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DOUGLAS UNITED NUCLEAR
MONTHLY REPORT

APRIL 1969

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AEC-RNL, RICHLAND, WASH
DOUGLAS UNITED NUCLEAR
MONTHLY REPORT
APRIL 1969

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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<td>H-1</td>
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SUMMARY

REACTOR PLANT OPERATIONS

Production Statistics

<table>
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<tr>
<th></th>
<th>C</th>
<th>KE</th>
<th>KW</th>
<th>N</th>
</tr>
</thead>
<tbody>
<tr>
<td>Input Production - Pu (KMWD)</td>
<td>41.8</td>
<td>26.3</td>
<td>118.6</td>
<td>102.6</td>
</tr>
<tr>
<td>- U-233 (Equiv. KMWD)</td>
<td>1.0</td>
<td>4.1</td>
<td>14.7</td>
<td>-</td>
</tr>
<tr>
<td>Time Operated Efficiency - %</td>
<td>80.8*</td>
<td>26.0</td>
<td>100.0</td>
<td>88.8</td>
</tr>
<tr>
<td>Steam Availability to WPPSS - %</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>87.9</td>
</tr>
</tbody>
</table>

*To time of deactivation at 4:10 p.m. on April 25.

C & K Reactors

Per AEC instruction, the C Reactor was shut down near month end for placement in standby status. This facility had been in service since startup on November 18, 1952.

Maximum power levels at KE (3910 MW) and KW (3980 MW) continued to be limited by inlet water piping brittle fracture considerations. The low TOE at KE reflects a scheduled outage for charge-discharge, maintenance, and project DAP-510 work on discharge chute clearing equipment.

At both K reactors the program to produce fuel grade plutonium is continuing and thoria core loads are being irradiated to make U-233. The latter program will require one additional E-Q core loading.

N Reactor

The input production of 102,558 MWD was a record N Reactor high for a single month. The by-product steam enabled the Washington Public Power Supply System to generate 479,080,000 KWH of electrical power, a new one-month record. Maximum reactor power level was 4000 MW, the administrative limit.

Two reactor outages occurred, both unscheduled. The first resulted from a trip due to a flow monitor sensing line failure; the second was necessitated by a fuel element failure.

Steam generator retubing continued on schedule with a July completion date for the work in Cell 2.
**FUEL AND TARGET FABRICATION**

**Production Statistics (tons)**

<table>
<thead>
<tr>
<th></th>
<th>For C &amp; Ks</th>
<th>For N</th>
</tr>
</thead>
<tbody>
<tr>
<td>Billets Extruded</td>
<td></td>
<td>61.0</td>
</tr>
<tr>
<td>Finished Fuel Produced</td>
<td>215.4</td>
<td>30.7</td>
</tr>
<tr>
<td>Thoria Canned</td>
<td>17.4</td>
<td></td>
</tr>
</tbody>
</table>

**C & K Fuels**

AlSi canning operations continued on the basis of three lines per day until April 28, at which time these operations were reduced to two lines per day to release personnel for staffing thoria wafer production on a full 3-shift basis. Bumpers or self-supports were attached to 90 percent of the fuel elements produced.

**N Fuels**

Output production of N fuels slightly exceeded forecast.

**TECHNICAL ACTIVITIES**

**C & K Reactors**

The testing program to determine the brittle fracture properties of K reactor inlet piping is about 50 percent complete. Fifteen of 32 fracture tests of wedge opening loading (WOL) samples have been completed by the Lawrence Radiation Laboratory. These 15 samples include material from the KE-D and KE-A risers, and show no more than a 10 to 15 percent scatter in toughness values. Crack growth rates completed by LRL on two WOL samples indicate higher rates than observed on other mild carbon steels; this may be partly due to the geometry of the sample rather than a basic difference in material behavior.

Loading changes made in the 10-kilogram plutonium irradiation test block during the April outage at KE Reactor are expected to increase support zone powers to normal levels, and to increase the powers of the PuAl columns.

Fabrication and characterization of the americium targets were completed. These targets will be irradiated for the production of medical grade Pu-238. The neptunium-aluminum matrix elements have been fabricated for irradiation in a column-cluster configuration. Fabrication of the neptunium-graphite matrix elements is scheduled for early completion; these elements will be used to evaluate techniques for reducing the Pu-236 contaminant in Pu-238.

Design criteria in support of the Project Proposal covering capital funded work associated with the deactivation of C Reactor have been prepared for submittal to AEC-RL. The capital work consists basically of provisions for (1) electric space heating where needed, (2) an evacuation warning alarm system, (3) permanent markers for closed burial grounds, (4) fire protection.
systems, and (5) two transformer stations. Also, detail design of piping modifications in 182-B has been initiated to permit adding one of the three last-ditch diesel-driven reactor coolant pumps to the export system. This unit will provide emergency coolant to the 200 Areas after the steam-driven pumps presently used become inoperative due to 184-B steam plant deactivation.

N Reactor

Examination of the Inconel tube removed from steam generator 4A after 5 years of service showed no significant increase in the minor corrosion observed two years ago on a tube examined after 3 years of service.

Test results performed on a Zircaloy process tube removed in November 1967 showed no deterioration in properties following irradiation to $0.9 \times 10^{21}$ n/cm$^2$.

Analysis of the I-131 concentration in N Reactor effluents discharged to the river, and an estimate of the dilution factor in the river, indicate that the resultant I-131 concentration after dilution is no greater than 0.001 of the 10 CFR 20 guidelines. It therefore has been recommended to AEC-RL that Part 7 of the Effluent Control Project, which covered construction of a rupture waste processing facility, not be built.

IRRADIATION SERVICES

In mid-April an in-reactor test section and other equipment were installed in C Reactor for immediate use in the first in-reactor experiment of the corrosion product transport studies being conducted by BNW for the AEC Naval Reactors Division. Several days of successful experimentation were achieved before C Reactor was deactivated. This facility will now be removed from C and installed in KE.

FEATURE REPORT

One of the problems caused by subsidence of the reactor graphite stacks is associated with the stainless bellows which conduct 3X system balls through the gas gap between the top biological and thermal shields. The appended summary report defines this problem, and describes the development and application of techniques for bellows severance and shrouding which have been used successfully at KE and KW.

GENERAL

In accordance with AEC-RL instructions for the FY 1971 Budget and Revision of FY 1970 Budget, preparation (prior to the reactor cutback announcement on April 11) had proceeded on the assumption that C Reactor would be deactivated and the two K reactors would be operated alternately. Now that only C has been affected, the operating budget is having to be completely revised.
The Company's contribution to the AEC-RL Capital Plant and Equipment Budget for FY 1971 and Revision of FY 1970 was delivered on schedule.

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

Charles D. Harrington
President
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:

The C Reactor was deactivated at 4:10 p.m. on April 25. The shutdown occurred about two days ahead of schedule because of an overbore fuel element failure.

The production of nondefense plutonium in the 8-10 percent Pu-240 range is continuing with enriched uranium at the K reactors.

Based on yield and quality data obtained from the test monitoring columns, one additional K reactor core E-Q load will be required. This is in addition to that shown in the April study forecast (DUN-5550).

OPERATING EXPERIENCE

Reactor Loadings

Front face maps showing the loadings at 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. Those shown for C Reactor are at the time of deactivation on April 25; fuel discharge from C was in progress at month end.
### Production Reactor Statistics

- **Time Operated Efficiency - %**
- **Average**
- **Power Level (MW) - Maximum**
- **Input Production - Pu - KwD**
- **Output Time Allocation - %**

<table>
<thead>
<tr>
<th>Reactor</th>
<th>KE</th>
<th>KW</th>
</tr>
</thead>
<tbody>
<tr>
<td>TOTAL</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Time</th>
<th>Output</th>
<th>Input</th>
<th>KE</th>
<th>KW</th>
</tr>
</thead>
<tbody>
<tr>
<td>9</td>
<td>7</td>
<td>100.0</td>
<td>86.8</td>
<td></td>
</tr>
<tr>
<td>9.3</td>
<td>36.9</td>
<td>3.960</td>
<td>3.0</td>
<td>2.31</td>
</tr>
<tr>
<td>10</td>
<td>200.0</td>
<td>0.300</td>
<td>3.0</td>
<td>2.31</td>
</tr>
<tr>
<td>19</td>
<td>1790</td>
<td>14.0</td>
<td>17.8</td>
<td>36.3</td>
</tr>
<tr>
<td>19.7</td>
<td>118.6</td>
<td>17.8</td>
<td>36.3</td>
<td>118.6</td>
</tr>
</tbody>
</table>

- **Diaphragm - Ppm**
- **PH**
- **Normal Operating Flow - Gpm**
- **Water To Reactor**
- **Helium Consumed - M Cu. Ft.**
- **Fuel Element Failures**
- **Fuel Change - (Tons) - Natural Uranium**
- **NEW TUBES INSTALLED**
- **VAN STONE**
- **NEW TUBES INSTALLED**
- **WATER LEAKS - TUBE**
- **Number of Outages**
- **OUTAGE TIME ALLOCATION - %**
Power Levels

Maximum power levels shown for the K reactors represent continuing operation under precautions taken to insure adequate coolant flow in the event of a brittle fracture failure of inlet piping. The KE power level was somewhat further limited by material charged under PTA-150, "PuAl Irradiations."

At C Reactor power level was restricted by the bulk outlet water temperature limit of 95°C.

Reactor Outages

Four reactor outages were experienced as summarized below. In addition, the KE Reactor was down 497.4 hours on an outage initiated late in March for scheduled maintenance and the charge-discharge of fuel.

<table>
<thead>
<tr>
<th>Date Down</th>
<th>Reactor</th>
<th>Outage Hours</th>
<th>Remarks</th>
</tr>
</thead>
<tbody>
<tr>
<td>April 3</td>
<td>C</td>
<td>38.9</td>
<td>Oil pump failure on the No. 4 process water pump at 190-C. Miscellaneous maintenance work was completed.</td>
</tr>
<tr>
<td>April 5</td>
<td>C</td>
<td>31.8</td>
<td>Correction of a rear Van Stone flange leak on tube 1369.</td>
</tr>
<tr>
<td>April 15</td>
<td>C</td>
<td>42.9</td>
<td>Installation of a corrosion product test facility in the C-1 loop. Miscellaneous maintenance and Production Test work was completed.</td>
</tr>
<tr>
<td>April 22</td>
<td>KE</td>
<td>33.5</td>
<td>1706-KE pump failure caused by a faulty bearing.</td>
</tr>
</tbody>
</table>

The C shutdown on April 25 has already been noted. The one-inch overbore element which failed was at 158 percent of goal exposure.

