samples of these rings have been scheduled for oxygen, nitrogen, hydrogen, carbon, and spectrographic analyses. This information will be used to determine if conditions other than cooling are affecting the hardness and the melting characteristics of the braze rings.

Component Manufacture

The on-site manufacture of inner Zircaloy cladding components for coproduct drivers is proceeding at a rapid pace with high yields. Target dates for the various phases of the program call for completion of all primary extrusions in late November, and all final extrusions by late

December. Based on these extrusion schedules, it is expected that the last of the required 2450 pieces will be delivered during January. The complete cladding manufacturing status report is included below.

Mark II Inner Cladding Component
Production Status

	<u>October</u>	<u>September</u>	<u>CYTD Total</u>
11-inch Primary Extrusions	30	0	74
7.5-inch Final Extrusions	49	19	107
Acceptable Cladding Delivered to Process	222	330	762
Piece Yield, Percent	92	93	90.5
Material in Process*			
Cut Pieces	580		
7.5-inch Billets	40		
11-inch Billets	6		

*This in-process material represents approximately 1180 acceptable finished components.

Sintered Pellet Sizing Operation - Target Fuels

Grinding, squaring and sawing of sintered pellets limited production of target assemblies during the month. In general, this reduction in output was caused both by a general shift in pellet hardness due to calcium carbonate additions for density control and a general deterioration of grinding and sawing equipment. The pellet hardness characteristics associated with additions for improved sinterability will continue because of the necessity to maintain pellet density within narrow limits, and equipment capability is being upgraded to meet the resulting requirements.

Sintering Time

Sintering time at the established process temperature has been reduced from ± 16 hours to ± 12 hours based on results of successful production test

evaluations. This reduced sintering time provides one additional sintering cycle per week from each furnace. This provides a net gain in sintering throughput capability of 17 percent with no loss in quality.

Pellet Density Control

Control of sintered pellet density is a primary mechanism for controlling isotope level per given volume in the reactor. Production experience, to date, shows an over-all sintered pellet density average of 77 percent of theoretical (2.62 g/cc). However, difficulty has been experienced in minimizing the range of sintered pellet densities obtained with varying powder lots. Specifically, it was not possible to obtain the required sintered densities of 78 ± 5 percent with some low-bulk density powders.

Attempts to improve powder sinterability by additives have shown that calcium carbonate (CaCO_3) is most effective with a minimum effect on product purity. Calcium carbonate decomposes to the oxide, CaO , at $\sim 900^\circ\text{C}$. It is believed that the oxide then enters into several possible complexes with available alumina and lithium aluminate. It is these reactions which aid sintering by providing intermolecular motion during the sintering process.

Development will be continued to the point where each incoming powder lot will be characterized as to ball milling time and CaCO_3 addition for required sintered density.

CONVERSION

The Washington Public Power Supply System generating plant continued to operate at approximately half load, as steam was available, during October. Under half-load operating conditions, some problems of instability were encountered with the automatic voltage regulating apparatus. Because of this, the automatic voltage regulation has been by-passed and a mode of manual control is being utilized. The manual operation was quite successful until October 16, when the Bonneville Power Administration experienced a voltage surge and the Supply System operators were unable to respond with manual correction in the time required. The result was a dual-turbine trip. The N-Reactor rode out the trip and continued operation. Corrective action to minimize the chance of recurrences of trips of this nature has been to limit power output to 400 Mwe, which will provide additional time for response to voltage changes while operating with manual voltage regulation. The generating plant had been routinely operating up to 430 Mwe.

The latest information from WPPSS/BPA indicates that the step-down transformer installation at Covington will be completed and ready for initial testing during December. This now places completion of N-4 testing in January 1967.

Dual-Turbine Trip

Successful dual-turbine trips from 400 Mwe were performed on October 12 and 16. A good measure of the severity of the test was reflected in the 20 psi pressure rise in the main steam header, followed by a gradual pressure reduction to setpoint. The pressure reduction was at such a rate that little or no positive rate of reactor power rise was detected on the nuclear instrumentation. As a result of excellent system tuning the transient effects, caused by the export steam load rejection, were minimized. In addition, the coordinated operation of the dump condenser steam admittance valves contributed to the success of the test.

Testing is now scheduled to be resumed on December 20 with a double turbine trip at 600 Mwe, and a 1-of-2 trip, a 2-of-2 trip, and a scram from 800 Mwe power level.

Export Steam Availability

An analytical investigation (cited in document RL-GEN-1184) was made of the availability of export steam to the turbines as a function of steam pressure, reactor power level, and the number of operating heat exchanger units. Curves were prepared representing available steam versus reactor power level for four, five, and six-loop operation with Mark I fuel.

TECHNICAL ACTIVITIES

1. BASIC PRODUCTION MISSION

1.1 Fuel Performance Evaluation

Five fuel failures (Nos. 24 through 28) occurred in October. All were in Mark I fuel. Two were due to fretting by the hard-sized buggy spring and three were end-cap associated. Data are given in this tabulation.

FUEL FAILURES

Failure Number	Tube Number	Date of Failure	Tube Power (Kw)	Exposure		Fuel Type	Type of Failure	Hours Loss	Prod. Loss
				Tube (MWD/T)	Element (MWD/T)				
24	0252	10-13-66	2800	2840	879	Mk I	Fretting	17.0	2833
25	0647	10-16-66	3865	2234	290	Mk I	Fretting	22.4	3733
26	1456	10-18-66	3710	2225	2740	Mk I	End Ass.	13.9	2316
27	1356	10-20-66	3900	2880	3267	Mk I	End Ass.	19.1	3183
28	0957	10-25-66	2944	2797	3447	Mk I	End Ass.	14.0	2333

There have now been 28 indications of fuel element failure in N-Reactor since its startup. Of these, all but one have been confirmed as actual ruptures. Among these 27 confirmed ruptures, 20 have occurred in Mark I fuels. Among these 20, 5 were caused by foreign object fretting, 6 were caused by buggy-spring fretting, and 9 were caused by end-closure failure.

No foreign object fretting failures have occurred since January 1966, and this lends support to the belief that this problem is under control. The general features of the buggy spring failures are believed to be understood, and in half or more of the cases failures can be associated with buggy springs which were faulty at the time of charging into the reactor. The final hard-sizing operation may account for most of these faults.

All six of the buggy spring failures in Mark I fuel were at the unlocked end and in short fuel. Four of the failed assemblies had only two sets of supports. The 18-inch and 24-inch lengths have three sets of supports, and the 12-inch and 15-inch lengths have but two.

Tests to date indicate that use of the W-type spring-stop support would eliminate this support problem. A priority effort to validate this concept is in progress. An intensive study is also under way to determine how to utilize more efficiently the fuel now in storage, some of which is in short lengths.

The failures which are associated with end closures are less well understood. Of the observed failures, one was definitely the result of a faulty end cap which contained an extrusion defect; another was ascribed to a porous braze, although a direct water path to the core was not proven conclusively. No definite cause has yet been assigned to the other seven failures although they bear some physical resemblance to the case in which a porous braze was the apparent cause. It is desirable to be able to forecast the number of fuel failures to be expected. A study of the available data by statistical methods suggests the probability that end-closure failures correlate with residence time in the reactor, or upon the tube-average exposure. A similar relationship appears to exist for the buggy-spring failures. These analyses are being refined and will be reported at a later date.

There were 178 columns of fuel (with average exposures exceeding 2200 Mwd/t) discharged to reduce the incidence of failures in fuels which have sustained high-level exposure conditions.

