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# DOUGLAS UNITED NUCLEAR MONTHLY REPORT

AUGUST 1969

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September 18, 1969



# DOUGLAS UNITED NUCLEAR

# MONTHLY REPORT

## AUGUST 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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## SUMMARY

### REACTOR PLANT OPERATIONS

#### Production Statistics

| | KE | KW | N |
|---|---|---|---|
| Input Production - Pu (KMWD) | 74.9 | 100.2 | - |
| - U-233 (Equiv. KMWD) | 11.9 | 14.7 | - |
| Time Operated Efficiency - % | 68.8 | 86.7 | - |
| Steam Availability to WPPSS - % | - | - | - |

#### K Reactors

Maximum power levels of 3800 MW at KE and 3865 MW at KW were achieved, being restricted mainly by the 95 C bulk outlet water temperature limit. The low production and efficiency at KE resulted from unscheduled outages caused mainly by fuel element failures.

Thoria core loads are being irradiated at both reactors to produce U-233. The plutonium made was largely fuel grade.

#### N Reactor

The reactor was shut down for an extended outage on July 20 to permit a number of piping system tie-ins required for GCP-411, "Effluent Control Program". This outage continued throughout the month of August, and the time was utilized to repair major primary system leaks and do other plant improvement work concurrent with the project effort.

### FUEL AND TARGET FABRICATION

#### Production Statistics (tons)

| | For KE & KW | For N |
|---|---|---|
| Billets Extruded | - | 30.4 |
| Finished Fuel Produced | 165.3 | 19.1 |
| Thoria Canned | 17.3 | - |

#### K Fuels

AlSi canning operations increased from two to three lines per day, five days per week, on August 11 to utilize personnel made available by the conclusion of the thoria canning program.

N Fuels

Input production was intentionally held below forecast to prevent excessive rejected materials resulting from clad dimple difficulty.

TECHNICAL ACTIVITIES

K Reactors

Work was completed on the development of basic information for use in deriving bases for tube power limits to be applied for a natural uranium loading in the K reactors. It is expected that preliminary bases will be available about the first week in September.

A sensitivity analysis has been completed to determine the effect on calculated maximum clad surface temperature that would be associated with failure of one K reactor vertical rod to enter the core. Also, the differences in flux peaking and shutdown transient effects due to a safety rod out of service during a riser loss accident have been calculated.

Brittle fracture program test work included two additional single-edge crack tests in progress, two additional samples to be fatigued in a tension-tension loading to simulate the effects of mean stresses of 4000 and 10,000 psi and drop tests to measure the nil-ductility temperature of KW Reactor B riser parent metal as specified in ASTM E-208.

The evaluation of Zircaloy process tubing removed from C Reactor after nine years of satisfactory operation showed a maximum of approximately 460 ppm hydrogen in the bulk metal as compared with 170 ppm, the maximum hydrogen content of any K tube examined to date.

The production test permitting on-reactor tests to investigate several parameters affecting a through-reactor decontamination was approved. Initial tests, scheduled for September, will consist of injecting preheated decontamination chemicals into single process tubes to determine the most efficient decontamination solution concentration, temperature, and time. Subsequent tests on a full-crossheader scale will determine flow control and distribution problems, together with the feasibility of using reactor decay heat to warm the solution.

Investigation of the cause of failure of the graphite matrix neptunium elements in the two core block columns continued. Radiometallurgical examination of four neptunium oxide-graphite wafer elements showed that dimensional instability of the wafers was responsible for clad failure with subsequent end cap separation. Graphite neptunium wafers will be examined radiochemically and fabrication process parameters will be investigated to attempt to identify the cause of the dimensional instability.

Oralloy fuel development and irradiation planning was started. The plans include demonstrating that fabrication, irradiation, and nuclear safety technology would be on an immediate production status in order to capitalize on the incentives for the potential economic and product flexibility advantages of oralloy use.

Reactivity tests on the lithium bearing spline samples procured for evaluation showed them to be about 50 percent as black as standard boron splines.

A 10 to 20 percent reduction in effluent activity, similar to that experienced during the winter months, was noted after equilibrium conditions were reached in the high turbidity test at KE Reactor.

## N Reactor

The CLUMSY code is being utilized to determine the limitations imposed on the nuclear protection systems in conditions for which one or more flux monitoring points has failed. A byproduct of the study could be the establishment of technical bases for the nuclear protection systems.

Thermal cycling tests at unbonded end closures have resulted in circumferential cracking of a significant number of test specimens. The brittle weld character was also evidenced by the impact tests where circumferential cracks propagated in the weld zone. The failure mechanism is believed to be the strain cycling or fatigue of the weld structure promoted by the presence of large alpha grains introduced during the thermal cycling to beta heat treat temperatures.

Ultrasonic inspection of the primary coolant system piping from cells 4 and 5 showed the piping welds examined to be free of significant flaws.

Two irradiated Mark I inner fuel element pieces were tested to determine the metal-water reaction rate under conditions simulating those imposed by the geometry restriction of the outer fuel piece and adjacent elements at each end, using a quartz tube to simulate the outer fuel piece.

Ten borescopic examinations were made on seven process tubes, with three tubes examined before and after through-reactor decontamination. The tubes showed normal tube wear and contained minor scratches from charge-discharge operations. Decontamination did not appear to have completely removed the surface film, and there did not appear to be any increase in nozzle corrosion as the result of decontamination.

## FEATURE REPORT

Discharged reactor fuels contain many potential exploitable materials of interest in a wide spectrum of industrial and medical research applications. The Transuranium Pilot Plant operated by Douglas United Nuclear provides the capability to demonstrate the fabrication of target elements containing these highly radiotoxic materials. The appended summary report describes the Pilot Plant and the target fabrication work performed to date.

## GENERAL

Effective August 1, a Reactor Standards Development Subsection was established in the Manufacturing Engineering Section, Operations Division. The subsection will develop maintenance and testing standards for the AEC's Division of Reactor Development and Technology.

Twelve metal handlers in the Fuels Section were given layoff notices at the completion of the thoria canning program. Eleven were finally placed in other jobs with the Company and other Hanford contractors.

A disabling injury to a pipefitter occurred in 100-B Area on August 6. No radiation exposures exceeded operational control.

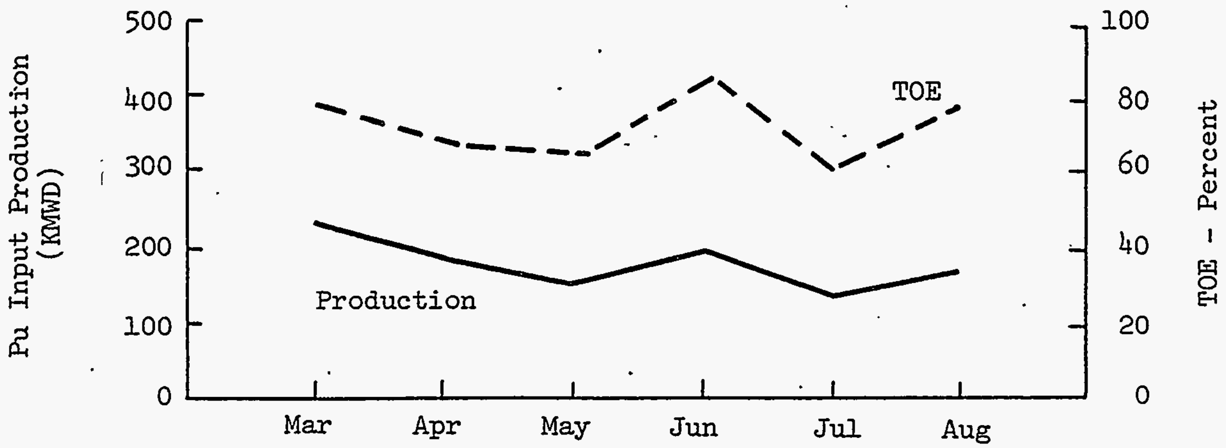
Charles D. Harrington
President

# REACTOR PLANT OPERATIONS - KE & KW

## PRODUCTION

### General

Production, power levels, efficiencies and related statistics for the two K reactors are tabulated below. Overall K reactor input production and time operated efficiency (TOE) for the past six months are shown on the following chart:




Thoria loads are being irradiated in both KE and KW to make U-233. The plutonium produced was largely fuel grade in the 8-10 percent Pu-240 range.

### Statistical Summary

| | KE | KW | Total |
|---|---|---|---|
| Input Production - Pu (KMWD) | 74.9 | 100.2 | 175.1 |
| - U-233 (Equiv. MWD) | 11.9 | 14.7 | 26.6 |
| Power Level (MW) - Maximum | 3,800 | 3,865 | 7,665 |
| - Average | 3,512 | 3,729 | 7,241 |
| Time Operated Efficiency - % | 68.8 | 86.7 | 77.8 |
| Number of Outages | 4 | 2 | 6 |
| Number of Startup Interruptions | 0 | 0 | 0 |
| Operating Coolant Flow - M gpm | 190.9 | 191.6 | 382.5 |
| Fuel Charged (Tons) - 94 Metal | 357.0 | 357.6 | 714.6 |
| - Natural U | 3.0 | 5.5 | 8.5 |

| | KE | KW | Total |
|---|---|---|---|
| Fuel Element Failures | 3 | 0 | 3 |
| Helium Losses - M cu. ft. | 190.1 | 114.3 | 304.4 |

## OPERATING EXPERIENCE

### Reactor Loadings

Front face maps showing the loadings in KE and KW are reproduced on the two pages which follow page B-5. The tonnages listed there are approximate; actual fuel totals are tabulated above.

### Power Levels

Maximum power levels shown above for the K reactors were restricted most of the month by the bulk outlet water temperature limit of 95 C. The precautions taken because of inlet water piping brittle fracture considerations restricted the KE level for a short period.

### Reactor Outages

KE Reactor was down for 64.5 hours at the start of the month, continuing work during an outage initiated in late July.

| Date Down | Reactor | Outage Hours | Remarks |
|---|---|---|---|
| August 4 | KE | 42.3 | Unexplained Panellit trip. Leak tested entire reactor and replaced one leaking tube. |
| August 7 | KW | 58.4 | Scheduled charge-discharge and maintenance. |
| August 12 | KE | 39.2 | Removal of a failed 94 Metal hot-die-sized fuel element (PTA-011) from tube 3074. |
| August 13 | KW | 40.5 | Scrammed by high speed scanner because of excessive humidity in the HSS plenum. |
| August 18 | KE | 48.9 | Removal of a failed 94 Metal fuel element from tube 4551. |
| August 24 | KE | 37.9 | Removal of a failed 94 Metal fuel element from tube 4951. |

## EQUIPMENT EXPERIENCE

### Vertical Safety Rods

At KE Reactor the pins in the remaining 12 flexible VSRs were checked for tightness and necessary repairs were completed as required. The average VSR hot drop time was 1.81 seconds, with all VSRs under 2.0 seconds.

At KW Reactor 28 pins were replaced in two rods. The pins in 12 VSRs remain to be checked.

### Turbogenerator

The No. 2 turbogenerator in 165-KE was removed from service on August 6 to correct alignment problems and severe arcing of the exciter commutator. The unit was realigned and test runs were made on August 28. Severe arcing of the exciter continued and the problem is attributed to faulty windings in the exciter rotor. The exciter rotor from the No. 2 turbogenerator at 165-KW will be installed in this unit, and the faulty rotor will be sent to Spokane for repairs.

### Deactivation - C Plant

Deactivation work at the C Plant is continuing. Processing's work in the reactor building, combined with steam and water facilities work, is estimated at 98 percent complete.

## PROCESS ASSISTANCE AND CONTROL

### Operational Physics

Although coverage was provided for six reactor startups during the month, there were no physics problems associated with the 94 Metal E-Q loads. With the current thoria irradiation program nearing goal, the reactivity coefficients and operating characteristics have been documented for future reference. Total control requirements for the forthcoming natural uranium loads will be calculated by a computer program, which is now nearly operational. Due to turnaround time on the computer, the program will not necessarily save time, but it will be particularly useful in calculating the minimum total control requirements for a variety of conditions. In addition, the program will assure a consistent approach for each calculation.

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

| | Reactor | |
| --- | --- | --- |
| | KE | KW |
| Effective Central Tubes (ECT)* | 2225 | 2243 |
| Flattening Efficiency** - August | 0.71 | 0.72 |
| - 12-mo. average | 0.72 | 0.71 |

| | Reactor | |
|---|---|---|
| | KE | KW |
| Maximum Operating Time Permitting Scram Recovery - Hours*** | 10 | 10 |

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

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

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

## Product Accountability Studies

Following the issuance by Oak Ridge of a 1969 $UO_3$ price schedule, the DUN burnout cost tables are being updated. The revised tables generated by the computer program and the text explanatory material have been completed in rough draft form for final review prior to publication.

## Production Fuel Performance

The two production K5E fuel element failures are described below:

| Tube No. | Charge Date | Failure Date | Exposure MWD/T | Failure Type |
|---|---|---|---|---|
| 4551-KE | 5/24/69 | 8/18/69 | 706 | End defect in fabrication |
| 4951-KE | 5/24/69 | 8/24/69 | 763 | Groove corrosion |

The failed element from tube 4551-KE exhibited characteristics of both core cracking and end closure defect failure mechanisms, although the latter is considered to be the primary mechanism. The core of this failure was fabricated by the recently-accepted solid induction heat treat process. The failed element from 4951-KE exhibited the typical self-support associated groove corrosion mechanism that has been observed in the recent HDS column failures.

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

| | 12 Months | 24 Months |
|---|---|---|
| Natural uranium | 11 | 8 |
| 94 Metal | 6 | 8 |

## High Speed Scanning System Safety Trip Limits

Study of the effects of variations of critical parameters pertaining to the zone system and rate-of-change system has been completed and reported (DUN-6103, "Safety Trip Limits for the High Speed Scanning System - K Reactors"). Parameters include coolant transport time, flow, RTD response time, RTD type (immersion or strap-on) and zone system trip points as a function of the rate-of-change system. In addition, the study pertains to a natural uranium load and a 94 Metal E-Q load. The previous study included limited variation of parameters.

## Use of Fuel Made for Smaller Reactors

Physics studies show that use of the 03 fuel (made for the smaller reactors) in a K reactor lattice will present no significant problem for speed of control and total control if the elements are only charged in the fringe region.

## 1969 Dual Reactor Tripout Test - Data Analysis

The dual K reactor tripout test was conducted on July 17. Tests were conducted successfully, following cleaning of the high pressure crosstie, and analysis of data thus far indicates that reliable top-of-riser pressure (TORP) decay data were obtained (KE as the stricken reactor) during the transient test (simulation of BPA failure at both reactors, accompanied by a secondary system failure at one reactor with two diesels supplying last-ditch flow). Steady-state test data indicated a measured flow from two diesel pumps of 32,900 gpm at KE and 35,500 gpm at KW during the simulation.

The tripout test results are being documented to provide a summary of test data, and also pressure decay curves, all the curves needed for backup adequacy calculations, last-ditch system pressure drop curves, diesel supply curves, reactor demand curves, a schematic of the hydraulic system and pressure tap locations, gage elevations and reactor loadings and load codes.

