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Battelle
monthly report

April, 1968

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MONTHLY REPORT
APRIL 1968

DOUGLAS UNITED NUCLEAR, INC.
Richland, Washington

Work performed under Contract No. AT-(45-1)-1857 between the Atomic Energy Commission and Douglas United Nuclear, Inc.

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REACTOR PLANT OPERATIONS

C & K Reactors

Reactor input production (Pu) was 269.0 KWMD, totaling 217.9 at the two K reactors and 51.1 at C Reactor. U-233 input production in the fringe loadings was 3,324 equivalent MWD. Overall time operated efficiency was 81.2 percent, averaging 84.6 at KE and KW and 74.5 at C.

The production of nondefense (high Pu-240) plutonium is continuing. The depleted uranium loads in the K reactors are scheduled for replacement with thorium about June 1. Most of the weapons grade plutonium being made is in 94 Metal.

The deactivation of B Plant is progressing on schedule.

N Reactor

Input production was 75.2 KWMD. Time operated efficiency was 75.0 percent and steam availability was 72.9 percent. The 3,500 MW administrative limit on reactor power level was maintained to restrict Mark II fuel stress severity.

For the third time in the past four months, steam supplied to the Washington Public Power Supply System enabled them to set a new second-highest monthly generating record (363,600 MWh).

The retubed steam generators in Cell 3 passed all hydrostatic pressure tests satisfactorily, and operability checks are in progress.

FUEL & TARGET FABRICATION

C & K Reactors

Fuels production totaled 206.9 tons of 94 Metal elements and 23.4 tons of natural uranium elements. Canning line time operated efficiency was 99.0 percent.

At month end, fuel core inventory was 987 tons, a 3.9 months' supply; finished fuel inventory was 883 tons, a 3.1 months' supply. These totals exclude all B Reactor fuel except the 94 Metal.

N Reactor

Input production totaled 260 extrusions, representing 100.2 percent of forecast. In addition, 30 miscellaneous development extrusions were made. Output of finished fuel assemblies totaled 29.7 tons, which was 106.1 percent of forecast.
TECHNICAL ACTIVITIES

C & K Reactors

The test evaluating 50-mil thick fuel element supports in KW Reactor was terminated as the result of a core-crack fuel failure. The remaining five columns were discharged at exposures about 18 percent below the goal of 2000 MWD/T. Preliminary examination of the failed element and those downstream from it revealed no support corrosion failures.

Although it has been demonstrated that anodized spacers inhibit the buildup of a hydride case layer in Zircaloy tubes, initial laboratory analyses indicate that the hydrogen pickup by the base metal may be continuing at the same rate as with unanodized spacers.

Design of the plutonium-aluminum fuel elements and the reactor loading for their test irradiation has been completed. The loading will contain about 15 Pu-Al columns initially, of which one or two will contain short monitor charges for use in checking the validity of buildup-burnout calculations.

Six striped charges were placed in KE Reactor for the test irradiation of high U-236 content 94 Metal. The two types of 94 Metal used contain 300 and 1000 ppm U-236, respectively. Discharges are scheduled at various exposures from 300 to 1,900 MWD/T.

A cost study of Pu-238 production from Np-237 indicated that additional fuel costs and loss of Pu-239 production accounted for nearly half of the total conversion cost exclusive of the neptunium cost.

The 4+4-tube one-inch overbore test block in C Reactor is operating well. Graphite temperature measurements in the block demonstrate that significant temperature reductions can be achieved by overboring.

Discharge of C Reactor effluent to the disposal trench continued at about 50,000 gpm. Water levels in wells in the immediate vicinity and within 100 B-C Area were essentially unchanged, but slight increases in level and temperature were noted in more distant wells. Temperatures in wells within 100 B-C Area showed some decrease during the month.

N Reactor

Recently completed thermal hydraulics studies included a series of boiling burnout heat flux measurements over the expected range of Mark IV operating conditions and also at extremes of heat flux. These measurements support the Mark IV process and safety limits.

A test was completed to determine the feasibility of protecting the bottom shield insulation layer against excessive temperature differentials as observed during the 4,800 MW probe. The test consisted of exchanging fringe control rods while adjusting other rods to maintain constant power. Test results indicate a reduction of more than 20 percent in the delta T across the insulation layer. The results also suggest that higher power levels can be attempted.
on a test basis by using the rods as fringe-protection, thus avoiding commit-
ment to a fringe blanket poison loading until routine operation at the higher
power is possible.

A report is in preparation outlining potential methods of meeting flux moni-
toring requirements of the Advanced Technology Case. A flux profile monitor-
ing system would comprise several traveling chambers, each of which would
relay signal outputs into a process computer which would then determine the
tube power limits in that region of the reactor. This system would enable a
closer approach to operational limits. A monitoring system consisting of 60
fixed chambers has also been developed which would detect rapidly any changes
in the flux profile caused by such anomalies as xenon cycling or improper rod
control movements.

In metal-water reaction tests, the measured rate of reaction between Zircaloy
and steam was only 58 percent of the values reported by Argonne for 1,100 C,
63 percent for 1,200 C, and 71 to 110 percent at 1,000 C. This indicates
less reaction at zirconium temperatures near the uranium melting point
(1,090 C) than was previously assumed.

ADVANCED OPERATIONAL PLANNING

Copies of the final report on MFC-13 (Co-60 application study) have been for-
warded to AEC-RL for review.

Program definition of the financial portion of CAGE Mod 2A is proceeding satis-
factorily.

A new computer program is being designed to prepare planning estimate output.

The submission of MINIMODEL data to AECOP has been completed. A total of
25 sets was supplied.

All revisions have now been made to slides for use in the upcoming presenta-
tion of the Long-Range Plan to the AEC Commissioners.

FEATURE REPORT

This month the appended special report describes the 44-tube Reactor Modern-
ization Test Facility which is now operating in the C Reactor with process
tubes and fuel elements about one inch larger than standard. The report
also summarizes the reactor safety and production advantages of process chan-
nel enlargement, and the incentives for its large-scale application.

GENERAL

The reduction of force in conjunction with the B Reactor deactivation and the
reduction of expenditures are continuing on schedule. Total employment was
1,999 at month end.
The decision received from the Federal Court in March requiring the Company to submit to arbitration the question of contract coverage for N Reactor employees has been appealed by the Company to the Federal Circuit Court of Appeals. The appealability of this matter is to be reviewed by the Court on May 20, with a decision expected shortly thereafter.

There were no disabling injuries, and no radiation exposures exceeded operational control.

Charles D. Harrington
President

DECLASSIFIED
REACTOR PLANT OPERATIONS - C AND KS

PRODUCTION

Reactor production, power levels, efficiencies and related statistics are tabulated on the next page.

Overall reactor input production and time operated efficiency for the past six months are charted below:

OPERATING EXPERIENCE

Reactor Loadings

Front face maps showing the loadings of the C, KE, and KW Reactors are reproduced on the three pages which follow page B-6. The tonnages listed are approximate: actual fuel totals are given on page B-2.

The special program for producing nondefense (high Pu-239) plutonium is continuing at the C and K reactors. Discharge of the remaining natural uranium in C Reactor being irradiated to 12 percent Pu-239 was achieved by midmonth with no fuel failures experienced. Natural metal currently loaded at C Reactor is being irradiated to about 8.5 percent Pu-239. At C and the Ks, the production of weapons grade plutonium is basically confined to 94 Metal.

The irradiation of depleted uranium at the K reactors for the production of 27 percent Pu-239 is scheduled for completion about June 1. Both K reactors then will be reloaded with thorium target elements and supporting enrichment to assure the delivery of 360 kg of "clean" U-233.
## PRODUCTION REACTOR STATISTICS - APRIL 1968

<table>
<thead>
<tr>
<th>REACTOR</th>
<th>c</th>
<th>KE</th>
<th>kW</th>
<th>TOTAL</th>
</tr>
</thead>
<tbody>
<tr>
<td>INPUT PRODUCTION - Pu - KMWD</td>
<td>51.1</td>
<td>122.7</td>
<td>95.2</td>
<td>269.0</td>
</tr>
<tr>
<td>- U-233 - Equiv. MWD</td>
<td>1287</td>
<td>1126</td>
<td>911</td>
<td>3324</td>
</tr>
<tr>
<td>POWER LEVEL (MW) - MAXIMUM</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>- AVERAGE</td>
<td>2390</td>
<td>4400</td>
<td>4400</td>
<td>11190</td>
</tr>
<tr>
<td>TIME OPERATED EFFICIENCY - %</td>
<td>74.5</td>
<td>93.9</td>
<td>75.3</td>
<td>81.2</td>
</tr>
<tr>
<td>OUTAGE TIME ALLOCATION - %</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>CHARGE - DISCHARGE</td>
<td>11.4</td>
<td>0</td>
<td>0.8</td>
<td>4.1</td>
</tr>
<tr>
<td>FAILED FUEL REMOVAL</td>
<td>0</td>
<td>3.3</td>
<td>0</td>
<td>1.1</td>
</tr>
<tr>
<td>WATER LEAKS</td>
<td>0</td>
<td>0</td>
<td>12.2</td>
<td>4.1</td>
</tr>
<tr>
<td>TUBE REPLACEMENT</td>
<td>1.4</td>
<td>0</td>
<td>0</td>
<td>0.4</td>
</tr>
<tr>
<td>OTHER MAINTENANCE</td>
<td>7.2</td>
<td>2.0</td>
<td>2.1</td>
<td>3.8</td>
</tr>
<tr>
<td>STANDARDS CHECK</td>
<td>1.6</td>
<td>0.8</td>
<td>1.1</td>
<td>1.2</td>
</tr>
<tr>
<td>PRODUCTION TESTS</td>
<td>3.2</td>
<td>0</td>
<td>7.5</td>
<td>3.5</td>
</tr>
<tr>
<td>PROJECT WORK</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>OTHER</td>
<td>0.7</td>
<td>0</td>
<td>1.0</td>
<td>0.6</td>
</tr>
<tr>
<td>TOTAL</td>
<td>25.5</td>
<td>6.1</td>
<td>24.7</td>
<td>18.8</td>
</tr>
<tr>
<td>NUMBER OF OUTAGES</td>
<td>1</td>
<td>1</td>
<td>3</td>
<td>5</td>
</tr>
<tr>
<td>NUMBER OF STARTUP INTERRUPTIONS</td>
<td>0</td>
<td>0</td>
<td>2</td>
<td>2</td>
</tr>
<tr>
<td>WATER LEAKS - TUBE</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>- VAN STONE</td>
<td>0</td>
<td>0</td>
<td>1</td>
<td>1</td>
</tr>
<tr>
<td>NEW TUBES INSTALLED</td>
<td>0</td>
<td>0</td>
<td>1</td>
<td>1</td>
</tr>
<tr>
<td>FUEL CHARGE - (TONS) - NATURAL URANIUM</td>
<td>136.3</td>
<td>288.0(^{1})</td>
<td>282.8(^{1})</td>
<td>707.1</td>
</tr>
<tr>
<td>- ENRICHED URANIUM</td>
<td>78.8</td>
<td>156.5</td>
<td>157.2</td>
<td>392.5</td>
</tr>
<tr>
<td>FUEL ELEMENT FAILURES</td>
<td>0</td>
<td>1</td>
<td>1</td>
<td>2</td>
</tr>
<tr>
<td>HELIUM CONSUMED - M CU, FT.</td>
<td>282.8</td>
<td>75.6</td>
<td>197.5</td>
<td>555.9</td>
</tr>
<tr>
<td>WATER TO REACTOR</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>NORMAL OPERATING FLOW - GPM</td>
<td>196,000</td>
<td>196,000</td>
<td>195,200</td>
<td>497,200</td>
</tr>
<tr>
<td>pH</td>
<td>6.58</td>
<td>6.51</td>
<td>6.52</td>
<td>-</td>
</tr>
<tr>
<td>DICROMATE - PPM</td>
<td>0.49</td>
<td>0.50</td>
<td>0.91</td>
<td>-</td>
</tr>
</tbody>
</table>

\(^{1}\) Includes 45.3 tons (at KE) and 45.7 tons (at KW) of special depleted uranium in the E-D loadings (PITA-048).
Power Levels

Power level at C Reactor was restricted by the 95 C bulk outlet water temperature limit. At both K reactors, power levels were restricted by the administrative limit of 4400 MW until late in the month when the bulk outlet water temperature limit of 95 C became limiting. Five-pump operation has continued at both KS because of the pump overhaul program.

Reactor Outages

The five reactor outages are summarized below. In addition, the C Reactor was down 68.2 hours and KW for 4.8 hours on outages initiated in March.

<table>
<thead>
<tr>
<th>Date</th>
<th>Down Reactor</th>
<th>Outage Hours</th>
<th>Remarks</th>
</tr>
</thead>
<tbody>
<tr>
<td>April 2</td>
<td>KW</td>
<td>89.9</td>
<td>Rear Van Stone flange leak on aluminum tube 0970. Equipment Maintenance Standards and Preventive Maintenance Checks were completed as required.</td>
</tr>
<tr>
<td>April 4</td>
<td>KE</td>
<td>43.9</td>
<td>Removal of a failed depleted uranium fuel element from tube 2884 (PITA-048). A flexible VSR was installed in channel 31 and miscellaneous other maintenance was completed.</td>
</tr>
<tr>
<td>April 8</td>
<td>KW</td>
<td>43.5</td>
<td>Removal of a failed Metal fuel element from tube 4276 (PTA-105). Two Zircaloy tubes were replaced for hydride studies.</td>
</tr>
<tr>
<td>April 11</td>
<td>KW</td>
<td>38.6</td>
<td>Unexplained Panellit trip on tube 3872. Miscellaneous maintenance work was accomplished.</td>
</tr>
<tr>
<td>April 14</td>
<td>C</td>
<td>115.5</td>
<td>Scheduled charge-discharge and maintenance, including the installation of new mattress plates in the discharge area and some VSR work.</td>
</tr>
</tbody>
</table>

EQUIPMENT EXPERIENCE

B Plant Deactivation

Deactivation work at B Plant is continuing on schedule with completion expected in June.