EQUIPMENT EXPERIENCE

Discharge Chute Clearing Equipment – KE Reactor

The installation of discharge chute clearing equipment was completed at KE Reactor, but problems encountered with fuel elements dropping under the conveyor belt instead of into the storage bucket prevented any beneficial use of this equipment. Corrective modifications are planned for May.

Horizontal & Vertical Rods – KE Reactor

At KE Reactor a new flat rod was installed in No. 1 HCR channel, No. 11 HCR (half-rod) was replaced, and Nos. 16 and 20 HCRs were replaced with regular round rods. Two VSRs were replaced with flexible rods.
Ball 3X Bellows Shrouding - KE Reactor

Shrouds were installed on the outlet bellows of seven ball 3X hoppers. This completes the known requirements for KE through August, 1970. (Bellows shrouding is the subject of the Feature Report in Section H.)

Rear Hardware Leak Repair - KE Reactor

A total of 467 rear face pigtails, Y block, nozzle and RTD leaks was repaired. The leaking rear crossheader expansion joints were replaced, six on the far side and two on the near side.

Front Hardware - KE Reactor

The front pigtails, venturis and crossheader weldolets were gauged and inspected on 1,590 tubes when the venturis were removed for installation of wide-mesh venturi screens per Design Change 1185. Ten crossheaders were isolated and visually inspected to check for suspected missing venturi screens. One screen was found in each of two crossheaders and two screens were found in one other crossheader. All other crossheaders were clean.

Steam Boilers - KW

A static pressure test on the superheater tubes of No. 1 boiler at 165-KW showed leakage around at least 20 of the 26 tubes where they are rolled into the outlet tube sheet. Bad pitting was discovered in the rolled area of one of the tubes, and this tube was plugged off. The other tubes were rerolled.

C-1 Loop Alterations - C Reactor

Several weeks of preliminary work were required prior to the April 15 outage for installation of the in-pile module and related equipment. All new installations were pressure tested and installed in compliance with code. The work proceeded on schedule and the results were satisfactory.

Deactivation - C Reactor

Deactivation work has started in the 182-B, 183-B, 183-C, and 184-B buildings.

PROCESS ASSISTANCE & CONTROL

Operational Physics

Ten loading changes (thoria to 94 Metal) were made to the 225-tube PuAl test block in KE Reactor such that the average PuAl column power is slightly over 900 kw; this is less than ten percent below the desired powers. The loading changes also served to increase the flattening 5 percent, and the total power generation in KE is now nearly equal to that prior to the test.

The C Reactor deactivation on April 25 followed an operating period which included four unscheduled outages. Optimum flattening had been achieved for the month and no operational physics problems were encountered.
Some operational physics parameters of interest are shown in the following table:

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Reactor</th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Effective Central Tubes (ECT)*</td>
<td>C</td>
<td>KE</td>
<td>KW</td>
</tr>
<tr>
<td></td>
<td>1615</td>
<td>2216</td>
<td>2237</td>
</tr>
<tr>
<td>Flattening Efficiency** - April</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>- 12-mo. average</td>
<td>0.86</td>
<td>0.71</td>
<td>0.71</td>
</tr>
<tr>
<td>Maximum Operating Time Permitting Scram Recovery - Hours**</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>10</td>
<td>10</td>
<td>11</td>
</tr>
</tbody>
</table>

*Reactor power level divided by the average power of the 10 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 production fuel element failures during the report period.

The following table shows production fuel failure frequencies, as number per million elements discharged, for the 12- and 24-month periods ending March 20:

<table>
<thead>
<tr>
<th>Reactor</th>
<th>12 Months</th>
<th>24 Months</th>
</tr>
</thead>
<tbody>
<tr>
<td>C Reactor - Natural Uranium</td>
<td>18</td>
<td>32</td>
</tr>
<tr>
<td>- 94 Metal</td>
<td>33</td>
<td>28</td>
</tr>
<tr>
<td>K Reactors - Natural Uranium</td>
<td>5</td>
<td>7</td>
</tr>
<tr>
<td>- 94 Metal</td>
<td>6</td>
<td>7</td>
</tr>
</tbody>
</table>

Product Accountability

Document DUN-5741, presenting equation fits for U-236 buildup utilizing the LEARN code, has been completed for issuance. Both "gratuitous" and "incremental" neptunium buildup from U-236 are considered.

U-233 Yield and Quality Monitoring

Five thoria elements from the 2586-KW monitor column discharged in October 1968, and two central thoria elements from a recently discharged fringe column, 0167-KE, were sent to BNW in early April for radiochemical analysis. The elements from 2586-KW were selected to have redundant geometry, and thus exposure,
to those from the 2886-KW and the 2786-KW 94 Metal analyzed earlier. The 2586-KW elements are now almost free of the protactinium content which complicated the 2886-KW results. The 0167-KE data will provide empirical bases for existing theoretical calculations of neutron and gamma spectra effects in fringe blanket columns. Results of these analyses have been received and are under study.
<table>
<thead>
<tr>
<th>Zone</th>
<th>Tons</th>
<th>Material</th>
</tr>
</thead>
<tbody>
<tr>
<td>Central</td>
<td>98</td>
<td>Natural Uranium</td>
</tr>
<tr>
<td></td>
<td>41 (Includes 14 in overbore block)</td>
<td>94 Metal</td>
</tr>
<tr>
<td>Ring</td>
<td>12</td>
<td>Natural Uranium</td>
</tr>
<tr>
<td></td>
<td>20</td>
<td>94 Metal</td>
</tr>
<tr>
<td>Fringe</td>
<td>25</td>
<td>Natural Uranium</td>
</tr>
<tr>
<td>Thoria Support</td>
<td>19</td>
<td>94 Metal</td>
</tr>
<tr>
<td>Thoria</td>
<td>4</td>
<td>Thoria</td>
</tr>
</tbody>
</table>

Loading Pattern - C Reactor

B-A

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<table>
<thead>
<tr>
<th>Zone</th>
<th>Tons</th>
<th>Material</th>
</tr>
</thead>
<tbody>
<tr>
<td>Central</td>
<td>237</td>
<td>94 Metal (for Thoria Support)</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>Natural Uranium</td>
</tr>
<tr>
<td></td>
<td>19</td>
<td>Thoria (&quot;X&quot; Designates Tubes)</td>
</tr>
<tr>
<td></td>
<td>1</td>
<td>Special Depleted Uranium (PITA-048)</td>
</tr>
<tr>
<td>Buckled</td>
<td>65</td>
<td>94 Metal (for Thoria Support)</td>
</tr>
<tr>
<td></td>
<td>6</td>
<td>Thoria (&quot;X&quot; Designates Tubes)</td>
</tr>
<tr>
<td>Blanket</td>
<td>55</td>
<td>94 Metal (for Thoria Support)</td>
</tr>
<tr>
<td></td>
<td>11</td>
<td>Thoria</td>
</tr>
</tbody>
</table>

Loading Pattern - KE Reactor

B-B
### Material Loading Table

<table>
<thead>
<tr>
<th>Zone</th>
<th>Tons</th>
<th>Material</th>
</tr>
</thead>
<tbody>
<tr>
<td>Central</td>
<td>237</td>
<td>9\frac{1}{4} Metal (for Thoria Support)</td>
</tr>
<tr>
<td></td>
<td>1</td>
<td>Natural Uranium</td>
</tr>
<tr>
<td></td>
<td>18</td>
<td>Thoria (&quot;X&quot; Designates Tubes)</td>
</tr>
<tr>
<td></td>
<td>1</td>
<td>Special Depleted Uranium (PITA-048)</td>
</tr>
<tr>
<td>Buckled</td>
<td>66</td>
<td>9\frac{1}{4} Metal (for Thoria Support)</td>
</tr>
<tr>
<td></td>
<td>6</td>
<td>Thoria (&quot;X&quot; Designates Tubes)</td>
</tr>
<tr>
<td>Blanket</td>
<td>53</td>
<td>9\frac{1}{4} Metal (for Thoria Support)</td>
</tr>
<tr>
<td></td>
<td>11</td>
<td>Thoria</td>
</tr>
</tbody>
</table>

**Loading Pattern - KW Reactor**

B-C
PRODUCTION

General

Reactor production (fuel grade Pu), 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>Input Production (Pu)</th>
<th>KMW D</th>
<th>102.56</th>
</tr>
</thead>
<tbody>
<tr>
<td>Electrical Generation (KWH)</td>
<td>Total</td>
<td>479.08 (per WPPSS)</td>
</tr>
<tr>
<td>Power Level (MW)</td>
<td>Maximum</td>
<td>4,000</td>
</tr>
<tr>
<td></td>
<td>Average</td>
<td>3,856</td>
</tr>
<tr>
<td>Time Operated Efficiency</td>
<td>%</td>
<td>88.8</td>
</tr>
<tr>
<td>Steam Availability</td>
<td>%</td>
<td>87.9</td>
</tr>
<tr>
<td>Number of Shutdowns</td>
<td>Scheduled</td>
<td>0</td>
</tr>
<tr>
<td></td>
<td>Unscheduled</td>
<td>2</td>
</tr>
<tr>
<td>Fuel Failures</td>
<td>1</td>
<td></td>
</tr>
<tr>
<td>Fuel Charge (Tons)</td>
<td>94 Metal</td>
<td>301.8</td>
</tr>
<tr>
<td></td>
<td>125 Metal</td>
<td>67.3</td>
</tr>
<tr>
<td></td>
<td>Natural U</td>
<td>0.4</td>
</tr>
<tr>
<td>Total</td>
<td>369.5</td>
<td></td>
</tr>
</tbody>
</table>
Helium Losses - M cu. ft. 516.1
Fuel Oil Usage - bbl. 17,600

OPERATING EXPERIENCE

Reactor Loading

The reactor loading at month end is shown on the front face map which follows page BN-4. The only charge-discharge during the month was process tube 2146 in which a fuel failure occurred.

Power Level

Power level was administratively limited at 4000 MW throughout the month.

The main steam header pressure (MSHP) was limited to some degree because of an imbalance in the reactor reactivity which resulted in higher than normal tube outlet temperatures on the right side of the unit. Even with this imbalance, it was possible to maintain the MSHP at between 117-118 psig for a major portion of the month, thus maximizing the electrical generation capability of WPPSS.