Last-ditch emergency coolant flow adequacy calculations using the tripout test data show coolant flow margins in terms of reactor power to be 4647 MW for KE and 4864 MW for KW.

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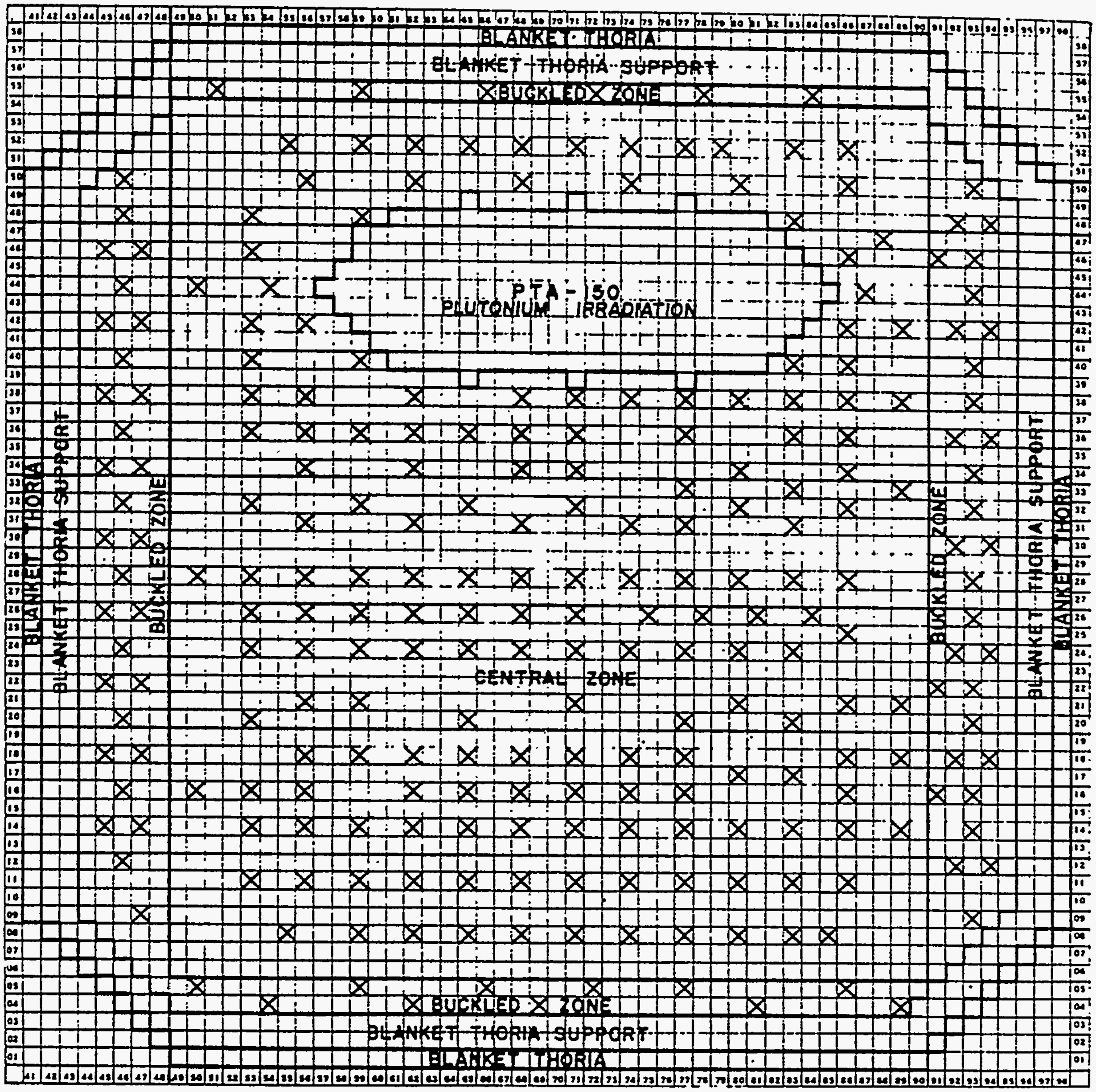


| Zone | Tons | Material |
|---|---|---|
| Central | 237 | 94 Metal (for Thoria Support) |
| | 2 | Natural Uranium |
| | 19 | Thoria ("X" Designates Tubes) |
| | 1 | Special Depleted Uranium (PITA-048) |
| Buckled | 65 | 94 Metal (for Thoria Support) |
| | 6 | Thoria ("X" Designates Tubes) |
| Blanket | 55 | 94 Metal (for Thoria Support) |
| | 11 | Thoria |

Loading Pattern - KE Reactor



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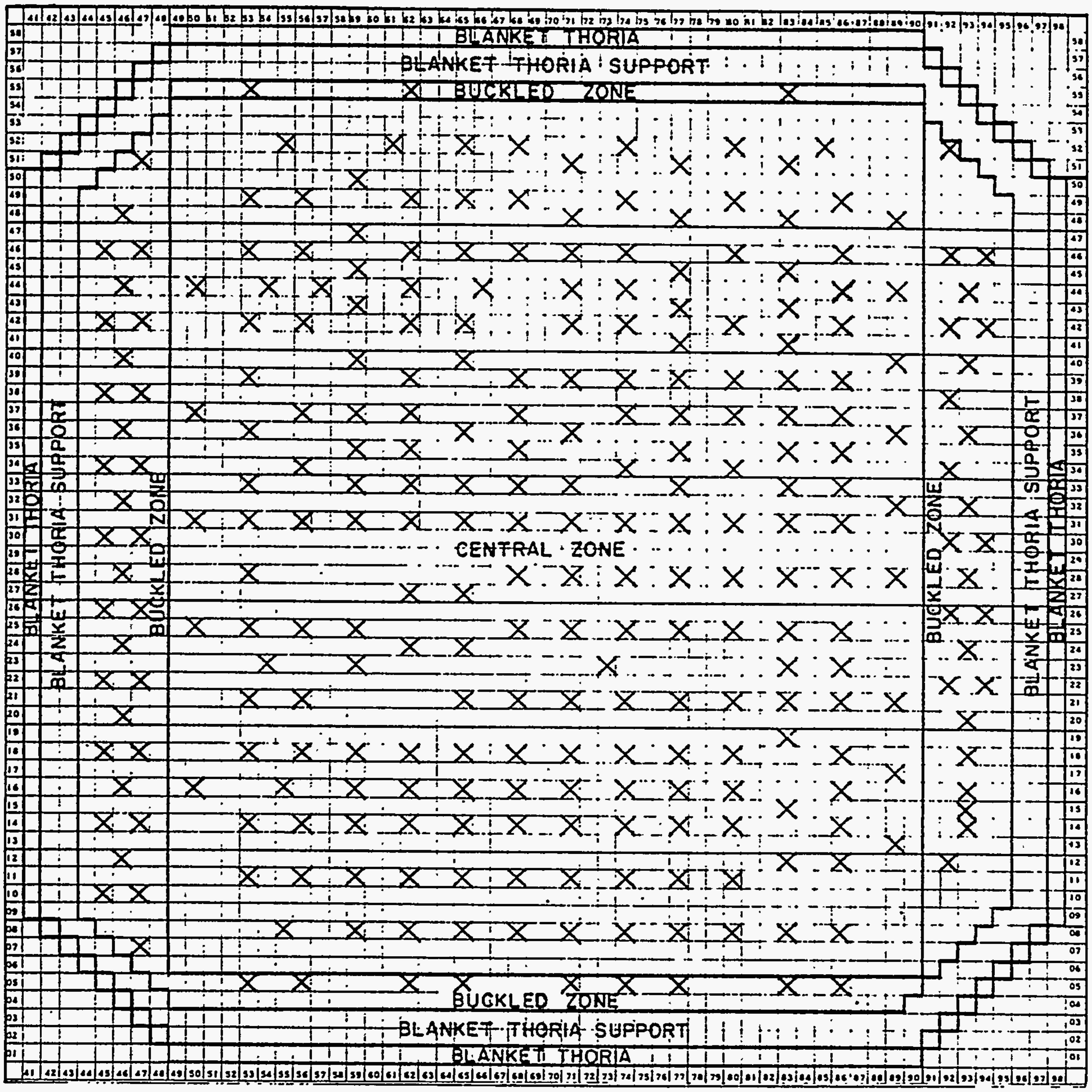



| Zone | Tons | Material |
|---|---|---|
| Central | 237 | 94 Metal (for Thoria Support) |
| | 1 | Natural Uranium |
| | 18 | Thoria ("X" Designates Tubes) |
| | 1 | Special Depleted Uranium (PITA-048) |
| Buckled | 66 | 94 Metal (for Thoria Support) |
| | 6 | Thoria ("X" Designates Tubes) |
| Blanket | 53 | 94 Metal (for Thoria Support) |
| | 11 | Thoria |

Loading Pattern - KW Reactor



[redacted]

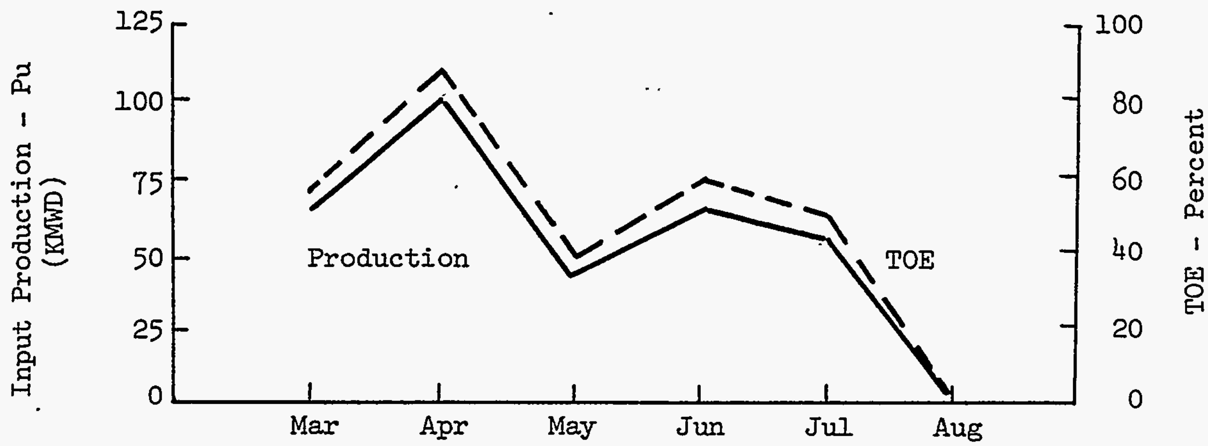
## REACTOR PLANT OPERATIONS - N

### PRODUCTION

#### General

The reactor remained shut down all month in continuation of the scheduled outage begun on July 20.

Input production and time operated efficiency (TOE) for the past six months are shown on the following chart:




#### Statistical Summary

| | | |
|---|---|---|
| Fuel Charge (Tons) | - 94 Metal | 309.8 |
| | - 125 Metal | 67.2 |
| | - Natural U | 0.4 |
| | Total | 377.4 |
| Helium Losses - M cu. ft. | | 0 |
| Fuel Oil Usage - bbl. | | 28,599 |

### OPERATING EXPERIENCE

#### Reactor Loading

The reactor loading at month end is shown on the front face map which follows page BN-6. As charge-discharge for the current outage was completed in July, there was no loading change during August.

## EQUIPMENT EXPERIENCE

### Boiler Experience

The CE-1 boiler was taken out of service for tube leak repair on August 1, and returned to service the next day.

The Foster Wheeler boiler was shut down for tube leak repair on August 6 and returned to service following repair on August 17. This boiler was again shut down for leak repair on August 21 and returned to service on August 31. Repairs consisted of correcting two mud drum leaks and one leak in a roof tube, replacing the ends of five steam drum tubes, and re-rolling ten steam drum tubes.

### Effluent Control Project

All scheduled tie-in work on GCP-411, Effluent Control Project, and the associated acceptance test procedures, were satisfactorily completed. The emergency coolant system will be operated in the same manner as before the tie-ins were made until the emergency raw water tank bypass is completed and tested. Excavation for the new dump tank was begun, and as a result the railroad serving the reactor building and the chemical waste loadout facility will be out of service until mid-September.

### Primary Loop

Extensive leak repair was completed on the primary loop. This and associated work included, but was not limited to, the following:

- Repacked 36 V-11 valves, and replaced 47 V-11 valve actuators without repacking the valves.
- Replaced 56 leaking V-12 valves.
- Repacked and regasketed PCRV-214-2.
- Repacked, gasketed, and seal welded four V-38 valves.
- Repacked V-26-1, V-23-1, V-9-8, PCRV-205-1, and eight CV-2 valves.
- Repaired leaks on V-16-2, RV-2-2, V-4-5, V-21, V-34-1, CV-1 & 5, and four V-1 valves.
- Installed new stem and plug and repacked PCRV-204-3.
- Replaced packing follower and leaking plug on V-18-1.
- Repaired leaks on 51 flow monitor sensing lines by replacing leaking sections.

Inspection of the primary loop at 300 psig pressure, following leak repair, revealed that a number of leaks still exist even though a marked improvement was evident.

A functional test of all 16 CV-2 valves (inlet header emergency line check valves) was made in an attempt to determine the cause of flow failure during the reactor piping decontamination. It was found that CV-2-11 would not come fully open, and investigation disclosed a small piece of extraneous metal wedged in the valve disc hinge. The metal was removed and the valve returned to normal. Upon rechecking, all CV-2s functioned satisfactorily.

Cell 2 was returned to service following steam generator retubing work completion. An attempt to pre-film the tubing in the steam generators was unsuccessful due to leakage from the primary piping in the cell through the isolation valves. This leakage prevented attainment of the required high temperature in the cell.

Cell 1 was isolated from the primary loop in preparation for steam generator decontamination. At month end, decontamination preparations were in HOLD status pending a decision to continue with steam generator retubing.

Secondary System

Extensive leak repair was completed on the secondary system by plant forces. Also, Design Change 3010, "WPPSS Feedwater Recirculation for Cleanup, 109-N Building, 100-N Area", was completed at the expense of WPPSS. With this modification, it will be possible for WPPSS to recirculate and clean-up water in their condensate return lines rather than dump to waste during plant startups.

Circulating Raw Water System

The overhaul of No. 3 CRW pump and motor was completed and the pump returned to normal service.

PT-125, "A-B Electrical Bus Hydraulic Independence Test", which included a number of transient tests, was run to verify safe operating conditions for the turbine generator upon loss of A bus power. New limitations on circulating raw water to the turbine generator surface condenser were established.

Emergency Raw Water System

The No. 1 low-lift diesel engine was completely overhauled (at the ITT/FSS shop) and a new crank shaft installed. It was returned to service and tested satisfactorily.

Fog Spray Strainer

Difficulty in welding-in the spool piece replacement for the damaged fog spray strainer body resulted in delays in scheduled reactor startup time. At month end the repair was not complete.

190 Roof Deluge System

Section 6 of the deluge system installed as part of a recent project was tested. Several faults were detected. As a result the system was placed on "manual" control rather than "automatic" until deficiencies can be corrected.