Effluent Retention Tanks - 107-C

On April 9 the 107-CW effluent tank was found to be leaking badly through the bottom. Examination of this tank and 107-CE showed deterioration in both, and the C Reactor effluent was diverted into the 107-B Basin. Repair of the C tanks is planned.
Confinement System - C Reactor

Twelve absolute filters in the "A" cell of Building 117-C were damaged late in March by water due to the filling of the "A" cell inlet water seal pit while the "B" cell inlet seal pit was being drained. Replacement filters are being procured, with installation planned for early May.

Rear Mattress Plates - C Reactor

Eighteen new mattress plates were installed in the metal discharge chutes at C Reactor. Also installed was an eight by ten foot mattress plate for the discharge of overbore fuel. A significant reduction in personnel exposure from fuel stuck in the old mattress plates is expected to result from this work.

Pressure Monitor System - KW Reactor

A design change was completed at KW Reactor which allows the Panellit system to reset without a reactor scram if the trip duration is less than 0.04 second. This change was made to alleviate a slight increase in pressure monitor trips which has occurred since operating on five process pumps. It is suspected that the lower flow with five pumps does not flush-seat all the tube charges processed during an outage, and that the seating of these charges after startup causes the momentary pressure fluctuation and scram.

Process Pump Change - 190-KE

A pump change was made at 190-KE during normal reactor operation on April 17. The No. 6 pump set was successfully "floated" on the line when it became necessary to shut the No. 3 set down because of excessive vibration. The extensive skill and careful planning which permitted this operation avoided an unscheduled reactor outage and possibly also expensive motor repair.

Rear Face Water Leaks - K Reactors

Inspection of the KW Reactor rear face disclosed that some of the water leakage problem is due to special cap installations where test-related instrument leads through the end cap make it impossible to correct the leakage without terminating the test involved. Some of these caps therefore are allowed to leak for considerable periods before they are removed. Leakage results in the deposition of salts and the acceleration of corrosion on rear face hardware.

Investigations are underway to find methods for removing the salt deposits within the process tube pattern without harmful effects on RTDs or other components. Also, the possibility of providing improved seals and gaskets is being explored.

REACTOR PERSONNEL TRAINING

Written recertification examinations are being given to the Nuclear Reactor Control Operators. These examinations are oriented principally toward the more recent changes at the reactors. Results of the exams to date indicate that the operators are satisfactorily acquiring knowledge required by new procedures and equipment.
Video tape equipment is being utilized to present the radiation exposure reduction program to Manufacturing exempt personnel. The program consists of three two-hour sessions.

**PROCESS ASSISTANCE AND CONTROL**

**Process Physics**

As the E-D loadings in the K reactors approach their late May-early June discharge goal, the high in-reactor plutonium inventory, because of increased moderator coefficient of reactivity effects, causes the potential for flux distribution oscillations to increase. No unusual reactor operating problems arose during April, however.

Flattening effectiveness continued somewhat below normal at KW Reactor, due to continued temporary operation without No. 2 HCR and to the relatively small long-term gains in the low-exposure fringe natural uranium. Flattening was considerably increased at KE due to long-term gains in the high-exposure fringe fuel and at C Reactor due to the even-row discharge of high-exposure fuel (for 12 percent Pu-240 plutonium production) and to loading adjustments which increased the power generation in the one-inch overbore block.

Cross-comparison of loading configurations on previous startups during the C Reactor startup of April 19 indicated that the apparent reactivity shortage of approximately three milli-k was probably due to overevaluation of spline effects and cold long-term gains in the high-exposure loading. Startup was therefore accomplished without the charging of additional enrichment, and the hot reactivity at the time of the subsequent samarium dip provided adequate flexibility for flux distribution control.

Operational physics data of interest are summarized below:

<table>
<thead>
<tr>
<th></th>
<th>Reactors</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>C</td>
</tr>
<tr>
<td>Effective Central Tubes (ECT)*</td>
<td>1582</td>
</tr>
<tr>
<td>Flattening Efficiency** - April</td>
<td>0.84</td>
</tr>
<tr>
<td></td>
<td>- 12-mo. Average</td>
</tr>
<tr>
<td>Maximum Operating Time Permitting Scram Recovery - hours***</td>
<td>11</td>
</tr>
</tbody>
</table>

*Reactor power level divided by the average power of the ten most productive tubes which are representative of the reactor loading.

**ECT divided by the number of power generating tubes.

***The maximum operating time subsequent to a cold startup following which a scram recovery could be made using the currently approved startup procedures.
Production Fuel Performance

There were no failures of standard production fuel elements during the report period.

The following table shows failure frequencies, as number/million elements discharged, for the 3-, 12-, and 24-month periods ending March 31:

<table>
<thead>
<tr>
<th>Reactor Type</th>
<th>U Type</th>
<th>3 Months</th>
<th>12 Months</th>
<th>24 Months</th>
</tr>
</thead>
<tbody>
<tr>
<td>Small Reactors</td>
<td>Natural U</td>
<td>13</td>
<td>38</td>
<td>24</td>
</tr>
<tr>
<td></td>
<td>94% Metal</td>
<td>0</td>
<td>29</td>
<td>12</td>
</tr>
<tr>
<td>K Reactors</td>
<td>Natural U</td>
<td>0</td>
<td>7</td>
<td>7</td>
</tr>
<tr>
<td></td>
<td>94% Metal</td>
<td>6</td>
<td>8</td>
<td>7</td>
</tr>
<tr>
<td>C Reactor</td>
<td>1 inch, Overbore</td>
<td>0</td>
<td>-</td>
<td>-</td>
</tr>
</tbody>
</table>

Reactor Effluent Activity Data

The following table shows the total kilocuries/month discharge for six significant radionuclides in the reactor effluent during March:

<table>
<thead>
<tr>
<th>Reactor</th>
<th>As-76</th>
<th>P-32</th>
<th>Zn-65</th>
<th>I-131</th>
<th>Cr-51</th>
<th>Np-239</th>
</tr>
</thead>
<tbody>
<tr>
<td>C</td>
<td>3.7</td>
<td>0.33</td>
<td>0.48</td>
<td>0.019</td>
<td>8.3</td>
<td>3.8</td>
</tr>
<tr>
<td>KE</td>
<td>3.5</td>
<td>0.44</td>
<td>0.41</td>
<td>0.033</td>
<td>4.3</td>
<td>3.4</td>
</tr>
<tr>
<td>KW</td>
<td>3.8</td>
<td>0.33</td>
<td>0.33</td>
<td>0.063</td>
<td>4.7</td>
<td>3.3</td>
</tr>
<tr>
<td>Total</td>
<td>11.0</td>
<td>1.10</td>
<td>1.22</td>
<td>0.115</td>
<td>17.3</td>
<td>10.5</td>
</tr>
</tbody>
</table>

The progress of experimental work on effluent activity reduction is described in Section D of this report, under R&D Mission 10.
Zone: Central

Tons:
- 97 Natural Uranium
- 58 (includes 14.94 Metal in overbore block)

Zone: Ring

Tons:
- 12 Natural Uranium
- 19 94 Metal

Zone: Fringe

Tons:
- 25 Natural Uranium

Zone: Thoria Support

Tons:
- 19 94 Metal

Zone: Thoria

Tons:
- 4 Thoria

Loading Pattern - C Reactor

B-A
<table>
<thead>
<tr>
<th>Zone</th>
<th>Tons</th>
<th>Material</th>
</tr>
</thead>
<tbody>
<tr>
<td>Central</td>
<td>45</td>
<td>Special Depleted Uranium (PITA-048)</td>
</tr>
<tr>
<td></td>
<td>102</td>
<td>$^{235}$U Metal (for depleted uranium support)</td>
</tr>
<tr>
<td></td>
<td>84</td>
<td>Natural Uranium</td>
</tr>
<tr>
<td>Ring</td>
<td>32</td>
<td>$^{235}$U Metal</td>
</tr>
<tr>
<td></td>
<td>26</td>
<td>Natural Uranium</td>
</tr>
<tr>
<td>Fringe</td>
<td>133</td>
<td>Natural Uranium</td>
</tr>
<tr>
<td>Thoria Support</td>
<td>22</td>
<td>$^{235}$U Metal</td>
</tr>
<tr>
<td>Thoria</td>
<td>4</td>
<td>Thoria</td>
</tr>
</tbody>
</table>

Loading Pattern - KE

B-B
<table>
<thead>
<tr>
<th>Zone</th>
<th>Tons</th>
<th>Material</th>
</tr>
</thead>
<tbody>
<tr>
<td>Central</td>
<td>46</td>
<td>Special Depleted Uranium (PITA-048)</td>
</tr>
<tr>
<td></td>
<td>100</td>
<td>94 Metal (for depleted uranium support)</td>
</tr>
<tr>
<td></td>
<td>3</td>
<td>125 Metal</td>
</tr>
<tr>
<td></td>
<td>82</td>
<td>Natural Uranium</td>
</tr>
<tr>
<td>Ring</td>
<td>33</td>
<td>94 Metal</td>
</tr>
<tr>
<td></td>
<td>25</td>
<td>Natural Uranium</td>
</tr>
<tr>
<td>Fringe</td>
<td>130</td>
<td>Natural Uranium</td>
</tr>
<tr>
<td>Thoria Support</td>
<td>22</td>
<td>94 Metal</td>
</tr>
<tr>
<td>Thoria</td>
<td>4</td>
<td>Thoria</td>
</tr>
</tbody>
</table>

Loading Pattern - KW Reactor

B-C
REACTOR PLANT OPERATIONS - N

PRODUCTION

General

Reactor production, power level, and related statistics are tabulated below. Input production, and time operated efficiency (TOE) for the past six months are shown on the following chart:

Statistical Summary

<table>
<thead>
<tr>
<th>Description</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Input Production (KMWD)</td>
<td></td>
</tr>
<tr>
<td>Total</td>
<td>75.2</td>
</tr>
<tr>
<td>Coproduct</td>
<td>48.3</td>
</tr>
<tr>
<td>Power Level (MW)</td>
<td></td>
</tr>
<tr>
<td>Maximum</td>
<td>3500</td>
</tr>
<tr>
<td>Average</td>
<td>3345</td>
</tr>
<tr>
<td>Time Operated Efficiency</td>
<td></td>
</tr>
<tr>
<td>%</td>
<td>75.0</td>
</tr>
<tr>
<td>Steam Availability</td>
<td></td>
</tr>
<tr>
<td>%</td>
<td>72.9</td>
</tr>
<tr>
<td>Number of Shutdowns</td>
<td></td>
</tr>
<tr>
<td>Scheduled</td>
<td>1</td>
</tr>
<tr>
<td>Unscheduled</td>
<td>2</td>
</tr>
<tr>
<td>Fuel Failures</td>
<td></td>
</tr>
<tr>
<td></td>
<td>0</td>
</tr>
<tr>
<td>Fuel Charge (Tons)</td>
<td></td>
</tr>
<tr>
<td>9-4 Metal</td>
<td>108.6</td>
</tr>
<tr>
<td>125 Metal</td>
<td>22.5</td>
</tr>
<tr>
<td>210 Metal</td>
<td>140.3</td>
</tr>
<tr>
<td>Total</td>
<td>271.4</td>
</tr>
</tbody>
</table>
Helium Losses - M cu. ft.  370.4
Fuel Oil Usage - bbl.  13,761

OPERATING EXPERIENCE

Reactor Loading

The reactor loading at month end is depicted on the front face map which follows page BN-5.

Power Level

Reactor power level continued to be restricted administratively to 3500 MW due to Mark II fuel stress considerations.

Reactor Outages

The three reactor outages and their principal causes were as follows:

<table>
<thead>
<tr>
<th>Date Down</th>
<th>Outage Type</th>
<th>Outage Hrs.</th>
<th>Cause</th>
</tr>
</thead>
<tbody>
<tr>
<td>April 1</td>
<td>Scheduled</td>
<td>135.3</td>
<td>Charge-discharge and plant maintenance.</td>
</tr>
<tr>
<td>April 6</td>
<td>Unscheduled</td>
<td>8.9</td>
<td>Low-flow problems in four process tubes. Replaced three transducers and uncocked one spacer.</td>
</tr>
<tr>
<td>April 26</td>
<td>Unscheduled</td>
<td>35.6</td>
<td>125-volt DC system ground in 105-N. Replaced offending solenoid and repaired packing leak on one primary loop dump valve (V4-6).</td>
</tr>
</tbody>
</table>

EQUIPMENT EXPERIENCE

Cell 3 Restoration

All of this Cell work is complete except a small amount of concrete repair.

Steam Generator Retubing

Cell 3 steam generator secondary side hydro tests (900 psig) and the Cell 3 primary piping hydro test (2740 psig) were completed during the month. Reassembly of the primary pump was in progress at month end. Control and
instrument system calibration and testing were nearing completion.

The strike by construction laborers resulted in no Cell 4 work during the April scheduled outage, thus delaying preparation of Cell 4 piping for use in decontaminating steam generator 4B.

**Resistance Temperature Detectors**

Thirty-six all-tube temperature monitor RTDs are inoperable on the central data logger system but readable on the auxiliary scanner. This is an increase of eight over the previous month. There are 11 unreadable RTDs for a decrease of one over the previous month.

Twelve zone temperature monitor RTDs are bypassed. It is suspected that three of these are controller problems. This would be identical to the number of failures experienced last month.

During the April shutdown, 33 ATTM RTDs, including 14 unreadable units, were repaired and four ZTM RTDs were returned to service.

**Reactor Crate**

The right side rear face shield and gas seal were inspected during the April scheduled outage. The cracks in the shield appeared about the same as before, with no new cracks noted. RTV No. 106 material was applied over the failed areas to reduce gas leakage. Gas loss during equilibrium operation averaged about 9,000 cubic feet per day.

**Horizontal Control Rod System**

Fifty-five gallons of a water-glycol base fire-resistant hydraulic fluid have been received for use in a test loop designed to observe the effect of this fluid on HCR components.

