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MONTHLY REPORT

JANUARY 1969

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February 18, 1969

MONTHLY REPORT

JANUARY 1969

DOUGLAS UNITED NUCLEAR, INC.

Richland, Washington

Work performed under Contract No. AT(45-1)-1857 between the
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SUMMARYREACTOR PLANT OPERATIONSProduction Statistics

	<u>C</u>	<u>KE</u>	<u>KW</u>	<u>N</u>
Input Production - Pu (KMWD)	35.3	90.6	100.2	68.8
- U-233 (Equiv. KMWD)	0.9	13.6	14.7	-
Time Operated Efficiency (%)	51.8	74.8	84.7	61.5
Steam Availability to WPPSS (%)	-	-	-	59.6

C & K Reactors

The process tube power limitations imposed in September at both K reactors because of inlet water piping brittle fracture potential, remained in effect, with 4040 MW the maximum power level attained. The low TOE at C Reactor reflects scheduled tube replacement and a lack of interruptible BPA power.

The nondefense plutonium program to produce Pu-240 in the 8-12 percent range is continuing and thoria loads are being irradiated at KE and KW. Weapons grade plutonium is being produced from natural uranium at C Reactor.

The prototype uncooled control rod was installed at KE Reactor on January 24 and is performing satisfactorily. This flat rod has hinged 40-inch segments, each consisting of a dysprosium oxide-nickel cermet core in a coextruded Inconel sheath.

N Reactor

Reactor power level was administratively limited at 4000 MW throughout the month.

Six reactor outages occurred, one scheduled and five unscheduled. Of the latter, two were due to flow monitor trips, one was caused by a fuel failure, one was due to an electrical ground, and one was necessitated by low flow indications on two process tubes along with higher than normal makeup rate to the primary loop.

All tubes have been removed from steam generators 2A and 2B, and installation of the new Inconel panels will start next month.

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FUEL AND TARGET FABRICATIONProduction Statistics (tons)

	<u>For C & Ks</u>	<u>For N</u>
Billets Extruded	-	42.7
Finished Fuel Produced	237.4	31.0
Thoria Canned	10.0	-

C & K Fuels

AlSi canning operations continued on the basis of three lines on the day shift, five days per week. Bumpers or self-supports were attached to 85 percent of the fuel elements produced.

N Fuels

A bonding problem in coextrusion resulted in reduced input production. The difficulty was corrected and production rate was back to normal by month end.

TECHNICAL ACTIVITIESC & K Reactors

Ten specimens from D riser at KE Reactor were shipped to W. W. Gerberich, consultant at the University of California, for initiation of the fracture toughness tests. These first ten will be fractured over the temperature range of -77 to 20 C to determine whether the sample design is adequate for measuring minimum values of fracture toughness.

Work on the matrix of distribution cases developed for the K tube limits studies is proceeding on schedule. The calculational model has been checked against the BNW Laboratory results with good agreement.

Investigation of the rod-in-tube fuel element failure experienced in the C overbore block last month showed that an incipient failure of an adjacent element also had occurred. Preliminary examination indicates erosion of the cladding beneath the self-supports. These two elements, and a third overbore element which also showed corrosion associated with its self-supports, will be sent to BNW Radiometallurgy for further examination. The fuel which failed apparently caused leakage of the process tube. Although only a small penetration was evident, it was associated with a raised spot in the tube about an inch in diameter. The tube section involved was also shipped to Radiometallurgy for examination.

Graphite dimensional results from thermal and radiation exposure in the ETR and K reactors are favorable, indicating that there is still some time to go

before a turnaround point will be reached in the Ks by the contraction and expansion phenomena.

The 10 kg plutonium irradiation test is expected to be loaded into KE Reactor during the February outage.

The Ames Laboratory has just completed measurements of direct interest on reaction heats involved in uranium-aluminum reactions. The data compare quite favorably with those reported by Russian investigators in 1958. Consideration is now being given to possible methods of determining the kinetics of the reactions.

A Consultant Agreement was negotiated with John A. Blume and Associates, Engineers, of San Francisco, California. This \$7,500 contract permits the initiation of a dynamic seismic analysis for the K reactors, and will provide a detailed cost estimate for the complete analysis. An initial site visit by John Blume representatives was made on January 22-23 for detailed discussions and a tour of K Area facilities.

N Reactor

The irradiation testing of Mark IV fuel elements continues to give favorable results. Visual observation of discharged test fuels has not shown any unusual characteristics with respect to support performance. Swelling of the fuel has been similar to that experienced with Mark I and Mark II elements. The January fuel loading included 41 full-length Mark IV columns charged into Zone Temperature Monitor tubes under Phase I of the program for converting N Reactor to this new fuel.

In the Zircaloy process tube monitoring program, the third precracked tube section is undergoing room temperature fatigue testing and has experienced 150,000 cycles between hoop stresses of 1,250 and 25,000 psi without failure. These results are consistent with previous findings that irradiation increases tube resistance to cyclic fatigue.

The first metal-water reaction test with long-cooled irradiated N Reactor fuel was performed in the 324-D hot cell on January 16. Although results were inconclusive, the test was useful for development of testing techniques.

FEATURE REPORT

This month the appended summary report covers the technical aspects of the flow stoppage which occurred in tube 3560 of KW Reactor last June. Included are the prediction of fuel and tube damage, the actual conditions of fuel and tube after removal from the reactor, and the potential application of information gained to the analysis of other coolant loss accidents. Restoration of the tube was described in the November 1968 Monthly Report.

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GENERAL

On January 15 the Company was advised by the AEC that, in response to President Johnson's FY 1970 budget, it has prepared tentative plans for placing C Reactor in standby and operating the two K reactors alternately. These plans were noted to be subject to review by the incoming Administration.

This announcement has resulted in considerable effort in making tentative plans for estimating the Company's funding situation for the remainder of FY 1969 and for FY 1970. A review of the timing, manpower, and cost effect of changes which would be required to meet the funding limitations was presented to Mr. Baranowski and his staff in Germantown on January 24.

Cancellation of a requisition for twenty-two Metal Handlers, in light of the reactor cutback announcement, nullified the planned addition of fifteen minority group members reported last month. Also dropped were plans to employ an Equal Employment Opportunity Specialist; this required cancellation of one confirmed and two open invitations to visit that had been extended to Negro candidates. Two of the ten new employees hired in January were Spanish-American. Of the two nonexempt promotions, one was a minority group member.

Company personnel had no disabling injuries in January, and no radiation exposures exceeded operational control.

Charles D. Harrington
Charles D. Harrington
President

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REACTOR PLANT OPERATIONS - C & KsPRODUCTION

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

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

The production of nondefense plutonium in the 8-12 percent Pu-240 range is continuing with both natural and enriched uranium at the K reactors. At C Reactor, the natural uranium load is programmed for weapons grade material and the enriched load for nondefense material.

OPERATING EXPERIENCE**DECLASSIFIED**Reactor Loadings

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

The K reactors remain charged with loads of thoria and 94 Metal, as required for the assured delivery of 360 kg of U-233.

Power Levels

Maximum power levels at the K reactors represented continued operation under limits resulting from the inlet piping brittle fracture potential. C Reactor's maximum represented operation at the bulk outlet water temperature of 95 C.

PRODUCTION REACTOR STATISTICS - JANUARY 1969

REACTOR	C	KE	KW	TOTAL
INPUT PRODUCTION - PU - KMW	35.3	90.6	100.2	226.1
- U-233 - EQUIV. MWD	855	13,608	14,672	29,135
POWER LEVEL (MW) - MAXIMUM	2,420	4,040	4,035	10,495
- AVERAGE	2,196	3,908	3,818	9,922
TIME OPERATED EFFICIENCY - %	51.8	74.8	84.7	70.4
OUTAGE TIME ALLOCATION - %				
CHARGE - DISCHARGE	5.0	10.5	0	5.2
FAILED FUEL REMOVAL	0	0	0	0
WATER LEAKS	4.5	0	6.0	3.5
TUBE REPLACEMENT	12.9	0	0	4.3
OTHER MAINTENANCE	3.6	4.4	5.9	4.6
STANDARDS CHECK	0.6	1.4	3.4	1.8
PRODUCTION TESTS	2.3	8.5	0	3.6
PROJECT WORK	0	0	0	0
OTHER	19.3 (1)	0.4	0	6.6
TOTAL	48.2	25.2	15.3	29.6
NUMBER OF OUTAGES	3	2	2	7
NUMBER OF STARTUP INTERRUPTIONS	0	0	0	0
WATER LEAKS - TUBE	0	0	1	1
- VAN STONE	1	0	0	1
NEW TUBES INSTALLED	48	1	1	50
FUEL CHARGE - (TONS) - NATURAL URANIUM	137.1	3.6	3.7	144.4
- ENRICHED URANIUM	78.2	360.4	359.6	798.2
FUEL ELEMENT FAILURES	0	1	0	1
HELIUM CONSUMED - M CU. FT.	269.4	105.2	143.1	517.7
WATER TO REACTOR				
NORMAL OPERATING FLOW - GPM	105,800	190,500	191,500	487,800
pH	6.58	6.69	6.69	-
DICHROMATE - PPM	0.48	0.50	0.92	-

(1) Includes 16.9 percent due to unavailability of interruptible electrical power in the BPA system.

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A three percent reduction in power level was experienced at C Reactor from January 10 through month end because of nine-pump operation (one being down for routine maintenance).

Reactor Outages

Seven reactor outages were experienced as summarized below. In addition, KW Reactor was down 32.9 hours on an outage initiated late in December.

<u>Date Down</u>	<u>Reactor</u>	<u>Outage Hours</u>	<u>Remarks</u>
January 3	KW	36.1	Reactor scram caused by an erroneous technical high-limit trip on the high speed scanner caused by a pin board fault.
January 5	C	143.1	Scheduled outage for process tube replacement; thirty-nine were replaced.
January 8	KW	45.2	Process tube water leak. Tube 3155 was replaced.
January 11	KE	56.7	Removal of a failed natural uranium fuel element from Water Treatment Test Facility tube 4456.
January 13	C	39.0.	Plugged loop header screen. Replaced eight process tubes.
January 19	KE	130.7	Scheduled outage for charge-discharge, replacement of the top section of A riser, and miscellaneous maintenance.
January 24	C	176.2	Reactor shut down due to the unavailability of interruptible electrical power in the BPA system.