WPPSS was able to generate 479,080,000 KWH of electrical power during the month, thus establishing a new monthly high in electrical output.

Reactor Outages

The reactor was down 21.7 hours at the beginning of the month for an outage initiated on March 31. The two outages in April can be summarized as follows:

<table>
<thead>
<tr>
<th>Date</th>
<th>Outage Hours</th>
<th>Remarks</th>
</tr>
</thead>
<tbody>
<tr>
<td>April 10</td>
<td>33.7</td>
<td>Unscheduled flow monitor scram due to a sensing line failure. Corrected sensing line leak and completed miscellaneous maintenance work.</td>
</tr>
<tr>
<td>April 26</td>
<td>25.3</td>
<td>Fuel element failure in process tube 2146. Minimum outage for charge-discharging this tube.</td>
</tr>
</tbody>
</table>

Fuel Failure

Examination of the Mark IV fuel from process tube 2146 revealed a blister-type failure on the ninth assembly from the rear of the charge. The blister was on the inner surface of the outer element, adjacent to the locking clip. The fuel exposure was about 25 percent of goal.
EQUIPMENT EXPERIENCE

Primary Loop

The April 10 safety circuit trip was caused by a leak in the high pressure impulse line from the venturi to the Barton transducer for process tube 3348. Initial examination indicates the leak resulted from stress corrosion.

Steam generator tube installation in Cell 2 continues on schedule with a July completion date. Work will start in Cell 1 or Cell 5 when Cell 2 has been finished and tied into the primary loop system.

Nuclear Flux Monitor System

Flux monitors continued to be a problem. Both subcritical monitor channels were out of service at month end. Galvanometer chambers 4A and 4B failed. However, 4A was later returned to service. The trips on channel C rate-of-rise circuit were reduced to compensate for the loss of chambers. Galvanometer chamber 1B failed on April 29; this caused a power supply fuse failure which resulted in a temporary loss of C channel rate-of-rise function. Chamber 1B was disconnected and the C channel returned to normal service with reduced trip settings.

Three flux monitor chambers that are suitable for installation in either the high level or galvanometer positions are now on hand and will be installed in the most appropriate locations. The cable on No. 2 subcritical chamber requires replacement because of the heat damage resulting from high-shield temperatures.

Secondary System

A reduction in the surface condenser vacuum for drive turbine No. 3 required placing the hogging injector in service to prevent loss of the turbine. The reason for the loss of vacuum has not been determined.

High Pressure Raw Water System

The No. 3 high pressure raw water pump was removed from operation because the inboard bearing of its electrical drive motor had failed. This resulted in the loss of a normally available spare motor-driven unit. Repair required dismantling of the motor and a subsequent class A overhaul.

After being cleaned, the motor stator was checked for resistance to ground. Readings indicated deterioration of the winding insulation. When repeated efforts to improve the low readings met with only limited success, the decision was made to rewind the motor.

The following pertinent facts were obtained during the stator rewind:

- The winding appeared to have been subjected to excessive temperatures in the past.
• The mica insulation used in the winding slots was cracked in several places and thoroughly impregnated with rust.

• A slight amount of distortion in the stator laminations was evident.

**Electrical Interruption**

On April 11 an A-bus outage occurred on the 480 V system. This electrical interruption was caused by an under-voltage trip induced by a surge during routine testing of the main 230 KV breaker. To prevent recurrence, the under-voltage trip relay was adjusted and additional relays have been procured to detect and annunciate when a low voltage condition exists.

**PROCESS ASSISTANCE & CONTROL**

**Operational Physics**

The rod configuration was switched to an outboard pattern when the excess reactivity dropped to 3.5 mk on April 10. The outboard pattern rather than the inboard pattern is used when the excess reactivity is near or below 3 mk so that more rods are in the reactor. No operating problems have been experienced with the Mark IV columns presently in the reactor. Except for a slightly lower reactivity effect, no physics difficulties are foreseen for the transition from Mark I to Mark IV fuel.

Flattening decreased 5 percent following the March discharge, during which nearly the entire right side of the spike ring was recharged with green fuel; therefore, the control rod density was higher on the right side in order to accommodate the higher localized reactivity in that region. The flattening efficiency is expected to improve during the summer as green spike fuel is charged in the remainder of the spike ring.

Some operational physics parameters of interest are shown in the table below:

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Effective Central Tubes (ECT)</td>
<td>781</td>
</tr>
<tr>
<td>Flattening Efficiency - April</td>
<td>0.78</td>
</tr>
<tr>
<td>- 9-mo. average</td>
<td>0.82</td>
</tr>
<tr>
<td>Maximum operating time permitting scram recovery - hours</td>
<td>24</td>
</tr>
</tbody>
</table>
### Fuel Code C

- **No.**: 155
- **Tubes**: Mk-IC (94 Metal Fringe)
- **Description**: (94 Metal Fringe)
- **PT-NR**: 76
- **No.**: 4
- **Description**: Fuel Meltdown Test

### Fuel Code D

- **No.**: 277
- **Tubes**: Mk-IC (94 Metal - High U-236)
- **Description**: (94 Metal - High U-236)
- **PT-NR**: 94+ 23
- **Description**: Mk-IV Demonstration

### Fuel Code E

- **No.**: 219
- **Tubes**: Mk-IC (94 Metal - Fringe - Central)
- **Description**: (94 Metal - Fringe - Central)
- **PT-NR**: 96 22
- **Description**: Mk-I from Upset-Forged Billets

### Fuel Code F

- **No.**: 1
- **Tubes**: Mk-IV (94 Metal - High U-236)
- **Description**: (94 Metal - High U-236)
- **PT-NR**: 01 12
- **Description**: Mk-I from Direct Cast Billets

### Fuel Code N

- **No.**: 1
- **Tubes**: Mk-IC (Natural U)
- **Description**: (Natural U)
- **PT-NR**: 07 47
- **Description**: Initial Full Length Mk-IV Columns

### Fuel Code X

- **No.**: 242
- **Tubes**: Mk-IA (125 - 94 Metal)
- **Description**: (125 - 94 Metal)
- **PT-NR**: 07 47
- **Description**: Blank Channel or Empty Tube

### Total

- **No.**: 895
- **Tubes**: Total
- **Description**: Total
- **No.**: 109
- **Description**: Total PTs
- **No.**: 1004
- **Tubes**: Grand Total

---

+Includes Mk-IV High U-236 Content Fuel and 2 Tubes w/Mk-IA 125-94 Metal Mk-IV 94 Metal.

**LOADING PATTERN - N REACTOR**

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

PRODUCTION

General

Production of AlSi-bonded and hot-die-sized fuel for the C and K reactors was 102.6 percent of forecast. Ninety percent of these elements had bumpers or self-supports attached.

Acceptable Elements Produced

<table>
<thead>
<tr>
<th></th>
<th>Finished Production (tons)</th>
<th>Yield - Percent</th>
</tr>
</thead>
<tbody>
<tr>
<td>AlSi-Bonded Fuel</td>
<td>206.5</td>
<td>95.9</td>
</tr>
<tr>
<td>Hot-Die-Sized Fuel</td>
<td>8.9</td>
<td>*</td>
</tr>
<tr>
<td>Thoria</td>
<td>17.4</td>
<td>96.5</td>
</tr>
</tbody>
</table>

*Finished HDS fuel sent to storage was insufficient to permit yield determination.

Month-End Inventories

<table>
<thead>
<tr>
<th></th>
<th>Tons</th>
</tr>
</thead>
<tbody>
<tr>
<td>Bare Uranium Cores</td>
<td>940*</td>
</tr>
<tr>
<td>Finished Fuel: AlSi-Bonded</td>
<td>1,303*</td>
</tr>
<tr>
<td></td>
<td>51</td>
</tr>
<tr>
<td>Thoria Elements</td>
<td>37</td>
</tr>
</tbody>
</table>

*These totals include 116 tons of cores and 134 tons of finished fuel made for the smaller reactors.

OPERATING EXPERIENCE

Overall operating efficiency of the AlSi-bonding lines was 99.3 percent. Downtime was assigned 57 percent to equipment malfunctions and 43 percent to operational causes. AlSi canning operations continued on the basis of three lines (day shift) five days per week until April 28, at which time production was reduced to two lines.

Shakedown of the new thoria wafer process was performed the week of April 21. Three-shift operation of the wafer sintering furnace commenced April 28. Canning operations will be performed on the day shift.
EQUIPMENT EXPERIENCE

One of the electrical lead cables to the coil of a canning line furnace failed on April 10. Because of oxidation at one end of the cable, some of the strands of wire were not conducting properly. Repairs were made before the both could freeze, and only 17 minutes of production were lost on the one canning line.

PROCESS ASSISTANCE AND CONTROL

Nothing significant to report.
FUEL AND TARGET FABRICATION - N REACTOR

PRODUCTION

Input Production

<table>
<thead>
<tr>
<th>Description</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Total billets extruded</td>
<td>288</td>
</tr>
<tr>
<td>Tons extruded</td>
<td>61.0</td>
</tr>
<tr>
<td>Percent of forecast</td>
<td>98.4</td>
</tr>
</tbody>
</table>

Output Production

<table>
<thead>
<tr>
<th>Description</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Total finished fuel assemblies</td>
<td>1,418</td>
</tr>
<tr>
<td>Tons output</td>
<td>30.7</td>
</tr>
<tr>
<td>Percent of forecast</td>
<td>102.3</td>
</tr>
</tbody>
</table>

Uranium Utilization - %

<table>
<thead>
<tr>
<th>Description</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>68.6</td>
</tr>
</tbody>
</table>

Month-End Inventories - Tons

<table>
<thead>
<tr>
<th>Description</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Bare uranium billets</td>
<td>218</td>
</tr>
<tr>
<td>Finished fuel</td>
<td>331</td>
</tr>
</tbody>
</table>

OPERATING EXPERIENCE

The primary reason for the below-forecast input production was the failure of one of the billet preheat furnaces. Temporary repair was made while awaiting procurement of a new grating for the furnace.

EQUIPMENT EXPERIENCE

The Heald borematic lathe used to counterbore uranium from fuel ends prior to end closure brazing was removed from service to correct a serious vibration problem. After leveling and alignment, neoprene covers were installed to protect the ways and slides from abrasive uranium chips. During the Heald outage all counterboring was performed on a Gisholt lathe.