Fifty-five gallons of a water-in-oil emulsion hydraulic fluid have been ordered for direct conversion of the test rod in 1705-N without any modifications. Problems are expected with existing pumps and filters when used with this class of fluids.

**PROCESS ASSISTANCE AND CONTROL**

**Reactor Physics Support**

N Reactor has been operating at 3,500 Mw, 715 MWe. Flattening efficiency is 80.2 percent, and excess reactivity held in the rods is 10 mk.

**Fuel Charging Studies**

Nine fuel charges during the last two outages have required higher-than-normal charging forces and have initiated a reassessment of fuel charging procedures, spacer design, and outer support systems of fuel assemblies.
A modified charging procedure recently put into practice will be re-evaluated during the May outage.

Examination of the seven Mark I-C fuel columns that required high charging forces during the April outage revealed fine Zircaloy shavings curled around the outer fuel element supports on as many as half of the supports in one column. The excessive number of shavings was judged to be too small to be potential fretters. Additional investigations are being made to evaluate the effects of support shoes (and shims), heavier fuels, and condition of the charging apparatus.

Studies of post-coproduct fuel spacers have shown that the 22.5-inch spacer has a greater tendency than earlier designs to become cocked in a monotube or process tube. Tests show that once a spacer cocks during charging, it can gall and severely scratch a tube. The post-coproduct spacers are being modified by aligning their supports and by increasing the distance between supports. Mockup tests of modified spacers show them to be much less prone to become cocked, and they create fewer and smaller shavings.

**Graphite Oxidation Monitoring**

Oxidation samples that had been in the reactor from June 1967 to March 1968 showed a maximum burnout rate (oxidation) of 0.27 percent per 1,000 operating days. This occurred about 80 inches from the front face of the graphite stack. Burnout values for samples on either side of the maximum burnout rate were all below 0.2 percent per 1,000 operating days. The lowest burnout rates were found for samples located in the downstream end of the graphite core.

**Primary System Radioactivity**

Front and rear elevator radiation readings were taken during the April outage. These and previous data are shown in the table below:

<table>
<thead>
<tr>
<th>Date</th>
<th>Front (mr/hr)</th>
<th>Rear (mr/hr)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Elev. Center</td>
<td>Elev. Apron</td>
</tr>
<tr>
<td></td>
<td></td>
<td>tors</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>10/3/67(1)</td>
<td>140</td>
<td>200</td>
</tr>
<tr>
<td>10/19/67(2)</td>
<td>12</td>
<td>24</td>
</tr>
<tr>
<td>2/8/68(3)</td>
<td>31</td>
<td>41</td>
</tr>
<tr>
<td>3/9/68(3)</td>
<td>20</td>
<td>37</td>
</tr>
<tr>
<td>4/5/68(3)</td>
<td>26</td>
<td>45</td>
</tr>
</tbody>
</table>

Notes:  
(1) Before reactor piping decontamination.  
(2) After reactor piping decontamination.  
(3) Four fuel failures after the decontamination.
The readings indicate that the contribution from the January fuel failures has decayed to the extent that the activity from activated corrosion products is now the principal contributor. When the readings were taken, on the five dates shown, the hours after shutdown were 912, 1296, 120, 120, and 120, respectively.
<table>
<thead>
<tr>
<th>Code</th>
<th>Tubes</th>
<th>Description</th>
<th>PT-NR</th>
<th>No.</th>
<th>Tubes</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>A</td>
<td>489</td>
<td>MK II Base (6% Pu-240)</td>
<td></td>
<td>4</td>
<td></td>
<td>Fuel Monitor Columns</td>
</tr>
<tr>
<td>B</td>
<td>162</td>
<td>MK IC (94 Metal Central)</td>
<td></td>
<td>76</td>
<td></td>
<td>Fuel for Meltdown Test</td>
</tr>
<tr>
<td>C</td>
<td>144</td>
<td>MK II Spike (6% Pu-240)</td>
<td></td>
<td>78</td>
<td></td>
<td>Increased Support Height</td>
</tr>
<tr>
<td>D</td>
<td>71</td>
<td>MK IC (94 Metal Fringe)</td>
<td></td>
<td>78</td>
<td></td>
<td></td>
</tr>
<tr>
<td>X</td>
<td>2</td>
<td>MK ICA (125 - 94 Metal Spike)</td>
<td></td>
<td>94</td>
<td>32</td>
<td>MK IV Demonstration</td>
</tr>
<tr>
<td></td>
<td>70</td>
<td>Total</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>938</td>
<td>Total PTs</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>66</td>
<td>Grand Total</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Loading Pattern - N Reactor

BN-A
FUEL AND TARGET FABRICATION - C AND K REACTORS

PRODUCTION

General

Production of AlSi-bonded fuel totaled 102 percent of forecast. Of the 230.3 tons produced, 168.8 tons of 73.3 percent were fuels with bumper or self-support rails attached.

Production of hot-die-sized fuel was limited to about 950 K5AE fuel elements processed for shakedown of the 313 line equipment.

Thoria target element production continued ahead of schedule. Initial loads for both K reactors are completed.

Acceptable Elements Produced

<table>
<thead>
<tr>
<th>Type</th>
<th>Tons Input</th>
<th>Elements to Storage - Tons</th>
<th>Yields - Percent</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Unrestricted Use</td>
<td>Upstream Use</td>
</tr>
<tr>
<td>AlSi-Bonded, Natural U</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>8&quot; Regular</td>
<td>33.7</td>
<td>22.3</td>
<td>-</td>
</tr>
<tr>
<td>8&quot; Bumper</td>
<td>-</td>
<td>0.8</td>
<td>-</td>
</tr>
<tr>
<td>8&quot; Self-Support</td>
<td>-</td>
<td>0.3</td>
<td>-</td>
</tr>
<tr>
<td>AlSi-Bonded, 94 Metal</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>6&quot; Regular</td>
<td>39.9</td>
<td>38.8</td>
<td>0.4</td>
</tr>
<tr>
<td>6&quot; Bumper</td>
<td>-</td>
<td>1.0</td>
<td>-</td>
</tr>
<tr>
<td>6&quot; Self-Support</td>
<td>153.3</td>
<td>158.9</td>
<td>7.8</td>
</tr>
<tr>
<td>Hot-Die-Sized, 94 Metal</td>
<td>5.7</td>
<td></td>
<td>-</td>
</tr>
<tr>
<td>Thoria</td>
<td>27.0</td>
<td></td>
<td>-</td>
</tr>
</tbody>
</table>

Procurement and Inventories

<table>
<thead>
<tr>
<th>Item</th>
<th>Tons Received</th>
<th>Tons Placed</th>
<th>End-of-Month Stock</th>
</tr>
</thead>
<tbody>
<tr>
<td>Natural U Cores</td>
<td>-</td>
<td>33.7</td>
<td>799.0</td>
</tr>
<tr>
<td>94 Metal Cores</td>
<td>264.4</td>
<td>198.9</td>
<td>148.0</td>
</tr>
<tr>
<td>Thoria</td>
<td>37.0</td>
<td>27.0</td>
<td>15.5</td>
</tr>
</tbody>
</table>

C-1
OPERATING EXPERIENCE

AlSi-Bonded Fuel

Time operated efficiency was 99.0 percent. This is the fourth consecutive month that 99 percent or greater efficiency has been achieved. Of the downtime, 0.6 percent was charged to operations and 0.4 percent to equipment. Canning line operations totaled 66 line-shifts at the rate of three line-shifts per day.

The total manufacturing yield was 96.3 percent. This is a record high yield and particularly noteworthy in view of the high proportion of fuel with self-supports attached. Inclusion-stain and weld reject categories were the main contributors to higher yields. Of the total rejects in each category, the percent reclaimed or restricted to upstream use was: marred surface - 43 percent, closure weld - 92 percent, rail weld - 100 percent, and AlSi slopover - 100 percent.

Hot-Die-Sized Fuel

Equipment break-in and debugging was continued on the 313 production line. The equipment performed well except for a few minor problems. An estimated yield of 94 percent was achieved through weld inspection. Following completion of a Process Work Request for 2000 elements using pierced-bottom cans, the operating personnel will be assigned to a fourth AlSi canning line to relieve a critical shortage of enriched fuel.

Thorita Target Elements

In the production of thoría target elements, total yield has improved since start-up in February. The April yield increased to 95.7 percent.

EQUIPMENT EXPERIENCE

One etch machine was removed from service for an overhaul of the indexing system. The rails were worn excessively on the ends, and the guides that hold the rails were badly worn and required complete replacement.
FUEL AND TARGET FABRICATION - N REACTOR

PRODUCTION

Statistical Summary

Input (Billets Extruded)

<table>
<thead>
<tr>
<th>Description</th>
<th>Input</th>
<th>Tons - Total</th>
<th>% of Forecast</th>
</tr>
</thead>
<tbody>
<tr>
<td>Mark I, Outers, 94 Metal</td>
<td>260</td>
<td>55.1</td>
<td>100.2</td>
</tr>
</tbody>
</table>

Output (Finished Assemblies)

<table>
<thead>
<tr>
<th>Description</th>
<th>Output</th>
<th>Tons - Total</th>
<th>% of Forecast</th>
</tr>
</thead>
<tbody>
<tr>
<td>Mark I, Outers and Inners, 94 Metal</td>
<td>1404</td>
<td>29.7</td>
<td>106.1</td>
</tr>
<tr>
<td>Mark I, Outers, 125 Metal Inners, 94 Metal</td>
<td>71</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Total Assemblies 1475 Tons - % of Forecast 72.7

OPERATING EXPERIENCE

Development Extrusions

Ten copper extrusions, fourteen extrusions of Mark IV geometry or for Mark IV components, five extrusions for Zircaloy-clad I&E fuel elements, and one Zircaloy rod extrusion for I&E end-cap stock were made.

Powder Braze Rings

Powder braze rings were used successfully in March on Mark I inner fuels, and were used on Mark I outer fuels in April with the following results:

Early in the month an attempt was made to calibrate braze Stations 1 and 2 with powder braze rings and to convert from cast to powder rings. Station 2 was calibrated and converted. Difficulties were encountered in calibrating Station 1. Thermocouple checks indicated that the induction coil was misaligned, and the freeze points of the powder rings were high, occasionally above the 970° C maximum temperature limit. The coil was re-characterized, but the station was held on cast rings. Station 3 also was maintained on cast braze rings.
Later, Station 2 began to exhibit continuously high freeze points with powder braze rings. The station was immediately recalibrated and converted to cast rings to assure meeting production requirements.

The differences between the freeze points of cast and powder rings are being investigated. Tests have since been run on Stations 2 and 3. Preliminary results indicate that the freeze points of the powder rings average 2 to 3 degrees higher than the cast rings. The use of the extra pieces of braze ring required to meet the minimum braze metal weight is possibly one of the major variables contributing to freeze point variations. Other factors also being considered are oxygen content of the rings, leak rates of the vacuum chambers, placement of calibration thermocouples, alloying effect of thermocouple metal, and other impurities in the braze rings.

Powder rings are now being used on Station 3. When the difficulties encountered in the use of these rings are resolved, powder rings will be used on all braze stations.

**Extrusion Unbonding**

Unbonding of the clad-to-core bond has been an occasional problem with both inner and outer extrusions the past few months. Three outer billets were not extruded this month, due to brittle fracture of the copper evacuation tube when the billet was removed from the preheat furnace. Immediately following the first of these evacuation tube failures, a study was initiated to determine the following:

- The correlation, if any, between copper evacuation tube embrittlement and extrusion unbonding.
- The cause of copper evacuation tube embrittlement.
- Other possible causes of extrusion unbonding.

The following areas investigated appear to be the most significant with respect to unbonding:

- It was found that the wrong I.D. weld overhang gauge was being used. This gauge allowed weld overhang to approach 0.087" (on the diameter) as compared to the acceptable maximum of 0.040". This diameter difference would result in an interference fit between the mandrel and the billet assembly I.D., possibly resulting in an I.D. weld fracture just prior to extrusion upset. Two of ten billets awaiting lubrication had excessive weld overhang.

- A billet rejected for loss of vacuum on the post-preheat vacuum test was examined and leaks were found in the rear I.D. weld. Metallographic examination of this weld is in progress. Some welding difficulties may be indicated, although nothing apparent other than this observation has developed.
Studies of the copper evacuation tube embrittlement showed that the embrittlement was not caused by chemical preparation or the presence of corrosive agents in the billet lubricant or preheat facilities.

Further tests have shown that a reducing atmosphere (which exists for production billets during preheating) is necessary to induce embrittlement of the copper evacuation tube. According to available literature, copper which contains cuprous-oxide exhibits intergranular cracking when annealed in reducing atmospheres. It is believed that this is the mechanism by which the evacuation tubes have been embrittled, since attempts to produce embrittlement in an oxidizing atmosphere have failed. Approximately 50 percent of the tubes annealed under process conditions (reducing atmosphere) have shown embrittlement.

Further work will be done to prove or disprove the cuprous-oxide reduction theory and to investigate weld integrity and strength.

EQUIPMENT EXPERIENCE

Heat Treat Temperature Readings

Audits made of heat treating indicated inaccuracies. The trouble appeared to be in the thermocouple slip ring or brushes that carry the thermocouple reading to the recorder. A temperature audit method has been devised to bypass these slip rings, and a new direct-reading heat treat recorder on order will not require slip rings and should eliminate this equipment problem.

Billet Preheat Furnaces

All types of extrusion billets are preheated in pit furnaces to the temperature required for extrusion. Of the four pit furnaces in service, two have been found to give unsatisfactory temperature gradients to the billets from top to bottom and side to side.

To improve the temperature gradients in the furnace, a sheet metal shell wrapped in asbestos cloth was installed to act as a secondary thermal baffle. This installation has been successful in reducing the billet-to-billet temperature variation from 25°F to 5°F. Another shell will be fabricated and installed in the second furnace.