EQUIPMENT EXPERIENCE

Riser Sampling - KE Reactor

Essentially duplicating work done on the D riser at KE in December, the dome section and 18 inches of straight pipe were removed from the A riser at KE as part of the sampling program for brittle fracture studies. A new pipe section and dome were installed, following which a thorough cleaning and inspection of the bottom interior of the riser were performed. A crossheader screen inspection

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after a three-hour flush showed that no foreign material had accumulated in these screens.

Horizontal Control Rods

Channel preparation was completed and a flat, uncooled flexible rod was successfully installed in HCR channel No. 2 at KE Reactor. During startup checks, some binding was experienced while withdrawing the first four percent of the rod.

Resistance Temperature Detectors - KE Reactor

Approximately 2100 immersion RTDs were checked for low resistance to ground during the mid-January outage at KE Reactor. Seven units on newly charged tubes were found with low resistance and were replaced. Subsequent checking of some of the lower readings during the ensuing operating period showed that the resistance values varied, making the resistance to ground methods of testing questionable. As an alternate plan, the mass replacement of RTDs in high failure frequency zones is being considered.

Packaged Boiler - D Area

The fire in the 184-CA packaged boiler was lost four times during the month. Three of the failures were caused by the loss of boiler feed water, which is supplied from the fire and sanitary water system, and the fourth was caused by a broken igniter electrode wire.

Funds have been made available to upgrade this facility for greater operating reliability.

Export System Freeze Damage

During the early part of January, freeze damage was sustained to the export water system Simplex valves. Extensive leaks in about half of the 44 valves were repaired.

PROCESS ASSISTANCE AND CONTROL

Operational Physics - SPR

The K Reactors continued to operate with good flattening and no operational physics difficulties. KW Reactor experienced some minor xenon cycling following a minimum outage startup with relatively high exposure fuel. Slight reactivity instability problems are not uncommon near the end of a fuel cycle, when the positive graphite temperature coefficient of reactivity is highest. Proper early action serves to dampen the oscillations before engineering limits are approached.

The C Reactor fuel load consists of about 50 percent excess B Reactor fuel as well as the normal fuel. In addition, the 62 tube block of zirconium process tubes were charged this month with K4NS fuel in place of the normal C5N fuel,

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which has been depleted. The reactivity transients for this hybrid fuel loading, which includes the 44-tube 1-inch overbore block with its 60-tube buffer zone, are quite similar to those for a normal fuel loading and no operational physics difficulties are anticipated.

Operational physics data of interest are shown below:

	Reactor		
	C	KE	KW
Effective Central Tubes (ECT)*	1613	2278	2268
Flattening Efficiency** - January	0.86	0.73	0.72
- 12-mo. average	0.83	0.72	0.70
Maximum Operating Time Permitting Scram Recovery - Hours***	10	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 currently approved startup procedures.

Product Accountability

Minor programming revisions have been made in the code which produces accountability tables. A revised set of tables for CMLF fuel is being prepared, but will not be issued as a part of the accountability tables document because of the short-term usefulness of data on the 80 Metal C overbore fuel.

Laboratory analysis of the enriched uranium samples from a thoria-support column has been completed, and the results are now being incorporated in assessment of the observed high U-232 content from the previously analyzed thoria samples.

Production Fuel Performance

The production fuel element failure which occurred at KE Reactor is described as follows:.

<u>Tube No.</u>	<u>Charge Date</u>	<u>Failure Date</u>	<u>Exposure</u> MWD/T	<u>Failure Type</u>
4456 KE	11/3/68	1/11/69	560	Hot Spot

This represented the first hot-spot failure since December 1966.

During CY 1968, a total of 16 fuel element failures were experienced at the C and K reactors. Of this total, 5 were from standard production fuel, 2 occurred in tubes containing thorium target elements, and 4 were sustained in depleted uranium fuel. The other 5 involved fuel being irradiated under Production Test evaluation studies. Also, 3 of the 5 standard fuel failures were associated with Production Test programs. Twelve of the 16 failures occurred at exposures in excess of that for weapons-grade Pu-240 concentrations.

The following table shows production fuel failure frequencies, as number per million elements discharged, for the 12- and 24-month periods ending December 20, 1968:

	<u>12 Months</u>	<u>24 Months</u>
C Reactor - Natural Uranium	26	33
- 94 Metal	41	28
K Reactors - Natural Uranium	-	4
- 94 Metal	7	8

Reactor Effluent Activity Data

Effective this month, the monthly reporting of total kilocuries per month for significant radionuclides in the reactor cooling water effluent is being discontinued because of adequate coverage elsewhere. This information is reported monthly to the AEC via other means, and is summarized quarterly in a separate report.

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<u>Zone</u>	<u>Tons</u>	<u>Material</u>
Central	98	Natural Uranium
	41 (Includes 14 in overbore block)	94 Metal
Ring	12	Natural Uranium
	20	94 Metal
Fringe	25	Natural Uranium
Thoria Support	19	94 Metal
Thoria	4	Thoria

Loading Pattern - C Reactor

B-A

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<u>Zone</u>	<u>Tons</u>	<u>Material</u>
Central	242	94 Metal (for Thoria Support)
	2	Natural Uranium
	16	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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<u>Zone</u>	<u>Tons</u>	<u>Material</u>
Central	244	94 Metal (for Thoria Support)
	1	Natural Uranium
	16	Thoria ("X" Designates Tubes)
	1	Special Depleted Uranium (PITA-048)
Buckled	66	94 Metal (for Thoria Support)
	5	Thoria ("X" Designates Tubes)
Blanket	53	94 Metal (for Thoria Support)
	11	Thoria

Loading Pattern - KW Reactor

B-C

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REACTOR PLANT OPERATIONS - N

PRODUCTION

General

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

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Statistical Summary

Input Production - Pu	- KMWD	68.79
Electrical Generation (KMWH)	- Total	321.2 (per WPPSS)
Power Level (MW)	- Maximum	4000
	- Average	3609
Time Operated Efficiency	- %	61.5
Steam Availability	- %	59.6
Number of Shutdowns	- Scheduled	1
	- Unscheduled	5
Fuel Failures		1
Fuel Charge (Tons)	- 94 Metal	300.3
	- 125 Metal	67.9
	- Natural U	0.4
	Total	368.6

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Helium Losses	- M cu. ft.	590.1
Fuel Oil Usage	- bbl.	23,486

OPERATING EXPERIENCE

Reactor Loading

The reactor loading at month end is shown on the front face map which follows page BN-6. Two hundred two process tubes were charge-discharged during the scheduled outage starting January 19. This included the charging of 42 columns of Mark IV fuel and discharging of one column of Mark IV fuel.

Power Level

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

The main steam header pressure was held at approximately 123 psig, enabling WPPSS to maintain electrical generation at 800 MW during the period prior to the January 19 outage. Reactor startup following the scheduled outage was accomplished with the main steam header pressure reduced to 115 psig, as necessitated by the transition operating characteristics of the Mark IV fuel. With the reduced steam pressure, a reactor power level of 4000 MWt was achieved and WPPSS was able to maintain electrical generation at approximately 745 MWe.

Reactor Outages

The six reactor outages are summarized below. In addition, the reactor was down 7.2 hours for an unscheduled outage initiated December 31.

<u>Date Down</u>	<u>Outage Type</u>	<u>Outage Hours</u>	<u>Cause</u>
January 3	Unscheduled	26.3	Fuel failure in process tube 1248.
January 4	Unscheduled	3.5	125V DC ground in 105-N.
January 19	Scheduled	212.2	Charge-discharge and general maintenance.
January 28	Unscheduled	0.7	Flow monitor trip on tube 3354.
January 28	Unscheduled	4.7	Flow monitor trip on tube 1052.
January 29	Unscheduled	31.9	Low flow on tubes 1242 and 2546 with excessive primary loop makeup.

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Fuel Failure

The Mark-IC fuel failure discharged from process tube 1248 was verified as an end-associated failure located six elements from the rear. Detailed examination is underway at BNW Laboratories.

EQUIPMENT EXPERIENCE

Package Boilers

Testing of the two CE package boilers was continued. Even though these boilers were not fully operational, they were used to supply limited amounts of steam, because of the extremely cold weather, during the one scheduled outage. Difficulty is still being experienced with their scanner systems.

Equipment Modifications

Equipment Modification Memorandum No. 27, which authorizes the changing of a circuit capacitor in the flow monitor controllers, was completed on 126 units during the month for a total of 150 completed to date. This modification reduces the triggering action, which prior to the modification resulted in a change of as much as six volts in the thyatron trip points, to a fraction of a volt and does not affect the controller trip point.

EMP-309, which authorizes installation of indicating lamps in the 105-N control room to indicate the open or closed position of the COND-243 valves, was brought to 95 percent completion during the month.

EMP-411, which authorizes installation of line drivers in GM tube holders in the gamma monitoring turrets, is now 25 percent complete. Turret No. 6 was completed.

EMP-427, which authorizes removal of the flow monitor channel bypass jacks from service and substitution of an alternate means of bypassing the channels, was completed.

Primary Loop Equipment

Primary loop repairs completed in January included: (1) twelve new Mark III operators were installed on V-11 valves, (2) three manual operators were installed on V-11 valves, and (3) a new piston was installed on the V-12 operator for process tube 1843. The rear connector diversion valves, V-12s, continue to be a problem. Following a functional test prior to the scheduled shutdown, there were 16 that would not operate, 4 that remained partially diverted, 6 that would not operate back to normal with 650 psi supply air pressure, 2 that were stuck in the field divert position, and 7 that were not tested at high pressure.

Supports for primary piping in Cells 1, 2 and 3 were repaired and the lines realigned. Hangers and framework in the pipe gallery were also repaired. This repair consisted of installing additional plates on the walls and

adding a support post. Following this work, it was possible to close the Cell 2 isolation valves a full thread farther than before. This reduced the in-leakage to the Cell significantly, although there is still slight leakage through the PCRV-202 valve.

A new cap installation tool was developed for installing caps on tubes with the connector valve (V-11) in the open position. This tool was used during the charge-discharge operation in conjunction with reduced flow conditions established by PT-111, Supplement 1, and it worked satisfactorily, although certain refinements are needed to direct water spray away from the operator when it is being used. Modification of the monotube adapter is also needed to reduce spray when the adapters are installed. The success of this tool permitted maintenance pipefitters to apply their efforts to repair of V-11s rather than to operation of V-11s.