1.2 Zircaloy Process Tubes

Work has continued on the evaluation of effects of service on process tube 1756. Most of the work has been completed and will be reported by Battelle Northwest Laboratories. Stress rupture tests remain to be completed. Results of the evaluation to date indicate that:

- a. Zircaloy strength was increased as a result of irradiation.
- b. No change in crack propagation has occurred as a result of service.
- c. Marks associated with rear face dummy spacers have no degrading effect upon tube strength, and showed no evidence of hydride formation at or below the service.
- d. Oxide film thickness measurements indicated no radiation enhancement of corrosion rate on either the inner-diameter or outer-diameter surfaces.
- e. An increase in hydrogen concentration from 15 to 40 ppm appeared in the section between 30 inches and 108 inches from the front rolled joint. The hydride level was well below the serious threshold. Possible hydrogen sources are being investigated and preventive steps are under consideration.

1.3 Graphite Studies

Dr. Peter Nettley, Assistant Director of the Reactor Materials Laboratory, Culcheth, England, visited Hanford to discuss graphite programs of mutual interest. The British effort is being reduced to a considerable extent. Dimensional changes in graphite have been resolved largely by a choice of operating temperatures. The maximum graphite temperature is below 500°C. This coupled with relatively low planned lifetime exposure negates any serious concern over stack stability. Radiation induced oxidation in carbon dioxide has been accommodated by using methane as an inhibitor. Thus, the British feel they have solved the graphite problem. Work in progress and planned includes additional work on radiation-induced creep, effect of in-reactor oxidation on strength and some very high exposure work.

1.4 Steam Generator Program

The program for adding phosphate to a 304 stainless steel tubed steam generator to arrest tube cracking is awaiting program acceptance and funding. In the meantime, steam generator tubing materials are being autoclave-tested in different phosphate environments to determine potential damage mechanisms, if any.

Four sections of an Inconel 600 tube removed from steam generator 4A after about a year's operation showed no indications of cracking. Minor surface etching (≤ 0.5 mil) was observed, but appeared to be no different from the condition of the tubes before installation.

Two samples of Inconel 600 tubing removed from a Tennessee Valley Authority Paradise plant steam condenser were also examined and showed no cracking. The pieces exhibited surface etching similar to the cell 4 tubing, but on the cooling water side of the tubing.

1.5 Univac Tube Limits Program

Process tube coolant outlet temperature limits were calculated for the All Tube Monitoring System for the following conditions:

- Inlet temperature from 320-420°F, incremented by 20 degrees.
- Flow ratio from 0.45 - 0.95, incremented by 0.05.
- Rear riser pressure 700-1600 psi, incremented by 100.
- Enthalpy imbalance factor of 0.80.

Limits for the Zone Temperature Monitoring System were based on:

- Inlet temperature from 320-420°F, incremented by 20 degrees.
- Flow ratio of 1.0.
- Rear riser pressure 700-1600 psi, incremented by 100.
- Enthalpy imbalance from 0.74 - 0.76, incremented by 0.01.

The major objective was to test all the program logic and to get an estimate of computer time for each calculation of maximum temperature. A total of 840 maximum temperatures were computed in one minute six seconds; which is about 80 milliseconds per run.

1.6 Shutdown Coolant Flow

An analysis was made of the adequacy of a single primary pump for shutdown cooling, assuming the pump to be driven by its pony motor at quarter speed, or 900 rpm. The maximum heat removal capacity was calculated to be about 3.9 percent of design reactor power. Since the 1-pump operation would not be used intentionally until at least a day after shutdown, the reactor power level would be decayed well below the heat removal capability of the postulated operation. However, the calculations indicated that the net positive suction head requirements of the primary pumps would be met minimally, assuming the pressurizer is vented to atmosphere and the liquid level is normal. The pressurizer level would be increased to its maximum height to gain about six feet of additional head.

1.7 Advance Study - Reactor Operation at 6000 Mw

A document (RL-GEN-1180) was issued setting forth improvements visualized for the reactor in the coming six to eight years. It is anticipated that with an adequate development program, power could be increased by 50 percent and full utilization could be made of the cross cooling tubes and spare tube channels for target irradiation. Plutonium, tritium and neptunium could be produced at attractive costs, with steam supplied at 8-10 cents per million Btu for an additional 400 Mw turbine-generator unit.

Based on a preliminary examination of the thermal hydraulics aspects of N-Reactor operation, it is concluded that 6000 Mw operation is possible provided the non-bulk boiling criteria could be removed. The first indications are that the non-boiling criteria can be removed for tube powers up to 7500 kw. The fuel element design can be a modification of either the Mark I or the Mark II designs. The Mark I dimensions must be altered to correct the current enthalpy imbalance, and an inner fuel cylinder must be added to the Mark II to permit the necessary tube power. Since bulk boiling generally produces flow instability, boiling burnout is probably the greatest concern. Thus, if bulk boiling is permitted, it must be shown that the instability flow reduction will not result in an oscillating flow, with premature burnout, or in a steady-state burnout. At this time, the lack of comprehensive experimental data precludes a reliable prediction of burnout in the 7500 kw tube. However, extrapolating known data does indicate that flow oscillations and burnout will not occur in the 7500 kw tube. Thus, until experimental data can be obtained, the best current estimate is that N-Reactor operation at 6000 Mw with some bulk boiling is possible.

Calculations were completed to define the broad parameters of a fuel loading designed to operate at 6000 Mw, and produce weapons-grade plutonium. The study postulates that the reactor process tube wall thickness will be decreased by 30 percent to provide space to accommodate an enlarged fuel necessary for improved conversion efficiency. This would require the development and use of special zirconium-columbium alloys for process tubes. Results of the study are summarized in the following tabulation.

Long-Range Fuel Design Estimate

	<u>Mark I Fuel, Design Power</u>	<u>Long-Range Fuel 150% Power</u>
Process tube inside diameter	2.712 inch	2.878 inch
Fuel geometry	tube-tube	tube-tube
Fuel weight	21.3 lb/ft	32.7 lb/ft
Maximum uranium temperature	450°C	600°C
Average fuel residence time	93 days	90 days
Plutonium yield	0.840 g/Mwd	0.92 g/Mwd
Feed enrichment (pile average)	0.974 + % U ²³⁵	1.032 + % U ²³⁵
Maximum graphite temperature	~ 620°C	~ 720°C
Maximum process tube temperature	600°F	590°F
Reactor control	adequate	adequate - slight supplement required
Confinement of fission product	adequate	design attention required

1.8 Nuclear Health and Safety

Criticality Safety

No violations of Nuclear Safety Specifications were logged during audits which were made during the month.

Safety Analysis Report

The safety analysis report covering handling of production quantities of Mark II fuel in the N-Reactor facility is approximately 50 percent complete.

Calculation of Underwater Criticality Excursion

Calculations to define the consequences of an accidental underwater criticality in the fuel storage basin are approximately 50 percent complete.