PROCESS ASSISTANCE AND CONTROL

Process Flow System

A system for matching the number of inner and outer fuels processed through the support welders, final etch, and autoclaving is being developed to improve material flow. It is planned for equal numbers of inners and 'outers to arrive at final inspection for assembly; assembly then can progress without extra
handling and sorting of fuel. Fuels at support welding that are "excess" (i.e., outers without matching inners or vice versa) are stored in racks until matching fuel arrives, with the buildup of fuel in the racks alternating between outers and inners.

Elimination of extra handling by the final assembly operators, and providing for longer processing runs before switching from one fuel geometry to the other, are main objectives of the matching system.
TECHNICAL ACTIVITIES - C & K REACTORS

RESEARCH AND DEVELOPMENT

The progress on research and development work conducted by Douglas United Nuclear is reported by Mission number and title, here for the B, C, and K reactors, and for N Reactor beginning on page DN-1. Unless otherwise noted, the Missions are as defined in the July 1967 issue of this report. No coverage is accorded the two Hanford R&D Missions (6 and 9) in which DUN does not participate.

Mission 1 - Basic Production

1-A. Fuel Development

Evaluation of 50-Mil Supports - KW Reactor

This test (PTA-105) was terminated April 10, when a fuel element failed in position 23 of one of the five remaining K5E columns with 50-mil supports. The failure resulted from extensive core cracking, and six pieces of the element were recovered for examination. Column exposure was approximately 1650 MWD/T, somewhat below the goal of 2000 MWD/T. A cursory examination of the failure, and of fuel elements from the downstream positions of the column involved, revealed no support corrosion failures. The fuel columns will receive a complete post-irradiation examination.

Occlusion Plating

In an effort to improve the secondary corrosion barrier characteristics, aluminum oxide, silicon carbide and tantalum pentoxide were occluded in a nickel matrix by direct-current plating. The initial conditions chosen were the same as the optimum conditions for silicon, i.e., pH 4.0, 360 ma/cm², rotating cathode at 60 rpm, and air agitation at 35 ml/min. The average particle size of the materials was approximately 2 microns, and about 85 percent of the particles were between 1.57 and 3.13 microns. The results are shown below, with the volume of the particles held constant:

<table>
<thead>
<tr>
<th>Sample No.</th>
<th>Material</th>
<th>Amount Occluded (Vol. %)</th>
<th>Particles (Grams/Liter)</th>
</tr>
</thead>
<tbody>
<tr>
<td>CC-136</td>
<td>SiC</td>
<td>15</td>
<td>228</td>
</tr>
<tr>
<td>CC-138</td>
<td>Ta2O5</td>
<td>20</td>
<td>622</td>
</tr>
<tr>
<td>CC-139</td>
<td>Al2O3</td>
<td>20</td>
<td>286</td>
</tr>
</tbody>
</table>

These results, along with the results obtained before for silicon, demonstrate that a large variety of material can be occluded in a nickel matrix. Pulse plating studies are not yet complete, but this technique is expected to double the amount of material occluded in the nickel deposit.
Fuel Performance Monitoring

PTA-093, the AlSi-HDS depleted fuel test, is proceeding without incident; as of April 1, a 550 MWD/T exposure had been accumulated.

The monitoring of regular AlSi-bonded and HDS fuel performance is proceeding by irradiation and evaluation of routinely fabricated fuel elements under PTA-011. There are currently six PTA-011 columns in C Reactor which are scheduled to be discharged by the end of April at exposures of about 1500-1550 MWD/T. Ten additional columns are available for recharging the monitor columns.

1-B. Zircaloy Hydriding

At the time that the change to anodized spacers was made in the Zircaloy process tubes at the K reactors, it had been shown that their use would stop the buildup of a hydride case layer on the inside surface of the tubes. This hydride layer had been gradually reducing the effective tube wall thickness. However, a question remained as to the effectiveness of anodized spacers in stopping also the hydrogen buildup in the base metal of the tubes.

Fourteen Zircaloy process tubes have been removed from the K reactors during the past three months in a program aimed at resolving this question. The initial results of laboratory analyses made on these tubes indicate that the hydride pickup by the base metal may be continuing at the same rate as with unanodized spacers. Analyses of the tubes removed to date will be completed, and two additional tubes will be removed from the KE Reactor in May to obtain further samples. Laboratory studies to define the mechanism of base metal hydriding are continuing.

Consideration is being given to running thermal cycling tests on sections of hydrided tubing from a reactor to see if such cycling will cause the case layer to move. Some tests along this line were carried out early in the investigation with negative results, but it may be that the test conditions were not meaningful.

1-C. Computational Techniques

HAMMER

The reprogramming of HAMMER to couple the thermal and epithermal fluxes directly has been completed and is in the process of being checked out.

The option in HAMMER which will generate a cross section library for a homogenized lattice is in the final stages of debugging. This option will allow a supercell case with the actual target lattice surrounded by homogenized drivers to be run directly on HAMMER.
EXTERMINATOR-II

EXTERMINATOR-II has been modified to store the final flux distribution of a case in Program Complex File. This modification permits the final fluxes for each case in a series of stacked cases to be stored on one flux save-tape which can be "edited" under CUR control.

The EXTERMINATOR-II analysis of the PCTR supercell experiments has shown that inputting a value of beta can lead to noticeable savings in computer time compared to that required utilizing the EXTERMINATOR-calculated value of beta. The range of beta established for this analysis is 1.4 to 1.6. However, the best value of beta is a function of the particular type of problem being studied.

Critical Mass Studies

Preliminary work is being done on the preparation of a document which will summarize all of the work done to date in correlating computer code results with experimental results. Below is a partial list of the fuels and enrichments for which experimental data are available:

- 4.89 wt% U-235 in metallic uranium
- 3.063 wt% U-235 in metallic uranium
- 3.00 wt% U-235 in metallic uranium
- 2.10 wt% U-235 in metallic uranium
- 1.02 wt% U-235 in metallic uranium
- 5.0 wt% Pu in Pu-Al fuel elements
- 1.8 wt% Pu in Pu-Al fuel elements
- 7.1 wt% oralloy in oralloy-aluminum fuel elements
- 5.4 wt% oralloy in oralloy-aluminum fuel elements
- PuO2-UO2 fuel elements of various concentrations and enrichments

1-D. High Pu-240 Program

Depleted Uranium Analysis

The analysis of a sample from a depleted element in a monitor column has been completed. There is an apparent discrepancy in the value reported for the total uranium, and this value is being rechecked. The total uranium value is necessary to verify the sample exposure. The relative plutonium assays, plutonium grams per ton, and percent Pu-240 agree very well with those calculated, based on the initial analytical results and "weasel" data.

This preliminary analysis indicates that the specifications for the high Pu-240 plutonium for ZPFR will be met at the scheduled discharge exposure of the depleted. More samples will be analyzed for cross checking.
Depleted Uranium Fuel Failure

On April 4, a second failure of a K5 depleted uranium element was experienced in KE Reactor under PITA-048. The column exposure was 2467 MWD/T. Like the first, this failure appeared to be the result of core cracking. The minor corrosion conditions associated with the 14th position, and a visual examination of the failure, were considered in assessing the failure mode. Visual examination of the remainder of the rupture column revealed no significant corrosion on the downstream pieces, or any apparent worsening of the small external fatigue marks on the cladding at the cap and base to core junctions.

Mission 2 - Coproduct

2-A. Lithium Spline Development

The Production Test has been written for the irradiation of lithium-aluminum splines at KW Reactor, and is being routed for approvals.

2-B. Small Scale E-N Test

Work is proceeding on this test (PTA-108) which is scheduled for charging into KW Reactor late in May. The test includes 10 columns, two of which will contain lithium aluminate targets. The driver fuel for the test is fabricated from 801 alloy 210 Metal. The test will provide preliminary data to evaluate fuel and target performance in 2.1 E-N loadings.

Mission 3 - Transplutonium Technology

3-A. Pu-Al Test Irradiation

Design of the Pu-Al fuel elements and the reactor test loading has been completed. The loading will contain about 15 Pu-Al columns initially, of which one or two will contain short monitor charges. Elements from the monitor charges will be used for checking the validity of the buildup-burnout calculations one to two years after test initiation.

Preliminary development work is under way on preparation of the Pu-Al core alloy and on the fabrication process for the Pu-Al. Target criteria and specifications have been completed. The current program will involve the fabrication and irradiation of about 400 target elements having cores containing 6.0 ± 0.2 w/o Pu.

Dummy target elements have been prepared for flow testing. Flow tests directed toward evaluation of hydraulics, and chattering or vibration effects of the light weight Pu-Al targets, are scheduled to continue for a minimum of one month.

3-B. Pu-Al Safety Analysis

Critical mass calculations, using HAMMER, were made for the Pu-Al elements to be used in the proposed Pu-Al irradiation test. The critical mass values
for optimum water moderation, spherical geometry, and water reflected are given below:

<table>
<thead>
<tr>
<th>Critical Mass</th>
<th>Safe Mass</th>
<th>Safe Number of Elements</th>
</tr>
</thead>
<tbody>
<tr>
<td>2.60 kg (Pu)</td>
<td>1.17 kg (Pu)</td>
<td>40</td>
</tr>
</tbody>
</table>

The Pu-Al element dimensions and Pu content are:

- Core Diameter: 1.30 in.
- Can Diameter: 1.48 in.
- Core Length: 8.90 in.
- Pu in Pu-Al: 6.0 wt%
- Pu per element: 29.74 g (42.0 g/ft)

Mission 4 - Pu-238

4-A. Test Irradiation

Analytical results are now completed on the 15 samples which were taken from the five dissolver batches processed by Battelle-Northwest. A document is being prepared for late April issuance which reports the detailed results.

A considerable amount of work is currently being done by both BNW and ARHCO on the measurement of Pu-236 contamination in Pu-238. The results of most of this work should be available by the end of April.

4-B. Irradiation of Uranium with a High U-236 Content

Six striped charges have been charged into KE Reactor under PTA-107. The two types of 94 Metal contain 300 and 1000 ppm U-236, respectively. Discharges are scheduled for 300, 600, 900, 1300, 1600, and 1900 MWD/T. The test is proceeding without incident, and is currently at an exposure of about 200 MWD/T.

4-C. Alternate Matrices

Thin-wall hollow pellets of pressed metallic magnesium have been prepared for evaluation of the effects of water penetration of targets containing such pellets. The pellet void space is designed to accommodate increased core volume caused by corrosion products, which is expected to minimize distortion of the target cladding during a cladding failure.

4-D. Neptunium Irradiation Cost

A cost study of Pu-238 production from Np-237 indicated that additional fuel costs and loss of Pu-239 production accounted for nearly half of the total conversion cost exclusive of the neptunium cost. Because of the cost sensitivity, refinements have been made to this portion of the PUPA code to permit more reliable parameter evaluation.
Conversion ratios and U-235 burnouts are being calculated by MOFDA for the drivers and the depleted uranium elements at the column ends for various target densities and target-to-driver exposures. These more precise values will be incorporated into the PUPA code.

**Mission 5 - Other Isotopes**

**5-A. Uranium-233**

The uranium isotopic assays of U-233 at 8 ppm U-232 in the cores of the C and K reactors have been calculated. The results are given below:

<table>
<thead>
<tr>
<th>Isootope</th>
<th>C ppm</th>
<th>K ppm</th>
</tr>
</thead>
<tbody>
<tr>
<td>U-232</td>
<td>8.0</td>
<td>8.0</td>
</tr>
<tr>
<td>U-233 (wt%)</td>
<td>98.27</td>
<td>94.48</td>
</tr>
<tr>
<td>U-234 (wt%)</td>
<td>1.8</td>
<td>1.5</td>
</tr>
<tr>
<td>U-235 (wt%)</td>
<td>0.03</td>
<td>0.02</td>
</tr>
<tr>
<td>U-233 (g/ton ThO2)</td>
<td>2141</td>
<td>1600</td>
</tr>
</tbody>
</table>

The above results assume no U-238 impurity initially in thoria and a cooling time such that essentially all Pa-233 has decayed to U-233.

**5-B. Cobalt-60**

One column of target elements containing Battelle-Northwest cobalt strips was discharged from C Reactor when the strips reached calculated activity levels of about 100 Ci/gm. These strips are undergoing calorimeter measurements. Irradiation of a second column of these strip targets is continuing.

The Production Test authorizing the installation of two Haynes Cobalt alloy tubes, one in C Reactor and one in a K, is being routed for approvals.

**Mission 7 - Target Space Enhancement**

**7-A. High Power Density Fuel**

150 Percent Power Demonstration Test (PTA-067)

Radiometallurgical examination of the 125 Metal element failure which occurred on February 25 at KW Reactor indicated the failure mechanism was water penetration through the female end weld bead due to mechanical damage of the bead. There was no evidence of core failure. During the March 26 outage, 26 columns of 125 Metal were discharged at an average exposure of 1137 MWD/T. Six weighed and measured columns of these 26 will be examined on a high priority basis to provide a basis for possible extension of the discharge exposure from 1500 MWD/T to 2000 MWD/T for the remaining 30 columns of 125 Metal under PTA-067.
Uranium Core Alloy Test

On March 30, the remaining 42 columns of test fuel authorized under PTA-098 were charged into C Reactor for a total of 155 columns. As of April 1, the original 113 tubes had an average exposure of 550 MWD/T. After the next startup, Panellit data will be compiled in an attempt to determine any indications of fuel swelling of the monitor columns. The first seven monitor columns will be discharged at an exposure of 800 MWD/T.

High Power Corrosion Test

PTA-138 has been written to cover the investigation of aluminum cladding corrosion behavior at conditions simulating 200 percent power operation. This test would authorize a six-column loading of C and J fuel elements in the small Zircaloy-2 tubes at C Reactor, probably in May or June. A unique feature of the test will be the measurement of the cladding temperature on one of the fuel elements in each column.

Alternate Aluminum Alloys

Evaluation of candidate cladding alloys is proceeding by fabrication and corrosion testing. Fuels have been fabricated by the hot-die-sizing process with each alloy. Metallographic examination of the fabricated cladding and bond layers is under way.