PROCESS ASSISTANCE AND CONTROL

Zircaloy Component Fabrication

The conventional method used to manufacture heavy-wall copper tubing includes primary extrusion or forging of an ingot to solid secondary billet size, OD turning and end facing. The billet is then drilled or pierced and extruded. Following extrusion, the butt end is cut or sheared off. A method being
developed for copper, which may be applicable to Zircaloy in the future, performs the billet piercing, extrusion and butt shearing steps in a single operation.

In this method, an ingot is primary-extruded to the proper OD from the 11-inch liner with no ID hole. It is then cut to length (about 12 to 13 inches), preheated to 1200 F, and placed in a 7.6-inch liner and upset at 200 to 400 tons to prevent shifting during the pierce operation.

The piercing mandrel is then driven through the solid billet until the tip emerges in the die land. The mandrel mover is stopped and the main ram is moved forward to complete the extrusion over the undersize mandrel. The piercing tip drops off and precedes the tube through the die. The mandrel mover goes forward with the main ram until the follower block is within 1/4 inch of the die. The main ram is then stopped and the mandrel mover driven forward forcing a sleeve through the copper butt. The mandrel is then retracted and the shearing sleeve removed from the butt with a 60-ton hydraulic press (or it can be ejected by forcing the sleeve completely through the butt by the mandrel mover assembly).

N Fuels Fabrication Brochure

An attractive brochure titled "N Reactor Fuels Fabrication Process" was completed and printed for unclassified use (e.g., in recruiting). This publication shows the process flowsheet and describes briefly the several steps involved.
Mission 1 - Basic Production

1-A. Brittle Fracture Program

Fracture Mechanics Testing Program

Preparation of samples for the various piping materials tests is complete. Three types of samples have been prepared for four tests. These include WOL (wedge opening loading), SEC (single edge crack) and NDT (nil ductility temperature).

DUN-5717, describing the materials testing program for the K reactor inlet piping, was issued. A summary of this program is given below:

- Thirty-two fracture tests of WOL samples have been planned, of which 15 have been completed. The 15 samples include material from the KE-D and KE-A risers and show no more than a 10 to 15 percent scatter in toughness values. It was originally intended to cover a temperature range from -60 C to 100 C; however, the material is sufficiently ductile (at least for the size of the sample) at the higher temperature that the test results are not meaningful. An interim report describing the results obtained to date is being prepared.

- To date, all fracture tests have been performed at the Lawrence Radiation Laboratory (LRL). Of the 17 remaining fracture tests, seven are to be run at LRL and the final 10 locally by Battelle-Northwest (BNW).

- Crack growth rate tests have been completed at LRL on two WOL samples. Preliminary results indicate higher growth rates than observed on other mild carbon steels. This may be partly due, however, to the geometry of the sample rather than a basic difference in material behavior. Six additional fatigue tests are planned for BNW with SEC samples. Tests will be performed in reactor process water at rates of 6 to 7 cycles/second and 1 to 2 cycles/minute to determine the influence of cycle rate on growth rate.

- Equipment is being installed for tests to determine what effect, if any, the process water environment has on defects in the piping steel at steady stress. Several structure modifications are necessary to the creep testing rigs which will be used to stress the WOL samples for this test.
Six flat-plate samples have been prepared to determine the NDT of the K piping steel for comparison with those obtained from Charpy tests. Following machining, a brittle weld bead is deposited on one side of the sample and notched part way through the bead. The sample is then struck by a falling weight in an approved NDT testing machine. The temperature at which the crack runs to one or both edges of the tension surface of the plate is defined as the NDT.

**Acoustic Monitoring**

The four wye joints in the inlet piping at the KE Reactor have been monitored for crack growth by installation of piezoelectric acoustic monitors. The wye joints on the A and B riser approach piping were inspected in December 1968 and March 1969, respectively. A signal source was found on the B joint similar to that reported in December for the A joint. The source will be located and examined with an ultrasonic tester, similar to the examination of the KE-A wye joint. The acoustic monitoring equipment was moved to the KE-C and KE-D wye joints and data obtained during April. These data are currently being analyzed.

1-B. Technical Bases for K Reactor Tube Power Limits

Work is continuing on the development of technical bases for K Reactor tube power limits. In the dual riser accident case, detailed technical bases are being developed in order to permit generation of process standards for tube power limits applicable to the E-Q loading. These will account for ribbed aluminum process channels, non-standard loadings, orificed process channels, and differences in flow rate among process channels.

For the single riser accident case, all preliminary scoping work has been completed and computer runs are being made to define worst-case post-accident power transients. The scoping phase primarily involved evaluation of scram delay times and determining the appropriate reference tube power distribution for evaluation of void-reactivity kinetics. It was determined that the scram delay time (time from the start of the accident to initiation of rod drop) utilized in the dual riser analysis will be suitable for the single riser case. However, a different reference power distribution will be required to account for differences between the two cases with respect to: (1) post-accident flow rate, (2) effects of numerical distribution of higher-powered tubes, and (3) significance of hydrostatic pressure in the outlet headers.

It is expected that development of comprehensive technical bases for the dual riser case will be complete early in May. By that time, the single riser case will be carried to the stage at which a correlation will be available to relate the principal reactor parameters (inlet temperature, bulk outlet temperature, exposure) to tube power limits.

Early in May, work will begin on the documentation of results obtained to that time in the safety analysis report of the brittle fracture program.
The principal objective of the tube power limit discussion in that report will be to demonstrate that adequate technology has been developed to enable operation of the reactors in such a way that supply piping failures would have no nuclear safety consequences. Subsequent efforts will be directed toward consideration of reactor loadings other than the E-Q.

1-C. Zircaloy Process Tube Hydriding

The influence of the internal case-hydride layers in K reactor process tubes on the stresses in the tube wall is being investigated. The experimental procedure consists of attaching strain gages on the outside tube surfaces to determine the changes in stress as the case layer is gradually removed electrochemically. Preliminary data indicate that the presence of a continuous case-hydride layer on the tube ID results in a tensile stress in the tube wall which may be several times the stress from internal pressure (normally about 2000 psi). The study is being continued to evaluate the magnitude of the stresses as a function of case thickness, and to determine their significance to tube life.

A statistical analysis of the hydride data has been made to determine if there is a correlation between hydride concentration and operating temperature or time. No correlation between either time or temperature and total hydrogen content could be found. The lack of correlation with time suggests that the bulk of the case-layer buildup occurs early in the life of the tubes. The linear relation between time and base-metal hydrogen content that has always been apparent was also observed, but again no relation to operating temperature could be discerned in the data.

It has been observed, however, that tubes with low operating temperatures (less than 80 C) have significantly lower hydriding rates than tubes operating above 100 C. The data analyses indicate that the temperature effect is small enough in the normal operating range to be masked by data scatter.

1-D. Physics Code Development

HAMMER Lattice Code Development

An investigation to determine the cause of the negative fluxes encountered when using the new HAMLET library indicates the program which prepares the cross sections to be at fault. Addition of the resonance scattering cross sections to smooth scattering cross sections, in an attempt to treat both resonance and smooth scattering, results in large and irregular scattering cross sections which may cause negative fluxes to be calculated. The suggested correction is to remove the resonance scattering contributions and ignore resonance scattering. The library is now being modified to test this solution.

The scattering cross sections which were missing from several isotopes in the thermal library have been added, and the library will be transferred to the PCF file in the near future.
DCODE Adaptation - K Reactors

Production of isotopes in a K lattice has not been well predicted by DCODE to date. To determine the reasons, several cases will be run on MOFDA and DCODE for both hot and cold conditions, and comparisons will be made of the C lattice against the K lattice. When a general problem area has been identified, Computer Sciences assistance will be requested for detailed programming support.

1-E. Corrosion Studies

Production Test 159, designed to compare the corrosion response of hot-die-sized fuel with AlSi-canned fuel, was charged into KE during the early April outage. Twenty-four columns were charged on row 34 in the central zone of the reactor. Four columns will be discharged at an exposure of 850 MWD/T, to verify continuance of this test, and the remaining 20 will be irradiated to 1200 MWD/T.

Mission 3 - Transplutonium Technology

3-A. Ten-Kilogram Plutonium Irradiation - PTA-150

Loading changes made in the test block during the April maintenance outage of KE Reactor are expected to increase support-zone powers to normal levels and to increase the powers of the PuAl columns. Additional adjustments are anticipated within the block as the heat generation rate in the PuAl columns decreases with exposure.

3-B. Plutonium Burning

Calculations are in progress for determining irradiation and production details for a 250 kg irradiation of 16 to 20 percent Pu-240 plutonium to a goal of 70 percent Pu-242 to be separated by January 1, 1975.

Plutonium isotopic yield and assay curves used for the two-thirds length PuAl columns are given in Figure D-1 (next page), which also shows the initial assumptions.

Mission 4 - Pu-238

4-A. Medical Grade Pu-238

The neptunium-aluminum matrix elements have been fabricated for the irradiation of neptunium in a column-cluster configuration. Fabrication of the neptunium-graphite matrix elements is scheduled to be completed shortly.

Pu-238 production parameters are being studied for various neptunium target designs to minimize diluents (especially aluminum) to alleviate waste management problems at Purex and to soften the neutron and gamma spectra for production of medical grade Pu-238. One example of the designs being studied is the annular NpO₂ element with a knockout graphite core, and another is a graphite cylinder with axial holes for NpO₂ near the outer surface.
Figure D-1. Pu Burning - Isotopic Assay & Yield vs. Exposure
Preliminary calculations have been made for the annular element and for one with a centered rod of NpO₂. The results show that, for equivalent neptunium g/ft, the Pu-238 production rate in the rod would be only half that in the annular element. The reduction would be due to the flux depression caused by concentrating the NpO₂. Adverse effects of this concentration would be higher Pu-236 concentration and less economical utilization of NpO₂.

4-B. Americium Irradiation

Fabrication and characterization of the americium targets are complete. Based on gamma ray intensity, the capsules contain about 0.1 percent Cm-243.

Mission 7 - Target Space Enhancement

7-A. Large Channel Facility - C Reactor

Following the early-April announcement of C Reactor deactivation, efforts in support of PTA-103, SUP3, for a 94 Metal-natural uranium replacement loading for the 44-tube overbore test block were discontinued. Hot-die-sized fuel elements from representative overbore columns at approximately 1200 and 1800 MWD/T exposures are being analyzed for performance and conversion data.

7-B. K Reactor Overbore Studies

Control system strength calculations for the various overbore fuels under flooding conditions are nearing completion using the recently debugged OVERKILL computer program for optimizing overbore calculations. Sufficient data have been generated to date to allow investigation of total control problems on a whole-reactor basis for partial overbore loadings, and efforts to that end are underway. K overbore physics work has been outlined, with the objective of documenting the technology status by the end of this fiscal year.