2. COPRODUCT MISSION

Coproduct Demonstration Program Summary

The implementation plan for the coproduct program has been redefined to reflect the delays introduced by the 45-day strike and the desire for additional testing of an improved target support system. A new planning schedule for the two reactor load demonstration program has therefore been developed. Rescheduled key action dates are shown below:

- Start fuel assembly October 1966
- Start of reactor loading December 1966
- Complete transition to coproduct March 1967
- First coproduct to basin March 1967
- Initiate shipment of targets to SRP June 1967
- Complete discharge of first reactor load of coproduct June 1967
- Obtain conversion ratio data from test fuel loadings June 1967
- Complete discharge of second reactor load of coproduct October 1967
- Initiate shipment of drivers to Purex January 1968

The only aspect in which technology for the coproduct fuel has not been essentially confirmed is the target support system. In the reactor, both thermal and irradiation-induced changes tend to expand the driver from the target. The spring must accommodate the annular expansion and still maintain rigidity to resist the hydraulic forces. To find the minimum spring forces to overcome the hydraulic forces that are present in the BNW Thermal Hydraulics Test Facility, assemblies with W-spring forces ranging from 10 to 30 pounds were exposed to loop conditions for 11 days. Detailed fuel examination revealed that two of the fuel assemblies with 10-pound force per spring showed evidence of slight fretting corrosion between the rigid support and the unlocked stop. Six elements assembled with springs predeflected to give essentially zero spring force withstood a 4-day exposure in the Thermal Hydraulics loop and showed no evidence of fretting. To introduce more conservatism and larger margins against failure, the W-type spring-stop support system will replace the buggy spring system. This decision, in part, is based on the fact that fuel assemblies using prototypical W-springs in the spring-stop support system have successfully withstood accelerated test loop operation conditions (flow rates over 30 percent above those to be expected in-reactor) for more than 60 days. Stress relaxation measurements on W-springs at 300°C continue to show a normal creep behavior of a slow decrease in force with increased time at temperature. Extrapolation to a time equivalent to a goal exposure of 1700 Mwd/t indicates a spring force of 30 pounds. After a full year at process temperature, the spring should exert a force of at least 25 pounds. During October, 42 columns of fuel assemblies using the W-type spring-stop support system were charged into N-Reactor.

2.2 Coproduct Extraction Studies

Extraction studies were finished at Savannah River Laboratories with the successful completion of the lower temperature extraction run at 750°C in the presence of excess aluminum. The results showed that 99 percent of the contained tritium was recovered. Extractions in plant equipment to confirm laboratory results and check out plant process equipment will be run during the first half of December 1966.

2.3 Mark II Target Support Flow Testing

Flow testing of Mark II fuel continued during the last month and is summarized in the following tabulation.

Mark II Target Support Flow Testing

<u>Test Series</u>	<u>Fuel Type</u>	<u>Target Support Type</u>	<u>Position Loaded</u>	<u>No. of Elements</u>	<u>Test Conditions</u>	<u>Hours at Test Conditions</u>
12	Mk II	Spring-stop (10 lb loaded springs)	Downstream	3	340 gpm 570°F	269.5
"	Mk II	Spring-stop (20 lb loaded springs)	"	5	"	269.5
"	Mk II	Spring-stop	"	4	"	430
"	Mk II	Spring-stop (hand-formed)	"	2	"	1,008

Testing was also started on Series 13 fuel which consists of the following types of target supports:

<u>Support Type</u>	<u>Number of Elements</u>
Spring-stop (Zero-loaded springs)	4
Spring-stop (Dogbone to inside diameter variations)	6
End spider	4

2.4 W-Spring Preload Gauge Development

It is essential to the proper operation of the coproduct fuel configuration that the target-driver assembly provides a preloading force on the W-springs beyond a specified value in order to ensure positive in-place force following the differential thermal expansions of the driver and target, and the reduction in the spring's effectiveness as a result of long-time temperature creep. In order to provide these requirements it will be necessary to obtain a measurement of the pre-loading on the W-spring in the target-driver assembled state. A development program to provide such a measurement device has been initiated. The program can be divided into several logical steps, each leading toward the ultimate achievement of a workable production line-gauging system.

A prototype gauge assembly is being built to determine W-spring preloading from loading and deflection measurements made at the end of the target. The deflection curve for the end W-spring is a function of a large number of target characteristics. This investigation has attempted to segregate these characteristics into the three regions of the deflection of the spring itself, the deflection of the support structure, and the deflection of the target as a simple beam. Additional complicating factors are the hysteresis presented in the deflection curve which causes the target to absorb force without actually moving until the friction forces involved in the peculiar suspension system have been overcome and motion can take place; and the fact that the spring has been stressed into the plastic region, with little recovery. Therefore, it may be difficult to discern the break point which is a measure of the preloading of the spring by applying a force at the end of the target and observing the deflection at the target end.

Now that a reasonable understanding has been obtained of the structural characteristics of the target element, the next step in the program will be to obtain statistical data on the variations of these characteristics among a number of elements. Ten target elements are being subjected to similar testing at both ends to evaluate the statistical variations of the support spring constant, the beam-flexure spring constant, and the W-spring constant itself.

2.5 Special Coproduct Test

A special supplement to Production Test No. 66 was issued to authorize:

- a. In-reactor testing of Mark II fuel utilizing spring-stop target supports.
- b. Increasing from 45 to 50 the number of Mark II fuel columns which can be irradiated simultaneously.
- c. Relocation of the original 45-tube coproduct block (to take advantage of higher flux values).

Forty-two tubes of Mark II spring-stop fuel elements were charged. Three tubes of tandem interference-fit fuel elements previously charged still remain in the reactor. A fourth tube (from the original block) was discharged October 3. Examination revealed no evidence of fretting corrosion.

2.6 Target Warp Testing

Fabricated target elements are tested in autoclaves at an elevated temperature and pressure for corrosion resistance of the Zircaloy components

and for end-weld quality. The emerged elements have been found to have varying degrees of warp. Warped targets inside fuel elements are not desirable under operating reactor conditions because of potential for local hot spots and diametral temperature differences developed which, if excessive, would further increase the warp. The available local burnout margins may possibly be diminished to levels below that of the design criteria. These points of operating thermal performance were investigated to determine the upper limits of acceptability for warp, based on annulus variation associated only with warp of the target and not of the drivers.

The case of a maximum powered tube, 5650 kw, for 4800 Mw reactor operation has been calculated. Results indicate that for a radial warp of 40 mils (corresponding to total indicated reading of 80 mils) at a location of maximum inner annulus heat flux of 785,340 Btu/hr-ft² the diametral coolant temperature difference is 5.1°F; 3.2°F above and 1.9°F below the normal coolant temperature for the given cross section. The numbers are too insignificant to warrant any concern. The highest local heat flux to burnout heat flux ratio reached in the tube for a low-flow trip on 10 percent tube flow reduction is 0.32. This is on the safe side of the design criterion value of 0.63.

2.7 Isotope Producing Control Rod

An isotope (tritium) producing control rod (rod number 23) was discharged and recharged with replacement elements on October 6, for the first time since the initial startup of N-Reactor. About 15 of the 40 elements had accumulated enough exposure to produce an estimated 1.9 liters of tritium per element. Tritium production for the entire rod was estimated to be about 40 liters. The special discharge equipment used to flush the target elements out of the tube worked excellently.

3. INTEGRATED SITE OPERATION

No significant activity during October.

4. NON-DEFENSE PLUTONIUM MANAGEMENT

Sorting to Get 20 Percent Plutonium-240

It was reported last month that some fuel sorting will be required to satisfy the commitment for 50 kg of plutonium containing 20% Pu-240. This will be required because the Mark I fuel will be discharged during the coproduct transition before reaching the exposure required to produce

20% Pu-240 in the entire column (4800 Mwd/t). Figures 4 and 5 indicate the size of the central segment of a column which will contain 20% Pu-240, and the amount of plutonium contained in the segment versus column exposure.

Sorting of fuel columns to obtain the committed product is not expected to be as complicated as originally indicated. Some 100-odd columns of high exposure fuel presently contain 2 lengths of fuel pieces. It may be possible, by judicious arrangement of discharge schedules, to sort these columns automatically with the fuel segregation equipment.

5. TRANSPLUTONIUM TECHNOLOGY

No significant activity during October.

6. PLUTONIUM-238 PROGRAM

Pu-238 Production Calculations

Work continued on the calculation of various features of producing Pu-238 by Np-237 irradiation in N-Reactor.

The production of neptunium from N-Reactor for a 10-year campaign producing plutonium with 20% Pu-240 and 12% Pu-240 was calculated using the CAGE computer program. The production was varied by utilizing Nuclear Fuels Services tails for blendstock and by varying the out-of-core time for reprocessing fuel. The results are plotted in Figure 6.