During the shutdown on January 29, it was found that the zone temperature monitor RTD for process tube 1242 had blown out. This accounted for the flow problem on tube 1242 and the high makeup rate. The Barton transducer for the flow monitor on process tube 2546 was replaced to correct the low flow indication on that tube.

Ball Safety System

All ball hoppers were functionally tested again during the scheduled outage. Two hoppers, Numbers 15 and 98, failed to operate when tripped. It appears the failures were caused by buildup of corrosion products in the trip mechanisms. The trip mechanisms of these two hoppers and 45 others were reworked and cleaned, and all hoppers tested again before refilling. Of the 47 hoppers reworked in January, 22 were repeats of those reworked in December, therefore a total of 67 have now been reworked. To reduce the humidity within the trip mechanism housing, a supply of service air was provided to each such housing.

Ball hopper No. 17, which failed to operate after installation last month, was removed and inspected. It was found that the gate shroud was out-of-round and caused binding of the gate. Repairs were effected and the hopper returned to service.

The ball test program resulted in a significant increase in helium consumption.

Electrical

DC grounds were a major problem during the month. Grounds occurred throughout the plant, and one in the 105 Bldg. resulted in a reactor shutdown. The trouble was traced to the right side V-28 valve. Isolation of power to the valve cleared the circuit for startup.

A fire in the motor control center compartment for the breaker supplying electrical power to the No. 1 lift station pump occurred during the outage on January 29. The fire was extinguished with dry chemicals and cleanup was effected. Cause of the fire was determined to be overwork of the breaker due to an operational procedure change effected several years ago.

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Confinement System

After the January 29 shutdown, confinement doors 221 and 505 were found to be unlocked and unlatched. Investigation revealed that water had collected in the door control boxes on the Zone I side, and had caused double grounds. The double grounding resulted in a signal to the circuit to open the doors, which did occur. All other confinement doors were checked for presence of water in the control boxes. Breakers CE 19 and CE 20 will be opened following the establishment of Zone I confinement in the future. The open breakers will prevent a recurrence of the inadvertent door operation due to grounds. This incident is being investigated to determine if there was a violation of Operating Safety Limits.

Testing

Special thermocouples were installed in selected penetrations of the shield walls by primary piping to determine the difference in temperature of the concrete shield with and without cooling water. If no cooling is needed to keep the concrete below its maximum allowable temperature, it may not be necessary to replace leaking cooling coils which are contributing to the high moisture in Zone I. The work will be done under Development Test 45.

PROCESS ASSISTANCE & CONTROL

Operational Physics

During reactor operation prior to the January 19 outage, the excess reactivity held in rods had decreased to 3.48 mk, but flattening remained higher than expected at 85 percent. The inboard rod configuration, to obtain axial as well as radial flattening, has been utilized successfully for the past four months.

Twelve additional Mark IA spike columns were added during the outage to bring the total number of spike in the ring to 244. Four of these columns were located in the corners inside the inner boundary of the presently defined spike ring. These columns were added to increase the excess reactivity held in rods for the upcoming operating period so that enough rods can be retained in the reactor at full power (and 90 psi MSHP) to provide reasonable control and flattening. The excess reactivity is expected to drop to 3 mk at the end of the next operating period, but it is believed full thermal power can be maintained.

Forty Mark IV columns displaced Mark IC columns to initiate the transition to the new fuel model. It appears at this time, after only a few days operation, that these columns are operating about as expected at 3 percent higher powers than the surrounding Mark IC fuel. The power generation should increase slightly in the near future as the plutonium buildup overcomes the earlier negative effects of samarium.

Operational physics data of interest are:

Effective Central Tubes (ECT)	857
Flattening Efficiency - January	0.85
- 6-mo. average	0.82
Maximum operating time permitting scram recovery - hours	24

N Reactor Control Rod Cleaning - Coolant Tubing Observations

Four horizontal control rods were cleaned with chromic acid during the December outage at N Reactor to restore normal coolant flow, making a total of 11 rods that have been cleaned in the past 3 months. Two of the rods, numbers 68 and 69, would not hold hydrostatic pressure after cleaning, and examination showed pinhole leaks in the mid-section aluminum coolant tubing. The pinholes were welded shut, the tubing replaced, and the rods returned to service.

During the upcoming February outage, mid-section tubing from two control rods will be removed for destructive examination, one before and one after cleaning, to determine the condition of the tubes and the cause of the penetration observed on rods 68 and 69.

The most probable cause of the pinholes is internal pitting that progressed through the coolant tube wall. The rods were not pressure tested before cleaning, and the leaks may have existed prior to the cleaning. It is also possible that pits penetrating the tube wall were plugged with corrosion products which were removed by the chromic acid cleaning. Destructive examination of the additional tubes will indicate the degree of pitting present, and whether or not replacement of rod coolant tubes needs to be considered.

ZTM Monitor Program Improvements

Changes were made to the input portion of the program to speed up execution time by eliminating three time-consuming loops and replacing them with a special indexing scheme.

Product Accountability

Accountability data for Mark IV fuel have been prepared for inclusion in the accountability tables document. Data for Mark I fuels previously appearing in graphical form are being prepared in tabular form for inclusion in the same document.

Burnout cost calculations for the N Reactor fuels have been completed, and the results are being published.

The Mark I-A lattice heating calculations have been completed and the results documented.

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28			C	C		X	X	X	X	E	A	A	A	A	E	E	E	07	E	E	E	E	E	E	E	E	E	E	X	X	X	X	C	C	C	28	
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Fuel Code	No. Tubes	Description	PT-NR No.	No. Tubes	Description
A*	62	MK-IC (94 Metal Fringe Normal)	76	6	Fuel Meltdown Test
C**	113	MK-IC (94 Metal Central)	94+	38	MK-IV Demonstration
D	175	MK-IC (94 Metal Fringe)	95	5	High U-236 Content Fuel
	1	MK-IC (Natural U)	96	31	MK-I from Upset-Forged Billets
E	278	MK-IC (94 Metal - High U-236)	01	12	MK-I from Direct Cast Billets
X***	242	MK-ICA (125-94 Metal)	07	40	Initial Full Length MK-IV Columns
	871	Total			
	133	Total PTs		1	Blank Channel or Empty Tube
	1004	Grand Total		133	Total

*50% NFS **100% NFS ***94 Metal-25% NFS

+Includes MK-IV High U-236 Content Fuel and 2 tubes w/MK-IA 125-94 Metal and MK-IV 94 Metal

LOADING PATTERN - N REACTOR

BN-A

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FUEL AND TARGET FABRICATION - C & K REACTORS

PRODUCTION

General

Production of AlSi-bonded fuel elements for C and K reactors was 102.3 percent of forecast. Eighty-five percent of these elements had bumpers or self-supports attached.

Thoria target element production continued on schedule.

Acceptable Elements Produced

	Finished Production (tons)	Yield - Percent	
		Current Mo.	FYTD
AlSi-Bonded Fuel	233.2	96.6	96.3
Hot-Die-Sized Fuel	4.2	-	48.3
Thoria	10.0	96.1	96.2

Month-End Inventories

	Tons
Bare Uranium Cores	1043*
Finished Fuel: AlSi-Bonded	1455*
Hot-Die-Sized	12
Thoria Elements	47.7

*These totals include 118 tons of cores and 207 tons of finished fuel made for the smaller reactors.

OPERATING EXPERIENCE

Overall operating efficiency of the AlSi-bonding lines was 98.8 percent. Downtime was assigned 42 percent to equipment malfunctions and 58 percent to operations. AlSi canning operations continued on the basis of three lines on the day shift, five days per week.

EQUIPMENT EXPERIENCE

Nothing significant to report.

PROCESS ASSISTANCE AND CONTROL

Nothing significant to report.

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FUEL AND TARGET FABRICATION - N REACTOR

PRODUCTION

Input Production

Total billets extruded	232
Tons extruded	42.7
Percent of forecast	87.1

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Output Production

Total finished assemblies	1,257
Tons output	31.0
Percent of forecast	100.0

Uranium Utilization - % 71.5

Month-End Inventories - Tons

Bare uranium billets	313
Finished fuel	313

OPERATING EXPERIENCE

Production was reduced significantly during January to solve major unbonding problems at extrusion. The reject rate was markedly reduced by incorporating the following modifications into the extrusion process: (1) front and rear end-plates were machined to a "lipped" configuration to provide a better inner welding surface; (2) copper-silicon inner sleeves were annealed prior to assembly for two hours at 1500 F; (3) billets were preheated in a stepwise fashion, to minimize temperature differentials between inner and outer copper-silicon sleeves during heat-up.

Input production had returned to normal at month end.

EQUIPMENT EXPERIENCE

The W-spring welder experienced a drift in the welding current which required a waiver to accept the fuel not within Quality Control Standards. No malfunction was found in the mechanical operation of the welder, and the difficulty appeared due to line voltage variation. During the investigation of weld quality and equipment malfunction, it was found that acceptable fuel could be welded when the current was maintained at 11,500 ($\pm 1,000$ amps) as compared to the standard of 11,500 (+ 500, - 1000 amps). Quality Control Standards were changed accordingly.

PROCESS ASSISTANCE AND CONTROL

The problem of loss of vacuum during billet preheating was virtually eliminated by relieving the stress on the rear inner weld. This was accomplished by placing a 0.040-inch copper-silicon spacer plate between the uranium billet and the rear preshape, thereby obtaining uniform weight distribution of the billet across the rear end plate. Relieving the stress in this manner prevents stress-corrosion cracking of the inner weld and subsequent loss of vacuum. None of the 117 billets using spacers lost vacuum during preheating as compared to six out of seventy billets losing vacuum without spacers. The permanent solution to this problem will be to change the preshape design to include a raised area the same thickness as the 0.040-inch space.

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TECHNICAL ACTIVITIES - C & K REACTORS

RESEARCH AND DEVELOPMENT

Mission 1 - Basic Production

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1-A. Brittle Fracture Program

Acoustic Monitoring

Following successful application of the acoustic monitoring technique on the A riser approach piping of KE Reactor, it was decided to expand this nondestructive test to the remaining junctions. Fabrication of equipment for the monitoring of each wye joint at both K Reactors is progressing, and the equipment is expected to be available in the near future. It is anticipated that the program will be completed in the next few months.