Horizontal Control Rod System

Thirteen HCR cooling lines were chemically cleaned during the outage. Rod 34 had indications of a slight leak prior to its cleaning; this leak persisted after cleaning and tube replacement, and testing revealed a tip leak. The rod is out of service. The packing was replaced in rods 26, 49, 82, and 107.

Ball Safety System

Ball channel 60 is used for graphite sample irradiation. Evidence of graphite moderator movement has been detected in the channel. This movement interferes with sample charge and discharge, and attempts to recover the installed samples were not successful.

Charge Machine

The charge machine was damaged when the W work platform went into a severe tilt condition. The tilt occurred during platform maintenance which required taking power off the hydraulic system. Repairs are underway and will require about two weeks.

D Work Platform

The D work platform went into a tilt condition while performing ATPs following Project work. In attempting to correct the tilt, the 28' left side landing grating was slightly damaged. The damage was repaired.

117 Building

The B cell mist separators were successfully decontaminated to permit making the scheduled DOP test. Radiation levels were reduced from 1500 mR/hr to 200 mR/hr, and dose rates for the test were 50 mR/hr. The test indicated the filters to be 99.998 percent effective.

Equipment Modifications

The following equipment modifications were completed:

EMP-237, "Main Steam Pressure Velocity Limiter Bypass", providing increased response to changes in the steam pressure set point and improved secondary system control.

EMP-345, "Main Steam Pressure Readout - 184-N", providing the 184-N control room with an indication of main steam header pressure for improved pressure control.

EMP-374, Addendum 1, "Permanent Installation of 24 Channel High Speed Oscilloscope", increasing the number of parameters monitored during reactor transients.

EMP-379, "Main Steam Generator Steam and Feedwater Transmitter Range Change", providing increased range on the steam and feedwater flow instrumentation for Cell 2. (Modification completed on Cell 2 only during this outage.)

EMP-411, "Gross Gamma GM Tube Line Drivers", providing line drivers in GM tube holders in the gamma monitoring turrets. (This EMP is 90% complete.)

EMP-424, "TV Rod Position Indicator Removal from Service", permanently removing the Astrodata TV Rod Position Indicator, which had been out of service for an extended period.

EMP-426, "Flow Monitor Power Supply Mode Indicator Additions", providing for separate mode indicators for the flow monitor discriminator, and the master control bias power supplies, to simplify operation of the Flow Monitor System.

EMP-434, "ERW High Pressure N2 System Upgrading", improving reliability of the high pressure nitrogen system.

EMP-439, "Ball Hopper Manual Locking and Cocking", providing the capability for one operator to complete hopper locking and cocking.

EMP-451, "Gross Gamma 28-Volt DC Failure Alarm", providing a failure alarm on the gross gamma 28-volt DC supply. (The alarm as designed did not function properly so further revision is required.)

DC-3001, "Modify Primary Coolant System Hanger 35".

DC-3013, "105-N Compressed Air Dryer and V-12 Purge Air System Installation", providing clean, dry, low-pressure purge air to both sides of the V-12 valve actuator to prevent entry of water into the cylinder and piston rod packing areas.

DC-3016, "Bracing for N Area Battery Racks", providing backing and anchor bolts to the N Area battery racks sufficient to withstand a design-basis earthquake.

DC-3028, "Horizontal Rod System Fluid Conversion", replacing the hydraulic oil of the left-side horizontal rods with a water-in-oil invert emulsion to minimize the potential for fire. (Further experience may indicate the desirability of converting the full rod system.)

In addition to the foregoing equipment work, a number of Minor Design Changes were completed.

## PROCESS ASSISTANCE AND CONTROL

### Operational Physics

The only direct operational support required during the outage has been to determine conditions under which certain rods could be out of service for maintenance.

A modified inboard rod pattern and withdrawal sequence has been formulated to eliminate the front and rear flux peaking occasionally observed on past start-ups with an inboard rod configuration. This peaking has necessitated about 10 percent decrease in the startup tube power limits which are used until the actual flux distribution is obtained with the traveling wire flux monitoring system. If the modified pattern and sequence eliminate the front and rear peaking, the startup tube power limits will be increased accordingly.

Reactivity coefficients for the Mark IV lattice have been calculated for the anticipated range of fuel exposures, and the operating requirements are similar to those of the Mark I lattice. Total control requirements will be somewhat more restrictive than for the Mark I lattice, in that the lattice flooding event is slightly worse than cold water injection. The power ascension rate is therefore expected to be slightly less.

### Nuclear Protection Systems Study

The CLUMSY code is being utilized to determine the limitations imposed on the nuclear protection systems in conditions for which one or more flux monitoring points has failed. Currently, trip limits believed to be conservative are imposed when part of a system is inoperative. The study is progressing well and preliminary results will be forthcoming in about two months. A natural byproduct of the study could be the establishment of technical bases for the nuclear protection systems. Limit curves can be obtained for the high average system, the individual high trip system and the rate-of-change system. These technical bases will be completed in several months.

[redacted]

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| | 41 | 42 | 43 | 44 | 45 | 46 | 47 | 48 | 49 | 50 | 51 | 52 | 53 | 54 | 55 | 56 | 57 | 58 | 59 | 60 | 61 | 62 | 63 | 64 | 65 | 66 | 67 | 68 | 69 | 70 | 71 | 72 | 73 | 74 | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 34 | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | 34 |
| 33 | | | | | | | C | E | C | E | E | E | E | E | E | E | E | E | E | E | C | C | E | E | E | C | C | C | | | | | | | 33 |
| 32 | | | | | | CE | C | E | E | E | E | E | E | E | E | 76 | E | E | C | E | E | E | 76 | C | E | E | CE | CE | CE | | | | | | 32 |
| 31 | | | | | E | E | E | X | X | X | X | X | X | X | X | X | X | X | X | X | X | X | X | X | X | X | X | E | C | C | | | | | 31 |
| 30 | | | | C | C | E | X | X | X | X | X | X | X | X | X | X | X | X | X | X | X | X | X | X | X | X | X | X | E | C | C | | | | 30 |
| 29 | | | CE | C | E | X | X | X | X | X | G | D | D | E | E | E | E | F | F | F | G | F | F | D | 01 | X | X | X | X | E | C | CE | | | 29 |
| 28 | | CE | CE | E | X | X | X | F | D | E | E | E | E | F | F | G | 07 | D | D | D | F | D | D | D | D | D | X | X | X | X | E | C | CE | | 28 |
| 27 | CE | CE | C | X | X | X | X | 07 | D | D | D | F | F | 07 | E | E | E | F | F | F | F | D | E | 07 | E | 07 | E | X | X | X | X | C | CE | CE | 27 |
| 26 | CE | CE | X | X | X | X | E | E | D | E | E | E | E | D | D | D | F | E | D | G | F | E | D | E | E | 94 | F | 94 | X | X | X | X | C | CE | 26 |
| 25 | CE | E | X | X | X | E | E | D | D | E | E | D | D | E | F | E | 07 | E | D | D | E | E | D | D | D | G | G | E | G | X | X | X | C | CE | 25 |
| 24 | CE | C | X | X | X | 07 | D | 07 | 01 | E | E | 07 | E | 07 | D | D | D | E | G | 07 | D | D | D | 07 | D | 07 | F | D | G | X | X | X | E | C | 24 |
| 23 | CE | E | X | X | X | 96 | 96 | 96 | E | F | E | E | 96 | E | F | E | E | E | 94 | E | D | E | E | E | D | D | E | E | G | X | X | X | E | C | 23 |
| 22 | C | E | X | X | X | E | E | E | E | E | 96 | 96 | F | F | E | E | F | F | F | G | E | E | D | E | F | D | F | E | E | X | X | X | E | C | 22 |
| 21 | CE | E | X | X | X | F | E | 07 | E | E | E | 07 | D | 07 | F | F | G | D | F | 07 | F | G | D | 07 | D | 07 | E | F | E | X | X | X | E | C | 21 |
| 20 | C | E | X | X | X | G | D | F | D | D | D | E | 96 | E | E | F | 96 | F | D | F | E | D | 96 | F | D | D | D | F | F | X | X | X | C | C | 20 |
| 19 | C | E | X | X | X | 96 | 96 | 96 | D | D | D | F | F | E | F | F | F | F | 96 | E | D | 96 | E | 96 | D | G | G | D | D | X | X | X | E | C | 19 |
| 18 | C | E | 94 | X | X | 07 | F | 07 | E | F | E | G | E | F | E | E | E | D | G | 07 | E | G | D | 07 | D | 07 | D | F | D | X | X | X | E | C | 18 |
| 17 | C | CE | X | X | X | F | D | E | E | E | E | D | E | F | F | E | G | F | F | G | E | F | E | D | D | D | E | D | D | X | X | X | E | C | 17 |
| 16 | C | C | X | X | X | D | E | ■ | N | D | D | D | F | F | F | E | 07 | F | F | F | D | D | D | D | E | D | D | E | D | X | X | X | E | C | 16 |
| 15 | E | C | X | X | X | 07 | E | 07 | E | D | D | 07 | E | 07 | E | E | D | D | E | 07 | E | D | E | 07 | D | 07 | E | E | E | X | X | X | E | C | 15 |
| 14 | C | E | 94 | X | X | 01 | E | 01 | E | D | D | D | 01 | 01 | 01 | E | 01 | E | F | 01 | E | 01 | 01 | E | 01 | E | F | 01 | 01 | X | X | X | E | C | 14 |
| 13 | CE | C | X | X | X | D | D | E | E | D | D | D | E | D | E | F | 07 | F | D | D | F | D | E | F | D | 94 | D | F | D | X | X | X | E | C | 13 |
| 12 | CE | E | X | X | X | 07 | D | E | F | D | F | 07 | E | 07 | E | D | E | F | 94 | F | D | D | D | 07 | F | 07 | D | D | E | X | X | X | E | CE | 12 |
| 11 | CE | C | X | X | X | E | F | F | F | F | E | E | F | F | F | F | D | D | D | D | D | D | F | D | D | F | F | D | F | X | X | X | C | CE | 11 |
| 10 | CE | CE | X | X | X | E | F | E | F | F | F | F | F | D | F | F | F | D | D | E | D | D | F | F | D | D | F | D | D | X | X | X | C | CE | 10 |
| 09 | CE | CE | X | X | X | X | E | F | F | E | E | 07 | F | 07 | E | E | D | F | F | 07 | D | D | D | 07 | F | 07 | D | D | X | X | X | X | C | CE | 09 |
| 08 | CE | CE | C | X | X | X | F | F | E | E | E | D | D | E | E | E | F | E | D | D | E | E | E | E | D | F | D | F | X | X | X | C | CE | CE | 08 |
| 07 | | CE | CE | E | X | X | X | X | E | D | E | E | E | D | D | D | 07 | E | F | E | D | D | D | D | D | F | X | X | X | X | E | CE | CE | | 07 |
| 06 | | | CE | CE | E | X | X | X | X | E | D | 07 | D | 07 | E | D | E | D | E | 07 | D | D | D | F | X | X | X | X | X | C | CE | CE | | | 06 |
| 05 | | | | CE | CE | C | X | X | X | X | X | X | X | X | X | X | X | X | X | X | X | X | X | X | X | X | X | X | E | C | CE | | | | 05 |
| 04 | | | | | CE | C | C | X | X | X | X | X | X | X | X | X | X | X | X | X | X | X | X | X | X | X | X | C | CE | CE | | | | | 04 |
| 03 | | | | | | E | 76 | C | E | E | E | E | E | E | E | C | E | C | 76 | E | E | E | E | E | E | E | C | CE | C | | | | | | 03 |
| 02 | | | | | | | CE | CE | E | E | E | C | C | C | E | C | E | ES | ES | CE | C | CE | E | CE | CE | CE | CE | CE | | | | | | | 02 |
| 01 | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | 01 |
| | 41 | 42 | 43 | 44 | 45 | 46 | 47 | 48 | 49 | 50 | 51 | 52 | 53 | 54 | 55 | 56 | 57 | 58 | 59 | 60 | 61 | 62 | 63 | 64 | 65 | 66 | 67 | 68 | 69 | 70 | 71 | 72 | 73 | 74 | |

| Fuel Code | No. Tubes | Description |
|---|---|---|
| C | 69 | Mk-IC (94 Metal - Fringe) |
| CE | 61 | Mk-IC (94 Metal - Fringe) |
| D | 163 | Mk-IC (94 Metal - High U-236) |
| E | 242 | Mk-IC (94 Metal - Fringe & Central) |
| ES | 2 | Mk-IC (94 Metal - Fringe) |
| F | 116 | Mk-IV (94 Metal - High U-236) |
| G | 21 | Mk-IV (94 Metal - Central) |
| N | 1 | Mk-IB (Natural U) |
| X | 241 | Mk-IA (125 - 94 Metal) |
| | 916 | Total |
| | 88 | Total PTs |
| | 1004 | Grand Total |

| PT-NR No. | No. Tubes | Description |
|---|---|---|
| 76 | 4 | Fuel Meltdown Test |
| 94* | 7 | Mk-IV Demonstration |
| 96 | 15 | Mk-IC From Upset-Forged Billets |
| 01 | 12 | Mk-IC From Direct-Cast Billets |
| 01 | 2 | Mk-IV From Direct-Cast Billets |
| 07 | 47 | Initial Full Length Mk-IV Columns |
| ■ | 1 | Graphite Samples Channel |
| | 88 | Total PTs |

*Includes Mk-IV high U-236 content fuel, 1 tube with Mk-IA 125-94 Metal and Mk-IV 94 Metal, and 2 tubes with Mk-IV-AA 125 Metal and Mk-IV 94 Metal.

Loading Pattern - N Reactor



[redacted]

# FUEL AND TARGET FABRICATION - K REACTORS

## PRODUCTION

### General

Production of AlSi-bonded fuel for the K reactors was 107.3 percent of forecast. Seventy-eight percent of these elements had self-supports or bumpers attached. Construction work continued on the hot-die-sizing facilities under project DCE-524.

### Acceptable Elements Produced

| | | Yield - Percent | |
|---|---|---|---|
| | Tons | Current Month | FYTD |
| AlSi-Bonded Fuel | 165.3 | 95.6 | 94.6 |
| Thoria (wafered) | 17.3 | 93.5 | 94.3 |

### Month-End Inventories

| | | Tons |
|---|---|---|
| Bare Uranium Cores | | 744* |
| Finished Fuel: | AlSi-Bonded | 1,303* |
| | Hot-Die-Sized | 39* |
| Thoria Elements | | 45.4 |

*These totals include 132 tons of cores and 138 tons of finished fuel made for the smaller reactors.

## OPERATING EXPERIENCE

Canning line efficiency of the AlSi-bonding lines was 99.0 percent. Downtime was assigned 30 percent to equipment malfunctions and 70 percent to operational causes. AlSi canning operations continued on the basis of two lines on the day shift, five days per week, until August 11 at which time a third line was placed in operation (utilizing personnel made available by the conclusion of thoria canning).

## EQUIPMENT EXPERIENCE

Nothing significant to report.

## PROCESS ASSISTANCE AND CONTROL

### Hot-Die-Sizing Process

Significant results were obtained from end-bond tests using high-power induction generators. Power sources up to 50 KW per station were and are being tested to determine relative heating rates, heat transfer phenomena, and operating techniques essential to rapid heating.