Increasing the U-236 content in the uranium by introducing Nuclear Fuels Services tails as blending material provides the major increase possible in the neptunium production from N-Reactor, to provide about 80 kg of additional neptunium over the campaign period. Increasing the goal exposure in going from a plutonium of 12% Pu-240 to 20% Pu-240 provides about 20 kg of additional neptunium when using Nuclear Fuels Services blendstock, but less than 10 kg when using Cascade blendstock with 2% U-235. Reducing the out-of-reactor time by a factor of two from 18 months down to 9 months increases the neptunium production by only 15 kg over a 10-year period when producing 20% Pu-240 with Nuclear Fuels Services blendstock whereas on a 12% Pu-240 campaign decreasing the out-of-reactor time by the same factor increases the neptunium production by almost 25 kg.

When using Nuclear Fuels Services material for blendstock, a small amount of enriched uranium from the Cascades must be used to make up deficits. As shown in Figure 7, the enrichment used has little effect on the production of neptunium for either the 12 percent or 20 percent material.

Figure 4. Fraction of Column to be Discharged to Yield 20% Pu-240 from Columns of Varying Exposure

Figure 5. Product Yield From Selective Discharge of Columns of Varying Exposure

When only Cascade material is used as blendstock, significant increases in neptunium production can be achieved by using enrichments higher than 2 percent. For example, 7 percent enriched uranium as blendstock would increase the neptunium production by 30 kg over the 10-year period.

A study is being performed to determine the optimum scheme for the production Pu-238 by the irradiation of neptunium targets in a super-cell array in N-Reactor. Current production at Savannah River Plant delivers a product containing 80% Pu-238 and approximately 20% Pu-239. By varying the concentration of neptunium atoms in the target, it is possible to vary the number of full reactor days required to achieve this level of product purity. As shown in Figure 8, the number of full power days varies from 98 for a 7 w/o neptunium target to 162 days for a 30 w/o target to achieve a product purity of 80% Pu-238. As the concentration of neptunium is increased in the target, the number of targets that must be fabricated per year decreases, the number of columns of standard fuel that must be displaced by neptunium targets decreases, and the number of times that a given quantity of neptunium must be processed through chemical processing decreases. However, on the negative side, the conversion efficiency of atoms of Pu-238 produced per atom of Pu-237 consumed decreases as the irradiation time is increased. Thus, to achieve the optimum scheme for irradiating neptunium targets a balance must be obtained between the decreased costs associated with higher neptunium concentrations in the target and the increased costs associated with the greater consumption of neptunium for a given Pu-238 production.

The number of fuel columns required to irradiate all of the neptunium produced annually in government production reactors would be 83 columns in 1968, increasing to 456 columns by 1975 were a 7 w/o neptunium target to be used as shown in the following tabulation.

Number of One-Half Length Process
Tubes Required for Np-Al Targets

Fiscal Year	Number of Process Tubes		
	7 w/o	15 w/o	30 w/o
1968	83	39	20
1969	139	65	33
1970	203	95	48
1971	259	121	61
1972	317	148	74
1973	359	168	84
1974	419	196	98
1975	456	213	107

Using a 30 w/o neptunium target, 20 columns would be required in 1968 and only 107 columns by 1975.

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Figure 6. Accumulated Production of Neptunium for 10-Year Campaigns in N-Reactor

Figure 7. Effect of Blendstock Enrichment on the Accumulated Neptunium Production for a 10-Year Campaign in N-Reactor

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Figure 8. The effect of Irradiation Time on the Contamination of Pu-238
Produced in Neptunium Targets

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Experimental Irradiation of Np-237

The first target column of the neptunium irradiation program was discharged early in October. The exposure on the surrounding Mark IA fuel columns was:

<u>Tube</u>	<u>MWD/Tube</u>	<u>MWD/T</u>
0756	309.5	1121
0855	314	1138
0857	316	1145
0956	318	1152

The reactivity checks on the test column and supporting Mark IA fuel gave the following results:

	<u>Reactivity Worth</u>
One column of Np-237, 70 g/ft, targets	-0.29 mk
Four columns of Mark IA fuel	+0.12 mk

From this it is concluded that 8 Mark IA columns will be required to support this particular neptunium target.

The return of the neptunium scrap (material from turning and reject elements) has been delayed by an accidental loss of about 700 grams of neptunium in a 180-gallon waste tank. A Monel swage lock in one line leading from the dissolver tank was dissolved by the nitrate acid solution as the tank was drained. The 180 gallons of solution were reprocessed and the neptunium material reclaimed.

The discharged Al-Np elements in the basin, as well as those in the three associated spike columns that were concurrently discharged, were weaseled and inspected in the basin. The weasel data appears to be anomolous indicating that the Al-Np elements were actually blacker than anticipated. This is currently being investigated. A review of the input oxide powder analysis shows that it contained neutron absorbing impurities such as Cs (10 ppm), Ag (2 ppm), U, Pa-239, and Pu. Whether these impurities were carried over into the master alloy is not certain at this time. However, the reactivity measurements would indicate that this was the case.

7. URANIUM-233 PROGRAM

Not a Hanford Atomic Products Department program at present.

8. COBALT-60 PROGRAM

The first cobalt fuel element to be irradiated in the graphite cooling system has been discharged. No problems were encountered during the discharge procedures and the sample piece is presently in the water basin in the right inner rod room. The estimated specific activity is 8 curies per grain.

9. POLONIUM-210 PROGRAM

No significant activity during the month.

10. HIGH POWER DENSITY FUEL PROGRAM

No significant activity during the month.

11. REACTOR MODERNIZATION

Not a Hanford Atomic Products Department program.

12. FUNDAMENTAL AND GENERAL STUDIES

Not an authorized program at present.

13. NUCLEAR SAFETY

Emergency Cooling System Studies

Continuing studies are being made of alternate emergency cooling systems for N-Reactor. Initial studies were centered on minimum cost systems to provide additional protection against specific incidents. The emphasis now is on the establishment of systems which would be independent of the present reactor cooling systems and would therefore provide more comprehensive protection. Two basic systems have been considered. One would essentially duplicate the existing emergency raw water (ERW) system, requiring a new river pumphouse, low-lift pumps, storage tank, and high-lift pumps. The other system would be similar, except that the storage tank and second set of pumps would be eliminated. Both systems would provide a flow of emergency cooling water sufficient to prevent fuel temperatures above the uranium alpha-beta transformation temperature during a transient from recirculation to

once-through cooling. The major deviation in the second system is that it would not have a storage tank as a means of providing preheated cooling water initially. A more extensive investigation of thermal stresses and primary system temperature requirements will be necessary. Two options have been studied which could be applied to either of the two cases. One option would be to provide a separate set of V3 control valves for emergency coolant entry into the primary loop. The other option would be an independent set of control circuits for opening the V3 inlet valves and the V4 dump valves.

14. COLUMBIA RIVER STUDIES

Not a Hanford Atomic Products Department program.

15. WASTE MANAGEMENT

Not a Hanford Atomic Products Department program.

16. OTHER ISOTOPES

No significant activity during the month.

17. HIGHLY ENRICHED FUEL PROGRAM

Investigation of the feasibility of oralloy fuel loadings has been accelerated. Calculations indicate that oralloy loading will definitely be feasible from a physics standpoint; continuing studies will attempt to define a near-optimum fuel geometry and concentration. Currently, an oralloy design of Mark II design dimensions has been found to be feasible; calculations have progressed to an exposure of 20,000 Mwd/t of fuel (equivalent to 38% U-235 burnout) (or 206 full-power days). This design, however, is low in productivity - approximately 80 percent as productive as Mark I fuel. Another calculation, with a more concentrated Li-6 target, achieves a productivity equivalent to Mark I fuel; calculations have only proceeded up to 16 percent burnout, or 104 days at full power. These designs, at most, will require a modest amount of supplemental control for the cold reactivity state.