Fracture Sample Specimens

Forty specimens are being machined from the D riser section recently removed from KE. This machining is in accordance with specifications prepared by W. W. Gerberich. Ten samples have been sent to Mr. Gerberich to date. These initial ten will be fractured over the temperature range of -77 to 20 C to observe whether the sample design is adequate to obtain minimum values of fracture toughness. The extent of any sample design revision required will depend upon the degree of brittle or ductile behavior observed.

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

The methods utilized in the K tube limit calculations were summarized in the December monthly report. Since that time, additional technical effort has gone into setting up the matrices of calculations for different conditions of flux distortion, inlet water temperature and specific power conditions, and computer cases have been started.

The calculational model has been checked against the BNW laboratory results with good agreement; decay transients and conductivities of the materials used in the test assembly were incorporated in this check.

A draft has been completed of a document to describe the calculational approach being used in the tube limits studies.

1-C. Zircaloy Process Tube Hydride Studies

Process Tube Examination

Two tubes removed from KW Reactor on December 19 for hydrogen analyses. The two tubes were 3661, which had operated at high outlet coolant temperatures with 210 Metal, and 3047, which had been grit blasted before it was installed on February 17, 1966. Sections from these tubes have been sent to the laboratory.

Preliminary data (base metal hydrogen content only) from three KE Zircaloy tubes which were sandblasted on March 27, 1967, to remove case hydride layers, and which operated until December 1968 with anodized spacers, indicate that further hydride buildup was prevented in areas where removal was completed. One of the three tubes examined showed no indication of additional hydride buildup during the period since sandblasting. The others had spotty areas of continued buildup indicating incomplete removal; the latter is believed to have been the case, in line with observations of sandblasted tubes which were destructively examined following the grit blasting demonstration in 1967.

Hydride Removal Techniques

A procedure for electrolytic removal of the internal hydride case layers on K reactor process tubes has been demonstrated on a laboratory scale. The laboratory technique in its present state of development involves anodic dissolution of the hydride case in a dilute HNO_3 electrolyte, followed by an HNO_3 -HF etch to remove the last vestiges of hydride. The primary virtue of the technique is its self-regulating nature; i.e., only hydrides would be dissolved during the electrolysis, whereas grit blasting is relatively indiscriminate and could lead to unnecessary removal of sound metal. A further expected feature of this technique is thoroughness of case removal, whereas grit blasting may leave islands of hydride behind.

1-D. Physics Code Development

The extensive CLUMSY modifications to handle the kinetics calculations for the K riser problem have been finished and incorporated into the code. Kinetics cases for determination of the K reactor tube power limit are being processed on an accelerated basis.

1-E. Corrosion Studies

Inhibiting Corrosion of K Area Emergency Coolant Pumps

Current-potential measurements on cast iron, brass and graphite were conducted in uninhibited filtered Columbia River water, and in filtered water inhibited with 500 ppm sodium dichromate, to determine whether or not dichromate ion would effectively control the galvanic corrosion in the K Area diesel-powered emergency coolant pumps. Dichromate is known to be an effective inhibitor of nongalvanically coupled steel at these concentrations. The use of dichromate was suggested since it could be used without increasing existing chemical costs by pumping the dichromate solution through the emergency coolant pumps and crosstie lines into the reactor cooling water where it is currently employed.

The data showed that the galvanic currents between cast iron and brass in untreated water were 4 to 30 times higher than in dichromate-treated water, and the galvanic currents between cast iron and graphite were 1.5 to 5 times higher in untreated water. The dichromate inhibitor was unable to completely overcome the driving force of the galvanic couple,

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however, and pitting occurred on the cast iron. Because of this pitting, the use of sodium sulfite, on which test measurements had previously been reported, is preferred over dichromate to control the corrosion.

Corrosion Resistance of Carbon Steel to Water Treatment Changes

Results of recent coupon measurements demonstrate that Hanford reactor plant water treatment tests and modifications over the period of March 1963 to August 1968 had little influence on carbon steel corrosion rates. The specific tests and changes were (1) reductions in sodium dichromate concentrations from 1.8 ppm to 1.0 ppm in both high- and low-flow systems, (2) reduction of pH from 7.0 to 6.6 in low-flow systems, and (3) changing from liquid aluminum sulfate to bauxite-sulfuric acid water treatment. Carbon steel corrosion rates were reduced by filming amine inhibitor additions in the D Area steam lines, and by addition of a proprietary dichromate-base inhibitor to the K reactor foundation cooling systems.

Aluminum Corrosion Mechanisms - Electrochemical Investigations

An electrochemical mechanism is among the several which have been proposed to explain the sporadic occurrence of ledge and groove corrosion on aluminum-clad fuel in the Hanford single-pass reactors. Laboratory galvanic measurements among several aluminum alloys and Zircaloy-2 revealed that potentials of over one volt could be obtained between some of the couples. When the temperature was increased from room temperature to near boiling, currents flowing between galvanically coupled samples were increased by a factor of over 100. In all cases the Zircaloy-2 was more noble than any of the aluminum alloys to which it was coupled, but for some aluminum-to-aluminum couples the currents reversed direction when the temperature increased.

Loop tests were conducted with potentials of 0, 0.5, and 1.0 volt impressed across 8001 and 1245 aluminum alloy samples exposed to 100 C and 130 C reactor process water. Pitting, which resembled groove pitting on in-reactor fuel cladding, was found on several of the samples after a 1-week exposure. These results, while not conclusive, indicate that one of the mechanisms for initiating pitting on reactor fuel cladding could be the flow of anodic currents originating from galvanic coupling.

Aluminum Corrosion Mechanisms - Erosion-Corrosion Investigation

A laboratory facility is being commissioned to investigate mechanisms of erosion-corrosion on Hanford aluminum alloy fuel claddings. Among the parameters to be investigated are pH, velocity, temperature, and sodium dichromate concentrations.

1-F. Graphite Studies

Burnout Monitoring

Graphite burnout results for samples removed in late 1968 from KE and C Reactors are summarized below:

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Reactor	Channel and Charge No.	Graphite Type	Monitoring Period	Percent Burnout per KOD* (Maximum)
KE	3066 18	TSGBF	2-16-68 to 12-10-68	2.0 near center line of stack
C	1960	KC	4-17-68 to 11-24-68	40.0 @ 1/4 the distance in from the downstream end
C	25	CSGBF	4-17-68 to 11-24-68	<2.0
C	1889	KC	4-17-68 to 11-24-68	~ 27.0 at center line of stack
C	16	CSGBF	4-17-68 to 11-24-68	<2.0

*KOD = 1000 operating days

These results are similar to those reported for the previous monitoring period (September 26, 1967 to April 17, 1968) implying that the oxidation rate has been consistent for more than the past year. The location of the burnout area, i.e., hottest temperature, has also remained essentially the same. The effect of graphite type on burnout rate is self-evident from the data. Pertinent to the above data is the fact that the C Reactor stack contains CSGBF graphite and the K reactor stacks consist of TSGBF graphite.

Graphite Expansion and Contraction Data

Specimens of K reactor graphite machined from the core of a K reactor are being irradiated in the ETR to evaluate the effect of continued irradiation on the radiation-induced distortion of graphite. These samples now have a total accumulated fluence (fluence in a K reactor plus fluence in the test reactors) of about 12.5×10^{21} nvt. Samples operating at 500 to 600 C are still contracting. Specimens which had operated at 1000 C, which had reached turnaround at 9×10^{21} nvt, and which had been returned to the reactor for operation at 600 C and 800 C to evaluate the effect of the reduced temperatures on the rates of expansion, were subsequently re-measured at 11×10^{21} nvt. Only a slight reduction in expansion rate was observed on the 800 C sample, but the 600 C sample showed considerable improvement. The K reactor graphite has accumulated exposure to 7×10^{21} nvt and is increasing at a rate of about 0.5×10^{21} nvt/year.

Mission 2 - Coproduct

2-A. Lithium-Aluminum Splines

The irradiation embrittlement of lithium-aluminum alloy splines was found to be too severe for their use in this application. This phase of the program has been dropped. The final report is being prepared.

A literature search has revealed that a lithium-bearing spline not subject to severe irradiation embrittlement may be made by cladding green compacts

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of materials such as lithium aluminate in aluminum by extrusion. Current plans are to procure small samples of this type of spline for small-scale irradiation tests to study irradiation embrittlement.

Mission 3 - Transplutonium

3-A. Ten-Kilogram PuAl Irradiation

Physics input for hazards analyses for the 10 kg plutonium irradiation test has been incorporated in the draft which has been routed for comment for planning purposes, and total control data for the PuAl lattice have been established. The loading in KE Reactor is currently expected to be accomplished during the February outage.

The special subcritical monitoring chamber sensitivity requirements depend greatly on the neutron flux and the gamma background; calculations of both of these are inaccurate. To better determine these quantities, a front-to-rear traverse through KE Reactor with one of the special chambers was performed on January 20. The traverse was successful in showing that the neutron count rate is sufficiently high and the gamma background sufficiently low to produce good results, even in a shutdown, partially discharged reactor.

Program difficulties have delayed the determination of spline worth in a PuAl column. Conservative values are available, and the computations for a more realistic value are scheduled to be completed before charging of the test load.

3-B. Plutonium-Burning Study

Studies were initiated to determine the capability of the K reactors to irradiate 250 kg of plutonium as feed for californium production at Savannah River. Preliminary results indicate that such an irradiation would probably be technically achievable, but further nuclear safety evaluations are required to establish operational feasibility.

Mission 4 - Pu-238

4-A. Americium Irradiation

The 150 grams of americium oxide (AmO_2) available for irradiation have been shipped to Livermore and are awaiting encapsulation. The canning components fabricated by DUN and sent to Livermore will apparently require additional machining. A revised rough draft of the authorization document has been completed and is presently being circulated for comments.

4-B. Neptunium Irradiation

The Production Test Authorization for a block irradiation to reduce Pu-236 contaminant in the Pu-238 product has been completed and is awaiting circulation for approvals. Test preparations are directed toward charging the test during the April thoria outage at KE.