Initial testing involved heating at various rates with existing end bonding equipment. With this system, the heat for bonding is derived from Inconel ram heads which are induction heated to 750 C during the heating cycle. Heating time is controlled to provide the necessary U-Ni-Al interface temperature, after which the power is shut off and heat is "soaked" from the ram heads.

Bonding at 16-50 KW was attempted to produce rapid heating of the interface, but results were negative since the interface heating at 50 KW was essentially no faster than at 16 KW. Although the ram heads are heated more rapidly at 50 KW, the rate of heat transfer from the heads to the fuel was the controlling factor. Faster heating rates could not be realized from larger generators with this system. The heating rate at the Al-Ni-U interface is relatively slow, and source temperature is necessarily low to avoid melting the cladding adjacent to the bonding dies. The temperature differential between the heat source (ram heads) and the interface is about 250 C.

To improve the heat transfer conditions, rams were installed which were electrically and thermally insulated. This permitted direct induction coupling with the bonding dies and increased heat transfer rates substantially. Bonding dies were also modified to provide greater strength at high temperature. Heating rates were higher with the increased power made possible by the insulated rams, indicating that more powerful generators could be of benefit if this concept were used. The small difference in source temperature and interface temperature permitted the use of higher bonding temperature without the risk of eutectic melting. It was also found that heating was more uniform and reproducible with the insulated rams and associated dies.

The combination of faster heating rates and higher bonding temperature made possible with insulated rams holds great promise for shortening bonding cycles. Tests are in progress to determine the effect upon actual bonding speed and quality. Selection of the proper power source will depend largely upon the results of this testing.

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# FUEL AND TARGET FABRICATION - N REACTOR

## PRODUCTION

### Input Production

| | |
|---|---|
| Total billets extruded | 138 |
| Tons extruded | 30.4 |
| Percent of forecast | 57.4 |

### Output Production

| | |
|---|---|
| Total finished fuel assemblies | 808 |
| Tons output | 19.1 |
| Percent of forecast | 86.8 |

### Uranium Utilization - % 59.2

### Month-End Inventories - Tons

| | |
|---|---|
| Bare uranium billets | 191 |
| Finished fuel | 323 |

## OPERATING EXPERIENCE

Input production was below forecast principally because the extrusion rate was intentionally reduced to minimize difficulty with "dimples" encountered in the inner cladding of outer-element tubing. The rate was limited to the number of extrusions which could be cut, copper stripped, and evaluated before production was resumed the next day. By month end, extrusions utilizing the butt-shearing technique were found to be essentially free of dimples, and the rate had been returned to normal.

## EQUIPMENT EXPERIENCE

A heavier lift train was installed on the beta heat-treat furnace conveyor. The hydraulic cylinder of this new lift train has adequate high-speed adjustable cushions on both the rod and heat ends of the cylinder. Installation of the new cylinder permitted the removal of troublesome cam-operated cushion valves from the hydraulic system. The conveyor mounting spindle and bearing systems were also redesigned providing better thrust and axial support. The conveyor system now has adequate power to lift 2400 pounds of fuel.



[redacted]

## PROCESS ASSISTANCE AND CONTROL

### Extrusion Mandrels

Tapering of the working area of extrusion mandrels has proven to be a satisfactory method of realizing a reasonable tool life when extruding materials such as Zircaloy. Evaluation of the tapered extrusion mandrel concept has been in progress for the past several months, and experience has shown a marked increase in mandrel life (upward of 500 percent).

Mandrels of a size used in the extrusion of Zircaloy tube hollows are generally tapered over the working area 0.0005 to 0.00075 inch per inch of length, while larger mandrels for the extrusion of outer clad shells have been manufactured with a taper of as much as 0.0015 inch per inch in length. The taper extends mandrel life by minimizing the contact area with the extruded tube after the extrusion passes operating temperatures, thereby preserving compressive yield strength in the tool during the critical period when the extruded Zircaloy tube may reach temperatures in excess of 1500 F. A further advantage of the taper is that it prevents the mandrel from sticking in the extruded tube at the completion of the extrusion.

The effect of the mandrel taper on the finished product is of little concern unless an extremely precise extrusion is required, as the taper in the extruded tube is generally no more than 0.0003-0.0004 inch per foot of tube length.

## TECHNICAL ACTIVITIES - K REACTORS

### RESEARCH AND DEVELOPMENT

### Basic Production

#### Brittle Fracture Program - K Inlet Piping

#### Technical Bases for K Reactor Tube Power Limits

Work has been completed in the development of basic information for use in deriving bases for tube power limits to be applied for a natural uranium loading. To provide for latitude in implementation of the limits, three basic tube power distributions have been defined, each representing a different degree of power flattening. Additionally, one distribution has been defined for each of the basic fuel types that will be loaded in the central zone and in the enrichment ring.

Flow and pressure transients have been determined for the dual riser accident case, using as a basis initial operation with six pumps and a top-of-riser pressure (TORP) of 405 psi. Work is in progress to develop the voiding characteristics that would correspond to the hydraulic data and each of the defined tube power distributions, under various combinations of reactor operating parameters. Since the time before the reactor loading change may be short, effort will be concentrated first on cases involving high inlet temperatures (above 10 C). Subsequently, the complete range of inlet temperatures (0 to 22 C) will be factored into the calculations. It is expected that preliminary bases for tube power limits for a natural loading will be available in the first week of September.

A sensitivity analysis has been completed to determine the effect on calculated maximum clad surface temperature that would be associated with failure of one vertical safety rod (VSR) to enter the core. The conditions chosen for the analysis were those utilized for the example presented in Supplement 5 of the K Reactor Hazards Summary Report (HW-74095 SUPP5) for a 1960-kw tube in the E-Q loading. Results are shown below:

| Condition | Maximum Calculated Temperature - C |
|---|---|
| Base case - all VSRs enter | 563 |
| Failure of one VSR to enter at least favorable location | 573 |

These results include consideration of the lateral effectiveness of the VSRs across the central zone of the reactor during the post-scram transient. The central portion of the rod pattern has been found to depress the flux to less than the average, with the result that, during a portion of the transient, the flux would exceed the average near the edges of the rod pattern. The calculation in the table was made in the location at which the flux would be peaked a maximum amount above the average.

As noted above, the differences in flux peaking and shutdown transient effects due to a VSR being out of service during a riser loss accident have been calculated. Longitudinal and side-to-side flux distributions following VSR insertion are such that the limiting case is that of the central rod in the outside bank.

The one-dimension kinetics program CLUMSY was run in the side-to-side direction to calculate distribution effects in that dimension. The rod poisoning effect was introduced as a step function at the time the VSRs would normally reach a lower limiting node in the reactor.

Because of voiding starting in the central part of the reactor, the flux distribution would be peaked toward the center prior to rod entry into the given horizontal node. Thus, the excursion in the columns near the outer rod bank up to the time of rod entry would be less than that in the center of the reactor.

The prompt drop and subsequent flux decay in the fringe following rod entry would not be as large as the central zone average reduction, however. Static BUCK code calculations indicate the difference during the initial transient to be less than half that amount. The front-to-rear effect of a central fringe rod out of service would be a small additional difference in local flux of less than 5 percent.

## Fatigue Tests

Two additional single-edge crack tests are in progress. Sample 62 (base metal from KE Reactor A riser) is being fatigued at 33 F as compared to room temperature for all other samples. Sample 106 (base metal from KW Reactor B riser) is being fatigued at one cycle per minute as compared to six to seven cycles per second for all other samples. Following these tests, two additional samples will be fatigued in a tension-tension loading to simulate the effects of mean stresses of 4000 and 10,000 psi.

## Drop Weight Tests

Drop weight tests were performed to measure the nil-ductility temperature (NDT) of KW Reactor B riser parent metal as specified in ASTM E-208. The results indicated that the NDT was between 20 and 30 F, which is further evidence that the piping materials do not meet an NDT plus 60 F criterion. As described in the K Reactor Hazards Summary Report, the integrity of the piping will be demonstrated by a proof test based on a fracture mechanics analysis of the 36-inch piping, since this material has been shown to be most limiting of the piping in the system with respect to brittle behavior.

## Pipe Whip Analyses

The process water coolant at each K reactor is supplied by six high-lift pumps discharging into 24-inch lines which join to form four 36-inch lines. The latter supply the reactor inlet face through four 36-inch risers. The system was designed with the intent that a break occurring in any one of these lines would not endanger the basic safety needs of the coolant system.

An engineering study evaluating the effects of pipe whip following a hypothetical instantaneous "guillotine" type break in the K reactor inlet piping has been completed, with the following results:

- Due to its size and wall thickness, the 36-inch pipe is very rigid. With the break assumed to occur in the longest unsupported section of the pipe, the maximum deflection in any one direction was very small (1/4 inch); the 36-inch pipe will remain in position on its structural supports and will not impact adjacent piping.

- The 24-inch pipes are attached to the structural steel pipe supports with 3/4-inch diameter U-bolts and support saddles. The concern was for the area where these lines tie into the 36-inch pipe. With a break at this point, the U-bolts would probably fail, allowing the pipe to move horizontally, striking an adjacent pipe. The existing pipe support can be modified to provide an adequate stop for this potential movement.

- A break at the elbow of the 24-inch discharge pipe from a high-lift pump would permit the pipe to move through the two-foot clearance and impact the adjacent pipe. The pipe support at this point can be modified to prevent excessive movement.

## Zircaloy Process Tube Hydriding

### Tube Analyses

The evaluation of Zircaloy process tube sections removed from C Reactor after nine years of service is continuing. Tube 1079-C, one of the two tubes of seven sampled showing high hydrogen absorption in initial measurements, was analyzed at approximate 12-inch intervals from the rear Van Stone to 66 inches upstream. The results are as follows:

| Distance from Van Stone, inches | Hydrogen, ppm Total | Hydrogen, ppm Base | Case Layer Thickness, mils |
|---|---|---|---|
| 3/4 | 2513 | 349 | 11.8 |
| 12 | 4710 | 458 | 15.6 |
| 24 | 4039 | 406 | 16.0 |
| 36 | 1979 | 326 | 10.0 |
| 48 | 2289 | 413 | 11.0 |
| 66 | 2500 | 315 | 7.5 |

As reported last month, this is the tube section with the broken Van Stone flange. Examination of the flange indicated that the tube had been leaking, but there had been no leak indication during reactor operation.

These are the highest hydrogen contents observed in process tube examinations to date. The maximum base metal hydrogen absorption observed in the examination of 45 tubes removed from the K reactors is approximately 170 ppm. Currently, the average base metal content of K reactor tubes is estimated to be 150 ppm.

Electrochemical Dehydriding

BNW has, through research sponsored by DUN, developed an electrochemical process to selectively remove zirconium hydride from Zircaloy process tubes. The process as demonstrated on a laboratory scale is essentially self-limiting and self-healing. Laboratory data indicate that the electrochemical process completely cleans the zirconium hydride case material from the tube base metal, and builds a protective film of zirconium oxide on the cleaned surface.

Pilot plant-scale tests have been initiated to develop and test the electrochemical process and equipment prior to on-reactor application. Prototype equipment has been assembled and two test runs with it have been completed. Initial observations indicate that the hydride case layer from the tube samples was successfully removed, but final evaluation will await detailed examination of samples.

Through-Reactor Decontamination

PTA-158 was approved permitting on-reactor tests to investigate several parameters affecting a through-reactor decontamination. Initial trials will consist of injecting preheated decontamination chemicals into single process tubes to determine the most efficient decontamination solution, concentration, temperature and time for conducting a through-reactor decontamination. Subsequent tests on a full-crossheader scale will determine flow control and distribution problems, the feasibility of employing reactor decay heat to heat the solutions, and the amount of radioactivity released (for waste management studies). The initial tests are scheduled for September.

Exposure to decontamination solutions of carbon steel piping containing a heavy buildup of corrosion product tuberculation simulating conditions in the K reactor front face piping indicated that removed tubercles could pose a serious problem during an actual reactor decontamination. The piping samples, including a section of tuberculed K reactor riser piping, were subjected to room temperature Turco 4306-C at 6 ounces per gallon for 15 minutes followed by a 1-week flush with process water. The procedure was repeated four times. For about one hour after each decontamination solution treatment, large amounts of tubercles were released and collected on an 8-mesh front crossheader screen. This test indicates that, to prevent clogging of front face screens, the front face piping must be cleaned before exposure to decontamination solutions.

Low-Dichromate Evaluation

The 15 downstream K5N fuel elements from tube 2067-KE, discharged August 1 after 934 MWD/T exposure, were visually examined for determination of their corrosion behavior in process water containing 0.5 ppm sodium dichromate. This tube was operating with one of the highest outlet temperatures (119 C) in the reactor and was selected because of the severity of the environment.

Fuel in positions 3 and 4 from the downstream end exhibited 5 to 10 mil deep erosion-corrosion (groove corrosion) on the lateral surfaces between the bottom rows of supports (hottest surface temperature area) while the remaining fuels exhibited little corrosion at this area. All fuels exhibited a small area of erosion-corrosion at the upstream ends of the bottom supports. The depth of this attack varied from 1 to 2 mils on elements 7, 8, and 10 through 15; 5 to 10 mils on elements 1, 2, 3, 5, 6, and 9; and 10 to 20 mils on element 4.

The corrosion attack in this KE tube is greater than has been observed on fuel elements discharged from KW Reactor after exposure in 1.0 ppm dichromate water. However, the difference could be accounted for on the basis of temperature. A direct comparison of fuel behavior in 0.5 and 1.0 ppm dichromate water will be obtained in September, following the first discharge of fuel for examination in the half-plant comparison test begun in June at KW Reactor. The charging patterns of the monitor columns of this test have been adjusted to obtain similar coolant outlet temperatures.

Computer Code Development

A change in storage requirements of the HAMMER code was accomplished which will circumvent the necessity of its being reprogrammed to be compatible with Computer Sciences Corp. system changes which will reduce available core space. Elimination of upscatter cross section determinations for high energy neutrons released the extra space without significant reduction in accuracy of calculational results. A second small change in the HAMMER program was the addition of an option for creation of a library of smeared lattices and their selection, thus simplifying the card deck requirements and program execution. Additional HAMMER studies established the size of the uncertainty band which might be associated with mesh point grid selection as ±1.5 mk for graphite lattices and ±2.5 mk for water lattices.

Hot-Die-Sized Fuel Irradiation (PTA-011)

The K5AEA hot-die-sized fuel element which failed in KE Reactor on August 12 was charged on March 27 and had an exposure of 1003 MWD/T. Groove corrosion caused penetration of the cladding at the upstream end of a downstream support, due apparently to a combination of temperature and coolant flow turbulence.

Product Flexibility

Pu-238

Neptunium Irradiation (PTA-163)

Investigation continues into the cause of failure last month of the graphite matrix neptunium elements in the two KW core block columns after 26 full-power days exposure (with an estimated 2 percent conversion of neptunium to Pu-238).