While the physics aspects of oralloy fueling appear tractable, engineering considerations have not yet been explored. Two obvious points of concern are 1) development of an appropriate fuel alloy, and 2) the change in relative fuel power with exposures, which reaches about $\pm 20\%$ at 20,000 Mwd/t. This may require remotely adjustable variable orificing to maintain high enough coolant utilization.

BURNOUT HEAT FLUXES FOR N-REACTOR
COPRODUCT FUEL ELEMENTS

SUMMARY

Reliable information about boiling burnout* conditions for N-Reactor coproduct fuel elements must be available before significant numbers of such elements are loaded into the reactor. This information is necessary to ensure that reactor operating conditions remain within limits which will prevent extensive fuel element damage via burnout.

The present knowledge of boiling heat transfer is not sufficient to allow predictions of boiling burnout conditions by theoretical means alone. Therefore, experiments were conducted in the Battelle-Northwest Heat Transfer Laboratory to define boiling burnout conditions for N-Reactor Mark II coproduct elements. Several models of the inner flow annulus, between the outer fuel tube and the inner target cylinder were used in this program to establish burnout heat fluxes over ranges of operating conditions and dimensions of interest for this channel. Results of these experiments validated the boiling burnout equation predictions, and confirmed that significant damage to the Mark II fuel will not occur because of burnout.

Boiling burnout experiments were conducted with four different electrically heated models, each having a different geometry of an inner coolant annulus in an N-Reactor coproduct element. Each of these models consisted of an unheated horizontal stainless steel tube centered in a HastelloyC tube which was heated by passing an electric current through it. Cooling water flowed through the annulus between the two tubes. The model design is illustrated in Figure 9. These four models provided three different sets of annulus dimensions, with two different heated lengths (two feet and eight feet) for one set of annulus diameters. Primary objectives of these experiments were to obtain experimental burnout data directly applicable to the inner annulus in the latest N-Reactor coproduct element design, to obtain information about the effects of dimensions on burnout conditions in this annulus, and to compare burnout behavior for cases with one and both annulus surfaces heated. No burnout experiments were conducted for the outer annulus which is between the outer surface of

* In this terminology, burnout is the creation of a gaseous film barrier at the interface of fuel and cooling liquid which may change thermal levels by orders of magnitude.

Section A - A

Figure 9. Diagram of Typical Electrically Heated Model

the fuel tube and the inner surface of the reactor pressure tube. Dimensions of this annulus in a coproduct element column are almost identical to those for a column of present tube-in-tube fuel elements. Experimental burnout data obtained earlier for the tube-in-tube elements are therefore applicable to the coproduct element outer annulus.

Operating conditions in these experiments covered the following ranges:

- Coolant Mass Velocity 0.5×10^6 to 6×10^6 lb/hr-ft²
- Pressure 1500 psig at the downstream end of the heated section
- Coolant Heat Content enthalpies ranging from 120 Btu/lb below to 140 Btu/lb above saturation (130°F subcooling to 26 w/o steam quality). The ranges covered varied somewhat with test section and mass velocity.
- Maximum Heat Flux varied from 1.5×10^6 to 2.9×10^6 Btu/hr-ft² for different models.

Results of these experiments are presented graphically in Figures 10 through 12. However, results of other experiments which were conducted at lower mass velocities are not detailed here since these were lower than the characteristics of N-Reactor operation. Results of the experiments performed validated the boiling burnout equation predictions, and confirm the validity of use of such equations for calculation of operating margins for the coproduct fuel.

Conclusions drawn from these experiments are summarized as follows:

- For the dimensions used, the number of heated surfaces had little effect on the burnout heat flux. One of the coproduct element models showed slightly (less than 10 percent) higher burnout heat fluxes than did a 2-heated-surface model of identical dimensions used in earlier burnout experiments for tube-in-tube fuel elements.
- Some general aspects of the dimensions on burnout heat flux for such annuli remain unclear. A more comprehensive experimental program would be required to more precisely define quantitative effects of dimensions. However, the data on the specific models tested adequately define the burnout heat flux limits for Mark II fuel.

Figure 10. Burnout Heat Flux Versus Excess Enthalpy at a Pressure of 1500 psig and Mass Velocity of 4.0×10^6 lb/hr-ft²

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Figure 11. Burnout Heat Flux Versus Excess Enthalpy at a Pressure of 1500 psig and Mass Velocity of 5.0×10^6 lb/hr-ft²

Figure 12. Burnout Heat Flux Versus Excess Enthalpy at a Pressure of 1500 psig and Mass Velocity of 100 lb/hr-ft²

- Experiments with the 8-foot-long model, which allowed higher steam qualities to be reached than did the 2-foot-long models, showed a flattening of the burnout heat flux versus enthalpy plots at higher enthalpies.
- A comparison of the experimental coproduct element burnout heat fluxes with values predicted by a burnout equation developed from earlier experiments for the tube-in-tube fuel elements currently being used showed that this equation could be applied satisfactorily to burnout calculations for the coproduct element inner annulus. Most of the experimental burnout heat fluxes within the acceptable range of the values predicted by the equation, and all values for the Mark II fuel were within the tolerance levels provided within operating safety limits for the N-Reactor under the planned operating conditions.

DISCUSSION

Experimental Apparatus

Boiling burnout experiments for N-Reactor coproduct elements were conducted in the Battelle-Northwest Heat Transfer Laboratory. In this facility, water is circulated through the experimental loop by a 250 gpm, 600 psi head pump. Coolant temperatures at the inlet to the experimental model are controlled by electric preheaters and a heat exchanger with a variable bypass. The flow to the model is controlled automatically by a recorder-controller which operates valves in two lines feeding and bypassing the model. Pressure at the downstream end of the model is maintained by a bleed system and injection pumps, and is controlled automatically at a pre-set level by a recorder-controller.

Two sources of direct current (DC) power are available for resistance heating of the models. These are a silicon rectifier bank capable of supplying 3840 kw continuous duty (32,000 amps at 120 volts), and a group of three motor-generator sets, supplying a total of 1250 kw (25,000 amps at 50 volts).

Experimental Models

Four different models were used in the experiments. All four of these models represent the inner coolant annulus between the outer fuel tube and the inner target cylinder of the coproduct element. The models were similar in construction and consisted of an unheated stainless steel tube centered in an electrically heated Hastelloy C outer tube. Electrical heads, welded to each end of the outer tube, were connected to the DC bus system. Heating was accomplished by passing an electric current

through the tube. A 2-inch thick layer of thermal insulation covered the model to restrict heat losses to negligible values from the electrically heated portion.

The design of these models is illustrated in Figure 9. The model was installed in a horizontal position with inlet and outlet lines connecting the coolant annulus to the high pressure loop. The outlet line exited from the top of the model (in a fashion similar to the actual reactor installation) to prevent steam being trapped in the upper part of the annulus under bulk boiling conditions. Ceramic supports in the model maintained the inner tube concentric within the outer tube, and insulated the inner tube from the electric circuit. Sheathed chromel-alumel thermocouples were welded to the outer surface of the heated tube at a number of locations to serve as burnout detectors. The number of thermocouples varied from 18 to 25 for the four models. Dimensions of the models are given in the following tabulation:

	Mark II Fuel			
	Model A	Model B	Model C	Model D
Heated Length, ft.	2	2	8	2
Inner Tube, Outer Diameter, in.	1.220	0.739	0.739	1.364
Outer Tube, Inner Diameter, in.	1.775	1.364	1.364	1.753
Annulus Thickness, in.	0.277	0.312	0.312	0.195

The dimensions of Model A were the same as those of a model representing the inner annulus of the present tube-in-tube fuel element, which was used in an earlier experimental program. In the earlier experiments, however, power was generated in both the inner and outer tubes. The primary purposes of the Model A experiments were:

- a. To determine whether changing the number of heated surfaces would affect the burnout heat flux, and
- b. In combination with the other coproduct element models, to provide information concerning the effects of annulus dimensions on burnout heat flux.