Initial corrosion tests are being conducted at 360°C in an autoclave, and at 130°C in a flow loop at 25 ft/sec. coolant flow in reactor process water. Samples are being exposed for periods of 10, 30, 60, and 90 days. The ten-day exposure specimens have been discharged and are being evaluated for corrosion penetration. The autoclave specimens showed no visually apparent accelerated corrosion. The ten-day flow loop specimens also showed no visually apparent excessive corrosion.

Preliminary visual observation of the 30-day autoclave specimens indicates minor blistering on the high silicon alloys. An AlSi control specimen apparently oxidized completely and showed considerable anisotropic growth. There was no apparent accelerated attack on the other alloys. Evaluation for corrosion penetration is being initiated on these specimens.

7-B. Reactor Modernization

General

Reactor modernization by process channel enlargement is the subject of this month's appended Feature Report.

Operation of C Overbore Test Facility

C Reactor was scrammed on March 30 by a leaking front nozzle cap on tube 3165 of the 44-tube overbore block. It was a spline cap, but investigation indicated the leak was the result of a blown-out O-ring gasket. There was
no plausible reason why the O-ring should have operated for a month and then suddenly fail. It was possible in the laboratory to blow the ring out when it was not properly installed. It also appeared that a little larger and stiffer ring should work better. Accordingly, all front O-ring seals were changed out to a larger and stiffer ring.

An additional loading adjustment in the vicinity of the Test Facility was accomplished during the C Reactor outage in early April. The average column power is now about 2080 kw, with a peak column power of 2340 kw. The highest ribbed tube fuel column outlet coolant temperature is now 105 C; the highest self-supported fuel column outlet coolant temperature is 110 C. The block continues to show significantly lower power in the center as compared to the periphery, and efforts to flatten the power will be made with splines and rod movement. The average exposure in the block was about 330 MWD/T on April 20.

Graphite temperature measurements at C demonstrate that significant temperature reductions can be achieved by overboring. The current temperature of thermocouples specially installed in the overbore block are comparable to temperatures elsewhere in the reactor, even though specific fuel power is 50-60 percent higher. It is estimated that had there been no increase in fuel power, overboring by one inch would have resulted in a temperature reduction of approximately 200 C.

Mockup Tests for K Reactor Overbore

Construction of a mockup simulating projected K reactor tube channel distortion will begin within the near future. Preliminary estimates of future channel distortion are based upon K reactor measurements and limited data from test reactor irradiations of K reactor graphite samples. It will probably be necessary to wait 4-6 months for additional test reactor sample data before meaningful mockup work can begin on tooling.

The present estimate of future channel distortion appears sufficiently severe that it is questionable whether the present methods of channel enlargement will permit more than a few years of trouble-free fuel charging once turnaround is reached. The time at which turnaround occurs, and graphite expansion begins, appears to be a critical factor since tube bending is expected at locations in the channel where straightening is not now possible with present overboring techniques.

7-C. Highly Enriched Fuel

J-metal fuel (seven percent oralloy in aluminum) to be used in the PCTR test is being shipped from KE Reactor to the 300 Area shops to be modified. This test will also include 94 Metal fuel and Li-Al targets. The current plan is to perform the experiment starting in June.

Mission 8 - Nuclear Safety

8-A. Heat Decay Test Evaluation

The computer program has been revised so that runs can be made economically
for times up to 2000 seconds after power loss. Films recording the transient flow rates during the test are to be viewed again to check the coolant flow rates between 100 and 500 seconds, because it is in this time interval that the greatest discrepancy between test and calculated values occurs. However, the discrepancy amounts to only 10 percent which could be handled as an error in the limit calculations.

8-B. Fuel Heating Tests

Six fuel heating tests with unirradiated aluminum clad fuel elements were run to further investigate the uranium-aluminum reaction and its potential effect on the accident predictions. Three fuel elements were heated in aluminum process tube sections, with results indicating that the aluminum tube added to the metal-metal reaction and resulted in almost complete uranium melting in some cases. No reactions were observed between the Zircaloy process tubes and aluminum or uranium. Some of the fuel element diameters after the uranium-aluminum reaction indicated that it would be likely for at least one element in a channel reaching 1000°C or higher to plug the channel.

Mission 10 - Columbia River Studies

10-A. Deionized Water Studies

Following the experiment in which unexpectedly high effluent activity was obtained when zirconium-clad fuel charges were exposed to process water, it was decided to see if the results from a previous experiment in which low effluent activities resulted from the use of Zr-clad fuel and deionized water could be duplicated. The data from this experiment are incomplete; however, effluent activities are decreasing. The effluent from the zirconium tube contains more activity than that from the aluminum tube, which is contrary to the previous results. This is attributed to the fact that a new aluminum tube was put in service and has not developed a complete surface-deposited impurity layer. The difference in effluent activity is thus attributed to differences in the activation products formed in the available deposits.

10-B. Water Treatment Variables

"Cat Floc," a positively charged polyelectrolyte, is now being used as the coagulant in place of alum in the Water Treatment Pilot Plant. The main objective in this test is to eliminate the aluminum that passes through the filter. It has been theorized that the residual aluminum may furnish the bulk of the material that forms the film on in-reactor surfaces, and as such, furnish impurities either through codeposition or exchange.

The initial experiment involved "Cat Floc" addition at a rate (ca 0.5 ppm) sufficient to maintain neutral surface potential. At first, Separan was added as a filter aid, but high filter effluent turbidities prompted elimination of this additive. Even so, the filter effluent turbidities remained higher than the normal 0.05 Jackson unit (J.U.). Effluent turbidity increased after backwash to about 0.4 J.U. and decreased through the run to 0.1 J.U., probably due to the formation of a "schmutzdecke."
10-C. Computer Study

Previous reports have discussed the use of the analog and digital computer in developing the model of the kinetics of activity buildup on reactor process tubes, on fuel elements, and in the coolant. In the initial work with the analog computer, it was found that the coefficients relating to the deposition and release of material from the surfaces were interrelated.

The relationship obtained is \( B = aA + b \), where \( A \) and \( B \) are the release and deposition coefficients, and \( a \) and \( b \) are assumed to be constant. Furthermore, \( a = \frac{m}{k_e} \) and \( b = \frac{m}{k_e} (k_e f k_i (1 - V) - \lambda_0) \), where \( m \) is a geometry factor for the system, \( k_e \) is the equilibrium constant which relates the concentration of the material on the film to an equilibrium liquid concentration, \( f k_i \) is the rate at which particles deposit on the surface by methods other than pure diffusion, \( 1 - V \) is the fraction of the material in particulate form, and \( \lambda_0 \) is a rate coefficient for material leaving the surface in particulate form.

In this particular study, the activation of arsenic, present in concentrations of less than one part per million, has been examined. The test facilities consisted of two tubes in KE Reactor charged with aluminum clad fuel. One tube was Zircaloy and the other aluminum.

It has been found that good fit between experiment (which contained large unexplained variations in the data) and theory, using the digital computer, is obtained when it is assumed that the arsenic could form two chemical species. The parameters calculated, based on this assumption, are:

For the aluminum tube:

- \( a \): 40, 85
- \( b \): \( 4.3 \times 10^{-10} \), 43

The data scatter was such that \( a \), for species (2), ranged between \( 4.3 \times 10^{-10} \) and 20.

For the Zircaloy tube:

- \( a \): 1100, 17
- \( b \): 38, 40

The calculations also indicated that species (1) in each tube was the same. A similar conclusion was drawn for species (2).

Using the above data, while realizing that some of the numbers are more qualitative than quantitative, the values for \( k_e \), \( k_f_i \), and \( \lambda_0 \) were calculated. Additional assumptions made were that \( f k_i \) and \( \lambda_0 \) are the same for both tubes for a given species, and that \( 1 - V = 0.02 \). The resultant values are:
<table>
<thead>
<tr>
<th>Term</th>
<th>Aluminum Tube</th>
<th></th>
<th>Zircaloy Tube</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Species (1)</td>
<td>(2)</td>
<td>Species</td>
<td>(1)</td>
</tr>
<tr>
<td>$k_e$ (cm$^{-1}$)</td>
<td>0.2</td>
<td>0.4</td>
<td>0 to 1.9x10$^4$</td>
<td>0.006</td>
</tr>
<tr>
<td>$f_{k1}$ (cm/sec)</td>
<td>0.15</td>
<td>0.08</td>
<td>0.15</td>
<td>0.08</td>
</tr>
<tr>
<td>$\lambda_0$ (hr$^{-1}$)</td>
<td>0.05</td>
<td>0</td>
<td>0.05</td>
<td>0</td>
</tr>
</tbody>
</table>

All of the values calculated for $k_e$ are much higher than the value of 0.0003 cm$^{-1}$ obtained from an independent experiment for aluminum. Also, the $k_e$ data for the Zircaloy tube are smaller than those for the aluminum tube, which is contrary to expectation.

The kinetic deposition factors, $f_{k1}$, are of the order of calculated values. The calculated factors were obtained using the assumption that the particles were of the order of 0.2μ in diameter.

No evaluation of the $\lambda_0$ values will be made at this time owing to insufficient data.

10-D. 100-BC Trench Disposal Study

Discharge of C Reactor effluent to the trench continued at the fairly steady rate of 50,000 gpm. Water levels in wells in the immediate vicinity and within the 100-BC Area stayed at essentially the same levels as noted last month, but slight increases in elevation and temperature were noted in more distant wells (the first definite indications of temperature increases in these wells).

Temperatures in wells within 100-BC Area showed some decrease during this report period, and the temperature of riverbank springs north of the disposal site continued to decrease slightly. The several possible causes for these temperature decreases are being investigated. Analytical results of weekly riverbank spring samples showed no significant change in radioisotope concentrations that might be attributed to the effluent disposal test.

10-E. Effluent Disposal by Cribbing

The effluent cribbing test facility in 100-BC Area has been discontinued temporarily due to excessive leakage from the 107-C waste tank from which reactor effluent has been diverted. Repairs to the tank have been scheduled; until these are made, this test facility will probably not be operated.

ENGINEERING AND TECHNOLOGY - REACTORS

Irradiation of O-Type Fuels in C Reactor

PTA-127, "Irradiation of Surplus B Reactor Fuel in C Reactor," was approved, and 20 mixed fuel columns were charged into C Reactor on April 2. Seven thermocouple probes were charged in test columns to evaluate temperature distributions. Data from these probes confirmed expectations showing maximum
R values of 1.00 to 1.12 at the end of the fuel columns. Coolant flow rates in the process tubes were essentially the same as those in normal full-length fuel columns. On this basis, charging of fuel columns containing equal lengths of C- and O-type fuel elements with an O-type mixer element between the column segments was authorized. About 200 fuel columns were charged with the mixed fuel pattern during the April 14 outage.

Measurement of Heat Decay in Freshly Discharged Fuel

Charging of the fuel elements, and installation of the rear nozzle flapper cap, were completed at KW Reactor under Supplement 2 of PT-IP-622-A (the radiological and physics investigation of fuel discharged from an operating reactor).

Replacing Carbon Dioxide with Nitrogen at C Reactor

A document describing the economic evaluation of substituting nitrogen for carbon dioxide at C Reactor for the production of 12 percent Pu-240 plutonium has been drafted. The study showed no economic incentive for the use of nitrogen for this application under current operating conditions.

C Reactor Last-Ditch Coolant System Flush Test

A flush test to determine the capability of the C Reactor diesel system to pump emergency coolant to the reactor was performed April 4. The results show that deactivation of B Reactor has resulted in an assured last-ditch system coolant flow increase of approximately 12 percent to C Reactor. Sixteen percent increase was expected, but did not materialize because of unexplained line pressure losses between 182-B and 105-C Buildings which were greater than measured previously.

Radiation Dosimeter Development Program

A program has been undertaken to develop pocket dosimeter devices suitable for reactor service conditions. This program was initiated after a survey of commercial units indicated certain inadequacies in all of those currently marketed. Generally these deficiencies were related to battery life, sensitivity, and alarm audibility under reactor maintenance conditions.

Priority in this program is being given to the development of the pocket dose rate alarm over the pocket dose alarm integrator. Work this period resulted in development of a more intense audio annunciator which produces a warbling tone over the frequency range of 200-2000 hz. Development and testing continues on two different count rate-alarm set point circuits. Prototype units are scheduled to be available for field testing during the next several months.

Iodine to River Monitoring System

With the completion of development effort on instrumentation which will provide quantitative data concerning the radioactive iodine-131 content of reactor effluent water which is discharged to the Columbia River, an engineering study was conducted to determine the requirements for installation of a
complete monitoring system at each of the operating reactors. The results of this study are reported in DUN-4014, issued April 8. Design criteria are in preparation in support of project proposal action.

**Tooling for C Reactor VSR Channel Renovation**

Final testing of all of the equipment required for VSR channel renovation in C Reactor has been completed. Operation of the machines has been demonstrated a number of times for plant personnel, including craft foremen. All of the tooling and machines are now in the shop for a final overhaul and painting. The footage of movie film shot during the most recent demonstration is being edited into a training film for use as a craft training aid.

**Thermal Shield Cooling Tubes – C Reactor**

Studies have been under way for some time to determine if a failure potential exists at C Reactor in the top thermal shield cooling tubes due to continued graphite subsidence. These studies have now been completed.

Test data obtained on a tube deflection mock-up indicate that no serious cooling tube failure is likely to occur at C Reactor. This same conclusion was previously reached relative to the top thermal shield cooling tubes in the K reactors following a similar series of mock-up tests in which the K situation was duplicated.

**Resistance Temperature Detector Study – K Reactors**

Previous reports have indicated that the basic cause of the immersion-type RTD failure problem at the K reactors is related to moisture penetration of the silicone rubber cable connector molding.

One of the principal difficulties with these RTD units is related to a breakdown of insulation resistance of the Kovar seal terminal pin in the presence of water. This insulation breakdown has been found in all of the units which have failed to date due to low resistance to ground (which comprises the bulk of the failures). Laboratory work has shown that the insulation resistance of this seal area may be satisfactorily restored by an outgassing of the seal while immersed in a silicone oil bath under moderate temperatures. It has also been determined that an outgassing treatment was not applied to these seals by the RTD manufacturer. The apparent failure of many of the repotted test units is also considered attributable to the seal deficiency.