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Mission 5 - Other Isotopes

5-A. Cobalt Tube Irradiations

A Production Test to authorize the irradiation of cobalt-alloy and cobalt-coated aluminum process tubes was written and circulated for approval. It is likely that the anticipated reactor shutdowns will cause the cancellation of this test, since the majority of the tube specimens were fabricated for the C Reactor, and the competition for test space in the K Reactors will be severe.

Mission 7 - Target Space Enhancement

7-A. Large-Channel Facility - C Reactor

Rupture of Overbore Fuel Element

On December 22, 1968, the fuel element in position 4 of one-inch overbore tube 2868 experienced a rupture. The rupture area was about the size of a quarter and was located at the upstream end of the downstream self-support. An incipient failure of the element in position 3 also occurred. A small blister was located adjacent to the upstream portion of the self-support on the upstream end of the element. Preliminary examination suggests that the failure mechanism was erosion of the cladding beneath the self-support. The fuel column exposure was 1025 MWD/T.

This fuel utilized a 1-1/8 inch long self-support which was not welded along its entire length. The support attachment process results in some upset at the ends of self-support and creates a crevice beneath weld tabs which could be susceptible to erosion. To further investigate this hypothesis, the two failed elements and a third element which also showed corrosion associated with its self-supports will be sent to Radiometallurgy for further examination.

The third element was irradiated in position 15 of tube 3266 which was discharged at an exposure of about 1400 MWD/T. The inner rod from element 3 and the outer tube from element 1 of this column will also be sent to Radiometallurgy to investigate the non-uniform corrosion which was observed. Although the corrosion was neither excessive nor extensive, it does represent the most severe examples of non-uniform corrosion which have been observed to date in the 1-inch overbore fuel.

The fuel from tubes 3266 and 3166, exposures about 1400 MWD/T, were also examined in the C Metal Examination Facility. With the exception of isolated occurrences of corrosion beneath the weld tabs of the supports (example element 15), this fuel revealed virtually no non-uniform corrosion.

Zircaloy-2 Overbore Tube Failure - Rupture Column

An overbore process tube failure occurred in conjunction with the fuel failure in Tube 2868-C in late December. The section of tube in which the rupture was located was examined in the viewing pit, and pictures

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were taken. It was then shipped to Radiometallurgy for detailed analysis. From the outside appearance of the failure location, it is believed that the tube conformed to a "bump" which formed on the outside surface of the fuel element, due to water entry and reaction with uranium, and that the top of the bump was flattened during discharge. The bump was about 1 inch in diameter and 150 to 200 mils high. Apparently, the rupture occurred at a point where there was no graphite backing-up the tube, perhaps at a block separation.

This event suggests two possibilities which may be unique to the overbore geometry: (1) a rupture may cause local distortion of the tube much more readily than in the smaller K reactor Zircaloy tubes, and (2) such a rupture can occur without causing a scram from the Panellit system.

The downstream 48-inch section of the tube was also sent to the laboratory for hydrogen analysis to determine whether or not the use of anodized spacers in this tube was effective against hydrogen buildup.

C Overbore Block Loading Calculations

Physics calculations on the 94 Metal and natural uranium loading of the C Reactor overbore block have been completed, and rough drafts on the loading pattern and charging procedures have been forwarded for review by other technical components associated with Production Test preparation.

The proposed loading was designed by use of EXTERMINATOR, and it requires only minor changes of the present loading in the buffer region. Calculated values indicate that the peak power in the new loading would not exceed that in the present loading, and that radial flattening within the block would be significantly improved relative to the current 80 Metal loading.

7-B. K Reactor In-Core Flux Monitor Burnout Calculations

Calculations of the expected lifetimes have been completed in support of equipment development studies for K Reactor in-core flux monitors. Predicted times to 90 percent burnout would be 209 days for fission chambers and 1185 days for the self-powered rhodium chambers. These numbers represent operating days at base power level. Documentation of these results and descriptions of the fluxes and flux energy spectra to which chambers would be subjected is planned.

Mission 8 - Nuclear Safety

8-A. Intermetallic Reaction Studies

Continued offsite contacts to locate and utilize existing facilities for studying the thermodynamics of the metal-metal reaction between uranium and aluminum have revealed that the Ames Laboratory has just completed measurements of direct interest.

Tabulated below is a summary of the Ames results to date, along with the results of the previous measurements (1958) reported by Russian investigators.

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There is very definite agreement for UAl_4 , especially considering that the methods of measurement were totally different.

HEAT OF FORMATION OF U-AL COMPOUNDS AT 298 K

<u>Compound</u>	<u>$-\Delta H_f$ (kcal/mole)</u>	
	<u>Ames</u>	<u>Russian</u>
UAl_2	--	22.3 ± 2.4
UAl_3	>25.7	25.2 ± 2.2
UAl_4	29.5 ± 2.0	31.2 ± 3.1

Consideration is now being given to possible methods of determining the kinetics of the reactions, which is equally important in ascertaining its effect on uncooled fuel temperature transients.

Mission 10 - Columbia River

10-A. Film Studies

Termination of Ferrous Sulfate Potassium Permanganate Test

Testing of the ferrous sulfate-potassium permanganate water treatment system in the Water Treatment Pilot Plant (WTPP) has been terminated. During the initial runs, iron concentration in the effluent was about threefold higher than that in the raw water being treated. Increasing the $KMnO_4$ concentration from 0.5 to 1.25 ppm and the pH from 6.6 to 7.6 resulted in a decrease in iron concentration relative to that of the raw water. Effluent samples analyzed by neutron activation techniques were inconclusive; however, there appeared to be some reduction in trace elements, especially Co and Ag.

A further increase in $FeSO_4$ and $KMnO_4$ concentrations to 6.76 and 2.50 ppm, respectively, was made on September 2, 1968, to determine if higher additive concentrations would more effectively remove trace elements. Analyses of iron concentrations in the effluent indicated the additives were still reacting completely under these conditions, but the unacceptably high manganese concentrations in the effluent lead to the decision to terminate the test. Other factors contributing to the decision were the high turbidity of the treated water, the interference in the chlorine analytical procedure caused by the manganese, and incomplete removal of parent isotopes.

Studies Indicating Origin of Process Tube Films

A study of the concentrations of inert elements and radioisotopes contained by the films on both fuel elements and aluminum process tubes of varying in-reactor exposures was completed. The amounts of radioisotopes contained on a unit surface at any particular location on the three tubes increased with longer exposure; however, the differences were less noticeable in the case of isotopes of shorter half-life. Inert element concentrations were quite constant at a given location in the three process tubes, and the linear distributions could be described as an exponential function of

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distance. The linear distribution of inert elements on a fuel column with 60 days exposure could also be described by this function, although higher concentrations were present. A different deposition pattern was exhibited by two fuel columns of lower exposure.

Film composition was uniform along the length of the various process channels, with relative values approximating the respective ratios of the parent isotopes in process water. Corrosion of aluminum process tubes was shown to contribute only minor amounts to the radionuclides contained by the films on the tube surfaces. It is therefore concluded that most of the materials comprising the in-reactor film originated as impurities in the cooling water.

Studies of Turbidity Effects

The influence of higher process water turbidity on effluent water radionuclide concentration is being investigated in a half-plant test at KE Reactor. The first stage of testing (at a turbidity of 0.1 JTU) was initiated on August 28, 1968, and completed on November 3, 1968. The second stage of testing (0.2 JTU) was initiated on November 4, 1968, and is still in progress. The control side of the reactor is operating with normal turbidity (0.03 - 0.06 JTU). The testing to date has produced no significant variance or trend in the radionuclide concentrations between the two sides of the reactor.

ENGINEERING & TECHNOLOGY - REACTORS

MFC-12 - New Capabilities of Production Reactors

The authorization for MFC-12 expired this month, and efforts were directed toward preparation of the necessary final documentation.

It has been agreed that the final reporting of MFC-12 will be accomplished as follows:

1. DUN and BNW will collaborate in the preparation of a combined report summarizing the significant aspects of MFC-12. Publication of this report is scheduled for next month.
2. DUN and BNW will each be responsible for the timely publication and distribution of individual project reports as required to adequately cover the complete details of each significant area of their individual work.

MFC-12 studies by both DUN and BNW personnel have considered a broad spectrum of possible concepts to expand utilization of the Hanford reactor complex.

Several of the concepts studies possess the unique characteristics and physics feasibility necessary to propose their continued pursuit. These proposals will be presented in the final documentation.

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Confinement Studies

As a part of the overall confinement improvement program at the K reactors, an accurate appraisal of the plant's seismic resistance is required. This will involve a dynamic analysis of critical structures and equipment.

Consultant Agreement CA-00074 was negotiated with John A. Blume and Associates, Engineers, of San Francisco, California, on January 9. This contract, in the amount of \$7,500, will permit initiation of the work and will provide a detailed estimate of the complete analysis. An initial site visit was made on January 22-23 by two John Blume representatives for the purpose of detailed discussions and a tour of the K Area facilities.

Uncooled Horizontal Control Rod

The prototype segmented uncooled horizontal control rod was received from the General Electric Company, Nuclear Systems Programs, in late 1968. The rod is made up of multiple 40-inch segments connected by hinged joints. The individual segments consist of a dysprosium oxide-nickel cermet core in a coextruded Inconel sheath.

The rod was installed in HCR channel No. 2 at KE Reactor on January 24 and is performing satisfactorily.

Project Engineering - Reactor Facilities

Project Status Summary

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

ENGINEERING & TECHNOLOGY - FUELS & TARGETS

Thoria Wafer Test

ARHCO has requested DUN to produce and irradiate up to 50 tons of wafered thoria elements for a large-scale dissolution test, and the 3722 Building is being modified to provide the necessary fabrication equipment. As part of this program, a test was conducted to evaluate the effect of heating rates on the characteristics of sintered wafers. Wafers were placed in a furnace and heated to 200, 400, 600, 700, and 800 C for one hour. Dimensions of the wafers were measured before and after heating. There was no measurable expansion in the wafers heated to 200, 400, or 600 C; some expansion occurred in the wafers heated to 700 C, and those taken to 800 C disintegrated from rapid binder evolution.