The diameters of Models B and C correspond to an early set of coproduct element design dimensions. Because of the long period (several months) required for component procurement and model fabrication, materials

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were ordered as soon as the design dimensions were reasonably firm to minimize delays in the experimental program. Continuing optimization studies resulted in a change in the final coproduct element dimensions after materials for these models had been received. Since the materials were on hand, the models were constructed and experiments run to provide further insight into the effects of annulus dimensions on burnout heat flux. The Model D fuel design is identical with that of the Mark II fuel which was actually chosen as the standard coproduct design for N-Reactor irradiation.

Experiments were conducted with the 8-foot long model primarily to investigate burnout behavior at higher steam qualities than could be reached with the shorter, 2-foot model. The attainable outlet enthalpy is limited by the inlet enthalpy, coolant flow rate, and total power generated in the model at burnout. The coolant entering the model was subcooled in all of these experiments, thus limiting the inlet enthalpy. By increasing the heated length, the total power for a particular heat flux and the outlet enthalpy were increased for a given flow rate.

As noted earlier, the dimensions of Model D fuel correspond to those of the latest coproduct element (Mark II) design. Experiments were conducted with this model to provide data directly applicable to the elements which will be used in the reactor. Data from experiments with this model, in combination with data from experiments with Models A, B and C, provide indications of the effects of dimensions on burnout heat fluxes. This information will be useful in determining experimental requirements for other coproduct elements which might be proposed for future use.

Only burnout at the inner surface of the fuel tube was investigated in the coproduct element burnout experiments. Heat will be transferred from the outer surface of the fuel tube to coolant in the annulus between the fuel tube and the reactor pressure tube also. However, the outer diameter of the coproduct element fuel tube is almost identical to the outer diameter of the outer fuel tube in the present tube-in-tube fuel element. Burnout data obtained earlier for the outer surface of the tube-in-tube fuel element is therefore directly applicable to the outer surface of the coproduct element fuel tube.

Experimental Procedure

The burnout experiments were conducted by increasing the supply voltage to the model in steps, while holding inlet temperature, downstream pressure, and coolant mass velocity constant. After each voltage step, readings were taken of flow, coolant inlet and outlet temperatures, system pressure, voltage drop across the model, and electric current. Readings of all thermocouples attached to the heated section were recorded by a digital data logging system. Also, the readings of several of these

thermocouples which were believed to be at the most probable burnout locations were recorded continuously on strip charts. Power increases were continued until one or more thermocouples showed abnormal temperature rises during a voltage step, usually 100°F or more at burnout, as compared with rises of less than 10°F for pre-burnout steps. When this point was reached, the supply voltage was decreased to prevent physical burnout of the model. Either the coolant mass velocity or the inlet temperature was then adjusted for the next set of runs.

The experiments were all conducted at a system pressure of 1500 psig (at the downstream end of the heated section). This is approximately the pressure which will exist along portions of the fuel column where burnout is most likely to occur. Heat fluxes, coolant enthalpies, and mass velocities in these experiments covered rather wide ranges, providing information for fuel elements both at normal operating conditions and for a variety of abnormal situations.

Data Handling

Some of the experimental data were recorded on punched paper tapes. The rest were read from gauges, and were recorded manually on data sheets for keypunching. The information was then processed by a digital computer to convert raw data into significant engineering parameters. For each run, the following items were calculated:

- heat flux
- mass velocity
- boiling length
- enthalpy at each thermocouple location
- steam quality at each thermocouple location
- model temperature at each thermocouple location

For all cases in which the outlet temperature was below saturation, heat balances were calculated by comparing the measured coolant temperature rise through the model with that calculated from the power input and mass flow rate. These heat balances for power runs, comparisons of inlet and outlet temperature measurements for zero-power runs, and comparisons of measured outlet temperatures with saturation temperatures for cases in which there was bulk boiling, were used to determine thermocouple errors and heat losses from the system. These two items were used to develop correction factors to be applied to the local enthalpies calculated from temperature measurements, powers and flow rates. After these corrections were applied, the local enthalpies are believed to be accurate within 5 Btu/lb.

After the calculations were completed, the data were examined to determine burnout locations and to provide a more refined determination of the burnout heat flux. Thermocouple readings were plotted against heat flux, generally producing a straight line plot in the nucleate boiling region preceding burnout. In most cases, burnout was signaled by a sudden departure to more than 100°F above the temperature predicted by extrapolating the nucleate boiling region lines, and in four instances, temperatures high enough to produce physical failure of the model were reached. In a few cases, however, a thermocouple would show a gradually increasing departure from this line over several runs. For these cases, an arbitrary burnout point definition was used. A line was drawn at 97-1/2 percent of the heat flux along the nucleate boiling portion of the curve, and was extrapolated to higher heat fluxes and temperatures. The point at which the data plot crossed this line was taken as the burnout point.

After the burnout points were all defined, conditions at burnout were tabulated. Plots were made of heat flux versus mass velocity and coolant enthalpy at burnout for each of the models.

Experimental Results

Plots of the experimental burnout data are shown in Figures 10 through 12. In these figures, burnout heat flux is plotted against excess enthalpy (local coolant enthalpy minus saturated liquid enthalpy) at the burnout location. Separate plots are provided for each mass velocity. In addition to data points from the experiments discussed in this report, each figure shows a line representing a burnout equation developed from earlier experiments with models of the tube-in-tube fuel inner annulus⁽¹⁾. This equation has the following form:

$$\frac{\phi_{BO}}{10^6} = 1.640 - 0.946 \left(\frac{G}{10^6} \right)^{-0.5} - \left[7.0 \left(\frac{G}{10^6} \right)^{0.5} - 5.0 \right] \frac{\Delta H}{1000} \quad (1)$$

Where:

ϕ_{BO} = burnout heat flux, Btu/hr-ft²

G = coolant mass velocity, lb/hr-ft²

ΔH = local enthalpy minus saturated liquid enthalpy, Btu/lb

In addition to the data plots in Figures 10 through 12 tabulations of operating conditions for each of the burnouts obtained in these experiments were made. Table I summarizes data for the Mark II fuel which has been chosen for N-Reactor coproduct loading. For each experimental burnout point, the table shows:

TABLE I
BURNOUT EXPERIMENT DATA FOR MODEL D

Heated Surface Inside Diameter: 1.735 inches
 Unheated Surface Outside Diameter: 1.364 inches
 Heated Length: 2 feet
 Annulus Thickness: 0.195 inch
 Pressure: 1500 psig

Mass Velocity (lb/hr-ft ²) (x 10 ⁶)	Inlet Temperature (°F)	Enthalpy at Burnout Point (Btu/lb)	Excess ^(a) Enthalpy (Btu/lb)	Burnout Heat Flux (Btu/hr-ft ²) (x 10 ⁶)	(ϕ_E/ϕ_C) ^(b)
0.51	589	701	+ 88	0.395	1.26
0.50	532	610	- 3	0.345	1.13
1.08	593	677	+ 64	0.594	1.01
1.03	563	642	+ 29	0.660	1.02
1.04	533	613	0	0.746	1.05
2.02	591	655	+ 42	0.798	1.04
2.03	559	621	+ 8	0.943	1.01
3.08	593	643	+ 30	0.863	0.97
3.08	563	612	- 1	1.091	0.99
4.06	590	633	+ 20	0.984	1.00
4.03	564	609	- 4	1.277	1.06
5.06	592	632	+ 19	1.087	1.07
5.06	562	604	- 9	1.512	1.15
6.05	591	630	+ 17	1.235	1.18
6.05	562	601	- 12	1.673	1.19

(a) Enthalpy at burnout point minus saturated liquid enthalpy.