Radiometallurgical examination of four neptunium oxide-graphite wafer elements (two aluminum-clad failed elements, one Zircaloy-clad failed element and one Zircaloy-clad nonfailed element) has shown that dimension instability of the wafers was responsible for clad failure with subsequent end cap separation. Average growth characteristics of the examined elements are shown below:

| | Wafer Diameter, Inches | | | |
|---|---|---|---|---|
| | Original | Final | Difference | Percent Increase |
| Aluminum clad (failure) | 1.498 | 1.513 | 0.015 | 1.0 |
| Zircaloy clad (failure) | 1.427 | 1.438 | 0.011 | 0.8 |
| Zircaloy clad (nonfailure) | 1.427 | 1.438 | 0.011 | 0.8 |

| | Wafer Stack Length, Inches | | | |
|---|---|---|---|---|
| | Original | Final | Difference | Percent Increase |
| Aluminum clad (failure) | 8.450 | 8.811 | 0.361 | 4.3 |
| Zircaloy clad (failure) | 8.406 | 8.506 | 0.100 | 1.2 |
| Zircaloy clad (nonfailure) | 8.406 | 8.553 | 0.147 | 1.8 |

Growth of the graphite matrix from irradiation damage will not satisfactorily explain these substantial dimension changes, as the expected irradiation-induced growth for the operating conditions associated with this irradiation would be less than 0.05 percent.

A graphite-neptunium wafer from a central element of each of the two core block columns has been transferred to BNW Radiochemistry for analysis. Extended alpha counting will be utilized to determine the Pu-236 contaminant level in the Pu-238, the primary objective of the test. Three samples are being analyzed from each of the two wafers.

In addition, investigation of fabrication process parameters is in progress to try to identify the mechanism responsible for this dimensional instability.

U-236 and Np-237 Buildup

A major portion of the computer program comparisons of buildup and burnout predictions for U-236 and Np-237 has been completed, and documentation of the compiled experimental results has been initiated. In addition to analyzing experimental results, the study has been informative in comparing the effectiveness of different computer programs in accounting for refined cross section and neutron energy spectrum effects.

Plutonium Burning Irradiation (PTA-150)

Tube powers for the PuAl columns under irradiation in KE Reactor have recently been running about 15 percent below the intended design level. Adjustments in the surrounding buffer zone will be made during the September outage to increase local power, and to facilitate the transition from E-Q in the surrounding block to primarily natural uranium in about two months. A MOFDA calculation for the exposure which will be reached in September indicates that the Pu-240 content will have exceeded 25 percent (the maximum attainable is about 60 percent).

## Oralloy Utilization

### Production Test Planning

Action was initiated on oralloy fuel development and irradiation planning. To capitalize on the incentives for the potential economic and product flexibility advantages of oralloy use, plans were made to demonstrate that the fabrication, irradiation and nuclear safety technology would be on an immediate production status. This can be achieved through routine use of oralloy as the enrichment ring material in the K reactors.

Conclusions reached in this planning were that a preliminary production test (10 to 20 columns of oralloy fuel) should be scheduled for loading in a K reactor by February 1970, and that concurrent efforts should proceed to obtain the necessary safety and procurement bases and approvals by July 1969 for the charging of a full E-ring with oralloy fuel by January 1971. Initial technical work has been concerned primarily with heat transfer and fuel stability considerations relative to a riser loss incident, both for solid and I&E elements.

### K Reactor Overbore

Documentation of the physics technology developed for the K reactor overbore study has been completed with the issuance of DUN-6059, "K Reactor Overboring - Reactor Physics Aspects."

### Lithium-Bearing Splines

Reactivity tests were made in the 305 Test Reactor on the lithium-bearing spline samples procured for evaluation. The results showed the test splines were about 50 percent as black as standard boron splines. In addition, corrosion tests were initiated in the 1706-KE laboratory. After two weeks in boiling water, only the spline with the dispersed LiMg alloy composition showed any reaction. At the ends of the sample where the LiMg is exposed, there were indications of swelling. The splines containing LiF and $Li_2CO_3$ show no sign of change in the corrosion tests to date.

## Environmental and Regulatory Technology

### Technical Bases Development for K Reactor RTDs.

Sensitivity calculations were completed using the CLUMSY code on the effect of varying immersion RTD time constants. These effects are being interpreted in terms of potential tradeoffs in either trip limit or rod withdrawal limitations.

The variable time constant of strap-on RTDs has been approximated in the past by an empirically determined double time constant. A study has been initiated to develop an analytical basis for their evaluation.

High Turbidity Coolant

A 10 to 20 percent reduction in effluent activity, similar to that experienced during the winter months, has been found after reaching equilibrium conditions in the half-plant high turbidity test at KE Reactor (0.3 JTU compared to a normal turbidity of 0.05 to 0.1 JTU).

Waste Management

Nothing to report.

ENGINEERING AND TECHNOLOGY - REACTORS

K Reactor Heat Decay Test

Under PTA-172 (DUN-5564, "K Reactor Full Flow Heat Test"), a heat decay test was performed August 7 at KW Reactor. Temperature transients were recorded from six thermocouple trains, two of which were located at the top, center, and bottom of the core, respectively. When the data become available, they will be analyzed to determine the post-accident power generation transient.

Personnel Dosimetry Development

The personnel dosimetry development program has been directed toward the procurement and/or development of radiation protection instrumentation which will provide improved exposure control of personnel and allow a faster response by personnel interpreting radiation intensity. Although there is considerable fixed area monitoring instrumentation, it was determined that radiation exposure to personnel could be more accurately controlled by the use of a miniature pocket dose rate alarm dosimeter and a pocket dose alarm dosimeter.

The market survey resulted in numerous commercially available instruments being evaluated for this application, but all were found to be lacking in the areas of reliability, presettable alarm points, energy dependence, operating life and alarm intensity. A development program therefore was initiated which has produced a selectable level pocket dose rate alarm dosimeter (PARD) which to date has met or exceeded the required performance criteria. Brookhaven National Laboratory and Battelle-Northwest are evaluating the performance of these PARD devices for applications throughout the AEC complex, with preliminary reports from these evaluations being favorable. Additionally, 20 of the PARD devices are being given field tests in DUN operating environments and have proved satisfactory to date.

A second device under development is a pocket dose alarm dosimeter (PADI). The development of the PADI has passed the layout stage and is currently being packaged for field evaluation. This unit has unusual capabilities for a device of this type in that it can be set to alarm covering up to six preselected doses, read-out the accumulated dose on an auxiliary monitor, and retain the accumulated dose information for periods up to 30 days.

Project Engineering

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

ENGINEERING AND TECHNOLOGY - FUELS & TARGETS

Long-Fuels Program

In June, 2500 of the 10.75 inches long Model K11E cores were ordered for delivery in August. In order to evaluate throughput capabilities and assess costs at National Lead of Ohio (NLO) for this long core model, AEC-CAO issued a Production Order for 8,000 cores so NLO could process at least four full shifts on their cross-transfermatic machine. Preliminary data received from NLO indicates that warp has not been a major problem as anticipated, and future orders will not require any more machining overstock than is normally used for the K5AEA universal core model. The major reject category reported so far has been OD to ID concentricity which ran from 5-6% on the first portion of the campaign but was reduced through machine adjustments to 1.9% for the latter part of the campaign.

NLO shipped the 2500 cores requested on August 22, and delivery at Hanford is expected in the first week of September. This will permit scheduled hot-die-sizing in mid-September so that charging, under a reactor Production Test, may be accomplished in mid-October.

The second portion of the long fuels program involves a number of K11E cores produced onsite via bare core extrusion on the 333 Building extrusion press. Current plans indicate that approximately 200 of these extruded long cores will be irradiated in the same reactor Production Test authorized for the long cores to be received from NLO.

## TECHNICAL ACTIVITIES - N REACTOR

### RESEARCH AND DEVELOPMENT

#### Basic Production

##### Unbonded End Closure Development

Thermal cycling tests (beta-heat treatment) of unbonded outer end closures have resulted in circumferential cracking of a significant number of the specimens. The brittle weld character was also evidenced by the impact tests where circumferential cracks propagated in the weld zone. Autoclave cycling tests, as yet, have failed to show detrimental properties. The mechanism of failure is believed to be the strain cycling or fatigue of the weld structure, which is promoted by the presence of large alpha grains introduced during thermal cycling to beta-heat treat temperatures.

Since oxygen and nitrogen substantially harden and embrittle welded Zircaloy, steps are currently being taken to improve the inert atmosphere which surrounds the end closure during welding. Additional fuels will be fabricated for impact tests to insure that the improved technique is capable of producing the quality weld desired. The beta-heat cycling tests will be discontinued because the weld microstructure is substantially modified and therefore this test does not represent in-reactor conditions.

##### Zircaloy Process Tube Surveillance

The examination of Zircaloy-2 process tube 0758, removed from N Reactor in February at an average neutron exposure of $1.45 \times 10^{21}$ n/cm$^2$, is continuing. Tube sections from the rolled joint areas in contact with the carbon steel inlet and outlet nozzles were sampled for hydrogen analyses.

At the inlet nozzle, hydrogen absorption in samples from the OD Zircaloy-2 surface in contact with the carbon steel wall was 31 and 33 ppm, with a value from the entire Zircaloy-2 wall cross section of 38 ppm. At the outlet nozzle, the corresponding hydride concentrations were 37 and 39 ppm on specimens in contact with carbon steel, and 42 ppm across the entire wall cross section. The values fall within the range of hydride concentrations reported previously for upstream and downstream out-of-flux locations and archive samples.

Six specimens for hydrogen analysis were taken from the vicinity of a tube butt weld located approximately 550 inches from the tube inlet, downstream from the end of the fuel charge. The hydride concentrations within the weld area were 26 and 31 ppm. Outside the weld area, the hydride concentrations were 48 and 49 ppm on one side, and 46 and 60 ppm on the opposite side. The sampling points outside the weld were located approximately 1 inch from the weld center line. The low hydride concentrations within the weld suggest that hydrogen migrated from the maximum weld temperature zone to cooler locations outside the weld area, thus contributing to the higher hydride concentrations there.

Zircaloy-2 Corrosion Studies

Information on in-reactor corrosion rates of Zircaloy-2 with small amounts of surface acid staining is necessary before permitting any large-scale, in-reactor exposure of N fuel elements with slight acid staining. Out-of-reactor measurements indicate acid staining causes a slight amount of accelerated corrosion (100 to 200 mg/dm$^2$) after which rates return to normal. Arrangements are being made to irradiate acid-stained coupons in the ATR at Idaho Falls.

In addition to the ATR tests, authorization to expose acid-stained coupons in N Reactor will be included in PT-NR-118, currently being written. Flow testing of reactor coupon holders showed there would be little flow disturbance by inserting the holders in a Mark IV fuel column.

Primary Piping Weld Inspection

Ultrasonic inspection of the primary coolant system piping from Cells 4 and 5 showed the piping welds examined to be free of significant flaws.

Mark IV High Exposure Testing

Lattice calculations show that each column of Mark IV irradiated to 7500 MWD/T exposure would require the reactivity support of two Mark I-A columns.

N Reactor Base Enrichment Proposal

The proposal to increase the base enrichment level of N Reactor from 0.947 to 0.970 percent has been documented as DUN-6176. The document concludes that the gain in operating flexibility with the higher base enrichment level can be achieved safely without additional cost.

Fuel Improvement

Under Production Test 94 SUP4, two columns of 125 Metal Mark IV-AA (spike) columns were charged to determine thermal hydraulic performance relative to that of full-length Mark IV fuel columns. Thermocouple train data have indicated that the heat and flow splits to the subchannels of these spike columns are not significantly different than for the base fuel. In fact, the maximum imbalance in subchannels of the spike fuel was approximately 2 percent less (8 percent for Mark IV-AA versus 10 percent for Mark IV). Thus, the Mark IV-AA spike appears to be compatible with a Mark IV base loading.

Process Surveillance Systems Study

The base case model for the surveillance system fault tree is being analyzed to develop parametric curves of system reliability and availability as functions of mission time. Also, estimates of the system prime electrical power requirements have been completed.

Primary Coolant Loop Technical Criteria and Bases

DUN-5954, "Technical Criteria and Bases for the Primary Coolant Loop High Pressure Injection System - N Reactor", was issued. Criteria for safety, capacity, and reliability are emphasized, including allowable response of the pressurizer level as a function of primary coolant loop outflow. Failure analysis of the system has been initiated to determine the reliability of the system relative to that specified in the safety and reliability criteria.

Product Flexibility

Nothing to report.

Environmental and Regulatory Technology

Metal-Water Reaction and Fission Product Release Studies

Two irradiated Mark I inner fuel element pieces were tested to determine the metal-water reaction rate under conditions simulating those imposed by the geometry restriction of the outer fuel piece and the adjacent elements at each end. A quartz tube was used to simulate the outer fuel piece. Both fuel pieces were heated to rupture temperature in approximately three minutes, and then held at 1070 C for 57 minutes. Each one ruptured initially near its longitudinal center.

In the first of these tests, SNH-7, the initial rupture occurred at 1075 C, and an estimated 15 percent of the uranium was extruded from the cladding. (Prior tests indicated extrusion should be expected at the test fuel exposures involved.) The uranium filled the 1/4-inch annulus between the fuel piece and the quartz tube for a length of about 2-1/2 inches, and filled the bottom third of the annulus for the full length. Two additional pressure-induced clad ruptures occurred after the initial rupture was apparently sealed. These ruptures were not accompanied by further uranium extrusion, indicating probable bond separation prior to rupture. The cladding was distended about 3/8-inch in diameter at some points. The uranium oxidation was slight compared to that in previous tests without the geometrical restrictions. No oxidation was visible on the uranium surfaces in contact with the quartz or cladding.

In the second test, SNH-8, the initial rupture occurred at 1065 C and an estimated 25 to 30 percent of the uranium was extruded out of the cladding. This amount of uranium completely filled the 1/4-inch annulus for about three inches in length, and filled the bottom two-thirds of the annulus for the full length. Two additional pressure ruptures also occurred in this test, but the amount of exposed uranium in the secondary ruptures was about half that of the SNH-7 test fuel piece. It appears the difference may have been due to the smaller amount of uranium left in the cladding after the initial rupture.

Preliminary analysis of the metal-water reaction data indicates: (1) the significantly larger amount of uranium extruded from the cladding in the

SNH-8 test, over that in the SNH-7 test, produced only a small increase in the exposed uranium surface area, and (2) the secondary cladding ruptures probably occurred about 17 minutes after the heating cycle started when an increase in hydrogen evolution rate occurred. The more linear release rate curve during the remaining 40 minutes suggests a continuing increase in exposed uranium surface area.

### Waste Management

Nothing to report.

## ENGINEERING AND TECHNOLOGY - REACTOR

### Short-Term Reduction of N Reactor Coolant pH

PT-NR-119, which was intended to determine the effect of a short-term reduction of primary coolant pH on activity distribution around the N Reactor primary system, was run during the 12 hours prior to the N Reactor shutdown on July 20. No significant change in activity in the system was observed. The time of the test was shortened from 24 hours to 12 hours to avoid large crud bursts.

The test began with a primary coolant pH of 9.9. The pH was dropped to 7.9 during the 12-hour test period. The concentration of radioisotopes of concern in the coolant increased during the test period, but not enough to give a significant removal by feed-and-bleed.