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TECHNICAL ACTIVITIES - N REACTOR

RESEARCH AND DEVELOPMENT

Mission 1 - Basic Production

1-A. Mark IV Fuel Development

Irradiation Testing

The Mark IV demonstration test fuels have continued to perform satisfactorily. The exposures for these fuels, as of January 1, were as follows:

<u>Fuel Designation</u>	<u>No. of Columns</u>	<u>Maximum Tube Average Exposure, MWD/T</u>
Thick clad (94 Metal)	3	2200
Base clad (94 Metal)	2	2270
Base clad, alternate alloy (94 Metal)	7	2500
Base clad (125 Metal)	2	2770
Modified design	1	2455
Standard design	22	1180

Visual observation of discharged test fuels has not shown any unusual characteristics with respect to support performance. Swelling of the Mark IV fuels has been similar to that experienced with Mark I and Mark II fuels.

The January fuel loading for N Reactor also included 41 full-length Mark IV columns charged into ZTM tubes. These particular columns are part of Phase I of the program for converting N Reactor to Mark IV (see November 1968 DUN Monthly Report). The additional weight of Mark IV columns above that of Mark I-C columns (840 pounds versus 755 pounds) has not affected charging pressures. Four of the 41 columns are monitor columns for evaluating fuel performance and also for obtaining thermal hydraulic data.

The Production Test presenting the operating characteristics and safety analyses for authorizing the first full-length column charging of Mark IV fuel was issued as DUN-5048. Technical conclusions have been reported previously.

Flow Loop Testing - Support Performance

Life testing of Mark IV test fuels has continued without incident. Loop exposures to report time were as follows:

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REF ID: A6454

<u>Fuel Type</u>	<u>Support System</u>	<u>No. of Elements</u>	<u>Loop Time, Hours</u>
26" Mark IV	End-spider	9	3245
26" Mark IV	End-spiders with one or more legs removed	5	2000
23" Mark IV	Spring-stop with fully rounded solids	3	2738
12" Mark IV	Same as above	6	50

The maximum number of hydraulically induced cycles experienced on the end-spider supports is 7×10^8 which is 70 percent of goal. The end-spiders with one or more legs removed have experienced a total of 4.3×10^8 cycles (1×10^8 cycles were experienced prior to leg removal). There have been no indications of fatigue cracks in any of the end-spiders.

The three Mark IV test fuels with fully rounded solids have been removed from the loop for detailed examination. Previously discharged fuels with this type of solid support have shown signs of clad depression at the support contact points. This situation, however, has not been confirmed on fuels discharged from the reactor. Additional testing is required to determine the reason for the discrepancy.

Physics Support of Mark IV Transition

Development of a method to predict the flux and power distributions of the Mark IV/Mark IV-A load is under way. Only moderate success has been achieved in "predicting" the current flux distribution, indicating that some further adjustments in the method are required.

Four columns of 125 Metal Mark IV are to be charged in the near future as a materials test. Because of the high relative power of the Mark IV-A to Mark I, it is possible that the use of 94 Metal ends, desirable for obtaining enthalpy balance data, could push the outlet temperatures of the test columns above limits. Calculations on the basis of current flux distributions indicate that the tube power of a test column with 94 Metal ends could run up to 15 percent above the level expected with dummy ends.

1-B. Zircaloy Process Tube Monitoring

Process tube 2455 mechanical-strength testing and examination is continuing. The third precracked tube section undergoing room temperature fatigue testing has experienced 150,000 cycles between hoop stresses of 1250 and 25,000 psi without failure. This section was cut from a high-flux zone in the process tube. These results are consistent with previous results which showed that irradiation increased the resistance of the process tubes to cyclic fatigue.

The remaining tests to be performed with sections from tube 2455 include replication of downstream-spacer fretting pits, stress-rupture testing of an irradiated section, studies of the effect of creep on crack propagation, and crack propagation tests using upstream sections with hydrogen buildup.

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1-C. Corrosion Studies

Surface Topography Studies

The complex mechanisms by which crud is generated, transported and redeposited in a reactor system are not well understood and have not been systemically investigated. The present studies represent an effort to define some of the system variables which directly affect crud deposition, corrosion product formation, and the relation between the two.

Coupon samples of Zircaloy-2, carbon steel, 304 stainless steel and Inconel have been exposed in both the loop and autoclave environments to determine the manner in which the experimental variables (exposure time, material and flow rate) affect the amount, texture, dispersion, particle size and adherence of corrosion product deposits.

Samples have been exposed for periods up to 2 months and micrographs have been obtained with the scanning electron microscope. The mild steel surface examinations are almost complete and the initial phases of corrosion product film formation can be demonstrated.

Nonmetallic Loop

A nonmetallic recirculating loop, constructed entirely of plastic, is being build to evaluate crud deposition mechanisms. The potential advantages of this loop include (1) restriction of the source of any corrosion products observed on the test surfaces to those surfaces or to a crud generator placed in the loop, and (2) easy loop section replacement making many different experiments possible with little cost or lost time.

1-D. Graphite Protection Studies

Results have been obtained from graphite samples irradiated in the Horizontal Control Rod (HCR) 74 experiment which was designed to provide length-change data for the N Reactor core material (TSX graphite). Four of the seven boats used to contain these samples were removed from the channel in July 1968 at a fluence of about 1.7×10^{21} n/cm² (2.18 Mev). Measurements on these samples show no significant contraction differences for the different bars nor for the different sample sizes. Weight changes of the size-effect samples averaged 0.022 percent per 10^{21} n/cm². Such a low oxidation rate indicates little need for concern at present operating conditions for N Reactor.

There is good agreement between the contractions of these samples and those irradiated in the GETR. However, there is other evidence that suggests that the reactor moderator and small test samples contract differently; the use of small sample data for the prediction of stack distortion must therefore be qualified accordingly.

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1-E. On-Line Data Processing

ISI-609 Process Computer

The computer as a data logger went back on line on January 13, 1969, three weeks ahead of schedule.

On-Line Computer Systems Study

Work on the systems study is continuing. Data acquisition requirements for the 181-N and 182-N buildings were completed during the report period.

1-F. Billet Fabrication Studies

A total of 25 columns of Mark I-C fuel are being irradiated under PT-NR-96 for evaluation of the upset forging billet fabrication technique. Nine columns have reached approximately 70 percent of goal, and are scheduled for discharge in March 1969. The remaining 16 columns have reached only 10 percent of goal.

Twelve columns of Mark I-C fuel fabricated from direct-cast billets were loaded into N Reactor in January.

Mission 2 - Coproduct

It was discovered during the report period that a large portion of the LiAlO_2 target elements picked for analysis at Savannah River had not been kept segregated as intended and were effectively unidentifiable. This required a replacement selection of the Mark II-A targets. The accuracy of yield results will be affected to the extent that the new selection may not have seen a 100 percent coproduct environment. Preparation for shipment is essentially complete.

Mission 8 - Nuclear Safety

8-A. N Meteorological System

The meteorological system has been operational since December 18. However, various problems have arisen as expected with any new data logging and EDP system of this magnitude.

The computer filing of the data has lagged, partly because of temporary unavailability of programmer assistance and partially due to the changing over of the executive system on the UNIVAC 1108.

The production code of MET 000 is being set up, and shortly the tapes will be processed without the need of help from the programmer.

8-B. Metal-Water Reaction and Fission Product Release Studies

Shakedown tests have been started in the 324-D hot cell utilizing unirradiated N Reactor fuel elements. The two tests completed to date indicated a need for rearrangement of the in-cell equipment. Further tests will be done with the new arrangement prior to initiating tests with irradiated fuel.

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The first metal-water reaction test with long-cooled irradiated N Reactor fuel was performed on January 16. The fuel was heated, in a flowing steam atmosphere, beyond the clad failure temperature, but slightly below the uranium melt temperature. Time-sequence photography was used during the test to develop techniques of correlating fuel physical changes with the measurements of the extent of metal-water reactions. The test was useful only for development of testing techniques, as the thermal conductivity meter used to measure the percentage of hydrogen evolved failed early in the test. In addition, there was early indication of fission product release near the normal in-reactor fuel operating temperature. This indicates that either mechanical damage from fuel discharge may have occurred, or cladding deterioration may have taken place during the 20-month storage in the N Reactor basin. In the past a detailed inspection of fuel integrity prior to these tests had not been considered necessary; it now appears this step will have to be added.

8-C. N Reactor and K Reactor Licensability Program

In accord with studies of the status and future of the reactor licensability program, the program has been rescoped to provide a more comprehensive look at the reactor safety status of the N and K reactors.

At the N Reactor, renewed emphasis will be given to accident mitigation studies in addition to the accident prevention approach taken by the Phase I part of the Effluent Control Project. Several studies now in progress will define the need for further plant improvements to be provided under Phase II of the Effluent Control Project. In addition, a previously scoped quality assurance program has been included..

To aid in the licensability studies now in progress, a categorization of N Reactor critical systems was issued.

ENGINEERING & TECHNOLOGY - N REACTOR

Project Engineering - Reactor Facilities

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

ENGINEERING & TECHNOLOGY - FUELS & TARGETS

Mark IV Spike Fuels

Ten Mark IV outer fuel tubes and seven Mark IV inner fuel tubes, both of 125 Metal, were extruded. These fuels will be processed to obtain 30 finished fuel assemblies with flat end caps and 30 assemblies with the chevron cap design. The fuels will be manufactured to the 21-inch length. Reactor charging is planned for March 1969 and will be authorized by a Production Test.

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The purpose of this test is to: (1) compare the flat and chevron end cap designs on Mark IV fuel, and (2) confirm the design of a Mark IV spike fuel utilizing 125 Metal for both the inner and outer tube. The current Mark I spike fuel uses 125 Metal in the outer tube and 94 Metal for the inner tube. A total of four columns are planned for this test.

Butt Shearing in the Co-Extrusion Process

One additional benefit has been found from the butt shearing development reported last month. The uranium in the end defects produced prior to the butt shearing development was returned to Fernald as Zirnlo feed. In this new process, the conical butt, after ejection from the press, contains the uranium which was normally found as the extrusion end defect. It also contains the copper-silicon from the preshape and the end plate. Since this uranium was found to be essentially free of contaminants, it can be returned to Fernald as remelt scrap which has a value of \$1.50 per pound more than Zirnlo feed. The conical butt defect of inner extrusions has an average weight of 10.2 pounds, of which 5.1 pounds of uranium can be salvaged in chunk form merely by exerting hydraulic pressure on the base while supporting the cone in a ring or sleeve which contacts the outer parameter. The salvageable copper-silicon from this process amounts to approximately 5 pounds.