(b) Ratio of experimental burnout heat flux to burnout heat flux calculated from Equation (1).

- heat flux
- mass velocity
- coolant inlet temperature
- coolant enthalpy at burnout location
- excess enthalpy at burnout location
- ratio of measured burnout heat flux to the burnout heat flux calculated from Equation (1)

An examination of the data shows that Equation (1), developed from the earlier experiments with the 2-heated surface models could be used to determine burnout limits for the coproduct element inner annulus with reasonable accuracy. Although different models exhibited somewhat different relationships between burnout heat flux and coolant enthalpy, most of the experimental burnout fluxes fell into the range of 1.00 to 1.25 times the values predicted by the equation. At coolant mass velocities of 2 million lb/hr-ft² or more, all experimental burnout heat fluxes were approximately equal to, or higher than, the values which would have been predicted by the equation.

The experiments with Model A showed no deleterious effects of the change from two, to one heated surface. Some other researchers have reported higher burnout heat fluxes for planes of two heated surfaces facing one another than for the plane of a heated surface facing an unheated surface, at comparable coolant mass velocities and bulk enthalpies. However, burnout heat fluxes for Model A, consisting of an unheated tube inside a heated tube, generally tended to be slightly higher than those for the earlier model having identical dimensions and both tubes heated. In most cases, the difference amounted to less than 10 percent. It is concluded that for the geometry, dimensions and operating conditions used in these experiments, the number of heated surfaces has little importance.

Although three different sets of annulus dimensions were used in the experiments, the results do not present a clear-cut picture concerning the effects of dimensions on burnout heat fluxes. Models A and D showed comparable burnout heat fluxes for the same coolant enthalpies and mass velocities. These two models had similar heated-surface diameters (1.775 inches for Model A and 1.753 inches for Model D), but the annulus thickness for Model D was about 30 percent less than that for Model A. On the other hand, Model B, which had a smaller heated surface diameter (1.364 inches) but only a 12 percent greater annulus thickness than Model A, consistently exhibited higher burnout heat fluxes at all mass velocities above 0.5 million lb/hr-ft². The burnout heat flux for

Model B also exhibited a much greater sensitivity to coolant enthalpy than did the burnout heat fluxes for the other models.

It was noted that burnout heat fluxes for Model C, which had the same annulus dimensions as Model B but four times the heated length, seemed to agree quite well with those for Models A and D rather than Model B in the low steam quality and subcooled regions.

A more comprehensive experimental program, using a number of different model sizes, would be required to separate effects of annulus thickness, heated surface diameter and length.

Experimental results for the 8-foot long Model C, which had been used to investigate burnout behavior at enthalpies above those which could be reached with the shorter models, indicate a "flattening" of the burnout heat flux versus enthalpy curves at higher steam qualities. Only subcooled and low steam quality burnouts could be obtained with the 2-foot long models. In these regions, burnout heat flux showed a linear decrease with increasing enthalpy, for a constant mass velocity. The data plots extrapolated to a burnout heat flux of zero at rather low steam qualities, with the slope of the burnout heat flux versus enthalpy plot depending on the mass velocity. It had been anticipated that the plots would flatten, with the burnout heat flux showing less sensitivity to enthalpy, at higher coolant enthalpies. (Such a flattening had been observed in earlier N-Reactor fuel burnout experiments with long tubular models.) The few burnout points obtained in the subcooled and low steam quality regions with Model C agree reasonably well with the burnout data for Models A and D and with the burnout heat fluxes predicted by Equation (1). In the higher steam quality region, however, the slope of the plot changed and the burnout heat flux showed considerably less dependence on enthalpy, as expected. In this region, the experimental burnout heat fluxes ranged well above those predicted by Equation (1). It may be noted in the data plots for higher mass velocities that the plots for Model C show a "breakaway" and flattening at enthalpies where plots for Model D show a continuation of the steeper curve. This difference in behavior has not been explained. It may be the result of differences in dimensions; length, annulus thickness, or heated-surface diameter. It is also noted that in the subcooled and low steam quality regions, burnout heat fluxes for Model C tended to agree with those for Models A and D, rather than with Model B which had the same annular dimensions as Model C. This difference between the behavior of Models B and C may be the result of some length effect. However, the number of experiments and length, diameter and annulus thickness combinations used were not sufficient to define effects of these parameters, or to allow quantitative estimates of burnout heat fluxes at high steam qualities for dimensions other than those of Model C.

The major conclusion from these experimental results is that Equation (1), previously developed from models of the tube-in-tube fuel inner annulus, should adequately predict burnout conditions for the

coproduct fuel inner annulus. The burnout heat fluxes predicted by this equation agree reasonably well with the experimental data, tending to be slightly pessimistic in some cases. The agreement extends to conditions well beyond those in the experiments upon which the equation was based; burnout heat fluxes up to 2.9 million Btu/hr-ft² as compared with 1.6 million, and coolant mass velocities up to 6 million lb/hr-ft² as compared with 4 million in the earlier experiments.

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*The special section of RL-GEN-888 10 summarizes information contained in BNWL-CC-729. Additional detail and experimental data points may be obtained from the parent document.

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ADMINISTRATIVE SUMMARY

OCTOBER 1966

EMPLOYMENT

Section	Permanent		Temporary		Total		Change		Total
	E	NE	E	NE	E	NE	E	NE	
General	1	1	0	0	1	1	0	0	2
Closeout Group	15	26	0	0	15	26	0	-2	41
Employee Relations	5	4	7*	0	12	4	-2	0	16
Finance	19	16	0	0	19	16	-1	0	35
Fuels	44	145	0	0	44	145	+2	+8	189
Plant	60	210	0	0	60	210	0	0	270
Project	58	17	0	0	58	17	0	0	75
Research and Engineering	<u>49</u>	<u>19</u>	<u>0</u>	<u>0</u>	<u>49</u>	<u>19</u>	<u>+2</u>	<u>0</u>	<u>68</u>
Total	251	438	7	0	258	438	+1	+6	696
Department Total					696				
New Change							+7		

*Technical Graduates

SAFETY

Days without a disabling injury	125
Hours worked without a disabling injury	403,106

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VISITORS

<u>Name</u>	<u>Company and Location</u>	<u>Contact</u>	<u>Date</u>	<u>Purpose</u>
Ken Jacobs	Marotta Valve Corp. Santa Ana, Calif.	N.J. Roy and P.K. Telford	9-28 to 9-29	Discuss hydraulic valve problems
E.L. Knoedler J.W. Siegmund	Sheppard T.Powell & Assoc. Baltimore, Maryland	W.D. Bainard	10-10 to 10-12	Discuss water treatment and corrosion; steam generator discussions
B.L. Eskaback	Travelers Insurance Co.	W.H. Dutton	10-11	Third party inspection of unfired pressure vessels
Pete Anderson	Simonds Machine Co. East Woodstock, Conn.	C.E. Gruhl	10-11	Design of target and driver support hardware
Dale Nevitt	True Cut Samplers, Inc. Golita, California	M.M. Hendrickson	10-11	Demonstrate new gas sampling system
W.A. Kirk	Frost Engineering Serv. Seattle, Washington	M.M. Hendrickson	10-11	Demonstrate new gas sampling system
J.D. Currie	Industrial Prod. Div. ITT	R.G. Lauer	10-14	Survey N-Reactor data display needs
R.A. Graham E. Baroch	Wah Change Corp. Albany, Oregon	J.W. Nickolaus	10-14	Zr-2 material needs for target support system
W.R. Conley	Neptune MicroFLOC Corp. Corvallis, Oregon	W.D. Bainard	10-17 to 10-19	Discuss operation of filter plant.