Front and rear elevator activity was not reduced by the test. Radiation measurements in the steam generator cells were unaffected except for Cell 1, where the area radiation readings dropped from 13 mR/hr to 7 mR/hr, and Cell 5 where the radiation readings increased from 6 mR/hr to 12 mR/hr. Radiation readings in the pipe gallery were unchanged.

The test showed that pH reduction prior to shutdown may remove some activity, but a much longer time than 12 hours is needed to accomplish significant activity removal.

### In-Reactor Tube Examinations

Ten borescopic examinations were performed on seven process tubes, with three of the tubes examined both before and after the through-reactor decontamination conducted on July 24. The tubes showed normal tube wear (downstream spacer fret marks) and contained the expected minor scratches due to charge-discharge operation. The decontamination did not appear to completely remove the surface films, nor was there any increase in nozzle corrosion as a result of decontamination. Replicas of the tube-to-nozzle joints were made before and after decontamination to further evaluate the effect of the process on corrosion at the Zircaloy-carbon steel interface. Results of this study are not yet available.

Emergency Cooling Backup during Modification of the Emergency Raw Water System

During phases of the effluent control program which involve removal of the high-lift diesel pumps from service, an earthquake-resistant backup to the emergency cooling system is required to provide for the contingency of simultaneous loss of A and B buses. Calculations determined that, after seven days in a shut-down condition, any of the following combinations of fog spray diesel pumps and low-lift diesel pumps would supply adequate cooling to the reactor through a high pressure flush line connected to the bottom of the inlet risers:

- Two low-lift pumps and two fog spray pumps
- One low-lift pump and two fog spray pumps
- Two low-lift pumps and one fog spray pump

Additionally, it was ascertained that the backup system would be capable of reestablishing liquid flow in the process channels in the event of cooling interruption sufficiently long to initiate boiling.

Additional analysis was made to determine the potential effectiveness of the graphite cooling system as a sole heat sink seven days after shutdown. Results indicated that fuel located within the graphite cooling system lattice would not reach melting or failure temperatures. Fuel at pre-shutdown specific powers of 100 kw/ft and 270 kw/ft would not exceed 750 F and 1150 F, respectively. However, two fuel elements at each end of a full-length column (34.8 feet) of Mark I or Mark IV fuel would lie beyond the graphite cooling system lattice and therefore would not receive adequate cooling. It was estimated that, with no cooling available through the process tubes, the end pieces would reach or exceed the melting temperature of 1994 F. However, the heatup rate would be extremely slow and the melting temperature would not be reached until approximately 30 hours after loss of cooling. Thus it would appear that, under the shutdown conditions, cooling could be restored prior to release of fission products.

Results of the foregoing studies have been reported in documents DUN-6077, "Primary Coolant Backup during ERWS Tank Modification", and DUN-5666, "Temperature Transients and Degree of Melting for Mark I and Mark IV Reactor Loadings".

Project Engineering

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

## ENGINEERING AND TECHNOLOGY - FUELS & TARGETS

Copper Recovery from Waste Effluents

Samples of copper stripping solution have been obtained for use in the copper removal studies directed toward establishing the feasibility of chemically plating copper out of solution. By use of demonstrated processes, copper can be chemically plated out of copper sulfate solutions by passing the solution over iron. Normally 1.5 to 2.5 pounds of iron are consumed per pound of copper precipitated, with recovery efficiency approaching 93%.

Current experiments have included attempts to precipitate copper out of nitric acid solutions onto iron. The pH of copper stripping solutions was adjusted from 0.5 to 2.5 by making additions of ammonium hydroxide. With the pH adjusted to within this range, chemical plating was possible. Approximately 1.7 pounds of iron was consumed per pound of copper precipitated. Thus far in the tests only 22% of the copper was reclaimed from solution. Additional testing is planned to determine methods for improving the efficiency of this process. The precipitation method does appear promising and is attractive because of low cost.

Electrolytic Copper Stripping

A process is being developed which is aimed at removing copper-silicon lubricants from Zircaloy surfaces by electrolysis. Early work is being conducted in an acidified copper sulfate electrolyte, using a DC welder as a power source.

The system under study is being used to strip copper-silicon from extruded Zircaloy components. The extruded tube is positioned vertically in the electrolyte and acts as the anode. A copper cathode is inserted into the tubular anode and another surrounds the OD. A current density of 50 amps per square foot has been satisfactory over the approximate six square feet of anode surface. The process proceeds at about 4 volts until the stripping operation is completed, at which time a rapid rise to approximately 10 volts occurs. The plated copper is in a finely divided or cement form, making removal from the cathode simple. The cement copper can be washed from the cathode in water and separated by sedimentation and decantation.

The major problem facing the process has been the possible buildup of silicic acid in the electrolyte which may reduce the life of the solution, and defeat the objective of lowering the volume of effluents sent to the settling pond. Results to date show the solutions to be longer-lived than the nitric strip solutions now employed. One batch of electrolyte has been used to liberate 1.1 pounds of copper per gallon compared to standard shop practice of 0.6 pounds of copper per gallon of nitric solution. Further use of this same electrolyte is continuing to determine its ultimate life. Work is now being directed toward characterizing the process in terms of solution composition, current density, and power efficiency.

[redacted]



## IRRADIATION SERVICES

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WITH DELETIONS

### ROUTINE IRRADIATIONS

One hundred fifty-five Quickie activation analysis capsules were irradiated in the K reactors for Battelle-Northwest (BNW).

Two BNW cooled tensile specimens were removed from the Snout facility spider at KW Reactor.



[redacted]

## ADMINISTRATION - GENERAL

### REACTOR STANDARDS DEVELOPMENT FOR AEC-DRDT

Effective August 1, a Reactor Standards Development Subsection was established in the Manufacturing Engineering Section, Operations Division. This Subsection will develop Maintenance and Testing Standards for the AEC's Division of Reactor Development and Technology. A $100,000 "purchase order" was received from Oak Ridge National Laboratory, which has overall responsibility for the program, to fund this work through the fiscal year.

This standards program, which will cover the design, construction, and maintenance of DRDT's sponsored water-cooled reactors, is intended to insure reactor safety and operational reliability by assuring a high degree of integrity in critical systems. The Company's responsibility will be to develop ground rules or criteria on maintenance and operational testing from which detailed standards can be developed for specific reactors. Although these standards are being designed specifically for AEC experimental and prototypical water-cooled reactors, it is expected that most of them can be applied to liquid-metal-cooled reactors.

### POLLUTION CONTROL COSTS

A request to supply a list of Douglas United Nuclear activities and related costs of pollution control efforts during FY 1969 was received from AEC-RL on July 29. This information was required for the congressional committee studying Federal Agency Pollution Control Programs. The costs provided were those associated with waste treatment, effluent sampling, burial ground disposal and surveillance, and gaseous releases to the atmosphere.

### APPROVAL LETTERS

At the end of August, final AEC-RL action was pending for the following requests:

| ATD Number | Subject | Date of Transmittal to AEC-RL |
|---|---|---|
| AP-39 | Pension Plan portion of letter entitled "Pension Plan, Salaried Savings Plan and Wage Savings Plan" | January 12, 1966 |
| ATD-175 Add. #1 | Overhead Charges Applicable to Diversification Work to be Used in Determining Credits to the AEC | July 30, 1969 |
| ATD-180 | Lay Off for Lack of Work Appendix "B" Modification MO-23 | February 24, 1969 |



[redacted]

YEAR-END ACCOUNTING AND REPORTING

All fiscal year-end financial statements and reports were submitted to the AEC in accordance with the due dates set forth in their letter to all contractors.

AUDIT SCHEDULE - FY 1970

A schedule of audits to be conducted during FY 1970 was prepared and reviewed with the AEC-RL Audit Branch. This schedule consists of twenty-six audits, including two which the AEC Controller's office requested all major integrated AEC contractors to conduct.

RECORDS MANAGEMENT

All General Electric Company records left on loan with Douglas United Nuclear on the takeover dates of November 1, 1965 and July 1, 1967 have been cleared, and GE stated officially on July 30, 1969, that the DUN obligations relative to the return of these records have been discharged.

EMPLOYMENT & PLACEMENT

Fuels Section Layoff

Principally because of completion of the thoria canning program, 12 Metal Handlers in the Fuels Section were given layoff notices early in the month. Eleven of the 12 employees were placed in other jobs, either with the Company or with other Hanford contractors. The remaining employee elected not to be placed on the project.

Employment Summary

DUN personnel totals and employee allocations as of July 31 and August 31 are shown in Appendix B.

SAFETY

No personnel radiation exposures exceeded operational control.

Month-end safety statistics were:

| | |
|---|---|
| Disabling injuries - August | 1 (see below) |
| - Year to date | 1 |
| Days since last disabling injury | 25 |
| Man-hours since last disabling injury | 219,000 |

The disabling injury occurred to a pipefitter in 100-B Area on August 6. A maintenance crew had been working for several days disconnecting water lines to various service buildings as part of the area deactivation program. On the morning of the accident two pipefitters and a welder were sent to disconnect and cap-off a three-inch fire and sanitary water line to one of the buildings. This crew had removed a four-foot section of the line between the isolation valve and the building. As the pipefitter was standing in front of the valve, aiding with caulking of the stub line and cap, the valve blew off striking him on the left leg and fracturing the fibula bone. Investigation revealed a breakdown in communications between the water plant and the maintenance supervisors, resulting in the craft personnel starting work on the line before it had been depressurized.

[redacted]

APPENDIX A

## PROJECT STATUS SUMMARY - REACTOR FACILITIES

| Number & Title | Authorized Funds - $ | Percent Complete Design | Percent Complete Construction | Status |
|---|---|---|---|---|
| **Single-Pass Reactors** | | | | |
| DCE-505, Boiler Control Improvements - 165-KE & KW | 410,000 | 100 | 85 | No construction activity this report period due to need for craftsmen on higher priority work at 100-N. |
| DAP-510, Discharge Chute Clearing Equipment - K Reactors | 220,000 | 100 | 92* | Installation of the remaining conveyor units has been delayed until basic equipment problems are resolved. *Reflects a 4% reduction from the July Report on the basis of a revised AEC-RL construction schedule. |
| DAP-516, Storage Building Addition, 105-KE & KW | 142,000 | 80 | 0 | Comments have been completed on all but the electrical drawings. |
| DAE-518, Effluent Radio-iodine Monitor - KE & KW Reactors | 100,000 | 100 | 23 | Plant forces continued fabrication of mechanical sampling equipment and electronic instruments. J. A. Jones Co. started field work August 20. |
| DCP-522, Modification of Reactor Coolant Crosstie Piping - 105-KE & KW | 163,000 | 100 | 100 | Completed with exceptions August 20. |

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PROJECT STATUS SUMMARY - REACTOR FACILITIES (contd.)

| Number & Title | Authorized Funds - $ | Percent Complete Design | Percent Complete Construction | Status |
|---|---|---|---|---|
| DAP-526, Deactivation of Hanford Production Reactor C. | 38,000 | 0 | 0 | AEC-RL issued Directive No. AEC-324 August 6 as an interim authorization for design and critical procurement. Purchase Requisition work sheets have been prepared by Vitro/HES for procurement of the electric heaters. |
| N Reactor | | | | |
| GCP-406, Improved Safety Platforms and Accesses - 100-N Area | 300,000 | 100 | 100 | Balance of reactor and Cell 2 platforms installed during August outage. Revised project proposal sent to AEC-RL on July 21 requests authorization of additional work to be accomplished with remaining funds. |
| GCE-408, W, C & D Elevator Safety Improvements - 105-N | 90,000 | 100 | 85 | Beneficial use of C and D elevators was obtained August 13. |
| GCP-411, Effluent Control Program - 100-N Area | 1,830,000 | 100 | 33 | Section I tie-ins completed August 18. Section II 182-N header installation started July 28 was completed August 20. Diesel tanks were set August 13. Section V, Emergency Dump Tank, review and approval of American Pipe's design information continues. Excavation in progress for the foundation. |
| DCE-519, Replacement of Bridge Crane and Hoist System with New Crane System - 105-N Storage Basin | 269,000 | 55 | 0 | Design was resumed August 15 to provide procurement specifications to be used as a basis for firm estimates for funding review. |

# APPENDIX B

## EMPLOYMENT SUMMARY
(with employee allocations)

| | 7/31/69 | 8/31/69 |
|---|---|---|
| CONTRACT PERSONNEL | | |
| 02 Programs | | |
| Douglas United Nuclear - | 1683 | 1655 |
| Assisting other Contractors | 14 | 14 |
| Total - 02 | 1697 | 1669 |
| Other Programs under AEC Contract | | |
| Assisting Other Contractors and WPPSS | 41 | 44 |
| Special Irradiations | 5 | 5 |
| Total - Other Programs | 46 | 49 |
| Total Contract Personnel | 1743 | 1718 |
| COMMERCIAL ACTIVITIES PERSONNEL | 18 | 16 |
| TOTAL FORCE | 1761 | 1734 |

FEATURE REPORT

## TRANSURANIUM TARGET ELEMENT FABRICATION

### INTRODUCTION

Present in all discharged reactor fuels are many potentially exploitable materials of interest in a wide spectrum of industrial and medical research applications. The recovery and processing of these materials can lead to practical uses in fields where their unique properties are capable of filling new demands, or in fields of competition with more conventional approaches. It is becoming increasingly evident that many future programs, notably those in medicine and space, will require both fission products and other heavy isotopes for use as primary materials in the achievement of program goals.

The Transuranium Pilot Plant operated by Douglas United Nuclear in the 300 Area provides Hanford with the capability to demonstrate the fabrication of target elements containing these highly radiotoxic reactor materials. This summary report describes this Pilot Plant and the target fabrication work performed therein to date. It also notes briefly the further work planned for this facility and the studies of larger-scale target element fabrication which are in progress.

### PILOT PLANT FACILITIES

#### Building Description

The general floor plan of the Transuranium Pilot Plant is shown on Figure 1, appended. The building (3708) is a concrete block structure with access via an airlock. To prevent contamination escape, the ventilation system maintains a progressively more negative pressure within the facility, with air drawn from the outside through the office and locker room to the work area (potentially contaminated zone), and finally exhausted back to the atmosphere after passing through one of two banks of absolute filters installed in parallel. Air at the exhaust stack, as well as air in the work area, is continuously monitored for airborne contamination and to ensure that adequate air flow through the facility is maintained.

Two independent criticality alarm monitoring systems are situated in the work area to detect the presence of any high-level radiation. Fire detection equipment is connected to the 300 Area central fire alarm system. To minimize the criticality hazard and the spread of contamination, there is no sprinkler system in the building.

#### Hood and Glovebox Equipment

The facilities for casting metallic cores are housed in a glovebox designed specifically for the melting, alloying, and casting of transuranium target materials. This hood contains a 25-pound aluminum capacity induction-heated tilt-pour furnace and two resistance-heated pot furnaces. Ceramic target forming is done in a glovebox which contains a 60-ton double-acting hydraulic press, a vacuum furnace, a ball mill, a balance, and a drying oven. To prevent enclosure failure due to excessive temperatures, both hoods are equipped with refrigeration units.