Mission 10 - Columbia River

10-A. Effect of Ionic Impurities on Effluent Activity

Tests to determine the effects of individual ionic impurities in reactor process water on effluent activity have been initiated. The process water is being simulated by adding the major ionic constituents to demineralized water, treating the resultant water in the Water Treatment Pilot Plant, and discharging the effluent to waste. As soon as this system is working smoothly, the WTPP effluent will be routed through the WTPF in-reactor tubes. Difficulties are being experienced currently with silica buildup on the filter beds. Normal backwashing procedures will not adequately remove the deposits.

10-B. High Turbidity Coolant

Data reported last month indicated that, during the period January 27 to March 3, the P-32 and AS-76 radioactivity levels were 15 to 25 percent lower on the side of KE Reactor operating at high (0.3 JTU) turbidity than
on the side of the reactor at normal (0.07 JTU) turbidity. Concentration measurements up to March 22 for AS-76 and three additional isotopes (Cr-51, Sc-46, Np-239) show similar trends. However, the latest data show no reduction of P-32 in the high turbidity coolant. The test is continuing to determine if the activity reductions will continue over a long-term period.

ENGINEERING & TECHNOLOGY - REACTORS

165-KW Smoke Density Monitor

The installation of equipment for the prototype monitor on the No. 1 boiler stack at 165-KW was completed, and the unit was used during boiler startup for the acceptance testing of boiler control modifications. During this initial operation, the monitor demonstrated very good response to changes of smoke density in the stack. Further testing of the monitor has been temporarily curtailed due to shutdown of the boiler for maintenance.

Foreign Object Detection in Crossheader Strainers

The mockup for use in K reactor crossheader strainer testing in 190-D Building was completed and is in operation. The audio noise level in the strainer from the pumps supplying the water to the test facility is considerably above the noise level on reactor. However, the noise from large metallic objects in the strainer is sufficient to permit their detection.

C Reactor Deactivation

This work includes the scope and design of modifications required to place the deactivated C Reactor and associated water plant facilities in a standby status. The following related documents have been completed and issued:

- DUN-5698, "C Reactor Plant Modification Engineering Report".
- DUN-5707, "Preliminary Engineering Report, Proposed Power Distribution and Electrical Services, C Reactor Plant Modifications".
- DUN-5708, "Preliminary Engineering Report, Evacuation Alarm System and Supervisory Control Revisions, C Reactor Plant Modifications".

Design Criteria in support of the Project Proposal covering capital funded work associated with the deactivation have been approved internally preparatory to submittal to AEC-RL for approval. The capital work consists basically of (1) procurement of electric heaters for those buildings or rooms requiring heat; (2) provision of evacuation warning alarm system; (3) installation of permanent markers for closed burial grounds; (4) provision of fire protection systems; and (5) installation only of two transformer stations.

Detail design of piping modifications in Bldg. 182-B to permit adding a diesel-driven pump to the export system pumping station has been initiated. It is planned that one of the three diesel-driven pumps which supplied last-ditch coolant to C Reactor will be used for this service. The required emergency water supply to the 200 Areas is currently supplied by steam-turbine-driven pumps. Steam for backup pumping power will no longer be available when 184-B steam plant is deactivated.
Columbia River Flow Forecasts

Due to low river flows in March, power reservoirs still hold below-normal amounts of water in storage. Roosevelt Lake has been drawn down to the lowest point in its history (1,168 feet msl); normal full reservoir elevation is 1,290 ft. msl. The river flow forecast (at Hanford), based principally on mountain snow cover and on the assumption that precipitation and temperature will be near average from the present time to the end of the forecast period, is as follows as of April 1 (expressed as thousands of acre-feet):

<table>
<thead>
<tr>
<th>Forecast Period</th>
<th>Runoff 1969</th>
<th>% of 15-Year Average</th>
<th>Runoff 1968</th>
</tr>
</thead>
<tbody>
<tr>
<td>Apr.-Sept.</td>
<td>80,500</td>
<td>106</td>
<td>69,046</td>
</tr>
<tr>
<td>Apr.-July</td>
<td>69,300</td>
<td>107</td>
<td>55,130</td>
</tr>
<tr>
<td>Apr.-June</td>
<td>53,500</td>
<td>106</td>
<td>39,602</td>
</tr>
</tbody>
</table>

Liaison has been established with the Portland River Forecast Center (PRFC) for obtaining projections of expected river flows at Hanford during the spring runoff season. The PRFC operates on a 24-hour, 7-day week basis during runoff season and provides forecast information to interested companies, utilities, and organizations via a private, unlisted telephone circuit. This communication system has been made available to Douglas United Nuclear on the basis that the unlisted telephone number will not be disclosed to unauthorized parties. As of April 1, the PRFC provided the following long-range crest stage/flow forecast for the Priest Rapids Dam Station:

<table>
<thead>
<tr>
<th>Flood Stage (feet)</th>
<th>Crest Stage (feet)</th>
<th>Peak Flow (cfs)</th>
</tr>
</thead>
<tbody>
<tr>
<td>422</td>
<td>420.5-422.5</td>
<td>370,000-420,000</td>
</tr>
</tbody>
</table>

Future river regulation and power generation planning programs, particularly as related to full implementation of Canadian Treaty Dam regulation, were discussed with BPA on April 7. Present, but preliminary, planning for regulation of the Columbia River indicates that river stage and flow through the Hanford Reservation will not pose any particular plant operating problems.

Project Engineering - Reactor Facilities

The status of approved construction projects relating to single-pass reactor facilities is summarized in Appendix A.

ENGINEERING & TECHNOLOGY - FUELS & TARGETS

HDS Process Test

A large-scale hot-die-size process test was completed except for data analysis. The results were: DECLASSIFIED.
• Machine cleaning of cladding components on the cap and can machine was successful. Refinements in procedure are necessary but the basic process parameters have been established.

• Testing of lengthened preheat furnace fuel cycles has shown this procedure to be acceptable. Increased throughput capability is realized by extending residence time and loading each carrier instead of loading alternate carriers. A 40-second fuel cycle was demonstrated and steps were taken to demonstrate a 30-second cycle. The success of this procedure was attributed to helium purging in the preheat furnace and refined contaminant control at plating.

• Testing of shorter end-bonding cycles has met with limited success. Small-scale tests are being conducted but the refinements appear to have no marked effect. Testing is underway to determine the capability of the pilot plant equipment to deliver greater power.

HDS Component Reclamation

Initial shipments of spireless cladding components exhibited galling on the base-hole surface. A cleaning procedure was developed which should clean the defect area sufficiently to permit effective closure of the defect during sizing. However, severe galling cannot be corrected so a reject rate which is higher than normal is expected. A large-scale test was initiated to determine the effect of this cleaning procedure, and the reject rate to be incurred by using these components.
TECHNICAL ACTIVITIES - N REACTOR

RESEARCH AND DEVELOPMENT

Mission 1 - Basic Production

1-A. Mark IV Fuel Development

Irradiation Testing

Eight more full-length (16-assembly) columns of Mark IV production fuel were charged in March, bringing the in-reactor total to 69 (including the twenty-one 13-assembly columns charged in July 1968).

Flow Loop Testing

Life testing of Mark IV test fuels is continuing in the flow loop:

<table>
<thead>
<tr>
<th>Fuel Type</th>
<th>Support System</th>
<th>No. of Assemblies</th>
<th>Loop Time (Hours)</th>
</tr>
</thead>
<tbody>
<tr>
<td>26&quot; Mark IV</td>
<td>End-spiders</td>
<td>8</td>
<td>4295</td>
</tr>
<tr>
<td>26&quot; Mark IV</td>
<td>End-spiders with one or two legs removed</td>
<td>4</td>
<td>5050</td>
</tr>
<tr>
<td>12&quot; Mark IV</td>
<td>Spring-stop</td>
<td>6</td>
<td>1037</td>
</tr>
</tbody>
</table>

The fuels were inspected twice this month and no failures were observed. The eight with uncut spiders have endured a total of $9.3 \times 10^8$ hydraulically induced cycles. The 12-inch Mark IV fuel has shown no signs of detrimental effects due to placing the outer supports nearer to the ends than is done on the standard 26-inch fuel.

Mark IV Reactivity Observations

Although the reactivity differences between the Mark I and Mark IV lattice calculated from the FLEX code were too small to show definite trends, recent MOFDA calculations have indicated a loss of 2 to 3 milli-k in converting from the Mark I to the Mark IV loading. This predicted loss is consistent with the predicted gain in conversion efficiency of around 3 percent. Detailed differences in Mark IV versus Mark I lattice parameters are shown in the table below:

<table>
<thead>
<tr>
<th>Exposure (MWD/T)</th>
<th>Computer Code</th>
<th>$f$</th>
<th>$p$</th>
<th>$\varepsilon$</th>
<th>$n$</th>
<th>$k$ (Amk)</th>
<th>Conversion Ratio ($\Delta%$)</th>
</tr>
</thead>
<tbody>
<tr>
<td>0</td>
<td>FLEX</td>
<td>+11.6</td>
<td>-12.6</td>
<td>+4.1</td>
<td>-2.1</td>
<td>-0.1</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td>MOFDA</td>
<td>+11.7</td>
<td>-15.3</td>
<td>+4.2</td>
<td>-1.7</td>
<td>-3.2</td>
<td>+3.7</td>
</tr>
<tr>
<td>1000</td>
<td>FLEX</td>
<td>+11.1</td>
<td>-13.0</td>
<td>+4.1</td>
<td>-0.4</td>
<td>-0.2</td>
<td>+2.7</td>
</tr>
<tr>
<td></td>
<td>MOFDA</td>
<td>+12.2</td>
<td>-15.8</td>
<td>+4.2</td>
<td>-0.1</td>
<td>-2.1</td>
<td>+3.5</td>
</tr>
<tr>
<td>2100</td>
<td>FLEX</td>
<td>+10.7</td>
<td>-13.2</td>
<td>+4.0</td>
<td>-0.3</td>
<td>-0.8</td>
<td>+2.7</td>
</tr>
<tr>
<td></td>
<td>MOFDA</td>
<td>+12.5</td>
<td>-16.7</td>
<td>+4.2</td>
<td>+0.7</td>
<td>-2.5</td>
<td>+3.7</td>
</tr>
</tbody>
</table>
HAMMER calculational cases are also being investigated. The above results indicate the principal difference to be in the calculation of captures in U-238.