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IRRADIATION SERVICESFUEL TECHNOLOGYMixed Carbide Fuel Development - United Nuclear Corporation

The two United Nuclear U-PuC fuel capsules have been received and are undergoing pre-irradiation tests. The capsules will be charged during the next KW Reactor outage. These irradiations are intended to evaluate mixed carbide fuel swelling and fission gas release characteristics, and are part of a program for development of fast reactor fuel.

MATERIALS DEVELOPMENTIrradiation Damage to Reactor Metals - Battelle-Northwest

Three Battelle-Northwest creep rate measurement capsules were discharged from side General Purpose facilities at KW Reactor.

One Battelle-Northwest tensile specimen capsule was discharged from the Snout facility spider at KW Reactor.

ROUTINE IRRADIATIONS

Seventy-nine Quickie activation analysis capsules were irradiated in the K and C facilities for Battelle-Northwest.

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ADMINISTRATION - GENERALPLANS FOR REACTOR CUTBACKS

On January 15 the Company was advised by the AEC that, in response to President Johnson's FY 1970 budget, it has prepared tentative plans for placing C Reactor in standby and operating the two K reactors alternately. These plans were noted to be subject to review by the incoming Administration.

This announcement has resulted in considerable effort in making tentative plans for estimating the Company's funding situation for the remainder of FY 1969 and for FY 1970. A review of the timing, manpower, and cost effect of changes, which would be required to meet the funding limitations, was presented to Mr. Baranowski and his staff at Germantown on January 24.

APPROVAL LETTERS

At the close of the reporting period the following requests had not been approved by AEC-RL:

<u>ATD Number</u>	<u>Subject</u>	<u>Date of Transmittal to AEC-RL</u>
AP-39	Pension Plan portion of letter entitled "Pension Plan, Salaried Savings Plan and Wage Savings Plan"	January 12, 1966
ATD-130	Exempt Overtime Appendix "B" Modification MO-13	December 11, 1968
ATD-156 Add. #4	DUN Education and Training Program--Tuition Refund Plan	November 13, 1968
ATD-175	Overhead Charges Applicable to Diversification Work to be Used in Determining Credits to the AEC	December 19, 1968
ATD-176	Conduct of Employees and Consul- tants of AEC Cost-Type Contractors and Certain Other Contractors	December 20, 1968

AEC COST IMPROVEMENT REPORT

The Semiannual Cost Improvement Report to the AEC was issued on January 17. Annualized savings of \$490,000 plus tangible savings of \$40,800 a year from the Suggestion System were reported.

EQUAL EMPLOYMENT OPPORTUNITY

Cancellation of a requisition for twenty-two Metal Handlers, in light of the reactor cutback announcement, nullified the planned addition of fifteen minority group members reported last month. Also dropped were plans to add an EEO Specialist to the Employee Relations staff; this required cancellation of one confirmed and two open invitations to visit that had been extended to Negro candidates.

Two of the ten new employees hired in January were Spanish-American. Of the two nonexempt promotions, one was a minority group member.

SAFETY

No personnel radiation exposures exceeded operational control.

Month-end safety statistics were:

Disabling injuries - January	0
Days since last disabling injury	92
Man-hours since last disabling injury	870,000

EMPLOYMENT SUMMARY

DUN personnel totals and employee allocations as of December 31 and January 31 are shown in Appendix B.

APPENDIX A

PROJECT STATUS SUMMARY - REACTOR FACILITIES

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Number & Title	Authorized Funds - \$	Percent Complete		Status
		Design	Construction	
<u>Single-Pass Reactors</u>				
DCE-505, Boiler Control Improvements - 165-KE & KW	410,000	100	48	Construction Schedule - Rev. No. 1 submitted to AEC-RL January 9 for approval. Equipment installation work resumed January 13.
DAP-510, Discharge Chute Clearing Equipment - K Reactors	190,000	100	70	Fuel conveyor completed and tested. Equipment for first reactor instal- lation will be ready in February.
DAP-516, Storage Building Addition - 105-KE & KW	142,000	5	0	Design schedule issued January 2 (approved December 17, 1968). Supple- mental information transmitted to the AEC-RL December 26 for Vitro's use. Vitro's design activity was stopped January 17 in line with production cutback planning.
DAE-518, Effluent Radioiodine Monitor - C, KE & KW Reactors	100,000	86	0	All Title II design drawings and specifications have been issued for comment. DUN comments have been submitted to Vitro/HES.
<u>N Reactor</u>				
GCP-406, Improved Safety Platforms and Accesses - 100-N Area	300,000	100	91	Installation of reactor platforms in progress as outage time permits. Con- struction schedule revised January 8, to extend completion to July 15, 1969.

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PROJECT STATUS SUMMARY - REACTOR FACILITIES (cont'd)

Number & Title	Authorized Funds - \$	Percent Complete		Status
		Design	Construction	
GCE-408, W, C & D Elevator Safety Improvements - 105-N	90,000	100	30	Project activity was halted December 13 after continued testing of pressure switches confirmed that performance was not satisfactory. Tests continued and an arrangement was found which provided satisfactory response time and repeatability. Vitro has been requested to provide a revised design based on successful tests.
GCP-411, Effluent Control Program - 100-N Area	1,780,000	100	0	Design was completed December 31, 1968, with exceptions. Total funds as per J. A. Jones' estimate released on first five sections. Construction started January 20. Project proposal revision in progress to extend completion date and reduce scope to fit available funds. Lump sum packages for the Emergency Dump Tank (V) and Diesel Fire Protection (IV) have been forwarded to J. A. Jones.
DCE-519, Replacement of Bridge Crane and Hoist System with New Crane System - 105-N Storage Basin	269,000	40	0	Representatives of American Monorail and Twin City Monorail visited the 105-N Storage Basin in connection with request for preliminary crane price quotations.

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DUN-5254

APPENDIX BEMPLOYMENT SUMMARY
(with employee allocations)

	<u>12/31/68</u>	<u>1/31/69</u>
<u>CONTRACT PERSONNEL</u>		
<u>02 Programs</u>		
Douglas United Nuclear	1843	1841
Assisting other Contractors	<u>14</u>	<u>14</u>
Total - 02	1857	1855
<u>Other Programs under AEC Contract</u>		
Assisting other Contractors and WPPSS	58	54
Special Irradiations	<u>10</u>	<u>11</u>
Total - Other Programs	68	65
Total Contract Personnel	<u>1925</u>	<u>1920</u>
<u>COMMERCIAL ACTIVITIES PERSONNEL</u>	4	5
TOTAL FORCE	<u>1929</u>	<u>1925</u>

FEATURE REPORTTECHNICAL EVALUATION OF 3560 KW INCIDENTINTRODUCTION

This report summarizes the theoretical computation of conditions in channel 3560 KW following flow blockage, the conditions found when the fuel and tube were inspected after removal from the reactor, and potential application of the information gained to analyses of other coolant loss accidents.

A low trip of the pressure monitor on process tube 3560 caused the KW Reactor to scram at 11:23 p.m. on June 19, 1968. The object that had lodged in the inlet venturi throat resulted in nearly instantaneous, complete flow stoppage. Following the scram, the pressure monitor remained at the low trip point for about one minute and then cycled between the low and high trip points. A gauge valved to the pressure monitor showed the rapid pressure oscillations were from 25 to 200 psig. Within a few seconds following the pressure trip, a high tube outlet temperature trip also occurred. The outlet temperature then cycled between 20 and 134 C. Seven minutes following the reactor scram, a fission product release was detected by the rupture monitor for header 35 and by the effluent cooling water gamma monitor. The release indicated the fuel had been damaged. Increased activity in the reactor gas system about 30 minutes after the scram indicated the Zircaloy-2 process tube had also been damaged.

Failure in attempts to force water into the process tube inlet, several hours after the reactor scram, indicated the fuel had plugged the tube tightly enough to prevent significant water leakage either through the tube or through the tube breaks into the reactor. This was supported by the lack of evidence of water in the reactor.

The KW Reactor had been operating only about 18 hours following a two-week outage for charging of an E-Q (enriched and thoria) loading when the flow stoppage occurred. The 51 pieces of 94 Metal in the tube 3560 had been freshly charged during the outage. This is the normal enriched metallic uranium fuel used in the K reactors, and it has aluminum cladding bonded with an aluminum-silicon alloy. The Zircaloy-2 process tube was originally installed in November of 1963.

Predictions and evaluations of the condition of the fuel and tube in process channel 3560 were carried out for guidance in removing the damaged fuel and for improving calculational bases for other postulated nuclear safety events. A more detailed report on the technical evaluation of the tube 3560 KW incident will be issued later as DUN-5000 Pt. 3.

PREDICTION OF FUEL AND TUBE CONDITIONS

Within a few hours after the tube 3560 KW flow stoppage, a preliminary estimate

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was made of the condition of the fuel in the channel. The ALOC (Accidental Loss of Coolant) computer program was used to make this estimate assuming an ideal flux distribution, high graphite temperatures, and other approximate input data because more detailed information was not readily available. This conservative calculation of the fuel might have approached the uranium melting point. However, the low fission product release indicated fuel melting probably had not occurred.

To obtain a more accurate estimate of the fuel and tube condition, specific input data matching the operating conditions of tube 3560 were needed. Measurement of the gamma flux profile of fuel removed from an adjacent tube (3559) provided a basis for calculating the initial decay heat power of each element in tube 3560. The 3560 channel graphite temperature was estimated from several recordings of temperatures at other locations, and the original tube-to-graphite clearances were obtained from drawings. These data were used to compute more refined temperature transients. The maximum temperature in the column fuel was computed to be about 850 C at the 35th element from the outlet end, nearly 300 C below the uranium melting point (1130 C). In the computed temperature transient for this element, the temperature would have risen to approximately 800 C in 5 minutes, remained above 800 C for 20 minutes and then cooled to below 660 C in about one hour. These computations included the significant assumption that the heat addition from the U-Al reaction, which had been observed in experiments, would not significantly raise the fuel temperature. A plot of the maximum predicted fuel temperatures is shown as the upper curve in Figure 1 (appended). The computed temperature transients were used in laboratory experiments which simulated the transient fuel conditions in tube 3560. Based on these experiments with unirradiated fuel, it was predicted that the fuel elements in tube 3560 which reached maximum temperatures above about 700 C would have generated sufficient uranium-aluminum reaction products to plug the tube during the heating cycle.