The target finishing facilities consist of a machining hood with a lathe, a glovebox housing the tungsten-inert gas welder, and an open-face hood suitable for the handling of material with little potential for gross spreading of radioactivity.

Other equipment includes a weld inspection booth, a helium leak detector to measure weld closure integrity, and sizing equipment to compress the cladding tightly onto the core for increased heat transfer. Proper tooling and gauging are of course provided as needed.

Since highly radioactive materials are handled in these facilities, the processing systems are adequately shielded to prevent excessive personnel exposure to radiation.

## OPERATIONAL CONTROLS & QUALITY ASSURANCE

Administrative and physical controls are imposed on all work performed in the Transuranium Pilot Plant to protect personnel against radiation, contamination, and criticality hazards. Administrative controls are outlined in various manuals and work procedures, and individuals are required to learn these procedures and to comply with them when working in the facility. The physical controls are imposed to prevent the mishandling of any fissile or radioactive materials during processing. Prior to being assigned in the Pilot Plant, each individual receives training to orient him with the applicable controls and operating procedures. Industrial safety standards are adhered to where applicable.

Quality of the finished elements is assured by strict adherence to the DUN Quality Assurance Plan in general, and more specifically to the Fuels Section Quality Control Program. Prior to the initiation of each target fabrication campaign, further process control and inspection and test procedures are designed and documented to assure that the fabricated elements will meet specifications.

## DEMONSTRATED TARGET FABRICATION PROCESSES

### Process Flowsheets & Unit Operations

Figures 2 and 3, appended, chart the sequence of processing steps used in the Transuranium Pilot Plant for the fabrication of metallic and ceramic target elements, respectively. Fabrication experience since work in this Pilot Plant was initiated (July 1968) is summarized below. Photographs of key unit operations are shown in appended Figures 4 through 11.

## Metallic Targets

### Plutonium-Aluminum Alloy

In mid-1968, 300 Pu-Al target elements were prepared for charging into KE Reactor (under PTA-150) as part of the program for "deep-burning" plutonium.

The Pu-Al cores for these elements were made by Battelle-Northwest, using about 6% by weight plutonium metal added to molten aluminum and held at temperature to achieve a homogeneous alloy. The alloy was then cast into graphite molds having dimensions near those of the final core. After cooling, the cores were machined to the final desired geometry, placed in "bird cage" shipping containers, and transferred to the Transuranium Pilot Plant.

There the machined cores were cleaned in perchloroethylene and inserted into aluminum cans. The can weld zone was decontaminated by swabbing with Q-tips dampened with alcohol, an end-cap was positioned on the core, and the assembly was evacuated, backfilled with helium and closure welded. The targets were then helium leak tested, hydrostatically sized (@ 1500 psi), and radiographed to ensure a weld closure of high integrity.

### Neptunium-Aluminum Alloy

In April 1969, twelve Np-Al targets were fabricated in the Transuranium Pilot Plant for irradiation (under PTA-163) in the medical grade Pu-238 demonstration program. To produce these elements, $NpO_2$ was reduced and alloyed with excess aluminum in the presence of cryolite ($Na_3AlF_6$) to dissolve the formed $Al_2O_3$. The resultant Np-Al alloy was cast into ingots which were analyzed, remelted to adjust the alloy composition, and then cast into preheated graphite molds to obtain the desired tubular core geometry. The cores were machined to final dimensions, then clad in aluminum components by the same method used for the solid Pu-Al alloy targets (except that an aluminum rod was put into the core ID of the Np-Al targets to prevent their chattering in the reactor process tube).

Exposure rates encountered during this fabrication varied from 1 R/hr at contact for 600 grams of $NpO_2$, to 13 mR/hr for a single Np-Al core at 16 inches.

## Cermet and Ceramic Targets

### $AmO_2$-Aluminum

In early 1969, 150 grams (12 targets) of americium oxide were received from the Lawrence Radiation Laboratory in the form of capsules with an $AmO_2$-aluminum cermet core hot-pressed into an aluminum alloy cladding. This material was obtained for irradiation (under PTA-171) as an alternate route for the production of "clean" Pu-238.

In March these capsules were further clad in aluminum cans of K reactor element geometry in the Transuranium Pilot Plant. Due to the extremely high radiation levels encountered (up to 70 R/hr at contact), special three-inch thick lead glass was used to minimize personnel exposure.

$NpO_2$-Graphite

In June 1969, eighteen target elements containing $NpO_2$ dispersed in a graphite matrix were fabricated for irradiation (also under PTA-163) as part of the medical grade Pu-238 program. The purpose of using graphite was to avoid the Pu-236 formed in the Pu-238 by the (gamma,n) reaction with aluminum in the Np-Al alloy.

These ceramic elements were prepared by the blending of $NpO_2$ and graphite powders followed by pressing into wafers, heating to outgas and anneal the wafers, and then assembling into cans. The cap end closures were then welded and inspected for integrity. Six of the targets were clad in aluminum alloy cans and twelve were clad in Zircaloy cans. Exposure rates were comparable to those encountered in fabricating the metallic Np-Al target elements.

## FUTURE PROGRAMS

In continuation of the program for demonstrating the production of medical grade Pu-238 at Hanford, arrangements have been made with AEC-RL for the procurement of 1000 grams of $NpO_2$ to be used in the further testing of various target design concepts. Included will be an investigation into the cause of the $NpO_2$-graphite element failures experienced in KW Reactor last month. In addition to the further work on that design concept, elements based on two other core geometries will be fabricated in the Pilot Plant: (1) a thin annulus of $NpO_2$ near the outer surface of the graphite core, and (2) long thin tubes of $NpO_2$ placed in a drilled solid graphite core. Materials and components for these elements are being procured.

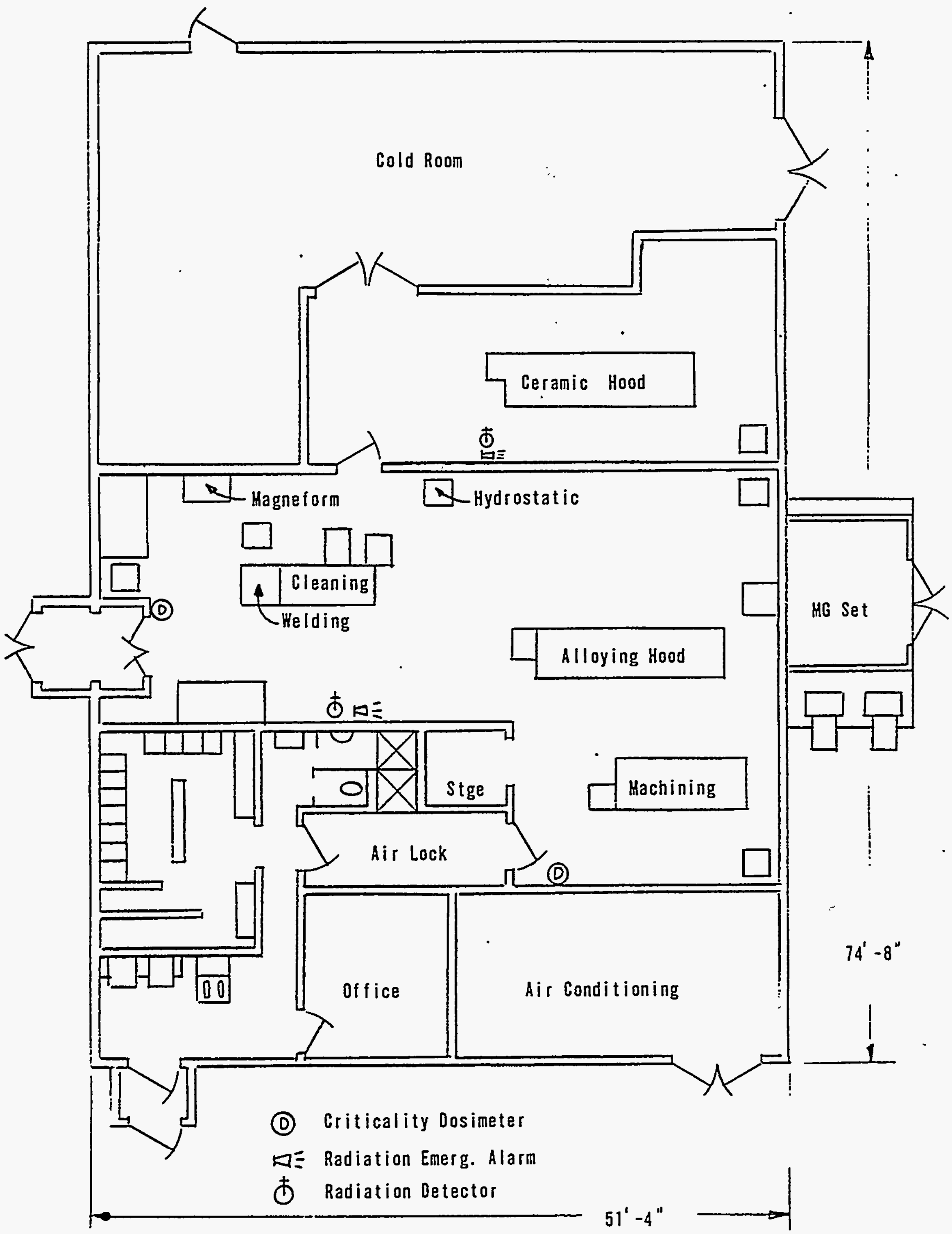


Figure 1. Transuranium Pilot Plant - Floor Plan (Bldg. 3708)

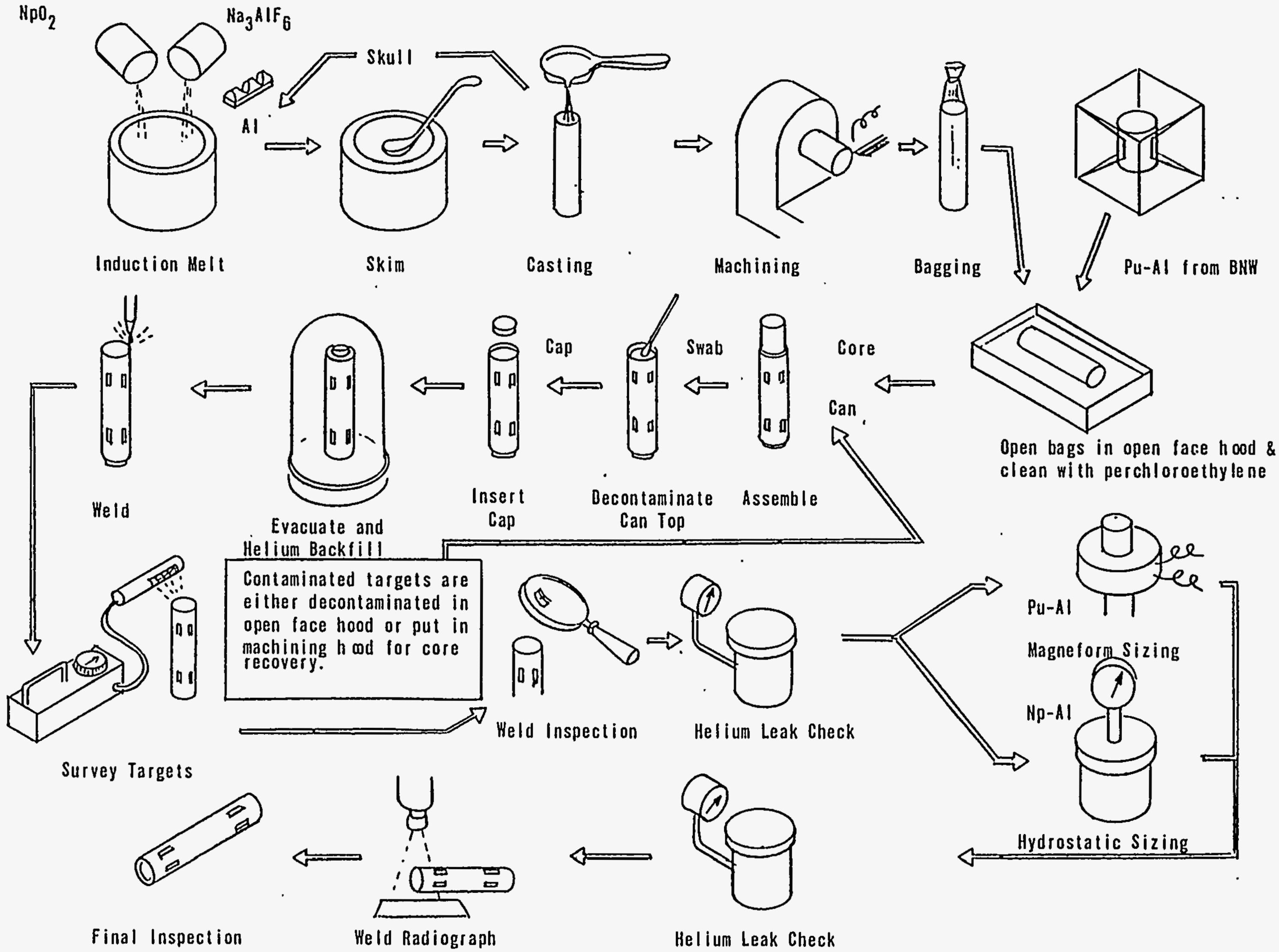


Figure 2. Process Flow Sheet - Metallic Target Elements
(Shows Np-Al alloying)