1-B. Zircaloy Process Tube Monitoring.

Zircaloy process tubes 1756 and 2455, removed from N Reactor at average exposures of 0.59 and $0.9 \times 10^{21} \text{n/cm}^2$, respectively, have been subjected to post-irradiation examination. Both had been fabricated by extrusion, followed by cold drawing to 30 percent reduction in area. Testing of the second tube discharged (2455) was completed in April. The results show that N Reactor Zircaloy process tube properties had not deteriorated as a result of irradiation to $0.9 \times 10^{21} \text{n/cm}^2$; indeed, resistance of the tubes to certain types of failure may have been enhanced by the strength increase experienced.

A third process tube, 0758, was removed in February 1969 at an average exposure of $1.45 \times 10^{21} \text{n/cm}^2$. This tube, fabricated by welding short extrusions together, received a final cold-work of 18 percent reduction in area by drawing. The tube has been cut into sections for testing.

Preliminary hydrogen analyses performed on 0758 tube sections located upstream of the fuel charge are in general agreement with analyses from tubes 1756 and 2455. That is, there is a front-to-rear hydrogen concentration profile, with low hydride contents near the tube inlet, increasing to a maximum in the section upstream of the fuel charge or flux zone and decreasing to low values throughout the flux zone and out-of-flux downstream sections. However, the preliminary data suggest that the concentrations in tube 0758 will be no higher than those found in tubes 1756 and 2455 which had less time in the reactor.

The data shown in the next table suggest that the rate of hydrogen absorption is decreasing with time, or variations between tubes are so great as to require a larger sampling to establish trends. Also, the fact that 0758 was fabricated by a different process could have affected its susceptibility to hydrogen absorption.

<table>
<thead>
<tr>
<th>Process Tube</th>
<th>1756</th>
<th>2455</th>
<th>0758</th>
</tr>
</thead>
<tbody>
<tr>
<td>Operating days</td>
<td>300</td>
<td>575</td>
<td>950</td>
</tr>
<tr>
<td>Hydrogen concentration, ppm</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>3-5 inches*</td>
<td>13</td>
<td>11</td>
<td>24</td>
</tr>
<tr>
<td>25-30 inches</td>
<td>41</td>
<td>66</td>
<td>52</td>
</tr>
<tr>
<td>50-60 inches</td>
<td>39</td>
<td>55</td>
<td>43</td>
</tr>
</tbody>
</table>

*Distance from the process tube inlet (at front rolled joint). The fuel charge begins at 137 inches from the tube inlet.

Further hydrogen analyses on sections from 2455 show that, in locations with relatively high hydrogen concentrations, there is a variation in hydride...
concentration across the tube wall. The results, tabulated below, from a sample location 46 inches from the tube inlet, indicate that two gradients exist, with higher hydrogen concentrations at both tube surfaces. Also, a circumferential difference was observed as shown by the gradients obtained at the 10 o'clock position.

<table>
<thead>
<tr>
<th>Circumferential Location (o'clock)</th>
<th>Hydrogen, Mg/mil*</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Tube ID</td>
</tr>
<tr>
<td>2</td>
<td>6.4</td>
</tr>
<tr>
<td>6</td>
<td>6.2</td>
</tr>
<tr>
<td>10**</td>
<td>6.1</td>
</tr>
</tbody>
</table>

*Average of duplicate analyses.
**Minimum occurred at point 200 mils from tube inside surface with a concentration of approximately 1.1 mg/mil. Midpoint is at about 120 mils from ID surface.

These data indicate the apparent entry of hydrogen from both sides of the tube. Analysis of the primary coolant shows that 20 to 30 cc H2/kg of coolant is continuously available to the inside surface. On the gas-phase side of the tubes, the wet helium environment supplies hydrogen by decomposition of water. Tube temperatures during operation are of the order of 370 F upstream of the fuel charge, 370 to 550 F across the fuel charge, and 500 F in the downstream spacer locations.

The hydrogen absorption observed in upstream locations may occur because damage or degraded oxide is repaired less rapidly at the lower temperatures. The presence of a neutron flux may also contribute. The oxide degradation hypothesis is being investigated in autoclave tests. In addition, the possible galvanic charging of hydrogen into the tube because of the presence of carbon steel spacers in the hydrided areas is being examined.

1-C. Corrosion Studies

Control Rod Pitting

Radiographic examination of 2 linear feet of seal section tubes of control rod 22 showed:

- No severe pitting had occurred in the area examined. The largest penetration observed was about 0.016 inch deep in the 0.065-inch thick wall.

- With improved penetrometer standards, the procedure should work for routine inspections of rod seal tubing.

Removal of two seal section tubes, one cleaned and one uncleared, is scheduled during May. The tubes will be used to calibrate the radiographic procedure, as well as to assess the effect of rod cleaning.
The aluminum shell in the tip section of rod 27 was examined borescopically after the isotope elements were discharged. Four minor pits were observed in about 12 feet of the rod.

**Examination of Inconel-600 Steam Generator Tube**

Ultrasonic examination of the tube removed from steam generator 4A has shown one internal indication about 0.012-inch deep. Sectioning and metallography of the tube at critical points—tube-to-tube sheet junction, tube sheet area, tube support areas, bend areas—has been completed. No cracking or surface attack was found other than the <0.0005-inch deep surface intergranular attack observed on the tube removed 2 years ago. The ultrasonic indication is being sought by repetitive grinding and polishing of the sample, but has not been located.

1-D. Control System Improvement

A digital simulation of selected primary coolant control systems has been developed for application to the plant improvement program. The simulation tool applied is the MIMIC digital program. The following systems are included:

- The high pressure injection pump speed control system,
- The pressurizer level control system (fill, spill and charge/discharge valves and their controllers),
- The pressurizer pressure control system,
- The seal water pressure control system.

1-E. Code Development

A least-squares fit program (Linear Regression Analysis) has been completed and checked out. It is intended for curve fitting of multiple point representations of the reactor and reactor checkout data. The program will fit polynomials of 20th degree or less, as well as some special exponential and transcendental functions.

1-F. Billet Fabrication Studies

Twenty-two columns of upset-forged Mark IC fuel remain in N Reactor under PT-NR-96. Nine columns which had reached goal exposure were discharged during the March outage. Post-irradiation measurements are in progress on the fuels from the two monitor columns.

The direct-cast billet test columns have reached an exposure of 1100 MWD/T, or 50 percent of goal.

**Mission 2 - Coproduct**

The lithium aluminum elements from isotope rod 27 were discharged in late March with no apparent difficulties. These target elements are to be examined.
for physical effects prior to their inclusion with other material for normal separations processing.

Mission 8 - Nuclear Safety

8-A. Metal-Water Reaction and Fission Product Release Experiments

An apparently successful metal-water reaction experiment was completed on April 15 in the 200-W laboratory. All instrumentation and equipment functioned satisfactorily. Fuel clad failure was observed at a temperature of 1030°C, and the hydrogen evolution rates were measured for 35 minutes following clad failure. Quantitative results of the experiment are not yet available.

8-B. Soil Studies

DUN-5602, "100-N Riverbank Springs I-131 Concentration Data," was issued. The document reports observations from 1965 through 1968 on the I-131 in riverbank springs near the N Reactor effluent crib. Available data on dilution of effluents by the river are presented, with an estimate of the contributions by N Reactor to the I-131 concentration at downriver points of uptake at Ringold and Richland. It is concluded that this contribution is no greater than 0.001 of the 10 CFR 20 guidelines. Accordingly, in a letter to AEC-RL dated April 16, it was recommended that the rupture waste decontamination facility (Part 7 of the Effluent Control Project) not be built.

In order to determine, on an analytical basis, the travel time of I-131 from the N Reactor crib to the riverbank, a computer program has been written which crosscorrelates the time-dependent I-131 concentration at the N springs. Since the resultant crosscorrelation is a function of the shift in time of the crib concentration with respect to the springs concentration, the shift for which the crosscorrelation function is a maximum (i.e., maximum correlation of the two concentration functions) will be the most probable travel time of I-131 from the crib to the springs.

Figure DN-1, on the next page, shows the normalized crosscorrelation for concentration data taken from August 1 to December 20, 1968. The normalization has no effect, since the only parameter of interest is the time shift at which the maximum positive value of the crosscorrelation occurs. It can be seen that the most probable I-131 travel time lies between 6 and 8 days.

ENGINEERING & TECHNOLOGY - N REACTOR

Ball Trip Mechanism

Development of a more reliable and maintainable ball trip mechanism is proceeding on a high priority basis. Two independent concepts are being pursued; however, both are intended to (1) employ the existing safety trip circuitry and (2) eliminate the existing hold-in coil, linkage, and latching mechanism.
FIGURE DW-1. Crosscorrelation of I-131 Concentration in N Crib with that in N Springs
One concept utilizes the existing air-operated cocking cylinder as the trip mechanism. The cylinder is modified to incorporate a fail-safe, single solenoid, air pilot control valve. Trip timing tests have demonstrated that the time for the gate to open is identical (0.39 seconds) to that of the present mechanical trip mechanism. Life testing of the new solenoid valve is presently in progress.

The second concept employs an electromagnetic clutch arrangement as the hold-in and trip mechanism. A clutch plate attached to the gate shaft is engaged to the clutch in the cocked position. When the safety circuit is de-energized, the clutch plate is released allowing the gate shaft and gate to open. Design drawings for the prototype unit are 75 percent complete.

**Flow Throttling and Orificing Device for V-11 Valve Elimination**

Design Test No. 50 presents a method of eliminating the need for V-11 valves at N Reactor. By inserting a sleeve in the inlet nozzle the function of the V-11 valves may be duplicated at the nozzle. A test fixture was fabricated from two nozzle sections and has been installed in the NPR test loop for evaluation of this design. To accommodate an orifice sleeve, the nozzle I.D. must be bored oversize. The resultant reduction in gasket seating area will be checked for successful sealing in the NPR test loop.

**Licensability Program - Seismic Studies**

As part of the licensability program for N Reactor, key safety system components are being analyzed and tested to provide substantiating evidence that they will function normally during an earthquake and maintain total reactivity control of the reactor. The components under study are those whose failure would have the effect of adding to the consequence of a loss of coolant accident or result in an uncontrolled release of excessive amounts of radioactivity or impede the safe shutdown and isolation of the reactor. The N Reactor systems which must function properly during a ground tremor include the Horizontal Control Rod System, Ball Safety System, Emergency Raw Water System and the Confinement System.