A computer program was later developed to show the fuel cooling effect of part of the cold water in the upstream end of the tube escaping past the fuel. The lower curve in Figure 1 shows the lower maximum temperatures that were computed with such cooling. Because this upstream cooling flow would have started before the fuel reached the uranium-aluminum reaction temperature, some reduction in maximum fuel temperatures would be expected. The pressure surges which continued for about 2 hours indicate that some water remained trapped in the upstream section of the tube.

The maximum process tube temperature was calculated to be about 50 C cooler than the maximum fuel temperature. Using the predicted maximum temperature of the 35th fuel element from the tube outlet, 850 C, as a basis, the maximum calculated tube temperature was 800 C. Although annealing would be expected, no significant tube damage would be expected at this temperature. However, it was apparent from the activity which leaked into the reactor that the Zircaloy-2 tube had been damaged. Annealing would reduce the tube strength. In addition, it was postulated that significant quantities of uranium and zirconium hydrides would be formed in the channel, creating a potential fire hazard during fuel removal. It was especially important that a hydride fire be precluded as it might otherwise ignite the uranium and release more fission products.

TUBE EXAMINATION

When the Zircaloy-2 process tube was inspected after removal, it was found to have several short longitudinal splits at the junctions between some of the fuel elements. The location of these splits was apparently due to the greater fuel swelling which occurred at the ends. Figure 2 shows a typical split at the junction of the 20th and 21st elements from the outlet end of the charge. The circumferential marks on the tube were made by a core drill used during recovery operation. Measurement of the tube at this split point showed about 7 percent elongation had occurred before the tube ruptured. This is in the range that would be expected for this 35 percent cold worked tubing if tested at room temperature after four and one-half years of irradiation.

The fact that the elongation would be expected to increase with temperature above about 400 C suggests that the rupture might have occurred before the temperature rise. This is not consistent, however, with the swelling expected with uranium-aluminum interaction above 660 C. An increase in the reactor gas activity about 20 to 30 minutes after the tube 3560 flow was stopped indicated the tube rupture occurred after the temperature increased substantially.

Photomicrographs of tubing samples taken at several points along the tube showed that 25 to 75 percent of the cold worked grains had recovered their normal (fully annealed) shape. Comparison of this with recovery of 35 percent cold worked tubing during laboratory annealing indicates that the tube had been heated to temperatures in the range of 600 to 700 C during the transient. Although the hydrogen content of the tubing had increased, it was still low enough that the tubing strength should not have been appreciably affected. It is apparent from these tube analyses that essentially all of the tube damage was caused by fuel swelling. Using the 50 C temperature difference predicted in the theoretical temperature transient computations as a basis, the maximum estimated fuel temperature would be in the range of 750 C.

FUEL EXAMINATION

Determination of the maximum temperature the fuel elements in tube 3560 had reached was based on laboratory tests in which unirradiated fuel was heated to simulate the predicted temperature cycle. These tests showed AlSi bond voids when the bond melting point of 580 C was exceeded. When the aluminum melting point of 660 C was reached the uranium and aluminum began to react, forming bulky reaction products. The next temperature gauge point was the uranium β to γ phase transformation temperature of 760 C. The growth of beta grains, which occurs when uranium cools through the 760 C temperature range, indicates that the phase change temperature was exceeded, but the growth gives little indication of the maximum temperature reached. Comparison of β grain size with that in experimentally heated fuel was used to estimate temperatures above 760 C.

The first four fuel elements removed from the inlet end of the tube had no damage and were easily removed. The first element removed from the outlet

end had exceeded the AlSi melting point. Figure 3 shows the AlSi voiding which occurs at 580 C. The second element removed from the outlet end of the tube had exceeded the AlSi melting point on the end adjacent to the first element and the aluminum melting point on the other end; this element is shown in Figure 4. The remainder of the fuel elements near the ends reached or exceeded the aluminum melting point and were progressively more difficult to remove. Most of the fuel was firmly stuck in the tube and was later removed from the reactor with the process tube. The 13th element from the outlet end, shown in Figure 5, is a typical example of the appearance of these elements after removal from the tube. The grooves near the ends of this element indicate that most of the uranium reactions occurred at the ends.

During removal of fuel from the process tube in the reactor storage basin, most of the reaction products were broken away from the uranium core. However, a few elements were taken to the Battelle Radiometallurgy Facility in sections of the tube. Examination by X-ray diffraction of the reaction products removed from these elements showed the products to be mostly UAl_3 and UAl_4 . UO_2 and UH_3 were also found in some of the reaction product samples. These latter products indicate that water was available to some elements for reaction with the uranium in the last part of the temperature excursion. This conclusion is based on the laboratory fuel heating experiments which showed little or no UO_2 and UH_3 were present after a heating cycle in a steam atmosphere. However, these experiments did not simulate the slower cooldown transient which occurred in the reactor. The shell of reaction products was observed in these experiments to tightly adhere to the uranium during the heating cycle until it cracked and broke away leaving the uranium relatively exposed during the cooling cycle below about 660 C.

The temperatures determined from the fuel examinations are plotted in Figure 6. The points were placed at the best estimate of the temperature of each element examined; the vertical segments show the estimated error range for each determination. The curves are the computed maximum fuel temperature profiles from Figure 1. A comparison of the maximum fuel temperatures determined from observations with the computed temperatures shows the observed temperatures are generally above the predicted values in the central region of the tube and below at the ends. The difference in the central region could be due to uranium-aluminum reaction heating. This heating could not be computed because reaction rate data are not available. Additional cooling at the ends would be expected because the pressure and temperature cycling indicated water was periodically flowing into the ends of the tube and steam was generated when it contacted the hot fuel. The low observed temperatures of the 40th and 41st elements from the outlet could not be fully explained. The effect of the control rod adjacent to these elements on the decay heating should have been accounted for in the gamma flux measurements on fuel from tube 3559. Other factors which were not accounted for in the computations were the lower initial fuel and graphite temperatures near the control rod. Other factors would probably be needed to fully explain these relatively low fuel temperatures.

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ACTIVITY RELEASE ANALYSIS

Charts redrawn from the activity monitors on the effluent cooling water, the reactor gas and the exhaust air systems are shown in Figure 7. These charts indicate gaseous fission products were released from tube 3560. Analysis of the water samples taken from both the liquid effluent system and the river showed about 20 Ci of iodine-131 had been released. This was about one percent of the calculated inventory of iodine-131 in tube 3560. The effluent-coolant gamma monitor reading was higher than could be attributed to radioiodines alone, indicating on the order of 3 percent of the noble gas fission products were also released with the iodine. Analysis of the reactor gas atmosphere and the exhaust air monitor recordings indicated noble gas fission products were initially released from tube 3560 into the reactor gas system about 20 to 30 minutes after the flow stoppage occurred. These gases leaked through the gas seals into the outlet face air stream and were exhausted through the confinement filters to the ventilation stack. The sharp peak in activity at 4 1/2 hours on these charts was the result of an increase in reactor gas pressure which was made to confirm that the source of activity in the exhaust air was the reactor gas and hence, leakage from tube 3560. No quantitative measure of the noble gases released was available. A total release of 0.51 Ci of iodine-131 was estimated to have occurred based on analysis of samples from the confinement charcoal filters. The release to the atmosphere was below the detection limits of the effluent air iodine-131 monitor. The gamma activity decay recorded on the reactor gas monitor charts after four hours matched the calculated decay of the noble gas fission products without the radioiodines. Since no quantitative measure of the activity released through the confinement exhaust system was available, a rough estimate of the noble gas release into the reactor was made based on the recorded gas activity. This calculation indicated about 15 percent of the tube 3560 inventory was released into the reactor. Maximum release values obtained in the laboratory fuel heating tests were much lower, indicating this estimate is probably high. Both the lack of evidence of iodine in the reactor gas activity decay and the small amount of iodine-131 found in the confinement charcoal indicate the reactor is an effective trap for iodine. Overall, the fission product release from tube 3560 KW supports the use of low release rates for unmelted uranium in accident analyses.

APPLICATION OF RESULTS

Evaluation of the tube 3560 KW incident both confirmed the present methods used in accident analyses and introduced evidence to support the use of less conservative assumptions in future accident analyses.

The relatively close correlation between the observed and calculated maximum fuel temperatures shows the ALOC computer program gives a reasonable estimate of the fuel temperature transients. Since the computer computations did not include chemical heating, the small differences in observed and computed maximum temperatures is interpreted as indicating the chemical heating up to the 850 C temperature range is not large compared to the decay heating. This evidence supports the use of a smaller estimate of the temperature rise due to chemical reaction heating than was previously assumed in accident analyses.

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Previous analyses of single tube plugging accidents have predicted uranium melting would occur.

The fuel swelling which occurred near the 660 C aluminum melting point showed a high potential for tube plugging. For this reason it appears that with AlSi-bonded elements a lower temperature should be used as the limit for the assumption that liquid coolant flow can be restored following a partial flow loss incident.

In recent accident analyses, the fission product release rates assumed for fuel which did not melt have been conservative estimates based on laboratory release rate measurements. The low release rate for iodine-131 found in the laboratory was roughly confirmed by the release from tube 3560. This supports the continued use of low fission product release rates from unmelted uranium fuel during the temperature excursion following a loss of coolant accident. The absence of a significant quantity of iodine-131 in the exhaust air emphasized the lack of release activity measurement capability. Unpredicted plateout and capture of the halogens upstream of the halogen monitor makes these isotopes an unreliable gauge for estimating the noble gas release. An activity monitor is being developed for use in the exhaust air duct downstream of the confinement filters.

Although evaluation of the tube 3560 KW incident has provided significant information for analyses of coolant loss accidents, the information is limited to the maximum temperatures reached in this incident. Data on the reaction rate of uranium and aluminum at higher temperatures and more data on the fission product release rates for molten fuel are needed to support analyses of the full range of coolant loss accidents. Collection of these data is being pursued as part of the Research & Development Program.

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Figure 1. Predicted Fuel Temperatures

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Figure 2. Typical Split in Process Tube

Figure 3. First Element with Band Melting

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Side View

End Views

Figure 4. Second Fuel Element from Outlet

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Side View

End View

Figure 5. Typical Central Fuel Element

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Figure 6. Comparison of Temperatures Determined from Observations to Predicted Temperatures

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Figure 7. Gamma Activity Monitor Charts