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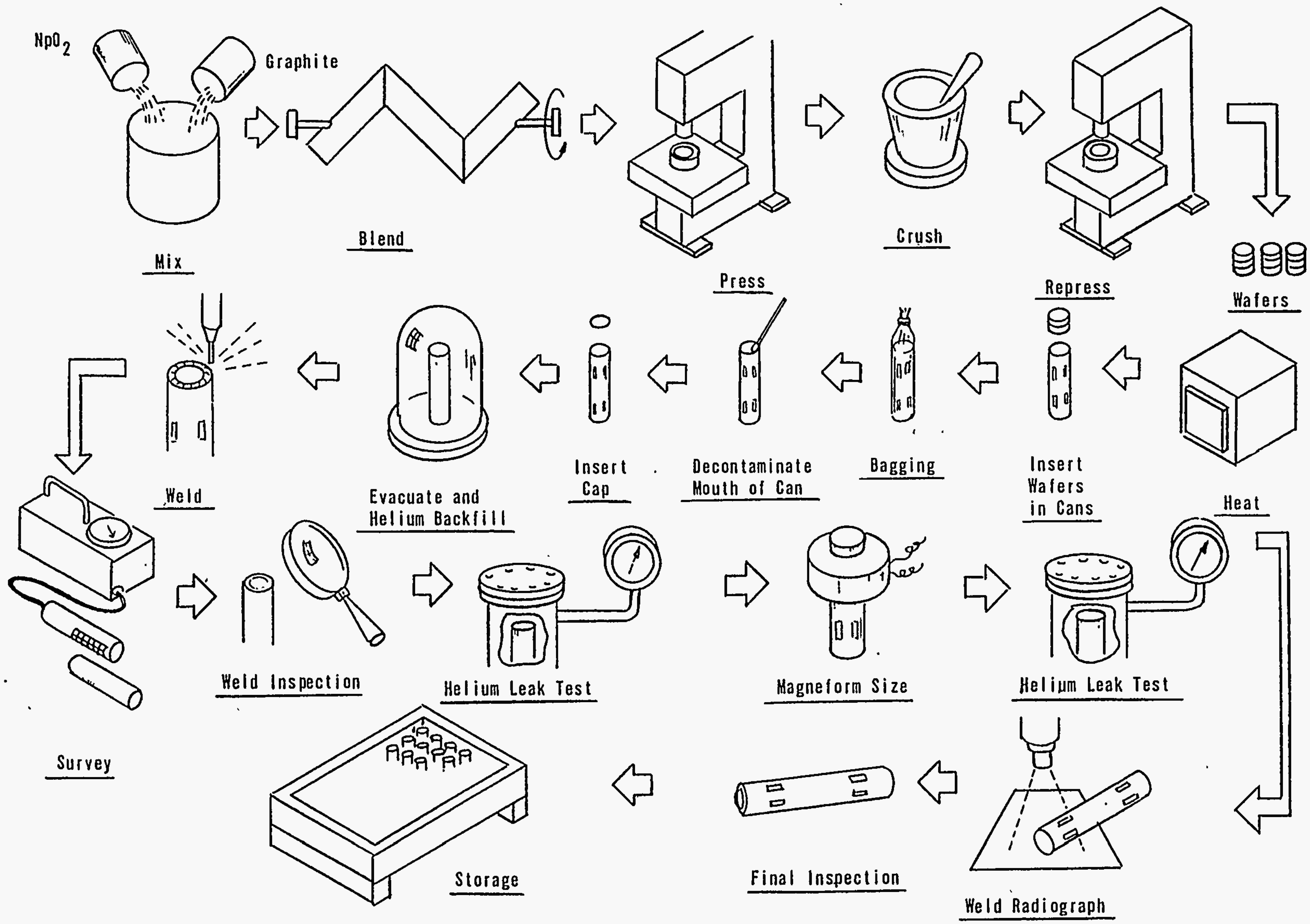


Figure 3. Process Flow Sheet - Ceramic or Cermet Target Elements
(Shows $NpO_2$ and graphite being mixed)

Figure 4. Pouring Alloy from Furnace into Casting Crucible

Figure 5. Casting Alloy from Crucible into Graphite Mold

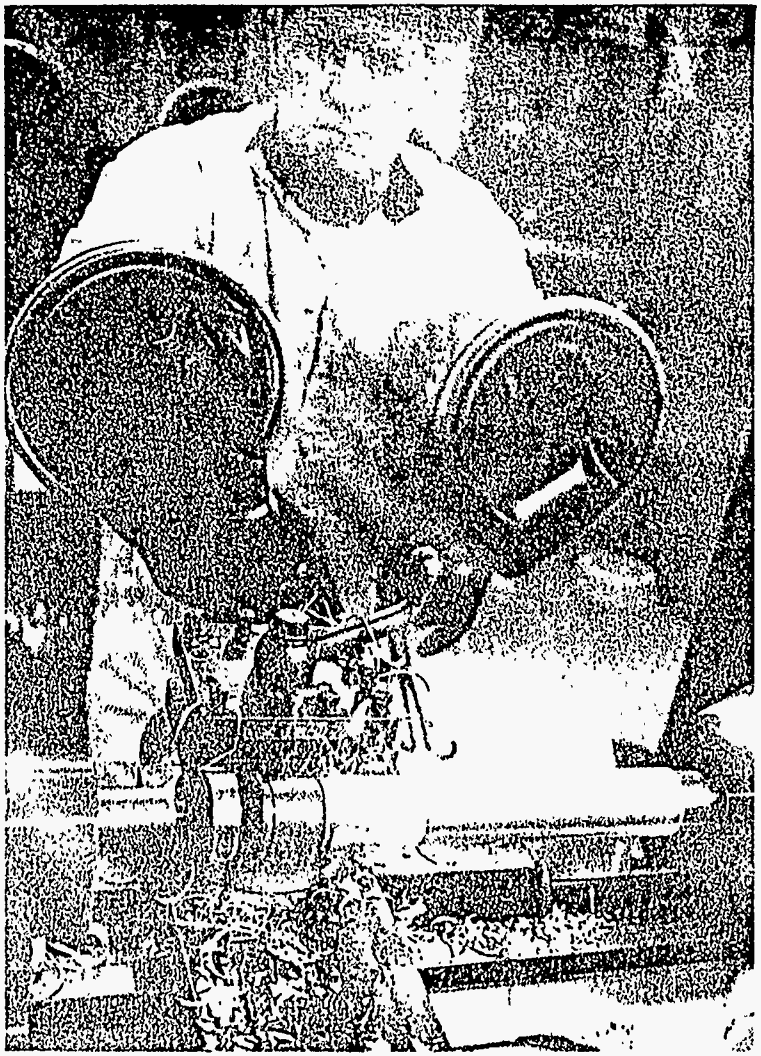

Figure 6. Machining Metallic Core in Lathe Hood

Figure 7. Bagging Material out of Hood (Bag is heat-sealed)

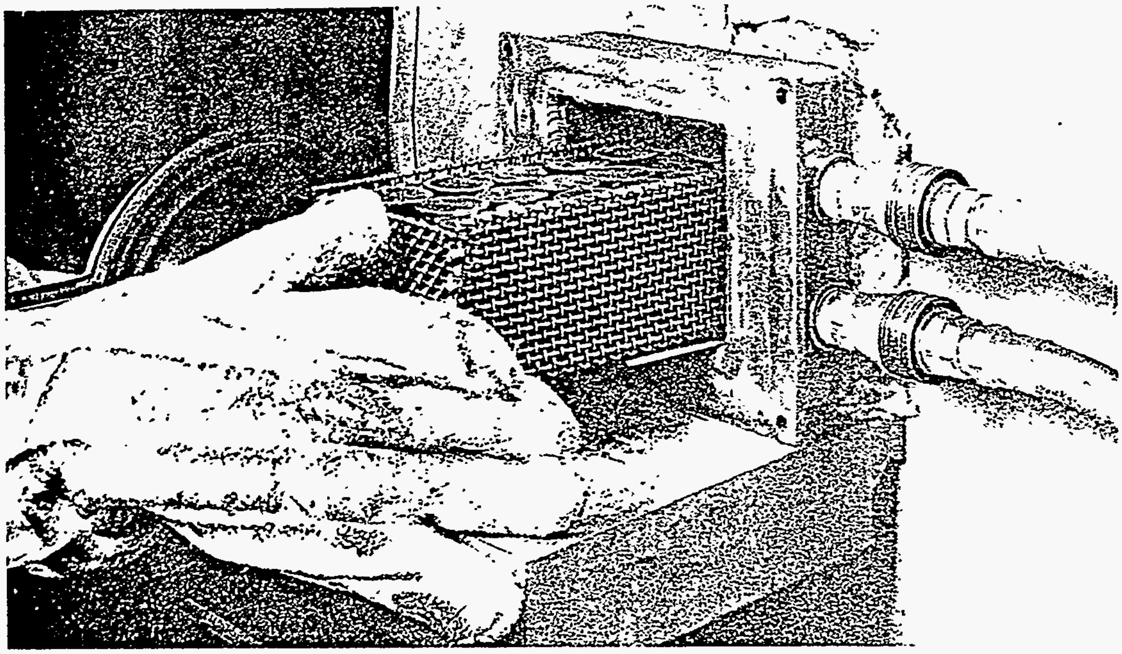

Figure 8. Inserting Basket of Wafers into Vacuum Furnace

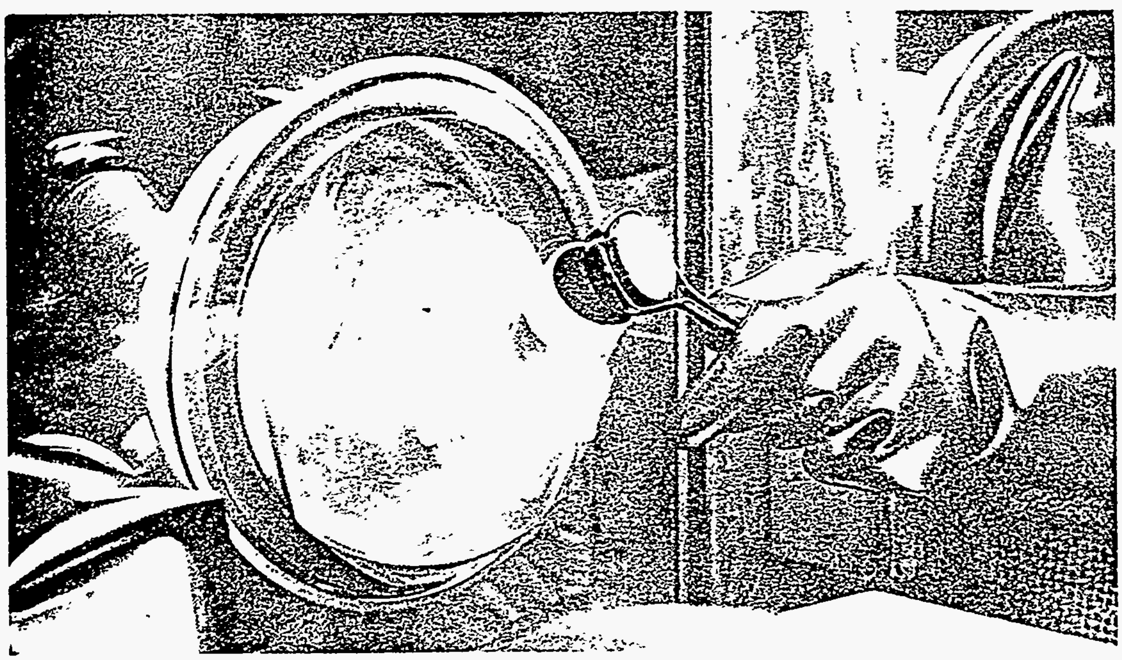

Figure 9. Inserting Wafers into Can



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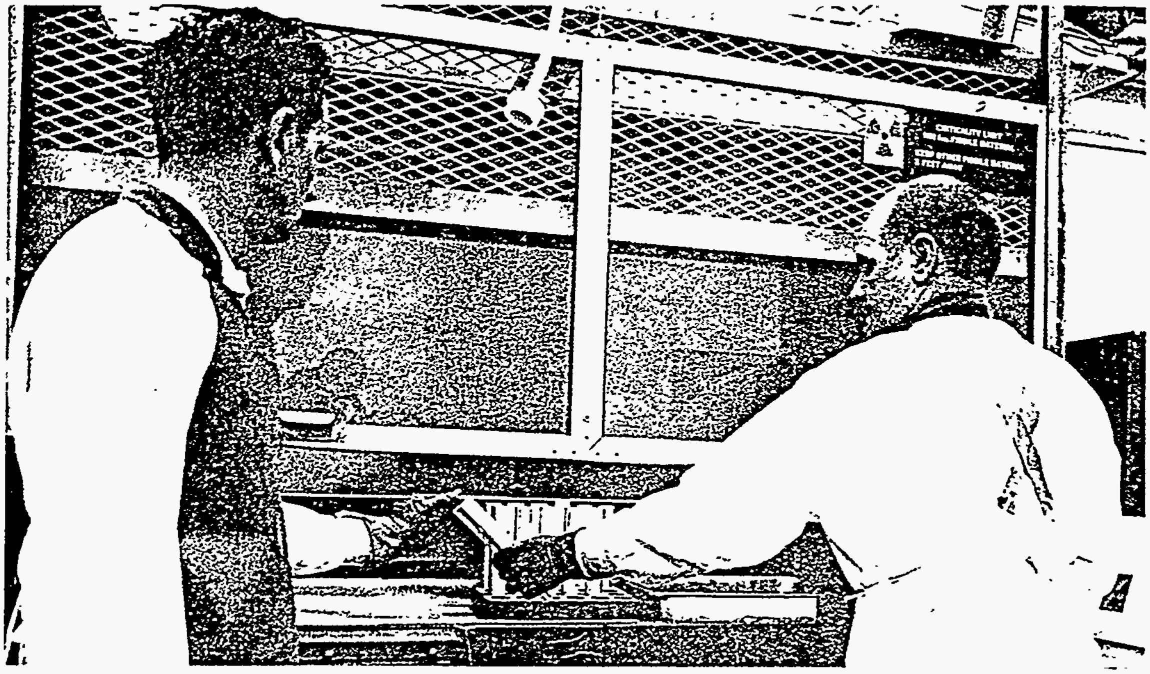

Figure 10. Placing Cap in Can
(Upright cores being dried after cleaning)

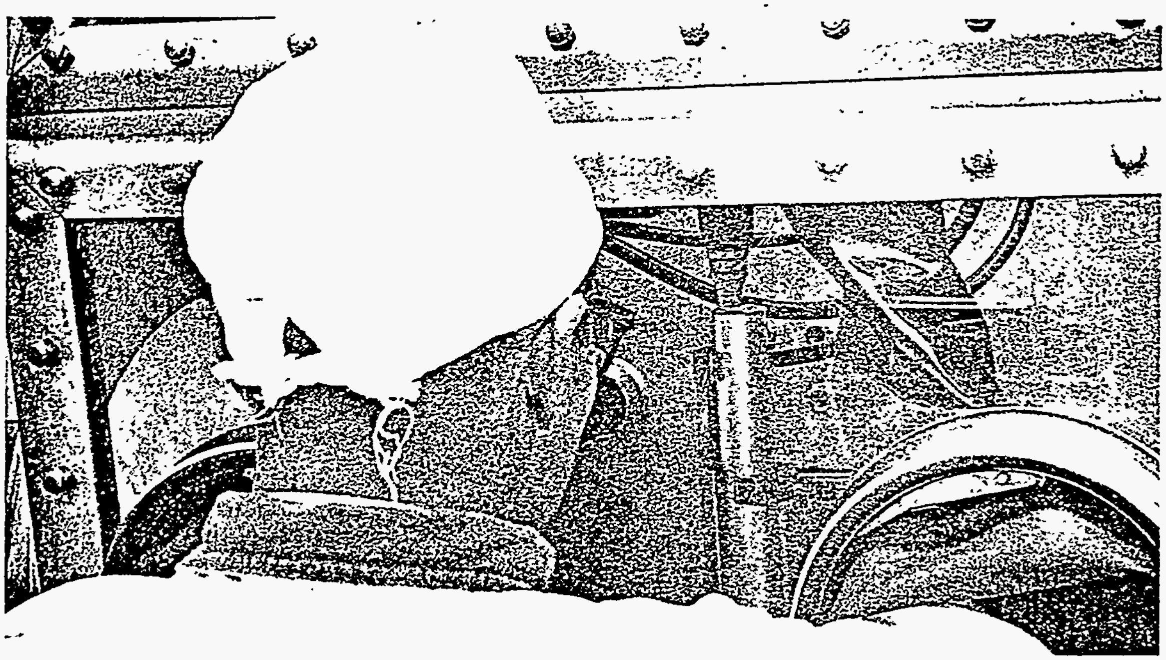

Figure 11. Inert Gas Welding of Cap End Closure
(Shield raised to show welder head)