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VISITS

<u>Name</u>	<u>Company and Location</u>	<u>Contact</u>	<u>Date</u>	<u>Purpose</u>
W.D. Bainard	Tennessee Valley Auth. Paradise, Kentucky	V. Shattuck G. Rubnik	9-30	Inspect Inconel-tubed heat exchanger; discuss water treatment and chemical cleaning
T.W. Evans	AIME 1966 Nuclear Metallurgy Symposium Delavan Wisconsin		10-3 to 10-5	Attend Symposium
"	Argonne National Lab. Chicago, Illinois	L. Baker	10-6	Discuss fuel technology
P.C. Althoff	Lundell Controls Chicago, Illinois	G.L. Hinote N. Benda	10-6	Discuss recordalarm and statalarm annunciator systems
"	Ther Electric & Machine Works, Chicago, Ill.	R.T. Thayer	10-7	Discuss maintenance of rectifiers
W.J. Dowis	AEC Headquarters Washington, D.C.	W.R. Voigt	10-11 to 10-12	Discuss N-Reactor operation
G.R. Klingler D. Marinos	Martec, Inc. Decatur, Ala.	D.D. Ooten	10-17 to 10-19	Inspect and approve preproduction Al coproduct canisters
R.C. Hoffman G.L. Swezea	Bonneville Power Adm. Portland, Oregon	G. Starr	10-10	Attend meetings on transient tests
D. Marinos	General Electric Co. Schenectady, New York	R.P. Carlson	10-20	Discuss metal handling

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VISITS (continued)

<u>Name</u>	<u>Company and Location</u>	<u>Contact</u>	<u>Date</u>	<u>Purpose</u>
J.H. Fastabend	General Electric Co. Manufacturing Services Schenectady, New York	R.P. Carlson	10-19 to 10-20	Discuss fuel handling
D.W. Darsow	Precision Products Co. Gresham, Oregon	C. Wilcox	10-21	Discuss production of Zr-2 supports for coproduct fuel
"	Pem, Inc. Albany, Oregon	C.A. Brown	10-21	" "
"	Wah Chang Albany, Oregon	R.A. Graham	10-21	" "
R.E. Spicka	General Electric Co. NED, San Jose, Calif.		10-24 to 10-25	Employment interviews
A.E. Engler	ASCE, Pacific Northwest Council Yakima, Washington		10-24	Meeting of Board of Directors
H.S. Davis	American Concrete Inst. New Orleans, Louisiana	ACI Committee 349	10-26 to 10-27	Meeting of ACI Committee No. 349
R.D. Hodgson	Northwest Electronics Co. Spokane, Wash.		10-27	Discuss availability of power line transient recording equipment
M. Lewis	ANS Meeting Pittsburgh, Penn.		10-30	National program committee meeting

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SIGNIFICANT REPORTS

<u>Report No. & Class.</u>	<u>Title</u>	<u>Author</u>	<u>Date</u>
RL-NRD-722 8 and S	Summary Status Report, Fuel Element Failures	A.E. Guay	9-30-66
RL-NRD-722 9 S	in N-Reactor		10-25-66
RL-NRD-799 9 S	Yield and Defective Rates, September 1966	W.H. Hodgson	10-66
RL-NRD-807 8 and S	Fuel Element Production and Inventory Report,	D.W. Darsow	10-24-66
RL-NRD-807 9	August and September 1966		
RL-NRD-830 SUP7 S	Spring-Stop Support, Coproducer Demonstration	E.E. Leitz	9-26-66
	Test, 2.1, Supplement 5		
RL-NRD-830 SUP8 S	Coproducer Demonstration Test (2.1), Production	E.E. Leitz	10-5-66
RL-NRD-830 SUP9 U	Test Change Authorizations No. 5 and No. 6		10-24-66
RL-GEN-1038 4 S	Coproduct Two Load Demonstration	E.E. Leitz	10-31-66
RL-GEN-1058 2 S	Target Yield and Defective Rates, September 1966	W.H. Hodgson	10-66
RL-GEN-1167 C	Autoclave Poison Rods	W.H. Hodgson	9-22-66
RL-GEN-1168 S	Coproduct Extraction Facility	R.L. Dickeman	9-21-66
RL-GEN-1177 U	Chemical Process for Cell 3 Decontamination	W.K. Kratzer	9-30-66
RL-GEN-1179 S	Near-Term Plans for Charging Coproduct Fuel	M.C. Leverett	10-3-66
RL-GEN-1180 S	Program Considerations, N-Reactor	R.L. Dickeman	10-3-66
RL-GEN-1184 U	Steam Availability Curves	J.A. White and A.J. Ebens	10-4-66
RL-GEN-1185 U	N-4 Test 5.4 - 400 Mwe Dual Turbine Trip from Reduced Thermal Power, Detailed Test Results	D.L. Renberger	10-4-66
RL-GEN-1186 C	Effect of Hydrides on Structural Integrity of Target Cladding from Redrawn K-Process Tube	R.R. Studer	10-5-66

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<u>Report No. & Class.</u>	<u>Title</u>	<u>Author</u>	<u>Date</u>
RL-GEN-1190 U	Trip Report, Visit to Paradise Steam Plant, Kentucky	W.D. Bainard	9-30-66
RL-GEN-1191 S	Coproduct Fuel Production Forecast	P.W. Hood	10-10-66
RL-GEN-1195 U	User's Manual for N-Heat II N-Reactor Lattice Heat Transfer Program for Univac-1107 Computer	G.H. Dickerson	10-11-66
RL-GEN-1196 U	Results from Accidental One-of-Two 185-N Turbine Trip at 400 Mwe Total Load	D.L. Renberger	10-12-66
RL-GEN-1197 S	Discharged Fuel Make-up During Coproduct Transition	R.L. Dickeman	10-18-66
RL-GEN-1203 S	Engineering Study, Weapons Grade Pu Production	D. Marinos	10-17-66
RL-GEN-1208 S	Weapons Grade Pu Production at N-Reactor	Process Design	10-20-66
RL-GEN-1209 U	Proposed Permanent Strainers for the N-Reactor Primary Coolant System	L.G. Henke	10-20-66
RL-GEN-1211 C	Safeguards Analysis Report, Coproduct Shipments in the GE-601 Cask	M.M. Hendrickson	10-24-66
RL-GEN-1215 S	Coproduct Shipments to Savannah River	R.L. Dickeman	11-2-66
RL-GEN-1216 S	Expansion of Coproducer Demonstration Test	R.L. Dickeman	10-27-66
RL-GEN-1218 S	Streamlining the N-Fuel Cycle	H.E. Jackel T.D. Naylor J.W. Riches	10-28-66
RL-GEN-1219 S	Change in Fuel Irradiation Program	R.L. Dickeman	10-27-66

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SUGGESTION PLAN PARTICIPATION

	<u>October</u>	<u>CY 1966</u>
Number of eligible employees	438	414 (average)
Number of suggestions received	46	377
Number of suggestions acted upon	19	483
Number of suggestions adopted	9	170
Net annual savings	\$170	\$27,972
Amount of Awards	\$100	\$ 3,825
Percent of awards to savings	58.8	13.7
Average amount of awards	\$ 11.11	\$ 22.50
Adoption rate per 1,000 employees (annualized)	-	493

INVENTIONS

Indication of Low Oil Level in an Internal Combustion Engine,
dated October 16, 1966, W. J. Morris

Design of Fuel for Nuclear Reactors, dated October 26, 1966,
by M. C. Leverett

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