A number of new laboratory and on-reactor test units have been prepared utilizing the seal outgassing technique. Based on accelerated laboratory testing, it now appears that a satisfactory repair technique can be devised utilizing an epoxy-based repotting compound in the connector cable terminal area coupled with a Kovar seal outgassing preparatory treatment. Preliminary tests have substantiated this conclusion. Fabrication of a larger number of units for both accelerated laboratory and on-reactor testing is in progress. If this repair method proves satisfactory, it will be necessary to repot both existing Stores stock and salvaged units from the reactors for reuse on a routine basis. This should provide sufficient extension of current RTD life to
permit the orderly development and demonstration of a longer-term replacement RTD for use at a later date.

An on-reactor inspection was made of the strap-on RTDs installed on the rear face of KW Reactor. Based on the results of this inspection and detailed evaluations of 32 units, it is concluded that no mass deterioration of the H film window in these RTDs should be expected in the near future. A number of the RTDs inspected showed varying degrees of deterioration due to water leak damage, but only two showed any significant amount of H film deterioration and these were in areas of large solids buildup. Development work has been initiated to devise a chemical or mechanical method of removing these solids from the process tube connectors so as to provide a clean surface for strap-on RTD installation. The inspection did indicate that some loosening of clamp tension was occurring on these RTD units due to permanent set of the rubber molding material. Laboratory work is under way on a new constant-tension spring clamp for these RTD units which will provide a constant pressure between the RTD body and the pigtail connector.

Project Engineering - Reactor Facilities

Project Status Summary

The status of approved construction projects relating to B, C, and K reactor facilities is summarized in Appendix A (attached).
Mission 1 - Basic Production

1-A. Mark IV Fuel Development

A column of Mark IV fuel assemblies with Zircaloy end-spider supports was discharged from the out-of-reactor flow loop after 225 hours exposure (about five percent of goal) to 310 gpm of 570 F water. The end-spiders showed no signs of fatigue cracking. Preliminary data from an instrumented assembly in the loop at low temperature and pressure indicate that an inner element vibrates from an excitation force of 10 to 15 pounds in a frequency range of 31 to 74 Hz, which brackets the calculated natural frequency of the system at 59 Hz. Confirmatory measurements at reactor temperatures and pressures are scheduled.

1-B. Thermal Hydraulics Studies

Battelle-Northwest's Thermal Hydraulics Laboratory completed a number of important experiments:

- They confirmed experimental calculations that temperatures are significantly higher under fuel supports than on uncluttered cladding. However, boiling burnout on a Mark IV test section occurred on uncluttered cladding.

- A series of boiling burnout heat flux measurements was completed over the expected range of Mark IV operating conditions, and also at extremes of heat flux approximately three times the expected in-reactor maximum. (Maximum laboratory heat flux was $1.8 \times 10^6$ Btu/Hr-ft$^2$.) These measurements support the Mark IV process and safety limits.

- A set of experiments to provide boiling heat transfer data for low quality 2-phase mixtures was completed. These data are inputs to the "trip-after-instability" analysis.

1-C. Process Control Requirement Following Fuel Failure

Process tube temperatures resulting from direct contact between a process tube and an operating Mark IV-M fuel element were calculated and are plotted as a function of lattice specific power in Figure DN-1 below. These temperatures are based upon a local primary coolant temperature of 460 F and can be applied to a different case by adding the difference between assumed coolant temperature and 460 F. The results pertain to a Zircaloy-2 clad uranium element in contact with a process tube. If the uranium were converted to an oxide, the calculated temperatures would be much higher.
Figure DN-1. Calculated Process Tube Surface Temperature Resulting from Contact by a Mark IV-M Fuel Element (coolant at 460 F)
The calculated values (1,650 F at current Mark IV specific power limit) appear sufficiently high to warrant either an evaluation of the credibility of the assumed accident and/or an evaluation of process tube integrity under the computed conditions.

1-D. Zircaloy Process Tube Tests

Stress-rupture testing of two sections of irradiated process tube 1756, which was removed from N Reactor in 1966, has reached the 6,000 to 8,000 hour level. (Testing is scheduled to stop at 10,000 hours.) Figure DN-2 depicts the elongation behavior to date. Note that there is a difference of almost a factor of two between the two samples, although their fast flux exposures differ by only 14 percent.

1-E. Coated Graphite Studies

Optical microscopic examination of a Marquardt Company coated TSX graphite sample which had been oxidized (920 C, 6,000 ppm H2O, 153 hours) to 1.1 percent weight loss revealed no differences in structural appearance from an unoxidized control. The SiC-graphite interface showed very little root structure, i.e., penetration of the coating material into the graphite pores. However, the bond between the coating and the graphite was quite strong, suggesting compound formation at the interface. Unexpected difficulty, because of their hardness, was experienced in cutting samples with a 3-mil thick coat for microscopic sample preparation.

1-F. Computer Code, DCODE

With a few minor exceptions, DCODE comparisons to other models (MOFDA, FLEX) and to experimental data indicate that the development phase of the code is over. The code therefore is being "frozen" in its present form, and its use is being permitted on varied problems so that some feedback may be obtained. The tabulation below presents a brief but current summary of both the good and the deficient points of the code as compared to other experimental models:

<table>
<thead>
<tr>
<th></th>
<th>Experiment</th>
<th>MOFDA</th>
<th>FLEX</th>
<th>DCODE</th>
</tr>
</thead>
<tbody>
<tr>
<td>N Mk I, 2100 MWD/T</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>U-235</td>
<td>0.735%</td>
<td>0.731%</td>
<td>0.728%</td>
<td></td>
</tr>
<tr>
<td>Pu-239</td>
<td>85.95</td>
<td>86.01</td>
<td>85.97</td>
<td></td>
</tr>
<tr>
<td>Pu-240</td>
<td>11.50</td>
<td>11.50</td>
<td>11.57</td>
<td></td>
</tr>
<tr>
<td>Pu-241</td>
<td>2.35</td>
<td>2.30</td>
<td>2.24</td>
<td></td>
</tr>
<tr>
<td>Pu-242</td>
<td>0.18</td>
<td>0.182</td>
<td>0.18</td>
<td></td>
</tr>
<tr>
<td>N-Mk I</td>
<td></td>
<td>1.0265</td>
<td>1.0272</td>
<td></td>
</tr>
<tr>
<td>C-22N</td>
<td>1.0162</td>
<td></td>
<td>1.0179</td>
<td></td>
</tr>
<tr>
<td>C3E</td>
<td>1.1191</td>
<td>1.1195</td>
<td></td>
<td></td>
</tr>
<tr>
<td>CMLE</td>
<td>1.0138</td>
<td></td>
<td>1.0164</td>
<td></td>
</tr>
</tbody>
</table>

DN-3
N-16: 10,325 psi (57,000 psi hoop stress); original section 27" long, located 19.75 feet from inlet end; exposure = 0.99x10^21 nvt, fast.

N-14: 10,737 psi (59,000 psi hoop stress); original section 23" long, located 31.25 feet from inlet end; exposure = 1.13x10^21 nvt, fast.

Figure DN-2. Stress-Elongation Relationship of Irradiated Process Tube No. 1756 at 300 C.
Poor Agreement

<table>
<thead>
<tr>
<th>Experiment</th>
<th>MOFDA</th>
<th>FLEX</th>
<th>DCODE</th>
</tr>
</thead>
<tbody>
<tr>
<td>N-Mk I, 2100 MWD/T</td>
<td>0.750</td>
<td>0.746</td>
<td>0.711</td>
</tr>
<tr>
<td>gM Pu/MWD</td>
<td>0.056%</td>
<td>0.050%</td>
<td>0.039%</td>
</tr>
<tr>
<td>Pu-238</td>
<td>576.</td>
<td>515.</td>
<td></td>
</tr>
</tbody>
</table>

1-G. Bottom Shield Protection Test

A test was performed to determine the feasibility of protecting the bottom shield insulation layer against excessive temperature differentials as observed during the 4,800 MW probe. The test consisted of swapping fringe control rods in bank 1, which lies just outside the first fuel lattice, while adjusting other control rods to maintain constant power. The test results indicate a reduction of more than 20 percent in the delta T across the insulation layer, and they suggest the possibility that higher power levels can be attempted on a test basis by using the rods as fringe protection, thus avoiding a commitment to a fringe blanket poison loading until routine operation at the higher power is possible. Swapping three fringe rods into the top and bottom fringe banks resulted in about a one percent decrease in reactor flattening efficiency.

1-H. Advanced Technology Case

A report is being prepared presenting the following potential methods of meeting the flux monitoring requirements of the Advanced Technology Case:

A Flux Profile Monitoring System: This system would consist of a traveling chamber that would obtain side-to-side flux traverses in ten axial graphite cooling tubes in eight vertical planes. Such chambers and equipment have been developed for use on power reactors and could be modified for adaptation to N Reactor geometry. Signal outputs from the chambers would be fed into a process computer which would determine a tube power limit applicable to all tubes surrounding the axial profile location. This system should allow operation up to within two and one-half percent of the specific power limit, as compared to the five percent approach allowed by the present TBNM system.

A Fixed Chamber System: This system would consist of 60 fixed chambers in the graphite cooling tubes using 12 front-face locations with five axial chambers at each face location. The signal from each chamber would be normalized to the profile determined by the traveling chambers. This system would be useful to rapidly detect any changes in the profile determined by the traveling chambers as caused by such things as xenon cycling or improper control rod movements. This system would also be connected to the reactor safety circuit to protect all fuel elements from approaching the burnout limit, and to assist in protecting other operating limits (e.g., process tube temperature, confinement, and fuel temperature). This would be accomplished through use of a power setback trip and a reactor scram trip in the safety circuit.
Mission 2 - Coproduct

2-A. Fuel Element Performance

Irradiation swelling and length change have been measured on driver elements from five Mark I1 monitor columns. These columns experienced average exposures between 2,255 and 4,339 MWD/T. A maximum swelling rate of 0.4 percent per 1,000 MWD/T was observed which is well within the previously reported maximum of 0.87 percent per 1,000 MWD/T.

The length change data indicate a non-linear relationship between length and exposure. It appears that anisotropic contraction occurs during the early stages of irradiation. Since swelling elongation occurs throughout the irradiation period, the net effect is one of initial shortening followed by a tendency to lengthen at high exposure.

Complete radiometallurgy examination of all seven Mark I1 fuel element failures confirms the hot-spot corrosion mechanism developed for the first three as described in DUN-3149 2, titled "The Causes, Mechanisms and Rates of Failing of Mark I1 Fuel Elements (Failures 39, 40 and 41)." The failure-free fuel performance which has been realized during almost three months of operation at reduced severity conditions indicates the correctness of the analysis.

Corrosion testing of actual Mark I1 fuel cladding in autoclaves showed predictable, fairly uniform corrosion at 400 C and 450 C. Tests at 475 C and 500 C showed highly non-uniform attack during the first day of exposure, and penetration of 25-mil-thick samples within one to five days. The reason for the non-uniformity of the attack is not yet clear. From United Kingdom test results, pressure is believed to accelerate the attack markedly. Experiments with heat transfer surfaces, simulating fuel cladding, also showed some acceleration.

2-B. Target Integrity

Basin water tritium levels have been persistently high during the past month even though target stripping had virtually stopped. This may indicate that stripped cores, stored in the basin, are being breached by corrosion of the aluminum claddings. However, cursory visual examination of older vintage stripped cores did not substantiate this. From basin sample analyses it is estimated that approximately 484 curies of tritium (186 cc) were released during March. For the first half of April, it is estimated that an additional 350 curies (134.5 cc) of tritium have been released in the basin.

Tritium concentrations in the primary loop are still in the range of normal levels for reactor operation, indicating an absence of in-reactor target failures during current operation.

Mission 3 - Transplutonium Technology

There were no significant developments on this program.
Mission 4 - Pu-238 Program

The irradiation of seventeen columns of fuel with high U-236 content continued under PT-NR-95. The remaining five columns will be charged during the July outage, so there will be no need to discharge them during the planned long operating run this summer.

Mission 5 - Other Isotopes

There were no significant developments on this program.

Mission 7 - Target Space Enhancement

There were no significant developments on this program.

Mission 8 - Nuclear Safety

Loss-of-Coolant Experiments

A pivotal area in the measurement of metal-water reactions on full-sized fuel elements is the capability to measure continuously the hydrogen evolution as sudden changes in fuel geometry occur. Battelle-Northwest has developed a continuous hydrogen measurement technique using a mass flow meter to measure total effluent gas flow, and a thermal conductivity meter to measure the percent of hydrogen. Using this technique, the first reaction rate laws have been developed for a full-sized unirradiated N Reactor fuel element. This is a major step in the R&D program and a significant contribution to the nuclear industry knowledge.

The measured rate for the reaction between Zircaloy and steam was 58 percent of data reported by Argonne for 1,100 C, 63 percent for 1,200 C, and from 71 to 110 percent for 1,000 C. These are favorable results, indicating less of a reaction at zirconium temperatures near the uranium melting point (1,090 C) than was assumed for the molten pool fuel model of N Reactor. Future utilization of this technique in tests with irradiated fuels (both below and above the uranium melting point) will provide needed answers on the effects of cladding distortions, fuel failure, and uranium melting on the reaction rate.

ENGINEERING AND TECHNOLOGY - N REACTOR

Process Tube Examinations

Process tubes 1550 and 0355, which experienced high charging forces during charge-discharge in April, were inspected with the TV probe and a borescope. Examination indicates that charging scratches are broader, but not necessarily deeper, in the bottom sections of these tubes than on tubes previously examined. Maximum scratch depth was 0.002 inch.
Process tube 2647 was examined, primarily for evidence of post-coproduct spacer vibration marks. There was no conclusive evidence of such fret marks.