The following safety system components have been tested, with results as documented in DUN-5574 and summarized below:

1. Horizontal Control Rod System
   a. Drive Mechanism
   b. Solenoid Valve Rack

2. Emergency Raw Water System
   a. V-3, V-4 Solenoid Control Valves

3. Ball Safety System
   a. Ball Hopper Gate
   b. Closing and Holding Solenoids
   c. Ball Drain Valve
   d. Ball Trip Mechanism Linkage
Since a large shaking table is not available at Hanford, the response of these critical components was studied by installing accelerometers on the equipment, shaking the system at varying frequencies, measuring component accelerations, and observing carefully any tendency for the system to function improperly. Maximum accelerations attained by the components during the shaking tests were greater than 0.80 g. The average acceleration spectra used in the design of the N Reactor plant correspond to a maximum ground acceleration of 0.2 g. The components operated successfully under the test conditions, providing strong evidence that they would operate successfully in the maximum earthquake considered for the N Reactor site.

Project Engineering - Reactor Facilities

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

ENGINEERING & TECHNOLOGY - FUELS & TARGETS

Coextrusion Bond Studies

Testing continued in an effort to determine causes of the severe clad-to-core unbond problem which occurred early this year. The present test was directed toward preshape design and materials. The test billets were divided into four groups, as follows:

A: 11 billets with copper-silicon preshapes on front end only (squared rear end)
B: 10 billets with copper-silicon preshapes on rear end only (squared front end)
C: 6 billets with copper preshapes on front end only (squared rear end)
D: 7 billets with copper preshapes on rear end only (squared front end)

All of the billets in the four groups were coextruded under normal process conditions and evaluated for bonding by the standard peel test. The reject rates were very low in comparison to those of the three previous campaigns which averaged 16.7 percent.

It is felt that this marked improvement was due to one or more of the following factors:

- Vacuum-annealed Cu-Si preshapes were used to eliminate outgassing in the billet assembly.
- A new uranium billet lot was used for the first time during this test.
- End-plate ODs were at the maximum specified value, resulting in tight fit-ups which may have contributed to OD weld quality.
IRRADIATION SERVICES

FUEL TECHNOLOGY

Fast Reactor Oxide and Nitride Fuels - Battelle-Northwest

Two capsules were irradiated in the Snout facility at KW in support of the LMFBR fast fuel development program. The first, a (U-Pu)O₂ fuel capsule, did not achieve core melting. The second capsule, a mixed (U-Pu)N fuel capsule, did not show the expected change in thermal conductivity. These capsules were the first of a series of irradiations.

Irradiation Effects on Fuels and Materials - Battelle-Northwest

Four capsules, two containing metallic uranium and two containing plutonium, were charged into a bottom front-to-rear test hole at KE Reactor. This irradiation is in support of a BNW-ANL study of the isotopic content of fuel samples.

MATERIALS DEVELOPMENT

Corrosion Product Transport Facility (CPTF) Operation - Battelle-Northwest

C Reactor was shut down in mid-April for installation of an in-reactor test section and for connecting up of the fourth module. These facilities were to serve as radioactive deposition and effluent water sampling stations. The work was successfully completed and the C-1 loop, which served to control the temperature of the in-reactor test section of the CPTF, was started up. The CPTF operated satisfactorily at test design conditions. This was the first in-reactor experiment of this program, and several days of operation were achieved before C Reactor was deactivated on April 26.

These experiments are part of a program being conducted for the AEC Naval Reactors Division, aimed at identifying and measuring the variables affecting corrosion product transport and deposition in a nuclear reactor system. With C Reactor now in standby, the CPTF will be removed from there and installed at KE.

Nuclear Graphite Studies - Battelle-Northwest

Three instrumented capsules containing boron carbide were charged into a General Purpose facility at KE Reactor. These capsules will be irradiated for three, six, and nine months, respectively. This test is being conducted to determine gas evolution and dimensional stability of a possible FFTF control rod material.

A BNW "cold-seeding" effect graphite capsule was irradiated in the KW Reactor Snout facility.
Haynes Metal Irradiation - Douglas United Nuclear

One capsule containing tensile specimen samples of a Haynes metal process tube was discharged from a downstream position of the Snout facility at KE. This test will assist in the determination of the physical properties of Haynes metal under irradiation.

ROUTINE IRRADIATIONS

Sixty-four Quickie activation analysis capsules were irradiated in the C and K facilities for BNW.

DECLASSIFIED
FY 1971 BUDGET AND REVISION OF FY 1970 BUDGET

In accordance with AEC-RL budget instructions, budget preparation (prior to the recent reactor cutback announcement) had been based upon the assumption that C Reactor would be deactivated and the two K reactors would be operated alternately. Now that only C will be affected, complete revision of the budget has been necessary. Because the announcement came late in the budgeting process, AEC-RL extended the due dates for budget submission by two weeks.

CAPITAL PLANT AND EQUIPMENT BUDGET

The DUN contribution to the AEC-RL Capital Plant and Equipment Budget for FY 1971 and Revision of FY 1970 was delivered on schedule.

CASH REQUIREMENTS

On April 2 the Company submitted to AEC-RL estimates of FY 1969 cash expenditures, by month, for April through June. For the month of April it was estimated that advances from Federal Reserve would total $2,532,000, and AEC-RL gave approval to withdraw this amount. Actual withdrawals were exactly $2,532,000 for the month and leave the Company with an approximate ending cash balance of $75,000.

AUDITING

An auditor is currently working with the AEC-RL appraiser on an appraisal of DUN contracting and procurement activities. This AEC-RL appraisal is being conducted concurrently with the Company's annual audit of this function.

Audit steps outlined in the "Program for AEC-Wide Audit of Employee Relocation Costs and Practices", issued by the AEC Controller, are well underway and a report will be submitted to AEC-RL in May. This audit applies to contractor employees "relocated" during CY 1969.

APPROVAL LETTERS

At the close of the reporting period, final AEC-RL action was pending for the following requests:

<table>
<thead>
<tr>
<th>AEC-RL</th>
<th>Subject</th>
<th>Date of Transmittal to AEC-RL</th>
</tr>
</thead>
</table>
EQUAL EMPLOYMENT OPPORTUNITY

Acceptance at Washington State University of the two potential Cooperative Education Trainees is virtually assured. Final acceptance of these two Negro high school graduates will increase to four the number of employees enrolled in the program. All will be enrolled in the School of Engineering. Possibilities are being explored with the University regarding expansion of the program to include the Business Administration Department.

SAFETY

No personnel radiation exposures exceeded operational control.

Month-end safety statistics were:

- Disabling injuries - April: 0
  - Year to date: 0
- Days since last disabling injury: 181
- Man-hours since last disabling injury: 1,800,000

EMPLOYMENT SUMMARY

DUN personnel totals and employee allocations as of March 31 and April 30 are shown in Appendix B.
## APPENDIX A

### PROJECT STATUS SUMMARY - REACTOR FACILITIES

<table>
<thead>
<tr>
<th>Number &amp; Title</th>
<th>Authorized Funds - $</th>
<th>Percent Complete</th>
<th>Status</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Single-Pass Reactors</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>DCE-505, Boiler Control Improvements - 165-KE &amp; KW</td>
<td>410,000</td>
<td>100</td>
<td>68</td>
</tr>
<tr>
<td>DAP-510, Discharge Chute Clearing Equipment - K Reactors</td>
<td>190,000</td>
<td>100</td>
<td>93</td>
</tr>
<tr>
<td>DAP-516, Storage Building Addition, 105-KE &amp; KW</td>
<td>142,000</td>
<td>5</td>
<td>0</td>
</tr>
<tr>
<td>DAE-518, Effluent Radio-iodine Monitor - C, KE &amp; KW Reactors</td>
<td>100,000</td>
<td>98</td>
<td>0</td>
</tr>
</tbody>
</table>

All construction drawings have been issued. Construction will be performed at KE and KW only. Instrument equipment to be fabricated by DUN; installation by J. A. Jones.
### PROJECT STATUS SUMMARY - REACTOR FACILITIES (contd.)

<table>
<thead>
<tr>
<th>Number &amp; Title</th>
<th>Authorized Funds - $</th>
<th>Percent Complete</th>
<th>Status</th>
</tr>
</thead>
<tbody>
<tr>
<td>DCP-522, Modification of Reactor Coolant Cross Tie Piping - 105-KE &amp; KW</td>
<td>163,000</td>
<td>99 10</td>
<td>Funding increased by AEC-RL Directive dated April 16 to cover fabrication and installation at both KE and KW. Shop fabrication and field preparations with emphasis now on KW. All materials are on site for one reactor with remainder due shortly. The reuse materials and equipment have been refurbished or cleaned and inspected.</td>
</tr>
<tr>
<td>N Reactor</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>GCP-406, Improved Safety Platforms and Accesses - 100-N Area</td>
<td>300,000</td>
<td>100 95</td>
<td>Installation of reactor platforms in progress.</td>
</tr>
<tr>
<td>GCE-408, W, C &amp; D Elevator Safety Improvements - 105-N</td>
<td>90,000</td>
<td>100 30</td>
<td>Revised design was approved April 8.</td>
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<td>GCP-411, Effluent Control Program - 100-N Area</td>
<td>1,780,000</td>
<td>100 5</td>
<td>Work progressing on Sections I and III. Revision of criteria and scope is in progress for the lump-sum portion for Diesel Fire Protection. The low bid for the Dump Tank is $358,000; the award was mailed on April 21.</td>
</tr>
<tr>
<td>DCE-519, Replacement of Bridge Crane and Hoist System with New Crane System - 105-N Storage Basin</td>
<td>269,000</td>
<td>40 0</td>
<td>The Vitro/HES project cost estimate for construction verified by J. A. Jones.</td>
</tr>
</tbody>
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### APPENDIX B

#### EMPLOYMENT SUMMARY
(with employee allocations)

<table>
<thead>
<tr>
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<th>4/30/69</th>
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<td><strong>CONTRACT PERSONNEL</strong></td>
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<td>Douglas United Nuclear</td>
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<td>Assisting other Contractors</td>
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<td>Other Programs under AEC Contract</td>
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<td><strong>Total Contract Personnel</strong></td>
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|                     |         |         |
| **COMMERCIAL ACTIVITIES PERSONNEL** | 17 | 18 |

|                     |         |         |
| **TOTAL FORCE**     | 1887    | 1879    |

G-3
FEATURE REPORT

BALL SAFETY SYSTEM BELLOWS SHROUING - K REACTORS

INTRODUCTION

The large graphite stacks of the Hanford single-pass production reactors are made up of individual blocks about four inches square by four feet long. Years of nuclear irradiation have caused contraction of these blocks and a general subsidence of the top of the stack. Graphite distortion being a function of both temperature and total neutron exposure, the stack subsidence is greatest at the center of the reactor and decreases to zero at the edges. This subsidence has created a variety of problems, one of which is associated with the Ball 3X nuclear safety control system.