Decontamination Side Effects

Some portions of the 1310 Building piping, which had been in contact with decontamination solutions during the filling and recirculation of the chemical waste storage tank, were examined for corrosive attack. The 24-inch fill line appeared unaffected, with no indications of pitting or localized attack. The 12-inch recirculation line next to the recirculation pump showed some small pits of about 0.625-inch diameter, but otherwise it appeared to be in good shape. The 1.5-foot-long spool piece that was removed to gain access to the line was covered with large pits up to 0.25-inch diameter and 0.1975-inch deep.

Project Engineering: Reactor Facilities

Backup Boiler Facility

The new schedule for boilout is now May 2. Acceptance Test Procedures prepared by Burns and Roe, Inc., have been received for comment. These ATPs will demonstrate the acceptability of this installation, and the capability of the system components to backup the existing boiler facility.

Project Status Summary

The month-end status of approved construction projects relating to N Reactor facilities is summarized in Appendix A.

ENGINEERING & TECHNOLOGY - FUELS & TARGETS

Uranium Billet Development

Uranium billets fabricated by the upset-forge route were received on a development order. Nine were rejected for exceeding dimensional specifications or because of poor surface quality. The remaining 36 are being coextruded and processed to gain production experience with forged billets. Three coextrusions were expedited through inspection, heat treat, and clad-and-bond stations. Data thus far from these coextrusions indicate that O.D., I.D., wall thickness variation, warp and bond are no different than for billets coextruded from standard process (drill-machine) uranium billets.

Further upset-forge preshape development work was performed by Reactive Metals, Inc., on 24 Mark IV inner billets. The uranium for this work was not high U-236, so these billets will not be shipped to Hanford for coextrusion. The main objectives of this forging campaign were to perfect process flow, to finalize tooling dimensions, and to improve surface quality. Based on preliminary results, it appears that the use of a water-cooled mandrel, and billet oxidation prior to forging, will improve I.D. surface quality which has been a problem on Mark I forging campaigns.
Mark I to Mark IV Billet Conversion

Several hundred Mark I outer billets presently in inventory will be converted to the Mark IV fuel geometry. All associated Mark I components, with the exception of the inner Zircaloy component, are also convertible. To reduce the billet I.D. from 2.800 inches to 2.510 inches, a sleeved billet technique is being tested wherein a uranium sleeve acts as a bushing between the billet and the Zircaloy inner clad. In the resultant coextrusion, this sleeve occurs as a 0.012 inch thick layer between the Zircaloy cladding and the uranium-uranium interface.

Four coextrusions, under PWR 298, have been made utilizing this technique. One coextrusion was unbonded in the Zircaloy cladding-uranium interface, due apparently to an evacuation problem.

Results of the other three coextrusions showed no significant difference in dimensional attributes from ordinary extrusions. Clad-to-core bonding was excellent. Metallographic examination of the uranium-uranium interface showed complete metallurgical bonding along the bond path. Sixteen more test extrusions are planned in May.

Mark IV Zircaloy Extrusions

Three extrusions of Zircaloy tubing have been made for manufacture of the inner cladding components for use in the Mark IV outer tube campaign in May. Extrusion parameters were as follows:

<table>
<thead>
<tr>
<th>Component</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Liner</td>
<td>7.580 inch</td>
</tr>
<tr>
<td>Die</td>
<td>2.575 inch</td>
</tr>
<tr>
<td>Mandrel</td>
<td>1.682 inch</td>
</tr>
<tr>
<td>Temperature</td>
<td>1200 F</td>
</tr>
<tr>
<td>Billet Reduction</td>
<td>16.52</td>
</tr>
</tbody>
</table>

These extrusions were produced to size on the I.D. and oversize on the O.D. to allow for machining to assure conformance to wall thickness specifications. The Zircaloy billets used were excesses from the previous Mark II inner clad program.

The "two step" technique was utilized, with one inch of graphite in the first step. In this procedure the press is stopped before the extrusion clears the die. The follower block is then replaced by another block with a small mandrel affixed to it. The container is then closed and the extrusion is completed. This technique allows extrusion of a longer billet by reducing the amount of graphite in the first step, and reduces loss of material in the rear end defect. Fifty-five pieces were realized from the three extrusions.

DN-9
ADVANCED CONCEPTS AND PLANNING

MFCs AND OTHER PROPOSALS

MFC-12 - New Capabilities for Production Reactors

Nothing to report this month.

MFC-13 - Study of Application for Hanford Produced Cobalt-60

Copies of DUN-3938, the final report on this MFC, have been forwarded to AEC-RL for review. Comments are being awaited to determine if a proposal should be prepared outlining the next phase of work.

Warm Water Utilization

The proposal to study the use of warm water for irrigation, prepared jointly by Douglas' United Nuclear, Battelle-Northwest, and Washington State University, with the cooperation of AEC-RL, has been forwarded to AEC Headquarters. AEC-HQ is still deliberating the extent to which they can support the program financially.

On April 28, at a meeting of the Pacific Northwest River Basins Commission (PNWRBC) in Portland, Oregon, Mr. Roy Harris, Director of the Washington State Water Pollution Control Commission, described the nature and scope of the Warm Water Irrigation Proposal during a panel discussion on the subject, "Power Looks at Its Environment." Later, summary information with associated cost data was formally transmitted to PNWRBC.

The Commission received the proposal as a committee-of-the-whole and placed it on the agenda for discussion at their next meeting. The composite proposal is being rewritten into an integrated prospectus for use in solicitation of funds. At present, $10,000 has been pledged by the Tri-City Nuclear Council and $3,000 by the Benton County PUD. It appears that the State of Washington will play the leading role in the funding solicitation.

SYSTEMS ANALYSIS

CAGE - Mod 2A

Program definition of the financial portion of CAGE Mod 2A is proceeding satisfactorily. Fuels cost relationships have been defined, and these relationships will be reviewed by appropriate plant management. A draft of the capital item input to CAGE has been written and reviewed with interested parties. It is planned that CAGE capital items will conform closely to those in the five-year capital budget and Long-Range Plan.

A case was prepared describing the first three years of reactor operation. It is intended that this case will provide the basis for "freezing" the first three years of all cases run, at least through December 1968. This action
implements the AOPG decision to consider FY 70 as the first study year. The case has been circulated for comment prior to being entered into the permanent library.

MINIMODEL

The submission of production data for four more operating modes has completed currently planned AECOP MINIMODEL input preparation. A total of 25 sets of data have been supplied.

Case Planner

A new computer program has been designed to prepare most of the planning estimate output directly and automatically from output from the planning part of the case planner program. Input preparation for the program has been completed except for Pu-241 data, beginning FY 70 inventories, and product demand schedules.

Plant Operating Alternates

The AEC requested a series of briefing tables on plant alternates. These tables were prepared and have been submitted to Headquarters. Separate tables were prepared for each reactor type (C, K, and N) covering the following production modes: defense plutonium; coproduct tritium; coproduct Pu-238; coproduct U-233; coproduct Co-60; and coproduct Tm-170. Each reactor mode was further subdivided by power level, reactor modification, and fuel enrichment. Technical status, R&D costs, capital costs, production capability, fuel throughput, and allocated unit cost were reported for each. The information was combined with related ARHCO separations plant data and issued in document DUN-AOP-116.

N Advanced Technology Evaluation.

The final draft of the N Reactor Advanced Technology Evaluation report was circulated for comments. Comments have been resolved and the final report is in preparation.

ADVANCED PLANNING

Long-Range Plan

All revisions have now been made to slides for the upcoming presentation to the AEC Commissioners. Twenty slides have been picked for that presentation, a few of which are for use as backup.

AECOP-Related Studies on Pu Values

The plutonium value study made here was discussed with AECOP, and there is excellent conformity between the work done at the two sites. AECOP also has arrived at a conclusion, which became apparent to us very recently, that the value of plutonium will be higher than the simple indifference value in the early years of the fast breeders when the breeders of a particular utility system are carrying more of the base load.
IRRADICATION SERVICES

ROUTINE IRRADIATIONS

The following routine irradiations were performed:

- Ninety-seven activation analysis samples were irradiated in the C, KE, and KW Reactor Quickie facilities for Battelle-Northwest.

- Two creep rate measurement capsules were charged into General Purpose facilities at KW Reactor, one from the X-level side and one from the rod room side.

- Nine graphite sample boats were removed from a test hole channel at KE Reactor.

- Four cooled tensile specimen samples were charged into a Snout facility at KW Reactor.
EMPLOYEE RELATIONS

The reduction of force in conjunction with the B Reactor deactivation and the reduction of budget dollars is continuing on schedule.

The decision received from the Federal Court in March requiring the Company to submit to arbitration the question of contract coverage for N Reactor employees has been appealed by the Company to the Federal Circuit Court of Appeals. The appealability of this matter is to be reviewed by the Court on May 20, with a decision expected shortly thereafter.

EMPLOYMENT SUMMARY

Douglas United Nuclear employment as of April 30 is summarized in Appendix B. The data reflect the continuing force reduction noted above.

PRODUCTION COMPUTER AND REMOTE TERMINAL

The acquisition proposal for a production computer was approved by AEC-RL. This facility is expected to result in annual savings of about $58,000 beginning next fiscal year. The Univac 9300 computer, to be purchased with FY 1968 capital funds of $145,210, will be delivered in September. It is a 16K memory core, tape-oriented system with data transmission capability. For the transmission and processing of unclassified data, the unit will be remotely connected with Computer Sciences Corporation's Univac 1108 system.

INSURANCE CLAIM FORMS REVISED

To help alleviate the continuing problem of employees submitting incomplete insurance claims, the claim form has been revised with key summary instructions contained in perforated extensions to the form. Connecticut General was very receptive to the Company's request for this change, and is considering the inclusion of this innovation on the forms furnished all companies covered under its group plans.

NUCLEAR MATERIALS ACCOUNTING SYSTEM

The new system for Nuclear Materials Accounting was made operational during the month, and will process nuclear materials data for the period beginning April 1 when the old system was terminated. Operating costs of the new system are expected to be about 25 percent of those under the old system, a saving of $11,000 annually.
APPROVAL LETTERS

At the close of the reporting period, the following approval requests had not been acted upon by AEC-RL:

<table>
<thead>
<tr>
<th>ATD Number</th>
<th>Subject</th>
<th>Date of Transmittal to AEC-RL</th>
</tr>
</thead>
<tbody>
<tr>
<td>ATD-151</td>
<td>Household Goods Moving Expense</td>
<td>April 23, 1968</td>
</tr>
<tr>
<td>Add. #1</td>
<td></td>
<td></td>
</tr>
<tr>
<td>ATD-152</td>
<td>Merit Salary Increases Exempt Employees</td>
<td>February 29, 1968</td>
</tr>
<tr>
<td>Add. #1</td>
<td></td>
<td></td>
</tr>
<tr>
<td>ATD-158</td>
<td>Newly Hired Recruited Exempt Employees; Appendix &quot;B&quot; Modification No. MQ-19 (Household goods limit increased from 8,000 to 11,000 pounds)</td>
<td>April 22, 1968</td>
</tr>
</tbody>
</table>

SAVINGS BOND DRIVE

As a result of the U. S. Savings Bond and Freedom Shares drive which is nearing completion, it is estimated that DUN employee participation will increase 100 percent.

LETTER OF CREDIT FINANCING

A Letter of Credit effective May 15 has been issued by AEC-RL in favor of the Company for $12 million. The effective date for letter of credit financing is still June 1; however, $12 million is being provided early so that two or three payment vouchers may be processed to determine the time required to credit DUN's account, and to discover any problems which may exist internally.
SAFETY

Month end safety statistics were:

Disabling injuries: April 0 0
CY to date 0 0

Days since last disabling injury 309
Man-hours since last disabling injury 3,300,000

No radiation exposures exceeded operational control.
## APPENDIX A

### PROJECT STATUS SUMMARY - REACTOR FACILITIES

<table>
<thead>
<tr>
<th>Number &amp; Title</th>
<th>Authorized</th>
<th>Percent Complete</th>
<th>Status</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>C &amp; K Reactors</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>DCE-505, Boiler Control Improvements - 165-KE and KW</td>
<td>$281,000</td>
<td>97</td>
<td>22</td>
</tr>
<tr>
<td>DAP-510, Discharge Chute Clearing Equipment - K Reactors</td>
<td>$190,000</td>
<td>100</td>
<td>2</td>
</tr>
<tr>
<td>DAE-512, Replacement of Turbine with Diesel Drive - 181-B Pump</td>
<td>$87,000</td>
<td>100</td>
<td>50</td>
</tr>
<tr>
<td>DAP-513, Deactivation of Hanford Production Reactor (formerly 509)</td>
<td>$80,000</td>
<td>100</td>
<td>38</td>
</tr>
</tbody>
</table>

Project Proposal Revision No. 1 was approved by AEC-RL March 28 authorizing $281,000 total expenditure and extending completion date to December 1. All construction drawings were completed and approved during the report period. Installation work continued approximately on schedule at the No. 3 KW boiler.

J. A. Jones Company is fabricating conveyors in the shop. Lost two weeks due to a construction strike.

A strike action from April 1-15 stopped work in that period. Since then progress continued in installation of the diesel engine drive.