This summary report describes (1) the difficulty anticipated with the stainless-steel bellows which conduct the balls through the gas gap between the biological and thermal shields, (2) the several corrective measures considered, and (3) the design development and in-reactor application of the bellows severance and shrouding techniques being used at KE and KW. Similar steps were planned for C Reactor, where the graphite contraction has proceeded more slowly, but that unit has now been deactivated.

THE BELLOWS PROBLEM

As shown in Figure 1, appended, each of the Ball Safety System assemblies at the top of the K reactors consists of a ball hopper embedded in the concrete biological shield, and a passageway through which the balls flow from the hopper into the associated vertical safety rod (VSR) channel. The ball entry bellows which bridges the gas gap is obviously vital to system integrity. In the K reactors as built, and as shown in Figure 1, this gap was 16-1/2 inches.

The biological shield is self-supporting, while the cast-iron thermal shield blocks rest on the graphite stack. The ball entry bellows was designed to accommodate the thermal expansion and the graphite distortion (growth) anticipated at that time. However, it was found that with higher fluxes and temperatures, graphite contraction rather than growth occurred. This contraction has caused a concurrent settling of the thermal shield and hence an elongation of the Ball System bellows.

It was realized that continued graphite contraction would result in either the bellows failing or the bellows supporting the thermal shield. In either circumstance there would be a discontinuity in the ball entry passageway into the VSR channel. If the bellows failed, balls could discharge between the biological shield and the thermal shield. If the thermal shield were to be supported by the bellows instead of resting on graphite, there would be a gap between the thermal shield and the moderator into which balls could penetrate. Failure of the balls to enter the VSR channel would result in insufficient reactor control.
during an emergency situation. Balls spreading over the graphite moderator, then falling into cracks or remaining on top, would alter the reactor physics characteristics, probably increasing fuel enrichment requirements and cost, and possibly even preventing reactor operation.

In mid-FY 1965 this problem was given serious attention and a program was established to provide a timely solution. Laboratory tests had established the load carrying capability and probable failure mode of the bellows. It was also established that, with a small factor of safety, and about an eight-inch elongation, the bellows would support the thermal shield load (assuming no failure of the bellows). Failure of the bellows was confined to tearing of the flange and/or failure of the cap screws (see appended Figure 2).

By mid-FY 1966, the problem had been studied and the several solutions noted below were under consideration. At that time the maximum subsidence of the graphite (and elongation of the bellows) was about 6-1/2 inches, and the continuing rate of subsidence at the top-center of the stack was estimated at 0.9 inch per year. It was deemed necessary to have a usable solution in hand by the time the bellows was stretched 7-1/2 to 8 inches. This meant that only 12 to 18 months were available in which to provide this solution.

The location of the bellows, about seven feet inside the reactor, greatly complicated the task. The only available access is through the adjacent VSR shield penetration which steps down to an opening about 4 inches in diameter, and which is offset about 8 inches from the bellows centerline. Consequently, consideration of bellows replacement or modification demanded unusual ingenuity in design of the concept and development of the associated tooling. Visibility for all work is restricted to the use of a borescope. Moreover, a beam of high energy gamma radiation projects from the reactor through the open VSR shield penetration.

CORRECTIVE MEASURES CONSIDERED

Several potential solutions received consideration during the evaluation of the bellows problem. Of these, the following four showed the most promise.

1. Abandon the existing ball hopper system and provide new step plugs and ball hoppers similar to those designed and installed in the smaller production reactors (B, D, F, DR, and H) after original construction. In this arrangement, the ball passageway through the shields and into the VSR channel in the moderator is the annulus between the VSR and its channel. Although this was a proven design, it had a very high cost and would somewhat restrict the ball flow rate into the reactor. The existing bellows would still have to be severed to prevent lifting of the thermal shield blocks.

2. Core drill through the biological shield so that the stretched bellows could be removed. A longer bellows would then be installed and the shield penetration plugged. Although this proposal would have retained the existing ball hopper system and rapid ball entry rate into the reactor, the installation cost was high. More importantly, excessive
time would have been required for tooling development.

3. Sever the stretched bellows and install a flexible metal liner through the existing ball entry passageway. The flexible liner would have extended from the ball hopper through one leg of the chute transition piece, through the severed bellows, and into the thermal shield. While the cost of this proposal was attractive, and the existing ball hopper system would have been retained, the lowered entry rate of balls into the reactor was a serious drawback. In fact, the use of this scheme in more than about 10 central zone channels would have required a reduction in reactor power level.

4. Sever the stretched bellows, install a stainless steel sleeve or shroud snugly around the severed bellows, and secure the shroud to the upper section of the bellows. This proposal retained the existing ball hopper system and its high rate of ball entry. The estimated installation cost was reasonable although the tooling design and development was seen to be complex.

Solution No. 4 was selected for development and use.

**SHROUD DESIGN AND DEVELOPMENT**

**Full-Scale Mockup**

All of the tooling for bellows severance and shrouding was designed to be inserted and operated through the VSR shield penetration adjacent to each ball system passageway. To check out and demonstrate this tooling, a full-scale mockup was constructed in Bldg. 189-D. This facility is shown in Figure 3, appended. The grating visible at the top of the front view represents the top of the 6'-10" thick biological shield.

**Bellows Saw**

Initial design and development efforts were devoted to the bellows cutting saw, since severance of the bellows was common to nearly all proposed solutions. The first saw was built under a design and fabrication contract with Hydranamics Systems Corporation of Seattle (see Figure 4, appended). The saw is inserted through the VSR opening with the blade extended downward, as pictured, and then the blade is turned 90° to the shaft axis for cutting. Power is supplied by a 1200 RPM air motor. For backup, a heavier saw, similar in design, was fabricated prior to the on-reactor operations. A severed bellows is shown in appended Figure 5.

**High-Strength Shroud**

The original shroud concept was based on the use of a preformed 301 stainless-steel shroud having very high yield strength (250,000 to 300,000 psi). With such a high yield strength, a preformed shroud can be reverse-wound into a small cylinder without any permanent deformation. Figure 6, appended, shows the preformed shroud and the installation tool cylinder onto which the shroud
is reverse-wrapped. The tool and shroud then are inserted through the VSR opening, and the cylinder is rotated 180° adjacent to the severed bellows. As the shroud is unwound, it returns to its preformed condition, fitting snugly around the bellows. The shroud is then clamped securely to the upper bellows section as shown in Figure 7, appended. The light and mirror of the borescope used for visual guidance during clamping can be seen at the end of the tool.

**Low-Strength Shroud**

Due to difficulty in procuring the high-strength shroud material, a backup method was devised which utilizes an easily procurable, low-strength stainless shroud material. The use of this metal requires that the shroud be formed around the bellows in-place. This is accomplished by rotating the shroud through a special set of forming rolls, after which the shroud is firmly attached to the top section of the bellows by a pair of self-tapping metal screws. The tools used for these two operations are shown in Figure 8, appended. A completed installation of a low-strength shroud is shown in Figure 9.

**ON-REACTOR SHROUDING EXPERIENCE**

A group of the five most critical bellows at each K reactor was chosen for the first phase of modification. These were units 44, 45, 46, 47, and 48 (see top-of-reactor map on Figure 10, appended). Two bellows were modified initially at each reactor under a Production Test Authorization to provide on-reactor testing of this concept. One high-strength and one low-strength shroud was installed at each reactor under this PTA. A few installation problems were encountered. It was found that positioning of the shroud prior to fastening is critical, and that installation of the clamps on the high-strength shroud is a tedious task requiring special skill. After these installations, it was apparent that either shroud type provided an adequate solution. Although the low-strength shroud was somewhat easier to fasten to the upper section of the bellows, the high-strength type was chosen for application during the remaining program because of its greater strength and stability.

During this first phase, engineers familiar with the new tools and techniques provided on-the-job training for the plant maintenance supervisors and craftsmen, and assisted during the installations. The bellows severing and shroud insertion operations were soon mastered, but final shroud placement and clamping are more difficult and require a certain "feel" and coordination between the tool and the borescope image. A mockup of a typical ball hopper and bellows assembly was built at KE Reactor and used in the further training of crafts people.

Because of the need for expediency in modifying the rest of the most extended bellows, ten more at each K reactor were shrouded as quickly as outage time and manpower permitted (see Figure 10). This work was completed in CY 1968.

The tooling for the bellows shrouding operation worked remarkably well, with only very minor changes required. The borescope was modified to provide a
clearer view and a variable voltage power supply was added to help control glare. Some of the difficulty experienced in attaching the shroud clamps, unless the shroud was in a preferred rotational orientation, was alleviated with a simple tool devised to engage the shroud slots and enable accurate positioning. Clamping can then be accomplished without undue difficulty by a qualified man.

During the past month, seven more bellows were cut and shrouded at KE Reactor. These are the ones shown on Figure 10 in the shaded blocks. They had been scheduled for modification by mid-1970, but were done now because the outage situation was favorable, and the people involved are still very familiar with the special manipulations required.

PLANS FOR FURTHER SHROUDING

While the program to date has successfully modified all K reactor bellows stretched to the 8-inch limit, plus the seven others at KE, the graphite moderator continues to contract. The subsidence is being monitored and future bellows severance and shrouding have been programmed. Based on present contraction rates, bellows 43 and 49 at KW must be shrouded by about April 1970 and bellows 61, 62, 63, 64, and 65 there will require this modification by August 1970. As will be seen in Figure 10, this will leave unsevered and unshrouded only the bellows at outer locations where the graphite distortion rate is significantly less.
Figure 1. Ball Hopper & Passageway Assembly
Figure 2. Bellows Flange Failure Modes
Figure 5. Severed Bellows
Figure 6. High-Strength Preformed Shroud & Installation Tools
Figure 9. Low-Strength Shroud - In Place
Bellows shrouding status:
- Central block: Completed at KE & KW.
- Shaded blocks: Completed at KE; Scheduled at KW.

Figure 10. VSR and Ball Hopper Locations - KE & KW