Work stopped April 1-15 due to strike action. Work has since returned to normal.
### PROJECT STATUS SUMMARY - REACTOR FACILITIES (contd)

<table>
<thead>
<tr>
<th>Number &amp; Title</th>
<th>Authorized</th>
<th>Percent Complete</th>
<th>Status</th>
</tr>
</thead>
<tbody>
<tr>
<td>C &amp; K Reactors (contd)</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>DAP-515, Replacement Steam Generating Facility - 100-D Area</td>
<td>$140,000</td>
<td>15 0</td>
<td>Design started. Purchase specifications prepared and bids are being requested for a building, pumps, tank and deaerator.</td>
</tr>
<tr>
<td>N Reactor</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>GAP-401, Upgrading Fire Protection - 100-N</td>
<td>$150,000</td>
<td>100 70</td>
<td>Fire Detection Services contract completion date of April 17 has been exceeded without any system being completed. Contractor still has design approval to be obtained for 105, 184 and 109 Bldgs.</td>
</tr>
<tr>
<td>GCE-405, N Reactor Temperature Monitoring System Improvements</td>
<td>$189,000</td>
<td>100 57</td>
<td>No change in status.</td>
</tr>
<tr>
<td>GCP-406, Safety Platforms and Accesses</td>
<td>$300,000</td>
<td>100 50</td>
<td>No scheduled work completed during April outage as result of the Laborers' strike.</td>
</tr>
<tr>
<td>GCE-408, W, C, D Elevator Safety</td>
<td>$90,000</td>
<td>100 0</td>
<td>Design tests in progress for selecting acceptable level detection device.</td>
</tr>
</tbody>
</table>
## PROJECT STATUS SUMMARY - REACTOR FACILITIES (contd)

<table>
<thead>
<tr>
<th>Number &amp; Title</th>
<th>Authorized</th>
<th>Percent Complete</th>
<th>Status</th>
</tr>
</thead>
<tbody>
<tr>
<td>N Reactor (contd)</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>GCP-409, Fuel Handling Improvements</td>
<td>$310,000</td>
<td>80</td>
<td>Approval for Revision 2 to Project Proposal not yet received from AEC-RL. Work is concentrated on high speed cart portion of project; that work is now 50% complete.</td>
</tr>
<tr>
<td>GAP-410, Decontamination Waste Loading Facilities</td>
<td>$65,500</td>
<td>100</td>
<td>Acceptance Test Procedure has been approved and design by Vitro/HES is complete. The J. A. Jones' subcontractor, HUICO, moved in on pipe excavation work April 15.</td>
</tr>
<tr>
<td>GCP-411, Effluent Control Program - 100-N</td>
<td>$590,734</td>
<td>79</td>
<td>Revised design schedule approved by AEC-RL March 29. $1,079,266 of authorized funds held in reserve by AEC.</td>
</tr>
<tr>
<td>DCP-514, Air Conditioning, Water Quality Lab., 109-N</td>
<td>$37,500</td>
<td>100</td>
<td>The air conditioner, two heating units and main roof duct have been installed. The Sheetmetal Workers' labor contract expires May 1; these negotiations could affect performance on this project.</td>
</tr>
<tr>
<td>DCP-517, Additional Storage Basin Cubicles, 105-N</td>
<td>$37,000</td>
<td>70</td>
<td>Boron frit order placed April 23 with American Porcelain Company. Bids for concrete cubicle components will be opened April 29. Delivery on or before July 1 has been requested.</td>
</tr>
</tbody>
</table>
### APPENDIX B

#### EMPLOYMENT SUMMARY

(as of 4-30-68)

<table>
<thead>
<tr>
<th>Division</th>
<th>Exempt</th>
<th>Non-Exempt</th>
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INTRODUCTION

Enlargement of the process channels in the production reactors to permit the use of larger diameter tubes and fuel elements was proposed in 1959. Advantages in both plutonium production and reactor safety were recognized, since thus reducing the moderator-to-fuel ratio causes an increase in conversion ratio and a decrease in reactivity upon loss of coolant.

The feasibility of process channel overboring was demonstrated in 1960 and 1961 when the diameters of 62 channels in C Reactor were enlarged by about one-half inch. Forty-four of these channels were in a block near the center of the reactor. The enlarged channels were fitted with new process tubes and fuel elements of larger diameter, and a substantial program of fuel and physics testing was conducted with the new configuration. This entire program was technically very successful. However, because of the decline in demand for weapons plutonium, action on large-scale application of process channel enlargement was tabled.

Subsequently it was recognized that enlargement of the central tubes of the B, C, and K reactors would also be important to their product flexibility. Making products other than Pu-239 involves the irradiation of targets other than U-238, and overboring enhances target space by displacing graphite rather than fuel. For this purpose, overboring by one inch instead of one-half inch was seen to permit greater flexibility in target selection. Development efforts therefore were undertaken to provide a Reactor Modernization Test Facility in C Reactor by further enlarging (to one-inch overbore) the 44 channels in the central block noted above. This work was successfully completed early in 1968, and the 44-tube Test Facility is now equipped with process tubes and fuel elements about one inch larger than standard.

The purpose of this report is two-fold: (1) to describe the Test Facility in C Reactor, including the channel overboring equipment, the enlarged process tubes, the hardware, and the fuel elements; and (2) to summarize the incentives and development activities associated with the enlargement of all central zone channels in the B, C, and K reactors.

THE REACTOR MODERNIZATION TEST FACILITY

Description and Objectives

This Test Facility has been installed in C Reactor to further refine the technology required to increase the performance and product flexibility of the B, C, and K reactors. The 44-tube block is located in the central zone, as shown on the appended Figure 1. Thirty-seven thin-walled Zircaloy tubes were installed, three having guide rails to permit control spline insertion.
Six channels are equipped with ribbed aluminum process tubes which also permit the insertion of control splines. One channel, without a process tube, is equipped with a graphite thermocouple stringer to enable the recording of graphite temperatures within the test block.

The specific objectives of installing and operating this Test Facility are as follows:

- Demonstrate the feasibility of one-inch enlargement of a representative number of process channels under the conditions which prevail in a production reactor.

- Demonstrate the installation feasibility and operational serviceability of the larger process tubes and hardware in a block arrangement of significant size.

- Confirm fuel designs and establish fuel performance characteristics.

- Evaluate various reactor operating modes.

Channel Enlargement Equipment

The development of process channel enlargement equipment and tooling originally was initiated to determine the feasibility of overboring the process channels of a B or C type of production reactor by one-half inch. That effort resulted in a number of specialized machines and cutting tools that were extensively tested on mock-ups simulating reactor shielding and graphite moderator stacking. The equipment consisted principally of the following:

- Centering Flange Cutting Machine - for boring through the centering flange on the reactor face and providing access to the shield opening.

- Tri-Bore Machine - for reaching into the shield opening to bore-out the cast-iron shield ring and thermal shield blocks, and to counter-bore the first graphite block of the moderator.

- Graphite Boring Machine - for overboring the process channels through the graphite stack.

- Centering Flange Welding Head - for automatic welding of a new hardware component to the centering flange.

Development effort in preparing for installation of the Test Facility in C Reactor resulted in many improvements in the channel enlargement equipment to increase speed, ease of operation, reliability, and versatility. The tri-bore machine (see Figure 2, appended) was made to handle also the cutting of the gas sleeve and the centering flange. Laboratory mock-ups were used to duplicate predicted moderator distortion for trial operation of the graphite boring equipment, and the trial installation and accurate dimensional probing of process tubes. The major steps in the development of the graphite boring bar finally used are illustrated in Figure 3, appended. Close-up views of
the drilling rig and 5-foot boring bar are shown in appended Figure 4.

All of this specialized equipment performed very well during installation of the Test Facility. Moreover, cycle times for the various operations were shorter than during previous channel overbore programs, thus reducing both reactor outage time and personnel exposure. New methods of channel examination with closed-circuit television viewing and tape recording were very helpful in evaluating the condition of the graphite channels. After enlargement, the general condition of the graphite was very good, with little damage that could be attributed to the overboring operation.

Process Tubes and Hardware

The thin-wall Zircaloy process tubes used in retubing the K reactors were formed by extrusion, as were also those used in the one-half inch overbore channels in C. However, the higher ratio of tube diameter to wall thickness required with the one-inch overbore for the Test Facility prohibited the extrusion of these larger thin-wall tubes. Accordingly, these Zircaloy tubes were fabricated by seam welding, in 24-foot lengths, and two sections then were fusion-welded together to obtain the required total length. This butt-welding was done by Douglas United Nuclear on-site, after attempts to contract the operation were unsuccessful.

The seam-welded method of Zircaloy tube formation also enabled the fabrication of special tubes with guide rail strips prewelded to the tube sheet blanks prior to forming. The guide rails were spaced to permit the charging of self-support fuel elements in an aligned column that would permit the insertion of control splines. As has been noted, three of these tubes were included in the 44-tube block.

The six ribbed aluminum tubes installed in the Test Facility were also procured on a best-effort basis from the only tube fabricator that responded to the request for quotations. After many attempts, satisfactory full-length aluminum tubes of the required size and shape were made; however, it was necessary to reduce the tensile strength specification slightly.

Hardware components to accommodate the larger process tubes were designed around the utilization of impact extruded aluminum nozzles. The latter were developed for increased integrity and serviceability over the cast aluminum nozzles used for many years on the Hanford production reactors. Compact flanged joints enabled the larger tubes to be accommodated within the confines of the lattice space and still retain room for maintenance of the balance of the assembly components. A degree of standardization was maintained with the existing smaller tube assemblies by using similar accessory components such as the flexible inlet connector, cap screws, and spline cap seals. Figure 5, appended, shows the partially completed Test Facility as viewed from the rear work platform.

The process tubes and the gunbarrels which serve as liners for the shield openings were easily installed in the Test Facility. This attested to the success of the overboring operations, and of the automatic fusion welding of the gas tube which forms the entry into the reactor shielding.
Fuel Elements

Hot-die-sized rod-in-tube fuel elements were developed and fabricated for use in the Test Facility. One of these new and larger fuel elements is shown with a standard I&E element in the appended Figure 6. The inherent flexibility of the hot-die-sizing process supported its selection for producing the rod-in-tube fuel.

Preliminary evaluation of the new fuel design was obtained by irradiating four columns in KE Reactor test channels which are equipped with N Reactor process tubes. The first column of test material was irradiated satisfactorily to 650 MWD/T. Subsequent charge have included exposure to 1140 MWD/T and average tube powers to 2820 kw. The final test column in KE Reactor involved irradiation of simulated fuel and target assemblies in which the outer tube was the fuel and the inner rod was the target. The performance of these assemblies was satisfactory.

The decision to use 80 Metal fuel for initial irradiations in the Test Facility enabled design and fabrication of the required elements to proceed. It was necessary to develop two fuel models, differing only slightly in dimensions, one for use in the Zircaloy process tubes and the other for use in the aluminum tubes. The fabrication of these fuels was completed, with two complete loads available for charging at the time the Test Facility was installed.

ADDITIONAL DEVELOPMENT ACTIVITIES

K Reactor Demonstration

Available data, in conjunction with that obtained in the Test Facility at C Reactor, are applicable also to the K reactors. However, the latter units differ from B and C in two respects: (1) the graphite in the Ks has suffered more damage because of higher integrated exposure, and (2) the closer lattice spacing of the Ks would require inlet crossheader modification to accommodate the enlarged tube hardware.

Channel enlargement equipment for the K reactors is being developed and will be demonstrated on a laboratory mock-up. It is then proposed to overbore a few K channels to the same enlarged diameter as at C to establish that the graphite in the K reactors is suitable for overboring, and to demonstrate the feasibility of the crossheader modification.

Physics and Thermal Hydraulic Tests

Laboratory work in support of the channel enlargement program, in addition to equipment development as described above, includes physics and thermal hydraulic studies. Tests with the one-inch overbore fuel and tube system are in progress in the Physical Constants Test Reactor (PCTR) to measure the basic lattice reactivity, and the changes in reactivity resulting from either coolant loss or moderator flooding. The thermal hydraulics behavior of this system also is being evaluated, using a full-scale process channel mock-up in a heat transfer laboratory. These tests will provide the data needed for
production and safety evaluations of the reactors with channels enlarged one inch.

**INCENTIVES FOR LARGE-SCALE PROGRAM**

Recent studies have shown that the growth of breeder reactors may be restricted late in the century by the availability of plutonium from thermal reactors as startup inventory for the new breeders. This limitation on nuclear power development could be partially alleviated by preproduction of non-defense plutonium in the production reactors. To do this economically, plutonium unit costs need to be reduced.

Overboring the 1500 central channels of B and C Reactors by one inch, and a lesser overbore of the 2500 central channels of the K reactors to the same enlarged diameter, is considered essential as the principal means for lowering unit costs to meet this possible new market.

For B and C Reactors, the overboring logically would be coupled with water plant modifications to gain increased cooling capacity. Production losses would be recovered within one year after the overbore outage, and lowered unit costs would permit recovery of the cost of overboring and water plant modifications within a little over two years. Corresponding times for the K reactors (without water plant changes) are three and four years, respectively, as reported in the reactor overbore study document (DUN-AOP-70).

The thin-wall Zircaloy process tubes for one-inch enlarged channels have an inside diameter (2.65 inches) which is almost the same as the 2.7-inch I.D. of the heavy-wall process tubes in N Reactor. Moreover, the rod-in-tube fuel elements being used are very similar in design to the N Reactor fuel. Overboring of B, C, and K reactors to accommodate the larger process tubes therefore could permit a degree of fuel standardization between N Reactor and the others which would have the effect of lowering Hanford fuel fabrication costs.

It has also been observed that the harder neutron spectrum prevailing in overbored reactors causes a substantial increase in production of scarce Np-237, which is in demand as a source material for Pu-238 production. This results both from an increased rate of n, 2n reactions with fast neutrons in U-238, and from an increased rate of capture of resonance neutrons in U-236.

In summary, the incentives for enlarging the central zone process channels of the B, C, and K reactors (including water plant modifications at B and C) are:

- Higher plutonium production rates, ranging from 20 percent increase at the K reactors to 60 percent increase at B Reactor.
- Reduction in unit cost of plutonium
- Increased safety of operation, since loss of cooling water would cause a loss of reactivity.
- Increased production versatility, and eased operation with oralloy drivers.

- A degree of fuel standardization between N Reactor and the other reactors.

- Higher neptunium production rates, the increase being on the order of 50 percent.
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Figure 1. Reactor Modernization Test Facility
(The 44-tube, one-inch overbore block in C Reactor)
Figure 2. Tri-Bore Machine and Tools Used to Enlarge Openings through Shields
Figure 3. Steps in Development of Graphite Boring Bar
Figure 4. Graphite Bridging Pipe and Porting Bar

 Deglassified
Figure 5. Reactor Modernization Test Facility - C Reactor (as viewed from rear platform during component assembly)